Docket No. 50-315

September 3, 1985

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

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 91 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated July 30, 1985, as supplemented by letters dated August 8, 1985 and two letters dated August 13, 1985.

The amendment revises the Technical Specifications to change the setpoints in the channels for overpower delta T, overtemperature delta T, and loss of flow trips and the reactor coolant temperature to protect against departure from nucleate boiling (DNB).

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

Sincerely,

/s/DWigginton

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

Enclosures:
1. Amendment No. 91 to DPR-58
2. Safety Evaluation

cc: w/enclosures
See next page

OK with change marked on new SER 8/30/85

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD-OR:DL
CParrish	DWiggnton;ps	SVarga		GLainas
8/2/85	8/2/85	8/2/85	8/ /85	8/29/85

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Donald C. Cook Nuclear Plant

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INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
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 July 30, 1985, as supplemented by letters dated August 8, 1985 and two letters dated August 13, 1985. 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. 91 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. These Technical Specifications are to become effective within 30 days of issuance.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 3, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 91 FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-5	2-5
2-9	2-9
3/4 2-17	3/4 2-17
3/4 3-17	3/4 3-17
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22

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 4
9. Pressurizer Pressure--Low	$\geq$ 1065 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2305 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*	$\geq$ 89.1% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_0 \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T-T") - f_2(\Delta I)\right]$

where:

ΔT_0 = Extrapolated ΔT at DESIGN THERMAL POWER

T = Average temperature, °F

$T"$ = Indicated T_{avg} at DESIGN THERMAL POWER 577.1°F

K_4 = 1.089

K_5 = 0.0177/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.0011 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

$f_2(\Delta I)$ = $f_1(\Delta I)$ as defined in Note 1 above.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation at RATED THERMAL POWER	4 Loops In Operation at DESIGN THERMAL POWER	3 Loops In Operation at RATED THERMAL POWER
Reactor Coolant System T_{avg}	$\leq$ 570.4°F	$\leq$ 579.7°F	$\leq$ 570.5°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq 1.386 \times 10^8$ lbs/hr	$\geq 1.386 \times 10^8$ lbs/hr	$\geq 0.9917 \times 10^8$ lbs/hr

*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.

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 Flow in Two Steam Lines-High				1, 2, 3##	
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line		15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low				1, 2, 3## **	
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops		14*
Three Loops Operating	1 T_{avg}/ operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		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 EITHER T_{avg}--Low-Low				1, 2, 3##**	
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops		14*
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		15
OR, COINCIDENT WITH				1, 2, 3##	
Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops		14*
Three Loops Operating	1 pressure/operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		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*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

*The provisions of Specification 3.0.4 are not applicable.

**Rod drop testing in accordance with specification 4.1.3.3, Rod Position Indication Calibration, and hot zero power physics testing may proceed prior to the correction of RTD Calibration Curves for cross-calibration results at the beginning of cycle.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operations may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-315

INTRODUCTION

Westinghouse Electric Corporation notified Indiana and Michigan Electric Company (licensee) on May 2, 1985 that they had found an error in their calibration allowance for Resistance Temperature Detectors (RTDs) manufactured by RdF. The NRC was notified of this error by a Westinghouse letter dated May 6, 1985. This error resulted from calibrtions performed by RdF that showed a non-compatibility with the presently accepted Westinghouse instrument uncertainty assumed in the analysis. Westinghouse has evaluated the impact of these calibration errors and has determined that certain changes to the Technical Specifications should be made.

The proposed changes to the D. C. Cook Unit No. 1 Technical Specifications would accommodate the use of any of the 16 RdF RTDs. Two of the RdF RTDs are already installed in Unit 1 as spares with the remaining 14 being installed during the current refueling outage as replacements for Rosemount manufactured RTDs. These RTDs are used for temperature input to Overtemperature ΔT, Overpower ΔT, Loss of Flow, DNB protection and Tavg low-low Essential Safety Features (ESF Actuation).

EVALUATION AND CONCLUSION - CALIBRATION ALLOWANCE AND CHANGES TO SETPOINTS

By letters dated July 30 and August 13, 1985 from M. P. Alexich to Harold R. Denton, the licensee proposed changes and provided justifications for these changes for the D. C. Cook Unit No. 1 Technical Specifications. These changes that time, plant startup was scheduled for August 10, 1985 and reaching mode 3 on August 18, 1985. In our review of the licensee's efforts to resolve this matter, we have determined that the submittals were timely and made on a best effort basis. The proposed amendment was noticed in the Federal Register on August 2, 1985 with comments due by August 16, 1985; less than the 30 days usually provided for comment but before the planned startup. Subsequent to that time, the licensee has experienced difficulty in startup with significant delays due to failures during a hydro test, repeating the integrated leak rate test after correcting value lineups, and performing unplanned crevice flushing on the steam generators. Startup is now scheduled after September 7, 1985. Although there has been some unanticipated delay in startup, the licensee responded promptly and reasonably to the unforeseen need for technical specification changes required to prevent anticipated delay in startup of the facility.

The proposed changes to the D. C. Cook Unit No. 1 Technical Specifications would accommodate the use of any of the 16 RdF RTDs. Two of the RdF RTDs are already installed in Unit 1 as spares with the remaining 14 being installed during the current refueling outage as replacements for Rosemount manufactured RTDs. These RTDs are used for temperature input to Overtemperature ΔT, Overpower ΔT, Loss of Flow, DNB protection and Tavg low-low Essential Safety Features (ESF Actuation).

EVALUATION AND CONCLUSION - CALIBRATION ALLOWANCE AND CHANGES TO SETPOINTS

By letters dated July 30 and August 13, 1985 from M. P. Alexich to Harold R. Denton, the licensee proposed changes and provided justifications for these changes for the D. C. Cook Unit No. 1 Technical Specifications. These changes resulted from the replacement of the Rosemount RTDs with RdF RTDs. The changes reviewed encompass the changes to Table 2.2-1, Functional Unit 7, 8 and 12; notes associated with Table 2.2-1; and Table 3.3-3, Functional Unit 4f. The changes channels (i.e., Overtemperature ΔT, Overpower ΔT, Low Flow). For the protection functions it was determined that the Safety Analysis Limit/Nominal Trip Setpoint relationship was sufficient to accommodate the changed uncertainties without causing changes to the Safety Analysis Limits or Nominal Trip Setpoint.

Utilizing the information noted above, an evaluation was performed which indicated that the current DNBR design limits are not impacted by the revised values. This evaluation demonstrated that installation of RdF RTDs (with increased and reallocated uncertainties over the currently installed RTDs) will not impact the Safety Analysis Limits assumed, nor the core limits utilized, in the plant's safety analyses. The only significant changes to the plant are the Allowable Values for several protection functions and the indicated Tavg value in the Unit 1 Technical Specifications. Nominal Trip Setpoints in the Technical Specifications remain unaffected.

Based on the above the staff concludes that the only impact on the plant, due to the installation of RdF RTDs, is the changing of the Allowable Values for the protection functions indicated and the staff has found these changes acceptable. Therefore, the staff considers the Technical Specification revisions noted above to be acceptable.

EVALUATION AND CONCLUSION - T_{avg} AND TESTS ABOVE P-12 SETPOINT

By letters dated July 30, 1985 and August 13, 1985, from M. P. Alexich to H. R. Denton, the Indiana and Michigan Electric Company proposed administrative controls for D. C. Cook Unit No. 1, to maintain the T_{avg} safety signal in the tripped condition when operating above the P-12 setpoint. This administrative action is only required for Cycle-9 until the licensee confirms that by replacing the RTDS (Resistance Temperature Devices) manufactured by Rosemount to those manufactured by RdF would not result in measurement uncertainties which exceed those assumed in the FSAR transient and accident analyses. This administrative control will assure that the consequences of a main steam line break will lie within the limits analyzed in the FSAR. Since both the safety injection actuation and steam line isolation signals are generated by a lo-lo average coolant temperature (T_{avg}) coincident with high steam flow, maintaining the T_{avg} signal in a tripped condition removes any uncertainty associated with the new RTDs. The staff finds the above administrative controls acceptable.

The licensee has also revised the T_{avg} value of the Technical Specifications from 570.5°F to 570.4°F (4 loop operation at Rated Thermal Power). This change is insignificant for the accident analysis of concern and the change to the Technical Specification is acceptable.

The licensee's proposal to add a note to page 3/4 3-22 to allow rod drop testing, rod position indication calibration, and hot zero power physics (above P-12) until the cross calibration results are available at the beginning of cycle is tentatively acceptable pending the licensee's development of a suitable Technical Specification on these tests under the section "Special Test Exceptions." This new Technical Specification has been discussed with the licensee and is requested to clarify and simplify the Technical Specifications. The proposed Technical Specification should be implemented before the beginning of the next fuel cycle (Cycle 10) and without the T_{avg} coincident channels in the tripped condition.

FINAL DETERMINATION - NO SIGNIFICANT HAZARDS DETERMINATION

In our review of the revised setpoints in the channels for overpower delta T, overtemperature delta T, and loss of flow trips and the reactor coolant temperature to protect against departure from nucleate boiling and in our review of the commitment to trip the low-low T_{avg} channels for Cycle 9 startup testing, we have determined that the licensee has used acceptable methods and procedures and that the revision of setpoints and insignificant change in T_{avg} will assure that the existing safety analyses for the D. C. Cook Unit No. 1 are unchanged. The revision to the setpoints has been analyzed with acceptable setpoint methodology and will be accomplished with established procedures. The setpoint methodology is capable of accounting for greater instrument uncertainty such that the instrument setting will produce the desired signals assumed in the safety analysis and to the extent other uncertainties are not changed by the resetting of setpoints, the safety analysis remains unchanged and valid. The change in T_{avg} of 0.1°F is so small as to be undetected in the safety analysis calculations. The results of calculations would not change as a result this 0.1°F difference. The commitment to place the low-low T_{avg} channels in the tripped condition was made when it was thought that the T_{avg} change would be greater. In any respect, the tripped condition although not significant because of the 0.1°F change in T_{avg} does assure without question that the steam line break analysis is unchanged. The proposed amendment does not change the previous safety analyses and therefore does not change the consequences or probabilities of accidents previously evaluated. The revision of setpoints in accordance with the approved setpoint methodology will assure that the desired signal is available as assumed in the safety analysis. This change is made to assure there is no reduction in the margin of safety. Likewise the 0.1°F change in T_{avg} is not discernible in the safety analysis results and, the proposed change is in keeping with the calibration and instrument settings. There is no reduction in the margin of safety with this change. The commitment to place the low-low T_{avg} channels in the tripped condition will provide half of a signal to protect against main steam line break. If the remaining half of the signal is received, the level of safety is maintained by the safety systems actuation to mitigate the steam line break. This would not change the margin of safety and does result in actions which have been previously evaluated. As with all of the proposed changes, we have determined that the amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. On all of these bases, the staff has made a final determination that the proposed setpoints, T_{avg}, and test exceptions involve no significant hazards consideration.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

CONCLUSION

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

Dated: September 3, 1985

PRINCIPAL CONTRIBUTORS:

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J. Guttmann
D. Wigginton