Docket Nos. 50-315
and 50-316

November 23, 1983

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

Dear Mr. Dolan:

The Comission has issued the enclosed Amendment No. 78 to Facility Operating License No. DPR-58 and Amendment No. 59 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 application transmitted by letter dated August 14, 1981, as supplemented by letter dated August 19, 1983.

These amendments revise the Technical Specifications with respect to decay heat removal capability and refueling cavity water level.

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

Sincerely,

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

Enclosures:
1. Amendment No. 78 to DPR-58
2. Amendment No. 59 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page



RSB:DSI
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10/26/83

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. DPR-58

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated August 14, 1981, as supplemented by letter dated August 19, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change in Technical Specifications is to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensee shall adhere to the Technical Specification for the systems, components, or operation existing at the time. The period of time during changeover of systems, components or operation shall be minimized or compensated for by suitable temporary alternatives.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

Attachment:
Changes to the Techical
Specifications

Date of Issuance: November 23, 1983

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. 59
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated August 14, 1981, as supplemented by letter dated August 19, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 59 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change in Technical Specifications is to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensee shall adhere to the Technical Specification for the systems, components, or operation existing at the time. The period of time during changeover of systems, components or operation shall be minimized or compensated for by suitable temporary alternatives.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

Attachment:
Changes to the Techical
Specifications

Date of Issuance: November 23, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. DPR-58

AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NOS. 50-315 AND 50-316

Revise Appendix A as follows:

**Unit 1**

| Remove Pages | Insert Pages |
|---|---|
| 3/4 4-1 | 3/4 4-1 |
| 3/4 4-2 | 3/4 4-2 |
| 3/4 4-3 | 3/4 4-3 |
| - | 3/4 4-3a |
| - | 3/4 4-3b |
| - | 3/4 4-3c |
| - | 3/4 4-3d |
| 3/4 9-9 | 3/4 9-9 |
| - | 3/4 9-9a |
| 3/4 9-11 | 3/4 9-11 |
| 3/4 9-12 | 3/4 9-12* |
| B 3/4 4-1 | B 3/4 4-1 |
| B 3/4 9-1 | B 3/4 9-1* |
| B 3/4 9-2 | B 3/4 9-2 |
| B 3/4 9-3 | B 3/4 9-3 |

* Included for convenience

**Unit 2**

| Remove Pages | Insert Pages |
|---|---|
| 3/4 4-1 | 3/4 4-1 |
| 3/4 4-2 | 3/4 4-2 |
| 3/4 4-3 | 3/4 4-3 |
| - | 3/4 4-3a |
| - | 3/4 4-3b |
| - | 3/4 4-3c |
| - | 3/4 4-3d |
| 3/4 9-7 | 3/4 9-7* |
| 3/4 9-8 | 3/4 9-8 |
| - | 3/4 9-8a |
| 3/4 9-10 | 3/4 9-10 |
| B 3/4 4-1 | B 3/4 4-1 |
| - | B 3/4 4-1a |
| B 3/4 9-1 | B 3/4 9-1* |
| B 3/4 9-2 | B 3/4 9-2 |
| B 3/4 9-3 | B 3/4 9-3 |

*Included for convenience

## 3/4.4 REACTOR COOLANT SYSTEM

### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

#### STARTUP AND POWER OPERATION

#### LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

#### SURVEILLANCE REQUIREMENT

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.5

## LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,
4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

## SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

## LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*
4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*
5. Residual Heat Removal - East,**
6. Residual Heat Removal - West,**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 188 °F unless 1) the pressurizer water volume is less than 62.00% of span or 2) the secondary water temperature of each steam generator is less than 50 °F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

**The normal or emergency power source may be inoperable in MODE 5.

***All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10 °F below saturation temperature.

## SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 25% of wide range instrument span at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

3/4.4.1 REACTOR COOLANT LOOPS

THREE LOOP OPERATION (N-1)

LIMITING CONDITION FOR OPERATION

3.4.1.4 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 46% 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 46% of RATED THERMAL POWER may proceed provided:

   1. The following actions have been completed with the reactor subcritical:

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

*See Special Test Exception 3.10.5.

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 $\Delta T$ channel.

2) Overtemperature $\Delta T$ 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.

