Docket File
50-316

REGULATORY DOCKET FILE COPY

JANUARY 11 1980

Docket No.: 50-316

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. 17 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated January 9, 1980 and supplemental January 11, 1980.

This amendment approves the ECCS analysis for the D. C. Cook, Unit No. 2 using the NRC approved February 1978 Westinghouse evaluation model and revises the total nuclear peaking factor (F_Q). This amendment supersedes the Order for Modification of License for D. C. Cook, Unit No. 2 dated June 6, 1978, accordingly, that Order has been terminated.

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

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 17 to DPR-74
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

DOR:RS
CHECK

TAC's 12472
11336

No legal objection to Notice and Amendment

8002250130

OFFICE	DOR:ORB#1	DOR:ORB#1	OELD	DOR:ORP	DOR:ORB#1	
SURNAME	CParrish	DWigginton:ms	R Black	WGammill	ASchwencer	
DATE	1/11/80	1/11/80	1/11/80	1/11/80	1/11/80	

#11

NRC FORM 318 (9-76) NRCM 0240

U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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

JANUARY 11 1980

Docket No.: 50-316

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. 17 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated January 9, 1980 and supplemental January 11, 1980.

This amendment approves the ECCS analysis for the D. C. Cook, Unit No. 2 using the NRC approved February 1978 Westinghouse evaluation model and revises the total nuclear peaking factor (F_Q). This amendment supersedes the Order for Modification of License for D. C. Cook, Unit No. 2 dated June 6, 1978, accordingly, that Order has been terminated.

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

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 17 to DPR-74
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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

Mr. Robert Masse
Donald C. Cook Nuclear Plant
P. O. Box 458
Bridgman, Michigan 29160

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, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 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

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. 17
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 January 9, 1980 as supplemented January 11, 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.

8002250145

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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 17, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The Order for Modification of License dated June 6, 1978 is hereby terminated.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 11, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. Revised pages are identified by Amendment number and contain vertical lines indicating the area of chance. The corresponding overleaf pages are also provided for document completeness.

Pages
3/4 2-5
3/4 2-7
3/4 2-8
3/4 2-17
B 3/4 2-1

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{1.99}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [3.98] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similiarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} \; [1+0.2(1-P)]$$

where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} :

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.87$ for all core planes containing control rods, and

2. $F_{xy}^{RTP} \leq 1.58$ for all unrodded planes above 6.2 ft, and ≤ 1.62 for all unrodded planes below 6.2 ft.

f. The F_{xy} limits of e, above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 $\pm$ 2%, 32.1 $\pm$ 2%, 46.4 $\pm$ 2%, 60.6 $\pm$ 2% and 74.9 $\pm$ 2%, inclusive.

