DISTRIBUTION:
~~Docket File-2~~
NRC PDR
L PDR
TERA-2
NSIC
ORB#1 Rdg
HDenton
DEisenhut
RPurple
JRoe
RTedesco
GLainas
TNovak
SVarga
SMiner
CParrish
OELD
AEOD
IE-3
ACRS-16
Gray File +4
BJones-8
BScharf-10
RDiggs
CMiles
JWetmore

Docket Nos. 50-315
and 50-316

JAN 2 8 1981

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

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 43 to Facility Operating License No. DPR-58 and Amendment No. 25 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 February 22, 1980.

These amendments revise the set point for the steam generator low-low level settings.

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

Sincerely,

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

Enclosures:
1. Amendment No. 43 to DPR-58
2. Amendment No. 25 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

RECEIVED
FEB 17 1981
U.S. NUCLEAR REGULATORY COMMISSION

8102190159

Staff Evaluation Not Reviewed
Notice Reviewed only.

OFFICE	ORB#1:DL	ORB#1:DL	C-ORB#1:DL	AD-OR:DL	OELD		
SURNAME	CParrish CP	SMiner/cb SM	SVarga	TNovak	RBlack		
DATE	1/1/81	1/13/81	1/14/81	1/21/81	1/26/81		

Docket

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

Docket Nos. 50-315
and 50-316

January 28, 1981

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

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 43 to Facility Operating License No. DPR-58 and Amendment No. 25 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 February 22, 1980.

These amendments revise the set point for the steam generator low-low level settings.

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

Sincerely,

Steven A. Varga

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

Enclosures:
1. Amendment No. 43 to DPR-58
2. Amendment No. 25 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

Mr. John Dolan
Indiana and Michigan Electric Company

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

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

Citizens for a Better Environment
59 East Van Buren Street
Chicago, Illinois 60605

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

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

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

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

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

Honorable James Bemenek, Mayor
City of Bridgman, Michigan 49106

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

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

Maurice S. Reizen, M.D.
Director
Department of Public Health
P. O. Box 30035
Lansing, Michigan 48909

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

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. 43
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 February 22, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8102190 167

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Page
2-5	2-5
2-6	2-6
3/4 3-26a	3/4 3-26a

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 - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 88,500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level - Low-Low	$\geq$ 17% of narrow range instrument span - each steam generator	$\geq$ 16% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq 0.71 \times 10^6$ lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span - each steam generator	$\leq 0.73 \times 10^6$ lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	$\geq$ 2750 volts - each bus	$\geq$ 2725 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	$\geq$ 57.5 Hz - each bus	$\geq$ 57.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 800 psig	$\geq$ 750 psig
B. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 17% of narrow range instrument span each steam generator	≥ 16% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196±18 volts with a 2±.2 second delay
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 17% of narrow range instrument span each steam generator	≥ 16% or narrow range instrument span each steam generator
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196±18 volts with a 2±.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 min. time delay	3596±18 volts with a 2.0 minute ± 6 second time delay

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. 25
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 February 22, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 25 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-5	2-5
2-6	2-6
3/4 3-25a	3/5 3-25a
3/4 3-26	3/5 3-26

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 - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	≥ 1950 psig	≥ 1940 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 93,750 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level - Low-Low	$\geq$ 21% of narrow range instrument span - each steam generator	$\geq$ 20% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq 1.47 \times 10^6$ lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span - each steam generator	$\leq 1.56 \times 10^6$ lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	$\geq$ 2750 volts - each bus	$\geq$ 2725 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	$\geq$ 58.2 Hz - each bus	$\geq$ 58.1 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 58 psig	$\geq$ 57 psig
B. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 20% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196 ± 18 volts with a 2 ± 0.2 second delay
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 20% of narrow range instrument span each steam generator
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196 ± 18 volts with a 2 ± 0.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 minute time delay	3596 ± 18 volts with a 2.0 minute ± 6 second time delay

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	Not Applicable
b. Reactor Trip (from SI)	Not Applicable
c. Feedwater Isolation	Not Applicable
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. DPR-58
AND AMENDMENT NO. 25 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

I. Introduction

By letter dated February 22, 1980 the licensee requested Technical Specification changes to the setpoint for steam generators low-low levels settings for D. C. Cook Plant Units Nos. 1 and 2. These new steam generator low-low level setpoints were provided in the licensee's letter of November 5, 1979 in response to IE Bulletin 79-21 concerning effects of containment temperature on safety-related level monitoring systems. Two level systems inside containment impacted by temperature are pressurizer level and steam generator level. This submittal was assigned to EG&G Idaho, Inc. (our consultants) for review and evaluation under our technical assistance program.

II. Discussion

Enclosure 1, Technical Evaluation Report, "The Effect of Containment Temperature of Liquid Level Measurements for D. C. Cook Units 1 and 2" was prepared for us by EG&G Idaho. EG&G concluded that the steam generator low-low setpoints for D. C. Cook Plant Unit Nos. 1 and 2 listed below are acceptable.