ACTION (Continued)

Below P-7:#

a. Startup and Power operation below P-7 may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. Hot standby, hot shutdown, and cold shutdown operation may proceed provided at least one reactor coolant loop in operation with an associated reactor coolant or residual heat removal pump; however, operation for up to 15 minutes with no pump in operation is permissible to accommodate transition between residual heat removal pump and reactor coolant pump operation.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 With one reactor coolant loop and associated pump not in operation, at least once per 7 days determine that:

a. The applicable reactor trip system and/or ESF actuation system instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions, and

b. If P-8 interlock setpoint has been reset for 3 loop operation, its setpoint is $\leq$ 76% of RATED THERMAL POWER.

4.4.1.4.2 Within 30 minutes prior to the start of a reactor coolant pump when any RCS cold leg temperature is $\leq$ 188°F, verify that:

a. The temperature of the secondary water of each steam generator is $\leq$ 50°F above the temperature of each of the RCS cold legs, or

b. The pressurizer water volume is less than 1116 cubic feet, equivalent to less than 62% indicated on the wide range level indicator.

#A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 188°F unless 1) the pressurizer water volume is less than 1116 cubic feet (62% of span or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

## 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

### LIMITING CONDITION FOR OPERATION

3.9.8.1 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

4.9.8.1 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.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each RHR loop.

## 3/4.9.10 WATER LEVEL - REACTOR VESSEL

### LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

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.

## STORAGE POOL WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore water level to within its limit within 4 hours. The provision of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

## 3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. 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 DNBR will be maintained above 1.30. 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 (51 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 188°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

## 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.

## 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

## 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

## 3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

## 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

## 3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

BASES

### 3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies 2) each crane has sufficient load capacity to lift a control rod or fuel assembly and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

### 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses.

### 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

### 3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

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

BASES

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

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

Water level above the vessel flange in MODE 6 will vary as the reactor vessel head and the system internals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods.

## 3/4.9.12 STORAGE POOL VENTILATION SYSTEM

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

## 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.

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

### STARTUP AND POWER OPERATION

### LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

### SURVEILLANCE REQUIREMENT

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.4

## HOT STANDBY

### LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,
4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

### SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*
4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*
5. Residual Heat Removal - East,**
6. Residual Heat Removal - West **

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 152 °F unless 1) the pressurizer water volume is less than 62.00% of span or 2) the secondary water temperature of each steam generator is less than 50 °F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

**The normal or emergency power source may be inoperable in MODE 5.

***All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10 °F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 25% of wide range instrument span at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

## 3/4.4.1 REACTOR COOLANT LOOPS

### THREE LOOP OPERATION (N-1)

### LIMITING CONDITION FOR OPERATION

3.4.1.4 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 $\Delta T$ channel.

2) Overtemperature $\Delta T$ 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. STARTUP and POWER operation may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. HOT STANDBY, HOT SHUTDOWN, and COLD SHUTDOWN 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

4.4.1.4.1 With one reactor coolant loop and associated pump not in operation, at least once per 7 days determine that:

a. The applicable reactor trip system and/or ESF actuation system instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions, and

b. If the P-8 interlock setpoint has been reset for 3 loop operation, its setpoint is $\leq$ 76% of RATED THERMAL POWER.

4.4.1.4.2 Within 30 minutes prior to the start of a reactor coolant pump when any RCS cold leg temperature is $\leq$ 152°F, verify that:

a. The temperature of the secondary water of each steam generator is $\leq$ 50°F above the temperature of each of the RCS cold legs, or

b. The pressurizer water volume is less than 1116 cubic feet, equivalent to less than 62% indicated on the wide range level indicator.

## CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

### LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

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

### SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be $\leq$ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

*Shared system with D. C. COOK - UNIT 1

## 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

### LIMITING CONDITION FOR OPERATION

3.9.8.1 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

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

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each RHR loop.

## 3/4.9.10 WATER LEVEL - REACTOR VESSEL

### LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

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.

## 3/4.4.2 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.

## BASES

### 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

### 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

### 3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

### 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

### 3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

## 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).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 152 °F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50 °F above each of the RCS cold leg temperatures.

## BASES

### 3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

### 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage rocks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

### 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

## 3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

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

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

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assemble. The minimum water depth is consistent with the assumptions of the accident analysis.

Water level above the vessel flange in MODE 6 will vary as the reactor vessel head and the upper intervals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods.