4. Core plane regions within $\pm$ 2% of core height ($\pm$2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}:

1. The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}, and

2. The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

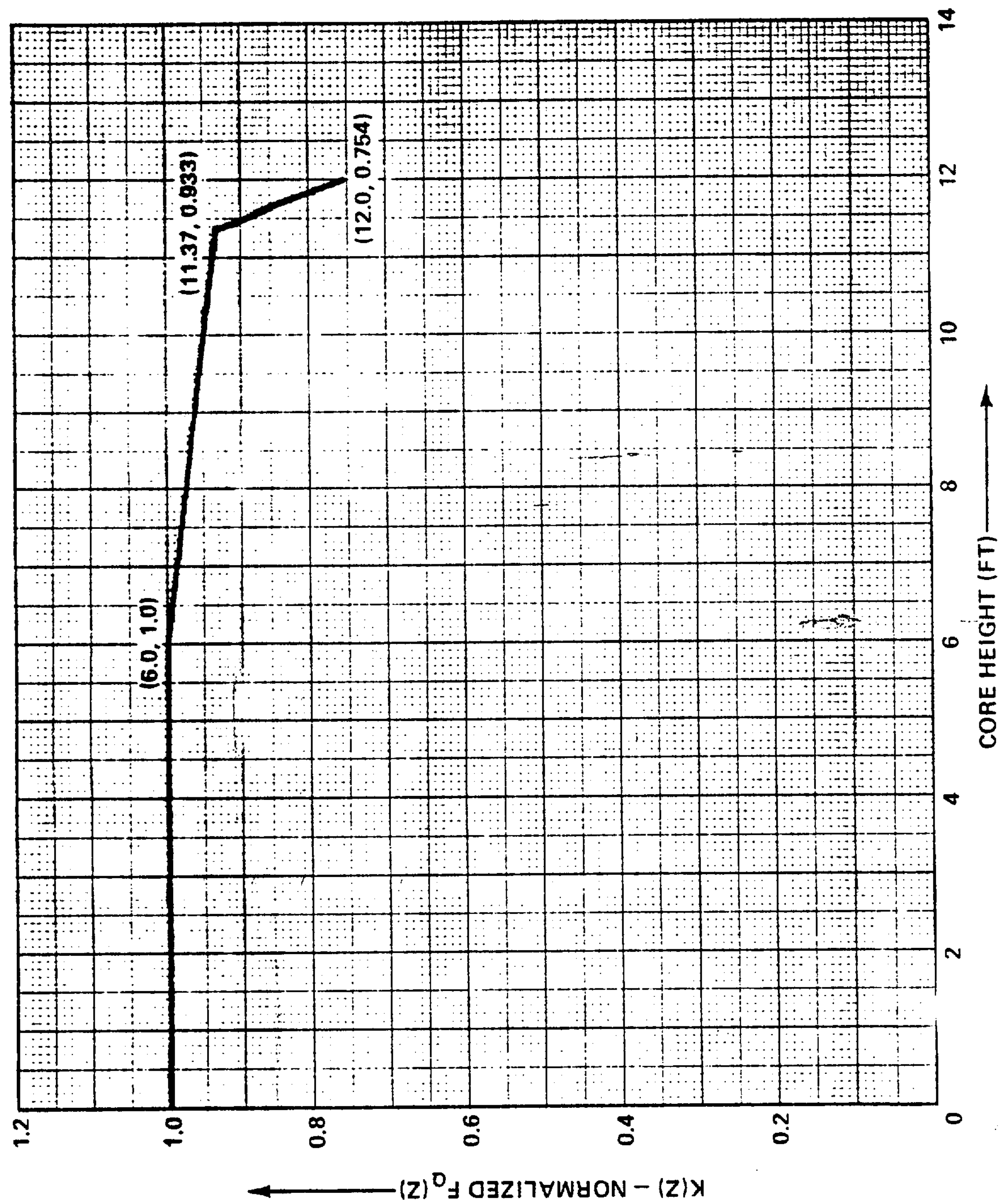

FIGURE 3.2-2 K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[1.99]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at ~~least~~ n=6 in-core flux maps covering the full configuration of permissible rod patterns above 94% of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

LIMITING CONDITION FOR OPERATION (Continued)

σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\overline{R}_j - R_{ij})^2 \right]^{1/2}}{\overline{R}_j}$$

The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system respectively.

The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 above 94% OF RATED THERMAL POWER#.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER one percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to 94% or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER to 94% or less of RATED THERMAL POWER within 15 minutes.

The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 1.99 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

POWER DISTRIBUTION LIMITS

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm 5\%$ target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 84% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of [illegible] OPERABLE excore detector outputs and provides an alarm message [illegible]ately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore cha[illegible]re outside the target band and the THERMAL POWER is greater than [illegible] RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 84% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated January 9, 1980, as supplemented January 11, 1980 (References 1 and 2 respectively), Indiana and Michigan Electric Company requested amendment to Appendix A to facility Operating License No. DPR-74 for Donald C. Cook Nuclear Plant Unit 2. Revised analyses are discussed in Section 2. Concomitant proposed changes to the Technical Specifications are summarized in Section 4.

The Large Break Loss of Coolant Accident (LOCA), was reanalyzed using the February 1978 currently approved Westinghouse LOCA-ECCS Evaluation Model (Reference 3). Additional analyses (Reference 1) were performed to assess the potential impact of recent concerns related to the LOCA-ECCS fuel clad models expressed in draft report NUREG-0630 (Reference 4).

2.0 LOCA-ECCS ANALYSIS

The current analysis (Reference 2) was performed with the approved (Reference 3) February 1978 version of the Westinghouse LOCA-ECCS evaluation model which is in compliance with Appendix K to 10 CFR 50. Results of the analysis are in compliance with the requirements of 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors.

Previous analyses for a spectrum of breaks were performed using the October 1975 Westinghouse LOCA-ECCS model. The ECCS analysis shown in the plant FSAR (Reference 5) provided the basis for operation with a peak to average linear heat generation rate, F_q, of 2.18. The discovery in April 1978 of an error in the prediction of the zirconium-water reaction calculation in the October 1975 model was accommodated by a reduction of the permissible F_q to 2.11. This value was determined by the staff and enforced by Commission order (Reference 6). Subsequent analyses by the licensee using the October 1975 Westinghouse model confirmed the adequacy of this value (Reference 7). Current analyses, performed

8002250153

using the February 1978 Westinghouse model, predict that the permissible F_q must be further reduced to a value of $F_q = 2.02$ in order to meet the requirements of 10 CFR 50.46. The licensee has proposed an additional reduction of F_q to 1.99 to accommodate recent concerns related to fuel cladding swelling and rupture. Pending resolution of this issue an F_q at rated power of 1.99 is to be incorporated in the plant Technical Specifications.

