March 13, 1997

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INCREASED STEAM GENERATOR PLUGGING LIMIT (TAC NOS. M92587 AND M92588)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 214 to Facility Operating License No. DPR-58 and Amendment No. 199 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 26, 1995, and supplemented September 26, 1995, August 2, 1996, and February 6, 1997.

The amendments revise the TS to allow operation of Cook Unit 1 at steam generator tube plugging levels up to 30%. Additional changes to increase operating margins for both Unit 1 and Unit 2 are also included.

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

Sincerely,

Original Signed by

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316
Enclosures: 1. Amendment No. 214 to DPR-58
2. Amendment No. 199 to DPR-74
3. Safety Evaluation
cc w/encl: See next page

DISTRIBUTION: See attached list

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

March 13, 1997

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated May 26, 1995, and supplemented September 26, 1995, August 2, 1996, and February 6, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation to occur within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John B. Hickman

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 13, 1997

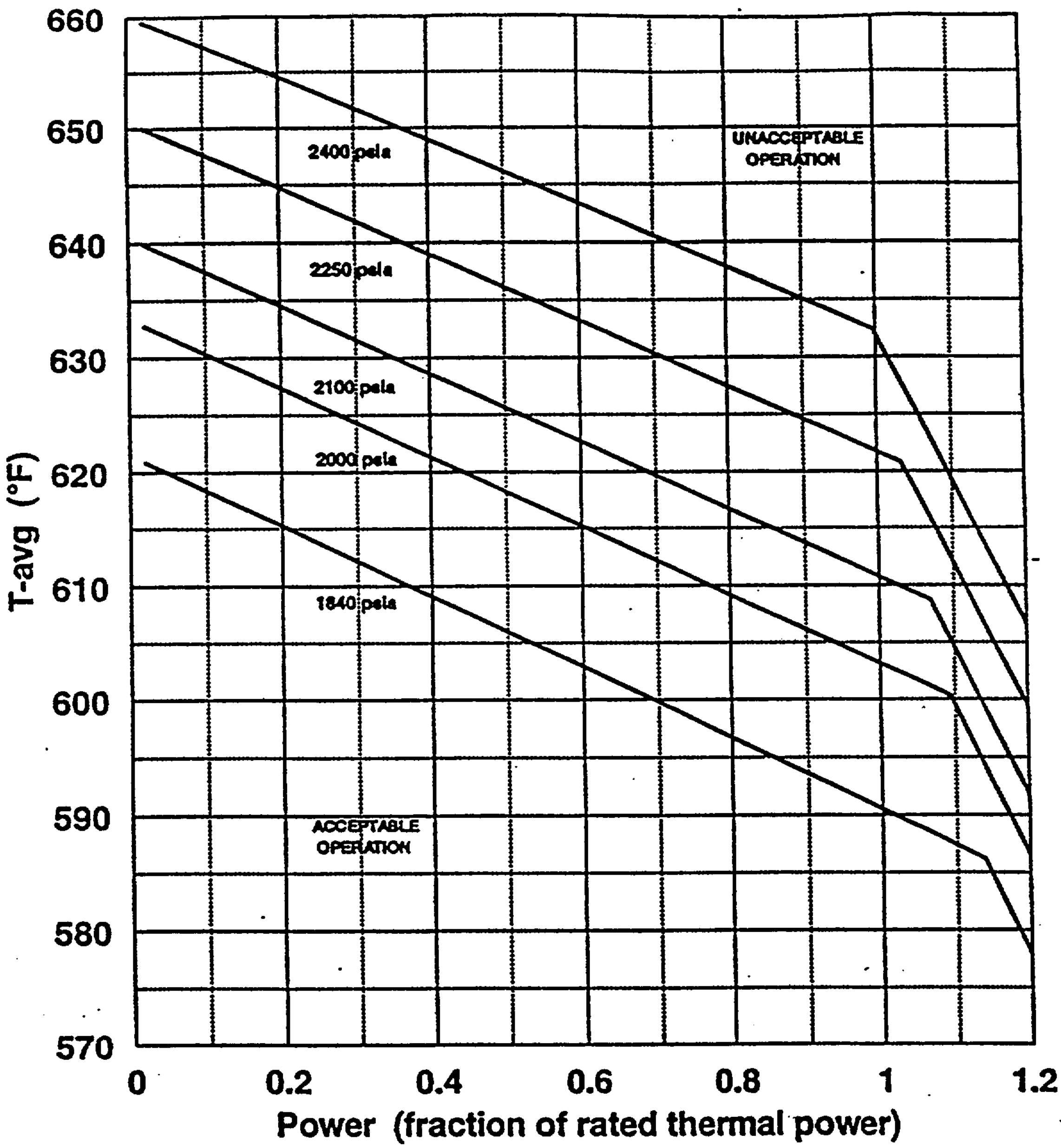
ATTACHMENT TO LICENSE AMENDMENT NO. 214

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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

