January 4, 1991

Docket No. 50-315

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SMeador	ACRS(10)

Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-58 (TAC NO. 77283)

The Commission has issued the enclosed Amendment No. 152 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 (TS) in response to your application dated July 23, 1990.

This amendment allows a decrease in the minimum measured flow requirements found in Table 3.2-1 of the TS. The Reactor Core Safety Limit Figure 2.1-1 and Table 2.2-1 Functional Unit 12 footnote are also revised to reflect this change.

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,

/s/

Timothy G. Colburn, Sr. Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 152to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRP345	PM/PD31:DRP345	D/PD31:DRP345	OGC
SMeador	TColburn	LMarsh	
1/2/91	1/2/91	1/2/91	1/2/91

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COOK AMEND 77283

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

January 4, 1991

Docket No. 50-315

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c/o American Electric Power Service Corporation
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Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

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See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
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 July 23, 1990, 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.

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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. 152 , 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

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 4, 1991

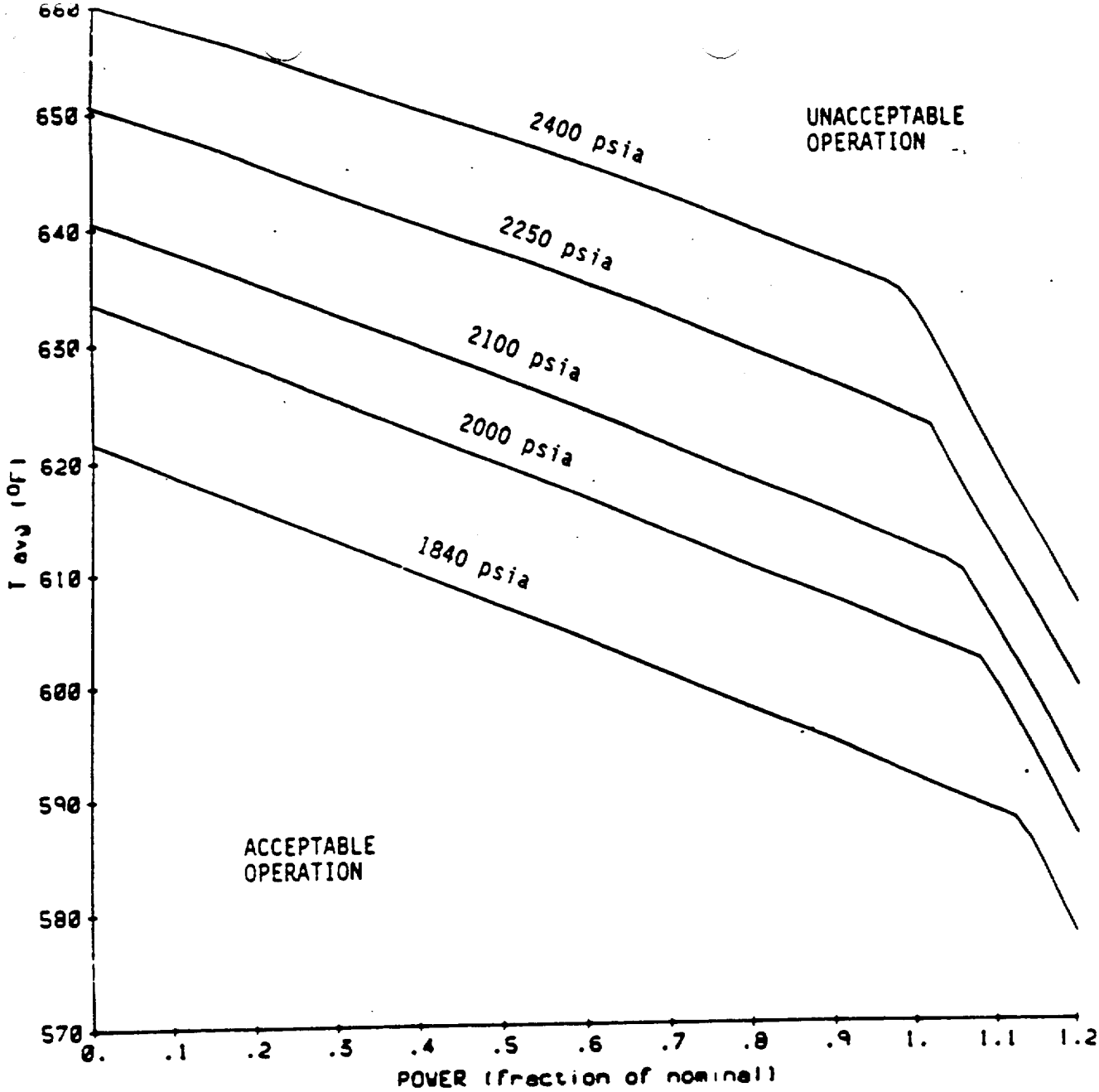
ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 152 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 marginal lines indicating the area of change.

REMOVE	INSERT
2-2	2-2
2-5	2-5
3/4 2-14	3/4 2-14



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN DEGREES F)
1840	(0.0, 622.1), (1.13, 587.3), (1.20, 577.5)
2000	(0.0, 633.8), (1.08, 601.4), (1.20, 586.0)
2100	(0.0, 640.8), (1.06, 609.8), (1.20, 591.3)
2250	(0.0, 650.7), (1.02, 621.9), (1.20, 598.9)
2400	(0.0, 660.1), (0.98, 633.7), (1.20, 606.2)

RATED THERMAL POWER = 3413 MWT

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 90,400 gpm per loop.

TABLE 3.2-1
DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	$\leq$ 570.9°F*
Pressurizer Pressure	$\geq$ 2050 psig**
Reactor Coolant System Total Flow Rate	$\geq$ 361,600 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letter dated July 31, 1990, the Indiana Michigan Power Company (the licensee) requested amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed amendment would allow a decrease in the minimum measured flow requirement found in Table 3.2-1 of the TS. The Reactor Core Safety Limit Figure 2.1-1 and Table 2.2-1 Functional Unit 12 footnote are also revised to reflect this change.

2.0 DISCUSSION

During the current DC Cook Unit 1 refueling outage, extensive steam generator tube testing was performed. This testing resulted in additional tube plugging that could reduce the total reactor coolant system (RCS) flow. The reduced flow is still calculated to be sufficient to meet the former RCS flow assumptions but the measurement fluctuations could result in failure to meet the existing Technical Specifications minimum measured flow (MMF) requirements. The MMF requirements assure that RCS flow meets the assumptions used in the nuclear steam supply system (NSSS) design calculations and the accident/transient analyses.

The Technical Specifications need to be changed to account for plugging of the Steam Generator tubes. The present RCS flow of 366,400 gpm will be changed to 361,600 gpm. To support these proposed Technical Specification changes the licensee submitted four documents :

1. D.C. Cook Significant Hazards Evaluation
2. Proposed Revised Technical Specification Pages
3. Westinghouse Evaluation of Reduced MMF.
4. Westinghouse ITDP Instrument Uncertainty Methodology (WCAPs 12568 & 12569)

The NSSS calculations and transient analyses assume certain values for the RCS flow. To successfully justify the Technical Specification changes, the previous analyses need to be assessed to ensure the MMF reduction will not impact the accident/transient analysis.

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3.0 EVALUATION

The licensee's Significant Hazards Evaluation determined (based on the other submittal evaluations) that the safety analyses remain valid for the reduction in the RCS flow and that the NSSS design calculations are unaffected by the reduction in the MMF requirement. Therefore, the MMF reduction does not create the possibility of a new or different kind of accident from any accident previously evaluated.

MMF is an important parameter in the NSSS calculations and transient analyses. These calculations and transients are divided into two categories :

1. Non-DNB (departure from nucleate boiling) transients which use thermal design flow (TDF).

2. DNB transients that use the Improved Thermal Design Procedure (ITDP) which uses MMF.

The Category 1 Non-DNB transients use the TDF value of 354,000 gpm. The new flow uncertainty value [361,600 (MMF)-2.1% (flow uncertainty) = 354,006.4 gpm] is greater than the TDF of 354,000 gpm. Since there is no impact on the TDF assumption, the NSSS design calculations are not affected by the change to the MMF requirement.

The Category 2 DNB transients are more complicated to justify. The MMF impacts the DNBR (departure from nucleate boiling ratio) analyses in three areas :

(a) DNBR design limits and margins
(b) Reactor Core Safety Limits
(c) Initial assumed RCS flow value

For (a), the assessments indicate the DNBR margin has been allocated to offset the penalty associated with the MMF reduction. The design limit DNBR values used in the safety analyses are more conservative than used in ITDP. This allows for DNBR penalties to be offset with available margin between the design and safety limit DNBR values. A reduction in core flow is a penalty with respect to the DNBR calculation. Therefore, sufficient margin is available and has been allowed to accommodate the MMF reduction.

For (b), the DNBR lines in Technical Specification Figure 2.1-1 remain unchanged for the MMF reduction, since the DNBR margin has been allocated as described in (a). However, the vessel exit boiling lines represent a physical limit. A reduction in the RCS flow results in these lines becoming more restrictive (i.e. for a given pressure and power, the exit boiling will occur at a lower RCS temperature). Therefore, the Reactor Core Safety Limits in Figure 2.1-1 has been revised. Also for (b), the Technical Specification Table 2.2-1 has been revised to denote the reduced MMF value. Functional Unit 12, Loss of Flow, now has a new value which is stated in its associated footnote.

For (c), the MMF reduction could affect the results of the analyses in the DNBR calculations and the system transient calculations. The assessments indicate that for the system transients the reduction in flow is sufficiently low, based on the Westinghouse Evaluation, to not impact the initial flow assumptions. To prove these conclusions, each of the six DNB accidents was assessed:

1. Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power
2. RCCA Misalignment/Dropped Rod
3. Loss of Reactor Coolant Flow
4. Loss of External Electrical Load
5. Excessive Heat Removal due to Feedwater System Malfunction
6. Excessive Increase in Secondary Steam Flow

Each of these events was addressed with the reduced MMF and determined not to impact the previous assumptions. Therefore, it is concluded that the transient results previously calculated for the events analyzed remain valid for the reduction in MMF to 361,600 gpm. A sample case was also run varying the RCS flow from 366,400 to 361,600 gpm. Varying the flow had very little impact on the RCS temperature, pressures, and time of reactor trip.

Westinghouse WCAP-12568 assesses the four ITDP parameters - (1) Pressurizer Pressure, (2) T avg, (3) Reactor Power, and (4) RCS Flow. The Westinghouse methodology is used to account for instrument uncertainties in the four ITDP parameters. The description of the four parameters and the resulting calculations are:

1. Pressurizer Pressure is a controlled parameter and the uncertainty reflects the control system. For this calculation, the numbers were also computed and validated by the staff. This assessment calculates an uncertainty of 34.6 psi.

2. T avg is a controlled parameter via the temperature input to the rod control system and the uncertainty reflects this control system. The first stage turbine pressure is used as a reference. This assessment calculates an uncertainty of 4.3 degrees F.

3. Reactor Power is monitored by the performance of a secondary side heat balance (power calorimetric) once every 24 hours. This calculation is called a precision flow calorimetric and is done every cycle. This calculation uses other calculations and plant measurements. This uses Primary Side and Secondary Side uncertainties. This assessment calculates an uncertainty of +/- 0.83 %.

4. RCS flow is monitored by the performance of a precision flow calorimetric at the beginning of each cycle. This calculation involves the primary or secondary side heat balance when the reactor is above 15% RTP (rated thermal power). This calculation is used for NIS (nuclear instrumentation system) verification to Technical Specification compliance. This assessment calculates an uncertainty of 2.1 %.

Westinghouse concludes that the above calculated uncertainties are reduced or at least the same values than previously used in the ITDP analysis. Therefore, the results are conservative.

In the proposed revised Technical Specification pages, Westinghouse refers to the MMF as total RCS flow which is not consistent with the text in these evaluations. However, this inconsistency is acceptable since the RCS total flow is more of a universally accepted term.

The Westinghouse Evaluation (WCAP-12568) assumes no venturi fouling and states that, if fouling is present and not cleaned, its effect must be treated as a bias in the calculations. The licensee has not addressed this venturi fouling. However, venturi fouling is not a problem since this calorimetric calculation is performed once a cycle and coincides with the complete calibration of the instrument loops. The text further states that the instrument drift effects are insignificant in the calculations. The staff disagrees with this statement. However, since the instrument loop is calibrated totally once each cycle and the calculation is done each cycle, then the drift error in this case is insignificant.

The reduction in the RCS flow impacts the NSSS design calculations and accident transient analyses. These transients can be divided into two categories - (1) the Non-DNB transients, and (2) the DNB transients. The Non-DNB transients use the thermal design flow in their calculations. For the Technical Specifications these analyses flow uncertainties are accounted for by the MMF and the calculation uncertainties being greater than the Thermal Design Flow, TDF. It is concluded, therefore, that the reduced MMF does not impact the Non-DNB transients.

The DNB transients use the Improved Thermal Design Procedure, (ITDP). The ITDP uses the MMF of 361,600 gpm. The ITDP uncertainties associated with the flow are incorporated into the DNBR limit. It is concluded, therefore, that the reduced MMF does not impact the DNBR design limits or margins, and that the Technical Specifications may be revised as proposed:

1. Safety Limit 2.1.1 - The Reactor Core Safety Limit Figure 2.1-1 is revised to the reduced RCS Flow.

2. Safety Limit 2.1.1 - The RTS Setpoints, Table 2.2-1, Functional Unit 12, is revised to the reduced RCS Flow.

3. Technical Specification 3/4.2.5 - Table 3.2-1 is revised to the reduced RCS Flow.

All the submittal documents of the Technical Specification change were reviewed in addition to the existing Technical Specifications for the proposed revised pages. All the numbers used in the calculations were also reviewed for consistency. The justifications were sufficiently documented and included adequate technical justification for the staff to conclude that the reduction in MMF is acceptable and that the former RCS flow assumptions in the safety analyses remain valid.

The licensee also rotated page 2-5 90 degrees and replaced mathematical symbols with words. This is a purely administrative change and is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in 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 or changes in surveillance requirements. 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 section 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.

5.0 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, 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.

Date: January 4, 1991

Principal Contributors: Jose Ibarra
Frank Orr