## 3/4.9.12 STORAGE POOL VENTILATION SYSTEM

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

## 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.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. DPR-58

# AND AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-74

# INDIANA AND MICHIGAN ELECTRIC COMPANY

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

# DOCKET NOS. 50-315 AND 50-316

## Introduction

By letter dated August 14, 1981, the Indiana and Michigan Electric Company (the licensee) requested revisions to the Technical Specifications for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The proposed revisions would assure that redundant decay heat removal capability is maintained and would provide a minimum water level over fuel assemblies during refueling operations. By letter dated August 19, 1983, the licensee provided supplemental information on the surveillance requirements on the residual heat removal (RHR) system.

## Discussion and Evaluation

NRC letter dated June 11, 1980, requested that all PWR licensee's amend their Technical Specifications with respect to reactor decay heat removal capability. The basis for this request was founded in a number of events that have occurred at operating PWR facilities where decay heat removal capability has been seriously degraded due to inadequate administrative controls utilized when the plants were in shutdown modes of operation. IE Bulletin 80-12 requested the licensee to immediately implement administrative controls which would ensure that proper means are available to provide redundant methods of decay heat removal. The proposed Technical Specifications are to provide long term assuance that redundancy will be maintained.

To assist in the review of the proposed Technical Specifications on decay heat removal capability, our contractor, EG&G Idaho, Inc., prepared the attached Technical Evaluation Report (TER) dated February 1981. The criteria used by EG&G Idaho to determine acceptability was the Standard Technical Specification which was also attached to the NRC letter dated June 11, 1980. This Safety Evaluation adopts that TER and the findings. We have further reviewed the the proposed surveillance 4.9.8.1 which states that the RHR flow will be monitored every 24 hours rather than the 4 hours in the model Technical Specifications. The licensee's letter dated August 19, 1983 addresses this change; our evaluation is as follows.

The RHR system provides the decay heat removal capability for the Unit following shutdown. With the reactor vessel head in place, there are other means to remove decay heat, i.e., the steam generators. Once the head is removed, the RHR system is the principal means of removing decay heat and some minimum flow should be available and monitored. The licensee notes that the shortest time following shutdown until the head is removed was 214 hours. The licensee's calculations with 2000 gpm flow shows that the RHR in this degraded mode (normal flow is greater than 3000 gpm) will still provide the required heat removal capability. The 2000 gpm flow was chosen for the calculation because it is the lowest flow that could go undetected in the control room for any length of time. The Donald C.Cook Nuclear Plant has a control room alarm on 2000 (and decreasing) gpm flow in the RHR.

We have reviewed the licensee's position and find it acceptable. It is further noted that upon entering Mode 6 refueling, the RHR must be operable and the initial surveillance checks to verify the 3000 (and greater) gpm flow would assure that more than sufficient flow was available at the time of greatest decay heat by the fuel in Mode 6. We find the licensee's proposed Technical Specifications to monitor the flow every 24 hours for 3000 (or greater) gpm flow to be acceptable.

By letter of August 15, 1980 we advised the licensee of changes that had been made in Westinghouse Standard Technical Specifications. These changes require at lease 23 feet of water over the top of the reactor pressure vessel flange during movement of fuel assemblies or control rods. This requirement assures there is sufficient water above the fuel to provide a temporary decay heat sink should the RHR system fail and to assure that fuel assemblies can be transferred out of the reactor pressure vessel with sufficient water coverage to prevent exposure of fuel handlers. In their letter dated August 14, 1981, the licensee proposed Technical Specifications to assure sufficient water above the fuel. In our review of the proposed Technical Specifications, we determined that the licensee's concern about 23 feet of water above the fuel at all times in Mode 6, refueling, was valid in that the water level will vary as the upper internals are removed or replaced. Since no fuel is being moved at this time, the water level could be significantly lower and still provide the protection to the workers above the pool. However, as fuel is being moved, we and the licensee agree that the 23 feet of water above the top of the reactor pressure vessel flange is appropriate. We have, therefore, proposed and the licensee has agreed to use the Standard Technical Specification wording and to supplement the bases of Technical Specifications 3/4.9-10 and 3/4.9-11 with the following:

> "Water level above the vessel flange in Mode 6 will vary as the reactor vessel head and the upper internals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods."

With these changes, we find the proposed Technical Specifications acceptable. In order to accommodate the changes, we have also relocated the Technical Specifications for three loop (N-1) operation; three loop operation is now a subset of the overall specification 3/4.4, Reactor Coolant System. No substantive changes were made in the N-1 specification and the Donald C. Cook Nuclear Plant is still prohibited from N-1 loop operation.

## 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 environmental 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) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 23, 1983

Principal Contributor:
D. Wigginton

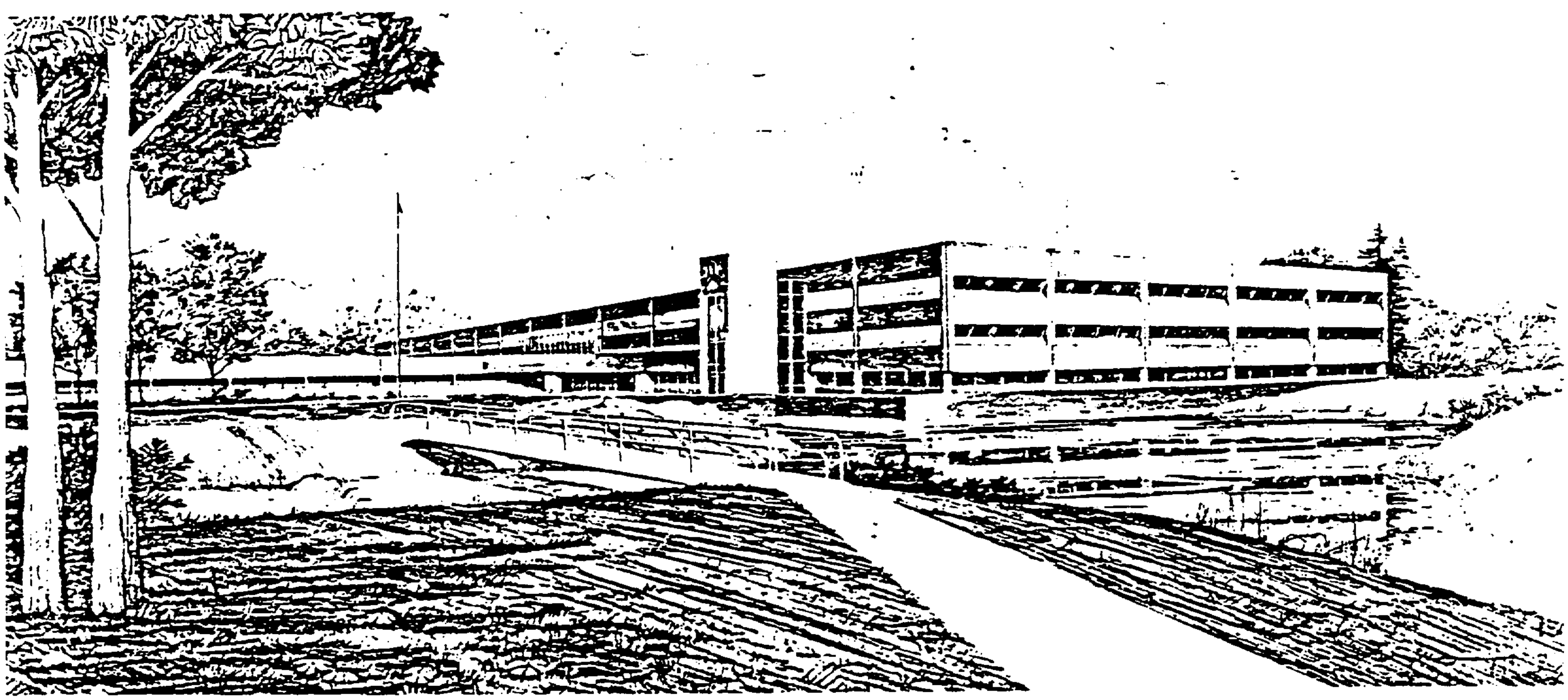
EGG-EA-5746

FEBRUARY 1982

# TECHNICAL SPECIFICATIONS FOR REDUNDANT DECAY HEAT REMOVAL CAPABILITY, DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

J. A. Steverson

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory



This is an informal report intended for use as a preliminary or working document

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EG&G Idaho, Inc.

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# INTERIM REPORT

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J. N. Donohew, Division of Licensing, NRC

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| This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final. |

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INTERIM REPORT

# TECHNICAL EVALUATION REPORT

## TECHNICAL SPECIFICATIONS FOR REDUNDANT DECAY HEAT REMOVAL CAPABILITY
## DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

Docket Nos. 50-315 and 50-316

February 1982

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Reliability and Statistics Branch
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TAC No. 42093
and 42094

## ABSTRACT

This report reviews the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, proposed technical specifications for redundancy in decay heat removal capability in all modes of operation.