REMOVE	INSERT
2-2	2-2
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
B 2-1(a)	B 2-1(a)
B 2-4	B 2-4
B 2-5	B 2-5
3/4 1-1	3/4 1-1
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
3/4 2-14	3/4 2-14
3/4 3-17	3/4 3-17
3/4 3-21	3/4 3-21
3/4 3-23a	3/4 3-23a
3/4 3-24	3/4 3-24
3/4 3-26	3/4 3-26
3/4 3-31	3/4 3-31
3/4 3-33	3/4 3-33
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
3/4 5-11	3/4 5-11
5-5	5-5
B 3/4 1-1	B 3/4 1-1
B 3/4 4-1	B 3/4 4-1
B 3/4 5-3	B 3/4 5-3
B 3/4 6-2	B 3/4 6-2



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN °F)		
1840	(0.02, 620.86),	(1.136, 586.17),	(1.2, 577.94)
2000	(0.02, 632.79),	(1.094, 600.31),	(1.2, 586.52)
2100	(0.02, 639.85),	(1.068, 608.72),	(1.2, 591.77)
2250	(0.02, 649.96),	(1.031, 620.83),	(1.2, 599.40)
2400	(0.02, 659.52),	(0.996, 632.42),	(1.2, 606.63)

Figure 2.1-1 Reactor Core Safety Limits

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is 1/4 Reactor Coolant System total flow rate from Table 3.2-1.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2 \left[\frac{1+\tau_1 s}{1+\tau_2 s}\right] (T - T') + K_3 (P - P') - f_1 (\Delta I)]$

where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 576.3°F)
	P	=	Pressurizer pressure, psig
	P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = 1.17$
$K_2 = 0.0230$
$K_3 = 0.00110$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For q_t - q_b between -37 percent and +3 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of (q_t - q_b) exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of (q_t - q_b) exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \; [K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2 (\Delta I)]$

where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 563.0°F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; K_6=0 for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
	S	=	Laplace transform operator
	$f_2 (\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

2.1 SAFETY LIMITS

BASES

4 Loop Operation

Westinghouse Fuel
(15x15 OFA)

(WRB-1 Correlation)

	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17
Design Limit DNBR	1.23	1.22
Safety Analysis Limit DNBR	1.40	1.42

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

*represents typical fuel rod

**represents fuel rods near guide tube

SAFETY LIMITS

BASES

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBRs will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steamline break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the reactor protection system.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

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

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

*See Special Test Exception 3.10.1.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the water level volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

*For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORATED WATER SOURCES - OPERATIONS

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

TABLE 3.2-1
DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	≤ 579.3°F*
Pressurizer Pressure	≥ 2050 psig**
Reactor Coolant System Total Flow Rate	≥ 341,100 gpm***

*Indicated average of at least three OPERABLE instrument loops.

**Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

***Indicated value.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1,2,3##	
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	14* 15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH					
T_{avg} -- Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3#	14*
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3#	15
e. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3#	14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3#	15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1,2,3	14*

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels greater than or equal to 1915 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 2 of 4 T_{avg} channels less than or equal to Setpoint. Setpoint greater than or equal to 541°F	P-12 allows the manual block of safety injection actuation on low steam line pressure causes steam line isolation on high steam flow. Affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset point, prevents or defeats the manual block of safety injection actuation on low steam line pressure.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a. Manual Initiation	---------- See Functional Unit 9 ----------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d. Pressurizer Pressure--Low	Greater than or equal to 1815 psig	Greater than or equal to 1805 psig
e. Differential Pressure Between Steam Lines--High	Less than or equal to 100 psi	Less than or equal to 112 psi
f. Steam Line Pressure--Low	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Radioactivity--High Train A (VRS-1101, ERS-1301, ERS-1305)	See Table 3.3-6	Not Applicable
3. Containment Radioactivity--High Train B (VRS-1201, ERS-1401, ERS-1405)	See Table 3.3-6	Not Applicable
4. STEAM LINE ISOLATION		
a. Manual	See Functional Unit 9	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	Less than or equal to 2.9 psig	Less than or equal to 3 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	Less than or equal to 1.42 x 10^6 lbs/hr from 0% load to 20% load. Linear from 1.42 x 10^6 lbs/hr at 20% load to 3.88 x 10^6 lbs/hr at 100% load.	Less than or equal to 1.56 x 10^6 lbs/hr from 0% load to 20% load. Linear from 1.56 x 10^6 lbs/hr at 20% load to 3.93 10^6 lbs/hr at 100% load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. Steam Line Pressure--Low	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	Less than or equal to 67% of narrow-range instrument span each steam generator	Less than or equal to 68% of narrow-range instrument span each steam generator

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure--High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	1, 2, 3
f. Steam Line Pressure--Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3,
c. Containment Pressure--High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	S	R	M	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional surveillance requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water.