Although no credit has been taken for the pressurizer level instruments in the safety analysis, the range of indicated pressurizer level already established is acceptable and will assure that the heaters remain covered and that the pressurizer does not go water solid. This acceptable range was established taking into consideration temperature effects on level instruments.

III. Instrument Trip Setpoints

	Trip Value	Allowable Values
Steam Generator low-low level		
Unit 1	≥ 17%	≥ 16%
Unit 2	≥ 21%	≥ 20%

8102190\72

IV. Evaluation

Based on our review of the consultant's technical evaluation, as described above, we have concluded that the level instruments inside containment have been properly evaluated for temperature effects. The higher setpoint values for steam generator low-low for D. C. Cook Unit Nos. 1and 2 are therefore acceptable. These values were discussed with the licensee and he agrees to them.

V. Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types of 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.

VI. Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probablity or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 28, 1981

REFERENCES

1. NRC Letter (K. Kniel) April 11, 1977 to Indiana & Michigan Power Company.

2. Indiana & Michigan Power Company Letter (J. Tillinghast) May 17, 1977 to NRC (E. Case).

3. NRC Letter (K. Kniel) July 8, 1977 to Indiana & Michigan Power Company.

4. Westinghouse Electric Corp. Letter (T. Anderson) June 21, 1978 to NRC (E. Case).

5. Indiana & Michigan Power Company Letter (J. Tillinghast) June 22, 1978 to NRC (E. Case).

6. Indiana & Michigan Electric Company Letter (J. Dolan) February 22, 1980 to NRC (H. Denton).

Enclosure 1

TECHNICAL EVALUATION REPORT
THE EFFECT OF CONTAINMENT TEMPERATURE ON LIQUID LEVEL MEASUREMENTS

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

Docket Nos. 50-315 and 50-316

August 1980

A. C. Udy
Reliability and Statistics Branch
Engineering Analysis Division
EG&G Idaho, Inc.

TAC Nos. 12981 and 12982

ABSTRACT

As an indirect result of an assumed loss-of-coolant accident, there is an apparent increase in the indicated water level for those systems whose sensors and reference legs are exposed to the elevated containment temperature. This report examines the effected systems, the effect on the initiation of safety systems, and the effect on the information displayed to the reactor operator.

CONTENTS

TECHNICAL EVALUATION REPORT
THE EFFECT OF CONTAINMENT TEMPERATURE ON LIQUID LEVEL MEASUREMENTS

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

1.0 INTRODUCTION

Based on the information supplied by Indiana & Michigan Electric Company (I&MECo), this report addresses the effect of the containment temperature on the steam generator and pressurizer water level detectors.

In June 1979, the Power Systems Division of the Westinghouse Electric Corporation (W) notified the Nuclear Regulatory Commission (NRC)[1] and W utility customers of corrections that should be applied to indicated steam generator water level and associated low water level protection system setpoints and emergency operating procedures. The problem identified was that, as the temperature of the level measurement reference leg increased due to a high-energy line break, the water column density decreased. This appears as an apparent increase in the indicated water level, which could result in delayed protection (reactor trip and auxiliary feedwater) signals.

On August 13, 1979, the NRC sent an IE bulletin (#79-21)[2] to all reactor facilities. Boiling water reactors and those facilities with construction permits were notified for information. All operating pressurized water reactor licensees were directed to take steps to evaluate the problem, take corrective actions needed, and to notify the NRC of all actions taken as a result of the evaluation.

I&MECo responded in a letter of November 5, 1979[3] and requested technical specification changes to Tables 2.2-1 and 3.3-4 on Feburary 22, 1980[4]. This report is a technical evaluation of the material submitted by, and actions taken by, I&MECo for the Cook units.

2.0 EVALUATION OF THE COOK NUCLEAR STATION, UNITS 1 AND 2

2.1 Review Guidelines. IE Bulletin No. 79-21 provided the pressurized water power reactor licensees with the following NRC guidelines:

1. Guideline 1 - The licensee is to review all liquid level measurement systems within containment. If the signals are used to initiate safety actions or to provide post-accident monitoring information, a description of the system is to be submitted.

2. Guideline 2 - For those systems identified by guideline 1, the licensee is to evaluate the effect of post-accident temperature on the indicated water level (in comparison to the actual water level, including all sources of error).

3. Guideline 3 - The licensee is to provide a listing of all safety and control setpoints used with the level instrumentation, and verify proper setpoint actuation throughout the range of ambient temperature (including accident temperatures).

4. Guideline 4 - If a change of setpoints is necessary to ensure safe action, the licensee is to describe the corrective action and state when the action was taken.

5. Guideline 5 - The licensee is to ensure that the operators are instructed on the potential for, and the potential magnitude of, erroneous level signals. The completion date for procedure changes and operator training is to be identified.

2.2 Description of Liquid Level Measurement Systems. I&MECo has identified the steam generator narrow-range water level (SG level) and pressurizer level (PZR level) systems as inside containment, and used to initiate safety functions or to provide post-accident monitoring information.[3] They provided the following description:

"The SG level reference leg is a conventional condensing pot open column system contained entirely within the lower volume of the containment. The PZR level high side reference leg is a sealed bellows type filled with distilled water and is contained partially in the lower volume and partially in the upper volume of the containment. The PZR level low side reference leg is a conventional open column system entirely contained in the lower volume of the containment."[3]

The FSAR indicates that the SG low-low level trip, on two out of three coincidence in any one steam generator, will cause a reactor trip. High pressurizer level serves as a backup to high pressurizer pressure on two out of three coincidence, to cause a reactor trip; however, the FSAR does not take credit for this function[3]. Both the SG level and PZR level provide post-accident monitoring information in the control room. Table 7.5-2 of the FSAR shows that the PZR level is designed to operate for 1/2 hour after an accident. Section 7.5.2 of the FSAR indicates that pressurizer pressure and level are the only transmitters inside containment that are required to actuate the Engineered Safety Features.

As a result of the W evaluation, I&MECo raised, on June 30, 1979, the SG low-low setpoints to 17% and 21% of the narrow range instrument span for Units 1 and 2, respectively.[3] The increase was chosen as recommended by W, to correspond the temperature limit at which the containment pressure-high setpoint would cause a safety injection.[3] I&MECo dismissed level bias associated with high SG water level at elevated containment temperatures as insignificant.

2.3 Evaluation of Liquid Level Measurement Systems. Guideline 1 requested the licensee to review all liquid measurement systems within containment. I&MECo has provided this review and descriptions of the pressurizer level and steam generator level transmitters.[3]

Guideline 2 requested the licensee to compare indicated and actual water levels in a post-accident environment. Reference 3 provides the requested comparison, and states that reference leg boiling will not occur.

Guideline 3 requested that the setpoints used on safety and control circuits be identified with verification of proper setpoint actuation thoughout the temperature range. Guideline 4 requested documentation of any corrective action taken. The SG low-low setpoints were increased at both units. The revised setpoints of 17% and 21% of narrow range instrumentation (Units 1 and 2, respectively) were incorporated in June 1979[3]. The increased setpoint is based on the W recommendation and analysis, and on the containment temperature expected before the high containment pressure reaches its setpoint for safety injection (which will cause a reactor trip).

Since no credit is taken in the safety analysis for the pressurizer level high trip, no changes were made to this setpoint. The FSAR (Section 7.5.3) states that the reference leg of the pressurizer level transmitter will not exceed 140°F. A corresponding temperature for the Steam Generator level transmitters is not established by the FSAR.

The signals identified and setpoints revised for the temperature range specified by I&MECo are within the analyzed limits of the FSAR and redundant instrument trip signals.

To ensure that the pressurizer does not become water solid, or that the pressurizer heaters do not become uncovered, I&MECo has established procedural limits to the allowable pressurizer level that takes the temperature effect on the level sensor into account.[3] The actions taken and the information documented satisfies guidelines 3 and 4.

Guideline 5 requires any procedure changes and operator training needed as a result of IE Bulletin No. 79-21 be scheduled. Reference 3 indicates that operator training and needed procedure changes have been completed.

3.0 SUMMARY

The material submitted by I&MECo for this review has been evaluated to the guidelines of IE Bulletin No. 79-21. The changes in operator training and procedural changes concerning the steam generator level and the pressurizer level signals satisfy the guidelines of the IE bulletin.

Changes to the steam generator low-low level trip setpoints satisfy the guidelines of the IE bulletin and the Westinghouse recommendations. The pressurizer level high trip setpoint was not changed since this was not credited with a safety function in the FSAR. This is acceptable as the trip is within the analyzed limits.

The NRC should approve the proposed technical specification changes for the steam generator low-low level setpoints of 17% for Unit 1 and 21% for Unit 2, with allowable limits of 16% and 20%, respectively.

4.0 REFERENCES

1. W letter, T. M. Anderson, to U.S. NRC, Victor Stello, "Steam Generator Water Level," NS-TMA-2104, June 22, 1979.

2. NRC letter to all power reactor licensees and construction permit holders, "IE Bulletin No. 79-21," August 13, 1979.

3. I&MECo letter, R. S. Hunter, to U.S. NRC, James G. Keppler, "Response to IE Bulletin No. 79-21," AEP:NRC:00271, November 5, 1979.

4. I&MECo letter, John E. Dolan, to U.S. NRC, Harold R. Denton, AEP:NRC:00313, February 22, 1980.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-315 AND 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 43 to Facility Operating License No. DPR-58, and Amendment No. 25 to Facility Operating License No. DPR-74 issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

The amendments revised the set point for the steam generator low-low level settings.

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

8102190 177

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

For further details with respect to this action, see (1) the application for amendments dated February 22, 1980, (2) Amendment Nos. 43 and 25 to License Nos. DPR-58 and DPR-74, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W. Washington, D.C. and at the Maude Reston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 28 day of January, 1981

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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