February 15, 1989

Docket Nos. 50-316

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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. 108 TO FACILITY OPERATING LICENSE NO. DPR-74 (TAC NO. 65677)

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated August 15, 1988.

This amendment revises moderator temperature coefficient, shutdown margin, and engineered safeguards actuation requirements to reflect the steamline break analysis performed by Advanced Nuclear Fuels Corporation for D. C. Cook Unit 2.

Copies of our related Safety Evaluation and Notice of Issuance are also included.

Sincerely,

original signed by

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 108 to DPR-74
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

5520 NAME: COOK AMD TAC 65677

LA/PD31:DRSP	PM/PD31:DRSP	(A)D/PD31:DRSP	OGC
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 15, 1989

Docket No. 50-316

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

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SUBJECT: AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-74 (TAC NO. 65677)

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See next page

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Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
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 August 15, 1988, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 108, 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

Dominic C. Dilanni/T.Q

Theodore Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 15, 1989

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 108 FACILITY OPERATING LICENSE NO. DPR-74

DOCKETS NOS. 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 marginal lines indicating the area of change.

REMOVE	INSERT
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 1-3b	3/4 1-3b
3/4 1-5	3/4 1-5
3/4 1-6	3/4 1-6
3/4 3-25	3/4 3-25
3/4 3-27	3/4 3-27
B 3/4 1-1	B 3/4 1-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 2.0% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN < 2.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 2.0% $\Delta k/k$:

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 increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, 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##, 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.

---

*See Special Test Exception 3.10.1

With $K_{eff} \geq 1.0$

With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

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.

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

4.1.1.1.3 Prior to blocking ESF Functional Units in accordance with footnotes # and ## of Table 3.3-3, SHUTDOWN MARGIN shall be determined to be greater than or equal to 2.0% Δk/k by consideration of the factors of 4.1.1.1.1e above. The Reactor Coolant System average temperature used in making this SHUTDOWN MARGIN determination shall be less than or equal to 350°F. This SHUTDOWN MARGIN shall be maintained at all times when the ESF functions are blocked in MODE 3.

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

a. In MODE 4:

1. Greater than or equal to 2.0% Δk/k when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

b. In MODE 5:

1. Greater than or equal to 1.0% Δk/k when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

APPLICABILITY: MODES 4 and 5

ACTION:

With SHUTDOWN MARGIN less than the above limits, immediately initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the above limits:

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 SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

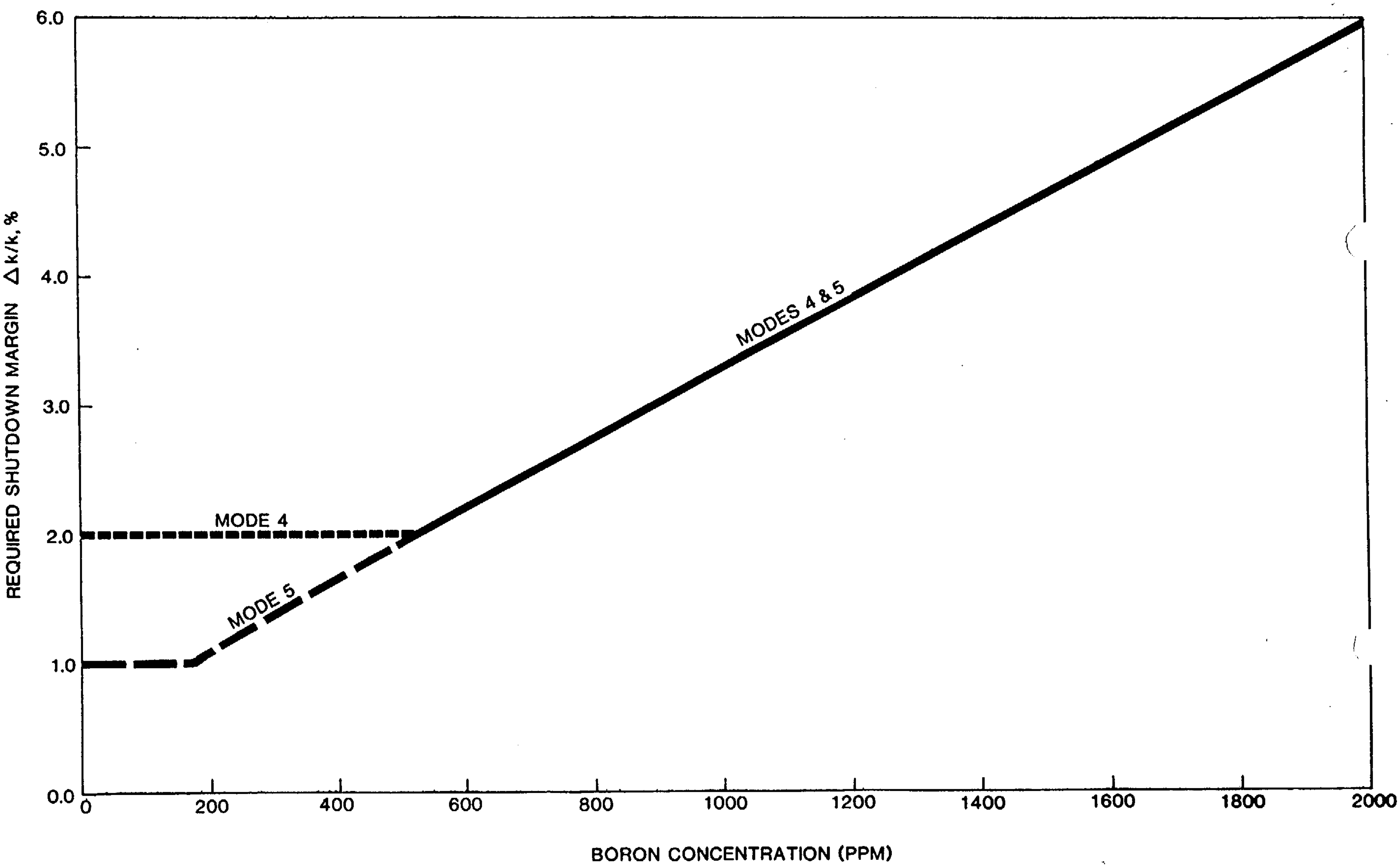

FIGURE 3.1-3 REQUIRED SHUTDOWN MARGIN

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the region of acceptable operation in Figure 3.1-2, and

b. Less negative than -3.5×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a above:

1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b above, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.0

\# See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. Measured MTC values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a) The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.4.a, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b) The MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm and the extrapolated MTC value compared to the EOL limit. In the event this comparison indicates that the MTC will be more negative than the EOL limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	$\leq$ A function defined as follows: A Δp corresponding to 1.47 x 10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 4.02 x 10^6 lbs/hr at full load.	$\leq$ A function defined as follows: A Δp corresponding to 1.62 x 10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 4.07 x 10^6 lbs/hr at full load.
	$T_{avg} \geq 541^{\circ}$ F.	$T_{avg} \geq 539^{\circ}$ F.
e. Steam Line Pressure--Low	$\geq$ 600 psig steam line pressure	$\geq$ 585 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	< 67% of narrow range Instrument span each steam generator	< 68% of narrow range Instrument span each steam generator

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 24.0*/12.0#
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 48.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 12.0#/24.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
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
h. Steam Line Isolation	≤ 10

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

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 for increased load events 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 2.0% Δk/k is initially required to control the reactivity transient and automatic ESF is assumed to be available.

Technical Specification requirements call for verification that the SHUTDOWN MARGIN is greater than or equal to that which would be required for the MODE 3 low temperature value, 350°F, prior to blocking safety injection on either the P-11 or P-12 permissive interlocks. This assures in the event of an inadvertent opening of two cooldown steam dump valves that adequate shutdown reactivity is available to allow the operator to identify and terminate the event.

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

In shutdown MODES 4 and 5 when heat removal is provided by the residual heat removal system, active reactor coolant system volume may be reduced. Increased SHUTDOWN MARGIN requirements when operating under these conditions is provided for high reactor coolant system boron concentrations to ensure sufficient time for operator response in the event of a boron dilution transient.

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 3000 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 3000 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE STEAMLINE BREAK ANALYSIS

AND AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated August 15, 1988, Indiana Michigan Power Company (the licensee) proposed an amendment to the Technical Specifications appended to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit 2. The proposed revisions are supported by the steamline break analysis performed by Advanced Nuclear Fuels (ANF). The analysis was performed using the methodology described in Reference 1 in response to comments by the NRC staff regarding previous ANF analyses. The analysis, described in report XN-NF-87-31(P), was transmitted to the NRC by the May 29, 1987 ANF letter and placed on the D. C. Cook docket by the licensee via a letter dated June 15, 1987.

The proposed Technical Specification changes are:

1. Increase in required shutdown margin from 1.6% delta-k/k to 2.0% delta-k/k to reflect the ANF analysis assumption.

2. Inclusion of a time response testing requirement of $\leq$ 10 seconds for the high steam flow/low-low T_{ave} steamline isolation function to reflect the ANF analysis assumption.

3. Reduction in allowable end of life moderator temperature coefficient from -3.9×10^{-4} delta-k/k/°F to -3.5×10^{-4} delta-k/k/°F to reflect the ANF analysis assumption.

4. Change in end of life moderator temperature coefficient surveillance due to the ANF analysis and desire to improve the surveillance basis.

5. Change in expression of high steam flow/low-low T_{ave} setpoint from percent full steam flow to lbs/hr to improve Unit 1/Unit 2 similarities.

2.0 EVALUATION

2.1 Steamline Break Analysis

The steamline break analysis performed by ANF is described in report XN-NF-87-31(P) (Ref. 2). The analysis utilizes the methodology which is described in report XN-NF-84-93(P), Steamline Break Methodology for PWRs (Ref. 1). This methodology was approved by the NRC staff in Reference 3. The ANF analyses utilize RELAP5, XCOBRA-IIIC, and XTG computer codes to predict the plant and core response to a steamline break. The analysis assumptions, plant modeling, and computer code interfaces are such that a conservative estimate of the plant and core response is predicted.

The analysis for D. C. Cook Unit 2 provided in Reference 2 included four possible transient scenarios. The scenarios included initiation of the steamline break transient from Hot Zero Power (HZP) or Hot Full Power (HFP) and with or without the availability of offsite power. The fuel response for the four scenarios was evaluated against minimum departure from nucleate boiling ratio (MDNBR) and centerline melt linear heat generation rate (LHGR) criteria. No fuel failures were predicted to occur based on either MDNBR or LHGR limits.

The HZP with loss of offsite power scenario was determined to be limiting with respect to MDNBR. The initiation from HZP results in a higher return to power. The coastdown of the reactor coolant pumps caused by the loss of offsite power reduces the return to power but the combination of power and reduced flow resulted in the lowest MDNBR for the cases analyzed. Reactor trip and safety injection are predicted to be actuated by the differential pressure between steam lines function. Main steam isolation is assumed to occur due to the high steam flow/low-low T_{ave} actuation signal. Delivery of borated water from the Emergency Core Cooling System (ECCS) is limited due to the assumed failure of one of two charging pumps and conservative modeling such as a stagnant reactor vessel upper head. Cooldown of the primary coolant and resultant power increase are maximized by the break flow model, feedwater and auxiliary feedwater modeling, stuck rod assumption, and other aspects of the methodology and plant specific analysis. The predicted MDNBR is above the Modified Barnett Correlation safety limit of 1.135, and therefore no fuel failure is expected to occur related to the DNBR critera.

The HZP with offsite power available scenario was determined to be limiting with respect to the centerline melt LHGR criteria. The initiation from HZP and continued operation of the reactor coolant pumps result in the maximum predicted power increase. Safety system initiations and conservative assumptions are similar to the HZP with loss of offsite power scenario. The analysis results predict a peak LHGR less than the 21 Kw/ft limit just prior to the borated ECCS water from one of two charging pumps reaching the core.

The ANF analysis of a steamline break for D. C. Cook Unit 2 has been found to be conservative. The methodology includes a conservative modeling of the break flow (Ransom-Trapp steam only) and adequately represents the asymmetric response of the reactor coolant system and reactor core. The upper head of the reactor vessel was modeled to maximize the possibility of flashing and related delay of ECCS delivery. Plant specific assumptions regarding feedwater and auxiliary feedwater delivery, failure of one of two charging pumps, stuck rod location, and initiation setpoints and delays associated with safety systems were found to be conservative with respect to either the current plant requirements or those Technical Specification amendments requested by the licensee.

2.2 Technical Specifications

The Technical Specification amendments proposed by the licensee are:

Shutdown Margin

The proposed Technical Specification revises the required shutdown margin for Modes 1 through 4 from a value of 1.6% delta-k/k to 2.0% delta-k/k. This change is in the more restrictive direction and is required to reflect the ANF assumption in the steamline break analysis. The proposed change has been reviewed and found to adequately incorporate the analysis assumption and is therefore acceptable.

Time Response Testing

The proposed Technical Specification adds an Engineered Safety Features Response Time requirement (Table 3.3.-5) for the steamline isolation function on high steam flow with coincident low-low T_{ave}. This addition is required due to reliance upon this function and specific actuation signal in the ANF analysis. The requirement of $\leq$ 10 seconds agrees with the ANF assumption and has been determined to be acceptable.

Moderator Temperature Coefficient (MTC)

The proposed revision in the allowable end of life MTC from -3.9×10^{-4} to -3.5×10^{-4} delta-k/k/°F is more restrictive and reflects the ANF assumption in the steamline break analysis. The proposed change has been reviewed and found to adequately incorporate the analysis assumptions and is therefore acceptable.

The proposed revision in the end of life MTC surveillance changes the criteria to which the measured MTC is compared upon reaching an equilibrium boron concentration of 300 ppm. The existing Technical Specification requires the measured MTC at 300 ppm to be less negative than -3.0×10^{-4} delta-k/k/°F. This value corresponds to the end of life analysis assumption of -3.9×10^{-4} delta-k/k/°F and ensures conservatism upon reaching the end of life core conditions. The required change in the end of life MTC limit to -3.5×10^{-4} delta-k/k/°F also required an inspection of the surveillance requirement. The licensee decided to use an extrapolation to end of life conditions and compare directly with the -3.5×10^{-4} delta-k/k/°F criteria instead of defining a value for 300 ppm which would ensure meeting the end of life criteria. The increased surveillance requirements (measure every 14 EFPD) associated with measuring an MTC more negative than the limit is maintained. The extrapolation to end of life for comparison to the actual MTC analysis assumption would also make the Unit 2 Technical Specifications more similar to those of Unit 1.

The proposed revision has been reviewed and found to adequately ensure the end of life MTC analysis assumption remains bounding for actual D. C. Cook Unit 2 cores. Maintaining the increased surveillance requirements ensures that required actions are taken prior to exceeding the end of life limit.

High Steam Flow/Low-Low Tave Setpoint

The proposed revision changes the expression of the steamline isolation setpoint from percent full steam flow to lbs/hr. The change results in minimal if any actual change in the differential pressure setpoints but allows the terminology for the setpoint to be similar for Units 1 and 2. The proposed change reflects the ANF assumption in the steamline break analysis and has been found acceptable.

2.3 Conclusion

The ANF analysis of a steamline break for D. C. Cook Unit 2 has been reviewed and found acceptable. The methodology, modeling, and selection of plant-specific assumptions has allowed the identification of limiting scenarios and conservative predictions which show that applicable acceptance criteria are satisfied. The proposed Technical Specification revisions have been found to be either required by the steamline break analysis or adequately supported by the analysis, and as such, have been determined to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment and Finding of No Significant Impact has been issued for this amendment (54 FR 6976, February 15 1989).

4.0 CONCLUSION

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

5.0 REFERENCES

1. XN-NF-84-93(P), "Steamline Break Methodology for PWRs," Exxon Nuclear Company, November 1984.

2. XN-NF-87-31(P), "Steam Line Break Analysis for D. C. Cook Unit 2, "Advanced Nuclear Fuels Corporation, May 1987.

3. Letter, A. Thadani (NRC) to R. A. Copeland (ANF), subject: Acceptance for Referencing of Licensing Topical Reports, ANF-84-93(P) and ANF-84-93(P), Supplement 1, "Steamline Break Methodology for PWRs," December 28, 1988.

Date: February 15, 1989

Principal Contributor: W. Reckley

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The United States Nuclear Regulatory Commission (the Commission) has issued Amendment No.108 to Facility Operating License No. DPR-74, issued to the Indiana Michigan Power Company (the licensee), which revised the Technical Specifications (TSs) for operation of the Donald C. Cook Nuclear Plant, Unit No. 2, located in Berrien County, Michigan. The amendment is effective as of the date of issuance.

The amendment revises the moderator temperature coefficient, shutdown margin, and engineered safeguards actuation requirements to reflect the steamline break analysis performed by Advanced Nuclear Fuels Corporation for D. C. Cook Unit 2.

The application for the amendment 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 amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on November 25, 1988 (53 FR 47782). No request for hearing or petition to intervene was filed following this notice.

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on February 15, 1989, at 54 FR 6976.

For further details with respect to this action, see (1) the application for amendment dated August 15, 1988, (2) Amendment No. 108to License No. 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, 2120 L Street, NW., Washington, DC 20555, and at the Maude Preston 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, DC 20555, Attention: Director, Division of Reactor Projects - III, IV, V, and Special Projects.

Dated at Rockville, Maryland, this 15thday of February1989.

FOR THE NUCLEAR REGULATORY COMMISSION

John Stang, Project Manager
Project Directorate III-I
Division of Reactor Projects - III, IV, V
& Special Projects