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

5.4 REACTOR COOLANT SYSTEM (Continued)

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is approximately 12,466 cubic feet at 0% steam generator tube plugging and 11,551 cubic feet at 30% steam generator tube plugging at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water.

b. A nominal 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. 1. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 plus or minus 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the safety analysis limit during all normal operations and anticipated transients. 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. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

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 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.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST 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, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. 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 a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be 11.49 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.49 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 199
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated May 26, 1995, and supplemented September 26, 1995, August 2, 1996, and February 6, 1997, 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:

<u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 199 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation to occur within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 13, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 199

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

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

REMOVE	INSERT
3/4 1-1	3/4 1-1
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
3/4 5-11	3/4 5-11
B 3/4 1-1	B 3/4 1-1
B 3/4 5-3	B 3/4 5-3
B 3/4 6-2	B 3/4 6-2

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{AVG} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

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

*See Special Test Exception 3.10.1.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 4300 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,

2. A minimum boron concentration of 2400 ppm, and

3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

*For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water,

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available.

With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST 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, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or steam line rupture. The limits of 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 a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 80°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be 11.49 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.49 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 199 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

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

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated May 26, 1995, and supplemented September 26, 1995, August 2, 1996, and February 6, 1997, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The proposed changes are primarily to support operation of Unit 1 at steam generator tube plugging (SGTP) levels up to 30%. In addition, the licensee has performed analyses and evaluations to support increased operating margins for Unit 1. Some of the margin changes have been proposed for both Unit 1 and Unit 2. Finally, one miscellaneous change has been proposed to make one Unit 1 specification more nearly like the corresponding Unit 2 specification, and one administrative change to maintain consistency of Unit 2 acceptance criteria. The September 26, 1995, August 2, 1996, and February 6, 1997, supplements provided clarifying information that did not expand the scope of the initial application and did not change the staff's proposed no significant hazards determination.

2.0 EVALUATION

2.1 Proposed Changes to the Safety Analysis

The primary purpose of the licensee's submittal is to request approval to operate Cook Nuclear Plant Unit 1 with SGTP levels as high as 30% in each steam generator. Since the analysis needed to support this request involved reanalysis or evaluation of most of the events discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR), the licensee performed the analyses such that additional operating margin was achieved in several areas. In addition, a number of proposed changes that can be supported for both units, a miscellaneous change, and an administrative change have also been proposed.

In addition to addressing an increased SGTP level of 30%, the following increased operating margins were also addressed:

(1) Reduction of Safety Injection (SI) and Residual Heat Removal (RHR) discharge pressure on recirculation - The RHR and SI minimum safeguards pump head curves were reduced by 15%, an additional 5% reduction from the current analysis degradation of 10%. The charging pump head curve degradation is maintained at the current value of 10%.

(2) The emergency diesel generator (EDG) start time was increased from 10 seconds to 30 seconds.

(3) To support increased delta-T drift, the margin between the safety analysis limits (SAL) and the nominal values of the K1 and K4 gains of the Unit 1 Over-Temperature Delta-T (OTDT) and Over-Power Delta-T (OPDT) setpoint equations were adjusted.

(4) An increase in the pressurizer code safety valve (PSV) setpoint tolerance from +/-1% to +/-3%.

(5) Decreased shutdown margin for Tavg greater than 200°F.

The analyses and evaluations as discussed in Reference 1 support all of these changes, and provides the necessary documentation to support the TS changes associated with the SGTP program. The results of the accident analyses and evaluations performed by the licensee for the SGTP program demonstrate that safe operation is maintained. A summary of the conclusions of the accident analyses performed by the licensee that serve as the basis for the staff's acceptance is provided below.