This progression of small changes have, in total, required a 9.5% reduction in the permissible F_q. The plant has, in fact, typically operated with an actual F_q of the order of 1.7, i.e. approximately 17% below the current reduced regulatory limit.

Original and revised analyses were performed for values of the discharge coefficient, C_D, of 0.6, 0.8 and 1.0. These analyses show that the double-ended cold leg guillotine pipe break with an assumed discharge coefficient of 0.8 is limiting. Similarly, all analyses have predicted that the plant is steam cooling limited. Recent analysis of the limiting break ($C_D = 0.8$) using the February 1978 model predicts an extended time to end of bypass relative to earlier analyses. This difference results in a reduced downcomer water level when the accumulator empties and hence delayed filling of the downcomer, delayed bottom of core recovery, earlier commencement of steam cooling and longer adiabatic heatup. These changes result in higher predicted peak clad temperatures which must be accommodated by a reduction in the permissible F_q. Calculations performed with the February 1978 model with assumed discharge coefficients of 0.6 and 1.0 do not show the delayed end of bypass.

This apparent anomaly is, in fact, due to the Appendix K to 10 CFR 50 definition of end of bypass, prediction of downward flow in the downcomer for the remainder of the blowdown period, and Appendix K requirement that all injected ECCS water prior to end of blowdown be discarded. It is this jump discontinuity inherent in the definition of an acceptable evaluation model, albeit conservative, that has led to the predicted singularity.

Changes of the model per se, and input to the model are enumerated in Reference 2. Model changes (October 1975 to February 1978 model) have previously been reviewed and approved (Reference 3). Input changes based on plant specific parameters such as increased safety injection flow, lower plenum volume revision, paint on containment surfaces, revision of rod backfill pressure have been made to demonstrate margin to 10 CFR 50.46 limits. These changes are based on the actual plant configuration and are hence acceptable. Reduction of the core inlet temperature to the nominal value is conservative. Input changes are predicted to result in a net impact on the predicted peak clad temperature of approximately 25°F. In contrast model changes have a net impact on the predicted peak clad temperature of approximately 100°F.

Additional calculations (Reference 1) have been performed to assess the potential impact of recent concerns related to the LOCA-ECCS fuel clad models included in draft report NUREG-0630 (Reference 4). Total adoption of the NRC fuel clad burst data would: (1) increase the burst node clad temperature, however, the plant would remain steam cooling limited, (2) increase the non-burst node PCT* by 96°F due to an assumed increased channel blockage from 20.4% to 75% (the maximum value for the NRC data), and (3) increase the peak non-burst node temperature at the peak non-burst node elevation of 7.5 feet by an additional 158°F due to behavior of the burst node at the 6.05 ft elevation. These changes would result in a net increase of the predicted PCT by 254°F. The maximum predicted PCT, using the February 1978 model results, is 2171°F. Hence 29°F margin to the 10 CFR 50.46 limit of 2200°F exists. The remaining penalty can be accommodated by a 0.23 reduction in F_q.

Westinghouse has submitted modifications to their standard ECCS evaluation model on two reload applications. These changes involve the slip and break flow models and have been approved for UHI plant application after extensive review. It is estimated that the total benefit of use of these models would be an increase of 0.38 units in F_q. If credit for horizontal slip is dismissed and an added uncertainty is assessed for translation of a phenomenon at blowdown to an effect during reflood, it is our current best technical judgment that application of these model changes would result in an increase of the permissible F_q of 0.20 (Reference 8).

Based on these considerations, we agree that the value of F_q at rated power of 1.99 (2.02 - .23 + .20) be incorporated in the Technical Specifications.

3.0 CONTROL OF F_Q

Constant axial offset control, CAOC, strategy calculations described in references 9 and 10 using revised values of planar peaking factors, F_{xy}, (References 9 and 11) show that, with a ±5%ΔI target band, during steady state and load follow operation, F_q at rated power will remain below a value of 2.11*K(z). K(z) is the normalized Fq as a function of core height. These calculations were performed to support control on ex-core detectors alone, assuming a permissible Fq at power of 2.11. Supplemental monitoring using the axial power distribution monitoring system, APDMS, is needed to control Fq to the revised permissible value of 1.99. The APDMS turn-on point is simply the ratio of 1.99 to 2.11 or 94% of rated power. Action must be taken to restore the axial flux difference to within the ±5% target band at powers 10% less than the APDMS turn-on power or 84%. The calculational methodology applied above has been previously reviewed and accepted.