## FOREWORD


This report is supplied as part of the "Selected Operating Reactor Issues Program (III)" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing, by EG&G Idaho, Inc., Reliability and Statistics Branch.

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# TECHNICAL EVALUATION REPORT

## TECHNICAL SPECIFICATIONS FOR REDUNDANT DECAY HEAT REMOVAL CAPABILITY DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

### 1.0 INTRODUCTION

A number of events have occurred at operating PWR facilities where decay heat removal capability has been seriously degraded due to inadequate administrative controls during shutdown modes of operation. One of these events, described in IE Information Notice 80-20,[1] occurred at the Davis-Besse, Unit No. 1 plant on April 19, 1980. In IE Bulletin 80-12[2] dated May 9, 1980, licensees were requested to immediately implement administrative controls which would ensure that proper means are available to provide redundant methods of decay heat removal. While the function of the bulletin was to effect immediate action with regard to this problem, the NRC considered it necessary that an amendment of each license be made to provide for permanent long-term assurance that redundancy in decay heat removal capability will be maintained. By letter dated June 11, 1980,[3] all PWR licensees were requested to: 1) propose technical specification (TS) changes that provide for redundancy in decay heat removal capability in all modes of operation, 2) use the NRC model TS which provide an acceptable solution of the concern and include appropriate safety analyses as bases, and 3) submit the proposed TS with the bases by October 11, 1980.

Indiana & Michigan Electric Company, New York, New York, submitted proposed revisions for decay heat removal to their technical specifications for Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2,[4] on August 14, 1981. The following discussion evaluates the proposed TS and notes any differences existing between them and the model TS provided by the NRC (Appendix A). The requirements are compared for equivalent modes of operation.

### 2.0 DISCUSSION

D. C. Cook, Units 1 and 2 are four-loop Westinghouse PWR plants. Specific sections of the Westinghouse Standard Technical Specifications[5] that apply to this task are:

- 3/4.4 REACTOR COOLANT SYSTEM
- 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION
- 3/4.9 REFUELING OPERATIONS
- 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The proposed D. C. Cook technical specifications are in very close agreement with the model TS provided by the NRC. Surveillance Requirement 4.9.8.1, however, states that at least one residual heat removal loop must be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 4 hours. The proposed TS require the flow rate of at least 3000 gpm to be verified at least once every 24 hours. All other Limiting Conditions and Surveillance Requirements in the D. C. Cook proposed TS are in complete agreement with those in the NRC model TS.

## 3.0 REFERENCES


1. NRC IE Information Notice 80-20, May 8, 1980.

2. NRC IE Bulletin 80-12, May 9, 1980.

3. NRC letter, D. G. Eisenhut, To All Operating Pressurized Water Reactors (PWR's), dated June 11, 1980.

4. Indiana & Michigan Electric Co. letter, G. P. Maloney to H. R. Denton, dated August 14, 1981.

5. Standard Technical Specifications for Westinghouse Pressurized Water Reactors, NUREG-0452-Rev. 3, Fall 1980.

# APPENDIX A

## MODEL TECHNICAL SPECIFICATIONS FOR REDUNDANT DECAY HEAT REMOVAL FOR WESTINGHOUSE PRESSURIZED WATER REACTORS (PWR's)

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# 3/4.4 REACTOR COOLANT SYSTEM

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

### STARTUP AND POWER OPERATION

### LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

### SURVEILLANCE REQUIREMENT

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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* See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,

2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,

3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,

4. Reactor Coolant Loop (D) and its associated steam generator and reactor coolant pump.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

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* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least $10^{o}F$ below saturation temperature.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENT

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop (D) and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal Loop (A),**

6. Residual Heat Removal Loop (B).**

b. At least one of the above coolant loops shall be in operation.***

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to (275)°F unless 1) the pressurizer water volume is less than ____ cubic feet or 2) the secondary water temperature of each steam generator is less than ____ °F above each of the RCS cold leg temperatures.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

APPLICABILITY: MODES 4 and 5.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENT

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to (  )% at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

## 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

### ALL WATER LEVELS

### LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) 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 REQUIREMENT

4.9.8.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to (2800) gpm at least once per 4 hours.

## LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

## SURVEILLANCE REQUIREMENT

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

* The normal or emergency power source may be inoperable for each RHR loop.

BASES

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to (275)°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than (  )°F above each of the RCS cold leg temperatures.

## 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140 F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the core ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.