<u>Large Break LOCA (LBLOCA)</u>

The LBLOCA analysis was reanalyzed for the impact of the increased tube plugging level, reduced Thermal Design Flow (TDF), loop flow asymmetry, revised ECCS flows, and the increased EDG start time. The LBLOCA analysis was not impacted by the pressurizer code safety valve tolerance increase, the revised K1/K4 values, or the decreased shutdown margin. The LBLOCA analysis was performed with the 1981 version of the Westinghouse ECCS Evaluation Model using the BASH computer code. Analysis assumptions included ECCS flow with the RHR cross-tie valves closed, a total peaking factor of 2.15, a hot channel enthalpy rise peaking factor of 1.55, and an accumulator temperature of 100°F. A full spectrum break analysis was performed at the nominal RCS conditions (initial RCS pressure of 2250 psia and initial hot leg temperature of 609.1°F) from which the limiting break discharge coefficient was determined. The limiting break was then reanalyzed at the reduced hot leg temperature and nominal RCS pressure of 2250 psia, and also at nominal hot leg temperature and an initial RCS pressure of 2100 psia. The above cases were all analyzed with minimum safety injection flow, which was determined to be limiting. The limiting break was determined to be Cd = 0.4 at the nominal hot leg temperature (Thot = 609.1°F) and a pressure of 2100 psia with minimum safety injection flow. The peak cladding temperature (PCT) was calculated to be 2164°F.

The LBLOCA analysis for 30% SGTP level, as discussed above, was performed using a cosine axial power distribution based on the Power Shape Sensitivity

Model (PSSM). After the licensee completed their analysis, effective October 30, 1995, the PSSM was replaced by an alternate axial power shape methodology (ESHAPE) which is based on explicit analysis of a set of skewed axial power shapes. Replacement of PSSM by ESHAPE was approved by NRC, and the use of ESHAPE methodology, in general, results in a more conservative PCT. Hence, the licensee performed an assessment (Reference 2) to determine the impact of this change on the magnitude of the PCT (2164 F) as submitted. In their assessment, the licensee included a "compensatory benefit" which was developed by Westinghouse to reduce or eliminate the PCT penalty associated with the change from PSSM to ESHAPE. This compensatory benefit results from the incorporation of the effects of leakage through the hot leg nozzle gap (HLNG) into the BASH methodology. Westinghouse submitted the HLNG methodology for staff review on July 26, 1995 (Reference 3), but the review has not yet been initiated. The Table below presents the results of the licensee's assessment:

METHODOLOGY	PCT
PSSM	2164 F
ESHAPE	2266 F
ESHAPE in conjunction with HLNG	2029 F

The above table shows that replacing PSSM by ESHAPE results in a PCT increase of 102 °F. If, however, the HLNG model is used in conjunction with ESHAPE, the PCT is 2029 °F. Since the change in PCT is significant due to the change of evaluation model from PSSM to ESHAPE, and the HLNG is not currently an NRC approved model, the licensee is required by Section 50.46(a)(3)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46(a)(3)(ii)) to submit a schedule for providing a reanalysis or take other action as may be needed to show compliance with 10 CFR 50.46 requirements. Use of the HLNG model without prior NRC review is permitted by 10 CFR 50.46; however, the staff reserves the right to review HLNG in the future.

In response to the staff's request, the licensee informed NRC by letter dated February 6, 1997, (Reference 4) that a revised submittal with an approved evaluation model will be made prior to the start up of cycle 18, which is currently scheduled to begin on April 20, 2000. At present, Unit 1 is coasting down in cycle 15. The proposed schedule for reanalysis appears reasonable. However, the staff may review the HLNG methodology. If the methodology is found to be unacceptable for inclusion in the LBLOCA, the licensee will be required to take immediate action to bring the plant into compliance with 10 CFR 50.46 requirements. An accelerated schedule for reanalysis of the plant with an acceptable evaluation model may also be required.

Small Break LOCA

The small break LOCA analysis was reanalyzed for the impact of the increased tube plugging level, reduced TDF, loop flow asymmetry, revised ECCS flows, and the increased EDG start time. The small break LOCA analysis was not impacted by the pressurizer code safety valve tolerance increase, the revised K1/K4 values, or the decreased shutdown margin. The small break LOCA analysis was performed with the Westinghouse small break LOCA ECCS Evaluation Model using the NOTRUMP code (including the recent model changes submitted in WCAP-10054-P, Addendum 2, Revision 1, which was approved by staff on August 12, 1996). The key analysis input assumptions include ECCS flows with the High Head Safety Injection (HHSI) cross-tie discharge valves closed, a total peaking factor of 2.32 and hot channel enthalpy rise peaking factor of 1.55. Other analysis input assumptions incorporated in the small break LOCA analysis are reduced hot assembly average power (Pha) and a power shape based on a reduced axial offset of +20%. A single break size analysis was performed at the previously-limiting break size of three inches. The calculation used the reduced temperature, reduced pressure operating condition. An evaluation of the break spectrum and the range of operating conditions concluded that the analyzed case would remain bounding with respect to peak clad temperature. The calculation was performed with minimum safety injection flow, which was limiting. The peak cladding temperature was calculated to be 1443°F.

LOCA Hydraulic Forcing Functions

LOCA hydraulic forces are relatively insensitive to specific SGTP levels. The D.C. Cook Nuclear Plant LOCA hydraulic forces were most recently analyzed for the Rerating Program. The RCS parameters used in the existing analysis conservatively bound the conditions at 30% tube plugging. Therefore, the existing LOCA forces analyses remain conservative relative to the SGTP Program.

Non-LOCA Analyses

The non-LOCA events were addressed by a combination of evaluations and analyses for the impact of the increased tube plugging level, reduced TDF, loop flow asymmetry, revised ECCS flows, pressurizer code safety valve tolerance increase, increased EDG start time, revised K1/K4 values, and decreased shutdown margin. The non-LOCA safety analyses were reviewed on the basis of both DNB and non-DNB acceptance criteria. All DNB event reanalyses were found to yield a minimum DNBR which remains above the limit value. The analyses demonstrate that all licensing basis criteria continue to be met and the conclusions presented in the UFSAR remain valid.

Steam Generator Tube Rupture (SGTR)

The SGTR event was analyzed for the impact of the increased tube plugging level and associated reduced TDF and loop flow asymmetry. The SGTR analysis was not impacted by any of the SGTP Program increased operating margins. The thyroid and whole body doses estimated for the Unit 1, based on the 30% SGTP evaluation, remain within a "small fraction" (10%) of the 10 CFR 100 exposure limit guidelines. Therefore, the conclusions of the UFSAR remain valid.

Post-LOCA Hot Leg Recirculation Time

The hot leg switchover to preclude boron precipitation and post-LOCA long term cooling are not adversely affected by the 30% SGTP Program. The proposed changes do not significantly affect the normal plant operating parameters, the safeguards systems actuation, the accident mitigation capabilities important to these events, or the assumptions used in the analysis of these events. The proposed changes do not create conditions more limiting than those assumed in the LOCA-related analyses.

Radiological Doses

A reanalysis of the offsite doses following a large break LOCA was performed for the increase in emergency diesel generator start time to 30 seconds. While there was a slight increase in the offsite thyroid doses, the doses are within the applicable limits. The source terms for LOCA and the fuel handling accident are unaffected by the increase in SGTP level or any of the other SGTP Program increased operating margins.

Post Accident Hydrogen Production

The licensee examined the effects of the Steam Generator Tube Plugging program on post-LOCA hydrogen generation and concluded that the values employed in the Rerating Program analysis remain bounding for the Steam Generator Tube Plugging (SGTP) Program.

Containment Analyses

The design pressure for the D. C. Cook primary containment is 12.0 psig as given in TS Bases for LCO 3.6.1.4 and 3.6.1.5 and discussed in FSAR Section 5.2.2.2 "Design Load Criteria". The previous analysis of peak containment pressure (licensee letter to NRC dated August 22, 1988) resulted in a value of 11.89 psig. As part of the Steam Generator Tube Plugging (SGTP) Program, the licensee recalculated the peak containment pressure. The revised value, based on calculations for the SGTP Program, is 11.49 psig. TS Bases for LCO 3.6.1.4 and 3.6.1.5 were revised to reflect this value. The analyses assumed a power level of 3413 (plus 2%) Mwt, which bounds the operation of both units.

The licensee reanalyzed containment integrity following a LOCA and a Main Steam Line Break to consider the impact of the increased level of tube plugging, the reduced thermal design flow, loop asymmetry, revised ECCS flows and the increased EDG start time.

The licensee also evaluated the short term containment analysis done for the Rerating Program and concluded that it bounds the case of 30% tube plugging.

For the LOCA, the licensee used previously approved methods to calculate the mass and energy released by a LOCA into containment and the resulting containment pressure. The LOCA mass and energy release calculations used the 1979 model which consists of the SATAN VI, WREFLOOD and FROTH computer codes. The results were then input to the LOTIC-1 computer program to perform the containment integrity peak pressure calculations.

In general, the peak containment pressure resulting from a LOCA decreases with increased steam generator tube plugging since the reactor coolant volume and fluid released are reduced, heat transfer across the steam generator tubes is reduced because the area is reduced (which decreases the energy of the escaping fluid) and the pressure differential upstream of the break is increased, resulting in a decrease in the break flow rate. However, for conservatism, the licensee assumed 0% tube plugging for the mass and energy release calculations in order to maximize energy released to the containment. The core rated thermal power assumed in these calculations was 3413 Mwt. The core rate thermal power assumed in the Rerating Program was 3425 Mwt. In a August 2, 1996, response to a staff question, the licensee stated that the calculated decrease in containment pressure (from 11.89 psig to 11.49 psig) is due to several reductions in margin included in the calculations performed for the Rerating Program. The revised calculations were performed with plant specific rather than generic values for the steam generator metal and the amount of fuel stored energy. In addition, more accurate data for mass and energy release during the post reflood period contributed to the decrease in margin which resulted in a lower value of peak containment pressure. The calculations still retain sufficient conservatism including the assumption of minimum safeguards and assumptions to maximize the melting of the ice bed since this depletes the ice in the shortest time.

The licensee also calculated the consequences of both a large and small Main Steam Line Break. The containment pressure calculated for a LOCA is more severe than that for the Main Steam Line Break. However, the Main Steam Line Break results in higher containment temperatures. The calculated peak containment temperature is 326°F which is within the Environmental Acceptance Criteria.

2.2 Proposed Changes to the TS

The proposed changes in the TS are discussed below.

TS pg. 2-2; Figure 2.1-1: Reactor Core Safety Limits were revised. This change was proposed to increase Unit 1 operating margin. The revised OTDT and OPDT setpoints are based upon new core thermal safety limits, which account for the effects of the Reactor Coolant System (RCS) parameter changes associated with the increased level of SGTP, using the methodology described in Reference 5. These setpoints were revised to increase the available margin between the safety analysis setpoint values and the nominal, or TS values, such that more delta-T drift could be accommodated between instrumentation calibrations during the fuel cycle. Presently, the power margin associated with the Rerating Program is being utilized to offset the delta-T drift that is being experienced during core burnup (i.e., the core power of 3411 MWt is supported by the analyses, but the plant is actually operated with a core full-power value of 3250 MWt). However, since the 30% SGTP parameters do not have this power margin available, there was a need to revise the OTDT and OPDT setpoints as part of the SGTP Program.

TS pg. 2-5; Table 2.2-1, Footnote: Design flow in footnote of Table 2.2-1 was redefined to 1/4 Minimum Measured Flow (MMF). This change was directly related to increased SGTP level. TS MMF is 1.025 times thermal design flow (TDF). The MMF employed in the licensee's DNB analysis is 1.019 times TDF. This was done to support a range of MMF's from 1.019 to 1.025 times TDF. Design flow in current TS Table 2.2-1 is MMF/4.

TS pg. 2-7; Table 2.2-1: The upper limit on T'(Indicated Tavg at Rated Thermal Power) in OTDT equation was increased to reflect licensee's analyses. This change was proposed to increase Unit 1 operating margin.

TS pg. 2-8; Table 2.2-1: The numerical value of K1 in OTDT equation was reduced from 1.32 to 1.17. This change and the change to f(delta-I) was requested to optimize Unit 1 operating margin. Some load rejection capability was sacrificed for instrumentation margin, increased allowance for core burndown effects on hot leg streaming, and an increase in the positive delta-I break point for the f(delta-I) penalty.

TS pg. 2-8; Table 2.2-1: f(delta-I) was changed to increase the region of positive delta-I which is without penalty. This change was requested to optimize Unit 1 operating margin. The previous discussion on reduction of K1 is applicable for this change also.

TS pg. 2-9; Table 2.2-1: The upper limit on T" (Indicated Tavg at Rated Thermal Power) in OPDT equation was decreased to reflect the licensee's analyses. This change was proposed to increase Unit 1 operating margin. Unit 1 is operated in a low temperature, low pressure mode to extend the life of the steam generators. Therefore, the licensee's analysis of the OPDT setpoint was analyzed with a low upper limit on T" to convert unused margin to operating margin.

TS pg. 2-9; Table 2.2-1: The allowable values in notes 3 and 4 were changed based on the licensee's calculation. These changes were proposed to increase Unit 1 operating margin.

TS pg. 3/4 1-1; Sections 3.1.1.1 & 4.1.1.1.1: The required shutdown margin was reduced. These changes were proposed to increase the operating margin of both the units. The new values were supported by the licensee's analyses, which included core response steam break (CRSB), steamline mass and energy release (SM&E) inside containment, and SM&E outside containment.

TS pg. 3/4 1-11; Section 4.1.2.3.1: The Centrifugal Charging Pump (CCP) surveillance was changed to be consistent with 10% degradation. The pump surveillance requirements were changed from "discharge" to "differential" pressure. These changes were proposed to increase the operating margin of both the units. The new surveillance criterion is supported by the licensee's analyses which included Loss of Coolant Accident (LOCA), CRSB, SM&E inside and outside containment. The value given for CCP applies to both units.

TS pg. 3/4 1-12; Section 4.1.2.4: The CCP surveillance was changed to be consistent with 10% degradation. The pump surveillance requirements were changed from "discharge" to "differential" pressure. These changes were proposed to increase the operating margin of both the units. See previous discussion for TS pg. 3/4 1-11.

TS pg. 3/4 1-15; Section 3.1.2.7: The minimum Refueling Water Storage Tank (RWST) temperature was reduced to 70°F. This change was proposed to increase the operating margin of both the units. The minimum RWST temperature for mode

5 and 6 is conservatively maintained at the same value as that required for modes 1, 2, 3, and 4.

TS pg. 3/4 1-16; Section 3.1.2.8: The minimum RWST temperature was reduced to 70°F. This change was proposed to increase the operating margin of both the units. The new value was supported by the core response large break LOCA performed by the licensee.

TS pg. 3/4 2-14; Table 3.2-1: The DNB temperature limit was increased from 570.9°F to 579.3°F. In addition, the MMF limit was reduced from 361,600 to 341,100 gpm which includes a 2.5% instrument uncertainty. These changes are directly related to increased SGTP level. Also, see the discussion for TS pg. 2-5.

TS pg. 3/4 3-17; Table 3.3-3: The Engineered Safety Feature (ESF) actuation logic to support 12% Auxiliary Feedwater (AFW) pump degradation was changed. This change was proposed to increase Unit 1 operating margin. The revised part of Table 3.3-3 incorporates the safeguards logic used in Unit 2. This will allow for the use of 12% AFW head degradation in Unit 1. All analyses, other than an "information only" feedline break analyses, have been performed by the licensee using the flow from an AFW pump with 12% head degradation. The safeguards logic itself will be modified via design change prior to implementation of these revised TS pages (i.e. before Unit 1, cycle 16). After this modification, the Unit 2 feedline break analysis using 12% degraded flow will bound Unit 1.

TS pgs. 3/4 3-21, 23a, 24, 26, 31, and 33; Tables 3.3-3, 3.3-4, and 4.3-2: Change ESF actuation logic to support 12% AFW pump degradation. These changes were proposed to increase Unit 1 operating margin. See the discussion above for the change in TS pg. 3/4 3-17.

TS pgs. 3/4 4-4, and 5; Sections 3.4.2 & 3.4.3: The pressurizer valve tolerance was increased. These changes were proposed to increase Unit 1 operating margin. The non-LOCA accidents were reanalyzed or reevaluated by the licensee based on a pressurizer valve setpoint tolerance of 3%. The licensee's analyses included loss of load, turbine trip, locked rotor/shaft break events, loss of normal feedwater, feedwater line break, and loss of all power to station auxiliaries.

TS pg. 3/4 5-5; Section 4.5.2.f.2, and 3: The RHR/SI pump surveillance requirements were changed to be consistent with 15% degradation, and the word "discharge" was replaced by "differential". These changes were proposed to increase Unit 1 operating margin. The new surveillance criteria are supported by analyses performed by the licensee.

TS pg. 3/4 5-5; Sections 4.5.2.f.2, and 3: The RHR/SI pump surveillance requirement was changed from "discharge" to "differential" pressure. This is an administrative change for Unit 2. The discharge pressure criteria in the current TS correspond to the same pump performance characteristics as the proposed differential pressure criteria. The change ensures that surveillance criteria use similar acceptance criteria.

TS pg. 3/4 5-5; Section 4.5.2.f.1: The CCP surveillance requirements were changed to be consistent with 10% degradation, and the word "discharge" was replaced by "differential" pressure. These changes were proposed to increase the operating margin of both the units. Refer to the discussion given for TS pg. 3/4 1-11 for information concerning the 10% CCP degradation.

TS pg. 3/4 5-11; Section 3.5.5: The minimum RWST temperature was reduced to 70 °F. This change was proposed to increase the operating margin of both the units. See discussion for TS pg. 3/4 1-16.

TS pg. 3/4 7-6; Sections 4.7.1.2.a & b: The AFW pump surveillance was changed to be consistent with 12% degradation. This change was proposed to increase Unit 1 operating margin. See discussion for TS pg. 3/4 3-17. The proposed surveillance criteria is identical to the criteria in the Unit 2 TS. These criteria correspond to the auxiliary feedwater flows used in all analyses for both units, except the "information only" Unit 1 feedwater line break. As noted in the discussion for TS pg. 3/4 3-17, after the changes to the Unit 1 safeguards actuation logic, Unit 1 will be bounded by the Unit 2 feedwater line break.

TS pg. 5-5; Section 5.4.2: The system volume was reduced to account for plugged steam generator tubes. This change is directly related to increased SGTP. Since the actual level of tube plugging may change each outage, a range of RCS volume corresponding to the range 0% to 30% plugging was specified: approximately 12,466 to 11,551 ft^3.

TS pg. B 2-1(a); Bases Section 2.1.1: The DNB values for fuel were changed. This change was proposed to increase Unit 1 operating margin. The values for DNBR for typical and thimble cells were revised. This change was related to the new thermal design and the new OTDT and OPDT protection trip setpoints. See also discussions for TS pages 2-2, 2-7, 2-8, and 2-9.

TS pg. B 2-4; Bases Section 2.2.1: The detail from the discussion of the OTDT protection trip was removed. This change was proposed to increase Unit 1 operating margin. The discussion of the proper normalization of T' and P' was removed. This information will be controlled administratively.

TS pg. B 2-5; Bases Section 2.2.1: The detail from the discussion of the OPDT protection trip was removed. This change was proposed to increase Unit 1 operating margin. The discussion of the proper normalization of T" was removed. This information will be controlled administratively.

TS pg. B 3/4 1-1; Bases Section 3/4.1.1.1 & 2: The required shut down margin was reduced. This change was proposed to increase the operating margin of both the units. See discussion for TS pg. 3/4 1-1.

TS pg. B 3/4 4-1; Bases Section 3/4.4.1: The DNB values for fuel were changed. The numerical value of "1.69" was replaced by the phrase, "the safety analysis limit". These changes were proposed to increase Unit 1 operating margin.

TS pg. B 3/4 5-3; Bases Section 3/4.5.5: The minimum RWST temperature was reduced to 70°F. This change was proposed to increase the operating margin of both the units. The conditions under which the reactor will remain subcritical were clarified. Specifically, large break LOCA was called out as the initiating condition and the control rods were assumed to be out instead of being inserted, except for the most reactive assembly. In addition, the explanation that a conservatively high value of the RWST temperature was included in the TS for Unit 1 was removed because the proposed value of 70°F was based on the licensee's analyses.

TS pg. B 3/4 6-2; Bases Sections 3/4.6.1.4 & 5: The peak containment pressure was changed to reflect licensee's analysis result. This change was proposed to increase the operating margin of both the units.

3.0 SUMMARY

The staff has completed its review of the documentation submitted by the licensee in support of the proposed changes in the TS and the associated Bases for the Units 1 & 2. On the basis of the evaluation presented above, the staff concludes that the proposed changes are acceptable. The staff concurs that post-LOCA hydrogen generation has been adequately addressed and finds the new containment analyses to be acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (60 FR 37095). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," WCAP 14285, Revision 1, May 1995.

2. Letter from E.E. Fitzpatrick to NRC, "Clarification of Power Shape used in LOCA Analysis," September 26, 1995.

3. Westinghouse Letter NTD-NRC-95-4477, Transmittal of Topical Reports WCAP-14404-P and WCAP-14405-NP, "Methodology for Incorporating Hot Leg Nozzle Gaps into BASH," N.J. Liparulo to USNRC Document Control Desk, July 26, 1995.

4. Letter from E.E. Fitzpatrick to NRC, "Request for Additional Information," AEP:NRC:1207C, February 6, 1997.

5. Ellenberger S.L. et al.,"Design Bases for the Thermal Overpower delta-T and Thermal Overtemperature delta-T Trip Functions," WCAP-8746, March, 1977.

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Date: March 13, 1997

March 13, 1997

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power
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500 Circle Drive
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SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INCREASED STEAM GENERATOR PLUGGING LIMIT (TAC NOS. M92587 AND M92588)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 214 to Facility Operating License No. DPR-58 and Amendment No. 199 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 26, 1995, and supplemented September 26, 1995, August 2, 1996, and February 6, 1997.

The amendments revise the TS to allow operation of Cook Unit 1 at steam generator tube plugging levels up to 30%. Additional changes to increase operating margins for both Unit 1 and Unit 2 are also included.

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

Sincerely,

Original Signed by

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316
Enclosures: 1. Amendment No. 214 to DPR-58
2. Amendment No. 199 to DPR-74
3. Safety Evaluation
cc w/encl: See next page

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AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK-UNIT 1
AMENDMENT NO. 199 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK-UNIT 2

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