4.0 TECHNICAL SPECIFICATIONS

Technical Specification changes were requested in references 1 and 11. Reference 1 changes superseded reference 11 requested changes where applicable. As a result of the revised analysis Cook 2 has submitted technical specification changes to support operation with an F_q of 1.99.

*PCT, peak clad temperature

Change No. 3 (reference 11) Revision to Section 4.2.2.2e. The Fxy limit at Rated Thermal Power is increased to an elevation-dependent value. The Fxy limits for Rated Thermal Power within specific planes shall be:

1. $Fxy^{RTP} \leq 1.87$ for all core planes containing control rods

2. $Fxy^{RTP} \leq 1.58$ for all unrodded planes above 6.2 feet.

   $Fxy^{RTP} \leq 1.62$ for all unrodded planes below 6.2 feet.

This proposed change is based on the analysis presented in Reference 9. The licensee has demonstrated by calculation that the use of these values of Fxy results in an F_q*(relative power) less than the 2.11 x K(z) limit, as discussed in Section 3.0.

Change No. 1 (Reference 1) Revision to Limiting Condition of Operation (LCO) 3.2.2 Figure 3.2-2 and Basis Item 3/4 2.1.

The change involves lowering the maximum allowable $F_q(z)$ limits, with the peak value reduced to 1.99. The K(z) curve is renormalized to the lower F_q. As discussed in Section 2.0, an Fq of 1.99 is necessary to meet the requirements of 10 CFR 50.46.

The K(z) curve shown in Figure 3.2-2 was derived from the small break analysis of record, and the revised large break analysis presented in reference 2. This curve is also consistent with the values used in Reference 9.

Since 1.99 is less than the Fq value of 2.11 for which safe operation with CAOC was demonstrated, the APDMS is required for operation above 94% power. Limiting Condition for Operation (LCO) 3.2.6 specifies the APDMS turn-on setpoint of 94%. LCO 3.2.1 is set 10% below the APDMS turn-on point, i.e. 84% power.

The Cook 2 technical specification (LCO 3.2.1, Figure 3.2-1, LCO 3.2.6 and 4.2.6) presently require APDMS turn-on at 94% power, and the upper limit of LCO 3.2.1 is 84% power. Therefore, no changes are required to these specifications.

Changes 4 and 5 of Reference 11 have been superseded (Reference 1).

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability 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 11, 1980

References

1. Letter from J. E. Dolan (AEP) to H. R. Denton (NRC), Serial No. AEP: NRC:00322B, January 9, 1980.

2. Letter from R. S. Hunter (AEP) to H. R. Denton (NRC), Serial No. AEP: NRC 00322C, January 11, 1980.

3. Letter from J. F. Stolz (NRC) to T. M. Anderson (Westinghouse), August 29, 1978.

4. Powers, Meyers, "Cladding Swelling & Rupture Models for LOCA Analysis," Draft Report, NUREG-0630, November 1979.

5. Final Safety Analysis Report, Donald C. Cook Nuclear Plant, Units 1 and 2, Docket 50-315 and 50-316.

6. Roger S. Boyd, Order for Modification of License (Donald C. Cook Nuclear Plant Unit 2) June 6, 1978.

7. Letter from John Tillinghast (AEP) to E. G. Case (NRC) Subject: ECCS Reanalysis, April 18, 1978.

8. G. N. Lauben (NRC) to R. P. Denise (NRC), "Review Status of Considered Revisions to Vendor ECCS Evaluation Models," memorandum, December 21, 1979.

9. WCAP-9566 "The Nuclear Design and Core Management of the D. C. Cook Unit 2 Nuclear Power Plant Cycle 2," August 1979.

10. WCAP-8385 "Power Distribution Control and Load-Following Procedures," September 1974.

11. Letter from J. E. Dolan (AEP) to H. R. Denton (NRC),Serial No. AEP:NRC: 00297, November 2, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 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. 17 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 No. 2 (the facility) located in Berrien County, Michigan. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specification limits for total nuclear peaking factor (F_q) for the Donald C. Cook, Unit No. 2.

The amendment also supersedes the Order for Modification of License for D. C. Cook Unit No. 2, dated June 6, 1978, accordingly, that Order has been terminated.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 9, 1980 as supplemented January 11, 1980, (2) Amendment No. 17 to 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, 1717 H Street, NW., 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 Operating Reactors.

Dated at Bethesda, Maryland, this 11th day of January 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors