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Docket No. 50-315

Indiana & Michigan Electric Company
Indiana & Michigan Power Company
ATTN: Mr. John Tillinghast
Vice President
P. O. Box 18
Bowling Green Station
New York, New York 10004

Gentlemen:

In response to your request dated February 3, 1978, supplemented by your letter dated April 17, 1978, the Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant Unit No. 1.

The amendment involves Technical Specification changes to incorporate modified calculational techniques for the fuel exposure dependent heat flux hot channel factor. The bases for the new calculational techniques are contained in Exxon Nuclear Company Document XN-NP-76-51, WREM-Based Generic PWR ECCS Evaluation Model ENC-WREM-II, Supplements 1 through 4, which the Commission has found acceptable.

In addition, the heat flux hot channel factor limit for Westinghouse Electric Corporation fuel has been modified in accordance with the Commission's exemption issued on May 19, 1978. We have also included a specification change relating to initial implementation of certain fire protection surveillance activities which we inadvertently left out of License Amendment No. 22 dated December 12, 1977. This change as well as other minor changes made to your proposals have been discussed with and agreed to by your staff.

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

5/30/78 3:45 Called D. Shaller and advised plant of issuance. He agreed to submit copy to plant. US mail E. Reeves

Sincerely,

Original Signed By

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

Enclosures:
1. Amendment No. 25 to DPR-53
2. Safety Evaluation
3. Notice

RDiggs
CMiles
JMcGough
JSalztman
Const 1

cc w/encl: See next page

OFFICE ➤			DOR:ORB#1	OELD	DOR:ORB#1	
SURNAME ➤			EAReeves:Tb	[illegible]	ASchwencer	
DATE ➤			5/23/78 5/26/78	5/26/78	5/26/78	

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 – 626-624

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

Docket No. 50-315

May 30, 1978

Indiana & Michigan Electric Company
Indiana & Michigan Power Company
ATTN: Mr. John Tillinghast
Vice President
P. O. Box 18
Bowling Green Station
New York, New York 10004

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In addition, the heat flux hot channel factor limit for Westinghouse Electric Corporation fuel has been modified in accordance with the Commission's exemption issued on May 18, 1978. We have also included a specification change relating to initial implementation of certain fire protection surveillance activities which we inadvertently left out of License Amendment No. 22 dated December 12, 1977. This change as well as other minor changes made to your proposals have been discussed with and agreed to by your staff.

Copies of the related 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. 25 to DPR-58
2. Safety Evaluation
3. Notice

cc w/encl: See next page

cc: Mr. Robert Hunter
Vice President
American Electric Power
Service Corporation
2 Broadway
New York, New York 10004

Gerald Charnoff, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, D.C. 20036

David Dinsmore Comey
Executive Director
Citizens for a Better Environment
59 East Van Buren Street
Chicago, Illinois 60605

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

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

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

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U.S. Environmental Protection Agency
Room 645, East Tower
401 M Street, SW
Washington, D.C. 20460

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

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

INDIANA & MICHIGAN ELECTRIC COMPANY
INDIANA & MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensees) dated February 3, 1978, as supplemented by letter dated April 17, 1978, 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 License No. DPR-58 is hereby amended to read as follows:

"(2) Technical Specifications

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Assistant Director
for Systems & Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 30, 1978

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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

Pages

3/4 0-2
3/4 2-1
3/4 2-2
3/4 2-4
3/4 2-5
3/4 2-6
3/4 2-7
3/4 2-15
3/4 2-16 (Replaced by pages 3/4 2-16, 3/4 2-17)
3/4 2-17 (Replaced by page 3/4 2-18)
3/4 2-18 (Replaced by pages 3/4 2-19 and 3/4 2-20)
3/4 2-19
3/4 2-20 (Added)
3/4 2-21 (Added)
3/4 3-39
B 3/4 2-1
B 3/4 2-2
B 3/4 2-5
B 3/4 2-6
B 3/4 3-3

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the facility shall be placed in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery. Exceptions to these requirements shall be stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION statements.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

SURVEILLANCE REQUIREMENTS (Continued)

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

The provisions of Specification 4.0.4 are not applicable to the performance of surveillance activities associated with fire protection technical specifications, 4.3.3.7, 4.7.9 and 4.7.10, until the completion of the initial surveillance interval associated with each specification.

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about a target flux difference in the range shown on Figure 3.2-4.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ±5% target band about the target flux difference and with THERMAL POWER:

1. Above 75% x T(E)** of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 75% x T(E) of RATED THERMAL POWER.

2. Between 50% and 75% x T(E) of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2
**T(E) is defined on Figures 3.2-3a and 3.2-3b and page 3/4 2-16

LIMITING CONDITION FOR OPERATION (Continued)

c) Surveillance testing of the APDMS may be performed pursuant to Specification 4.3.3.6.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 6 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

b. THERMAL POWER shall not be increased above 75% x T(E) of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The indicated AFD shall be considered outside of its ± 5% target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. POWER OPERATION outside of the ± 5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days with all part length control rods fully withdrawn. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

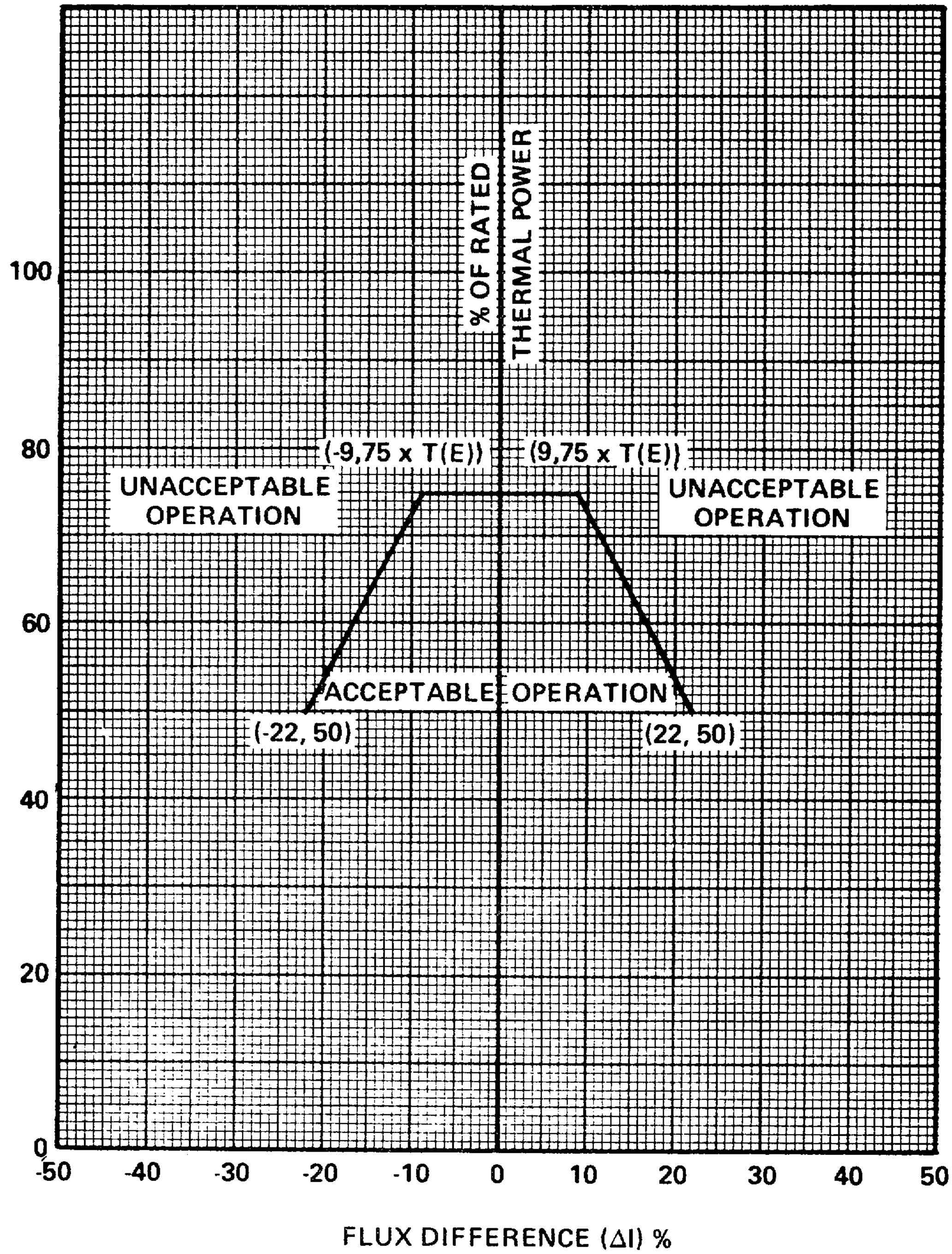

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z,\ell) \leq \frac{[F_Q^L(E_\ell)]}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z,\ell) \leq 2 [F_Q^L(E_\ell)] [K(Z)] \text{ for } P \leq 0.5$$

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

$F_Q^L(E_\ell)$ is the exposure dependent F_Q limit for rod ℓ and is defined on Figure 3.2-3a for Exxon Nuclear fuel, Figure 3.2-3b for Westinghouse fuel and page 3/4 2-15. E_ℓ is the maximum pellet exposure in assembly ℓ. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.
F_Q is defined as the $F_Q(Z,\ell)$ with the smallest margin or the greatest excess of the limit.

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

a. Comply with either of the following ACTIONS:

   1. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly 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 exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritical.

   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; THERMAL POWER may then be increased provided F_Q 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 $F_Q(Z, \ell)$ 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

SURVEILLANCE REQUIREMENTS (Continued)

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

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. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing either bank "D" control rods or any part length rods, and

2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane 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 18.4 ± 2%, 36.6 ± 2%, 54.7 ± 2% and 72.9 ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

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

1. The $F_Q(Z,\ell)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}^{L}, and
2. The effects of F_{xy} on $F_Q(Z,\ell)$ shall be evaluated to determine if $F_Q(Z,\ell)$ is within its limit.

4.2.2.3 When $F_Q(Z,\ell)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z,\ell)$ 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.

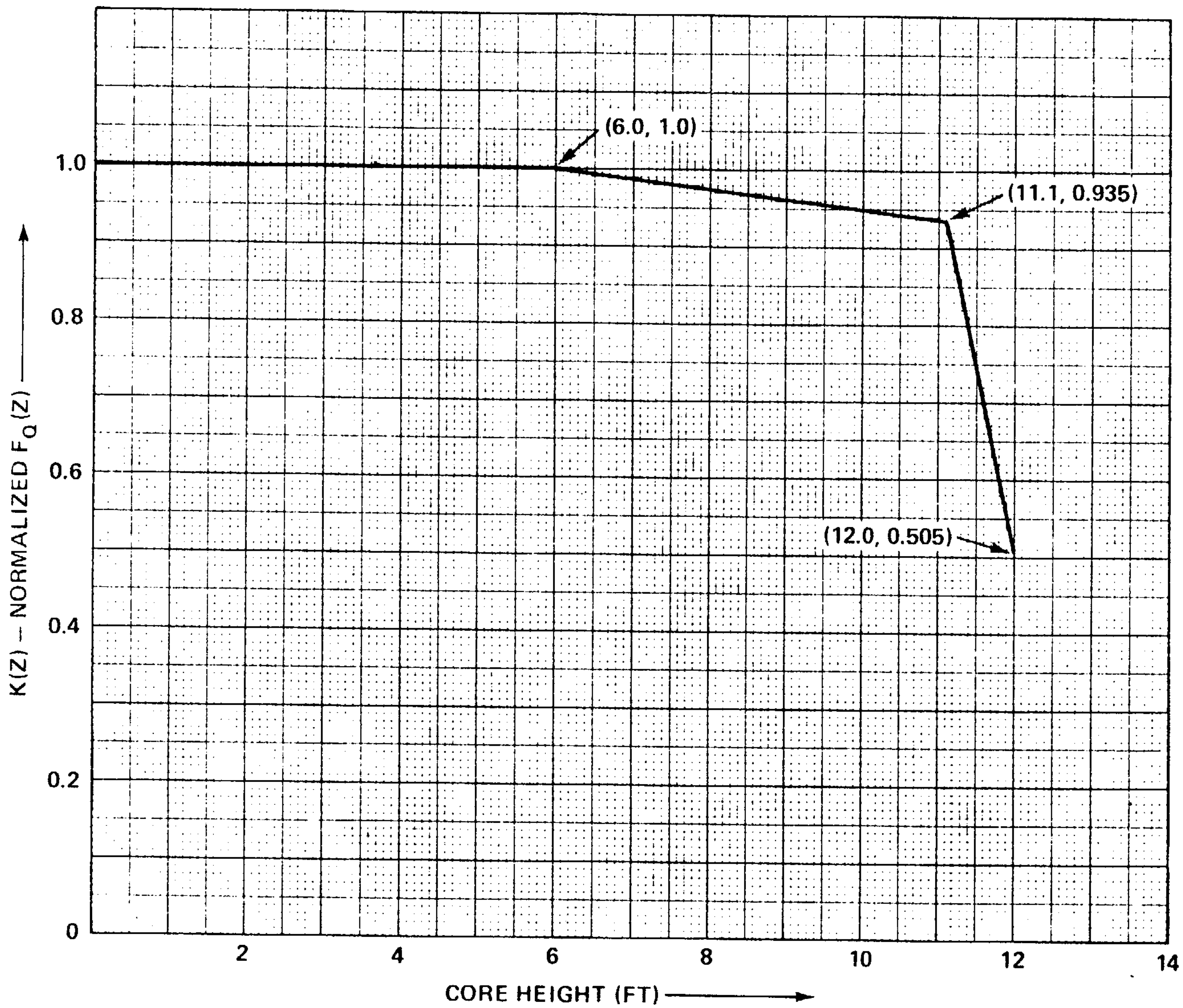

FIGURE 3.2-2 K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

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

$$[F_j(Z)]_S = \frac{[1.95]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)\,F_p}$$

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 84% x T(E) of RATED THERMAL POWER in accordance with:

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

Where:

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

R_{ij} and its associated σ_j may be calculated on a full core or a limiting fuel batch basis as defined on page B3/4 3-3 of basis.

e. $F_{Qi\ell}^{Meas}$ is the limiting total peaking factor in flux map i. The limiting total peaking factor is that factor with least margin to the $F_Q^L(E)$ curve defined in Figure 3.2-3a for Exxon Nuclear Company fuel and in Figure 3.2-3b for Westinghouse fuel.

LIMITING CONDITION FOR OPERATION (Continued)

$T(E)$ is the ratio of the exposure dependent $F_Q^L(E)$ to 1.95 and is defined in Figure 3.2-3a for fuel supplied by Exxon Nuclear Company and in Figure 3.2-3b for fuel supplied by Westinghouse Electric Corporation.

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

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.

g. F_p is an uncertainty factor for Exxon fuel to account for the reduction in the $F_Q^L(E)$ curve due to an accumulation of exposure prior to the next flux map. This correction is only required when $T(E)$ for the limiting fuel segment is less than 1.0. The following F_p factor shall apply:

$F_p = 1.0$	for $T(E) = 1.0$
$F_p = 1.0 + 0.009 \times W$	for $T(E) < 1.0$ where W is the number of effective full power weeks (rounded up to the next highest integer) since the last full core flux map.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODE 1 above 84% x T(E) 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 84% x T(E) 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 84% x T(E) or less of RATED THERMAL POWER within 15 minutes.

The APDMS may be out of service: 1) when incore maps are being taken as part of the Augmented Startup Test Program or 2) when surveillance for determining power distribution maps is being performed.

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.6 at the following frequencies.

1. At least once per 8 hours, and

2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above 84% x T(E) of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

1. At least once per 8 hours, and

2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above 84% x T(E) of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

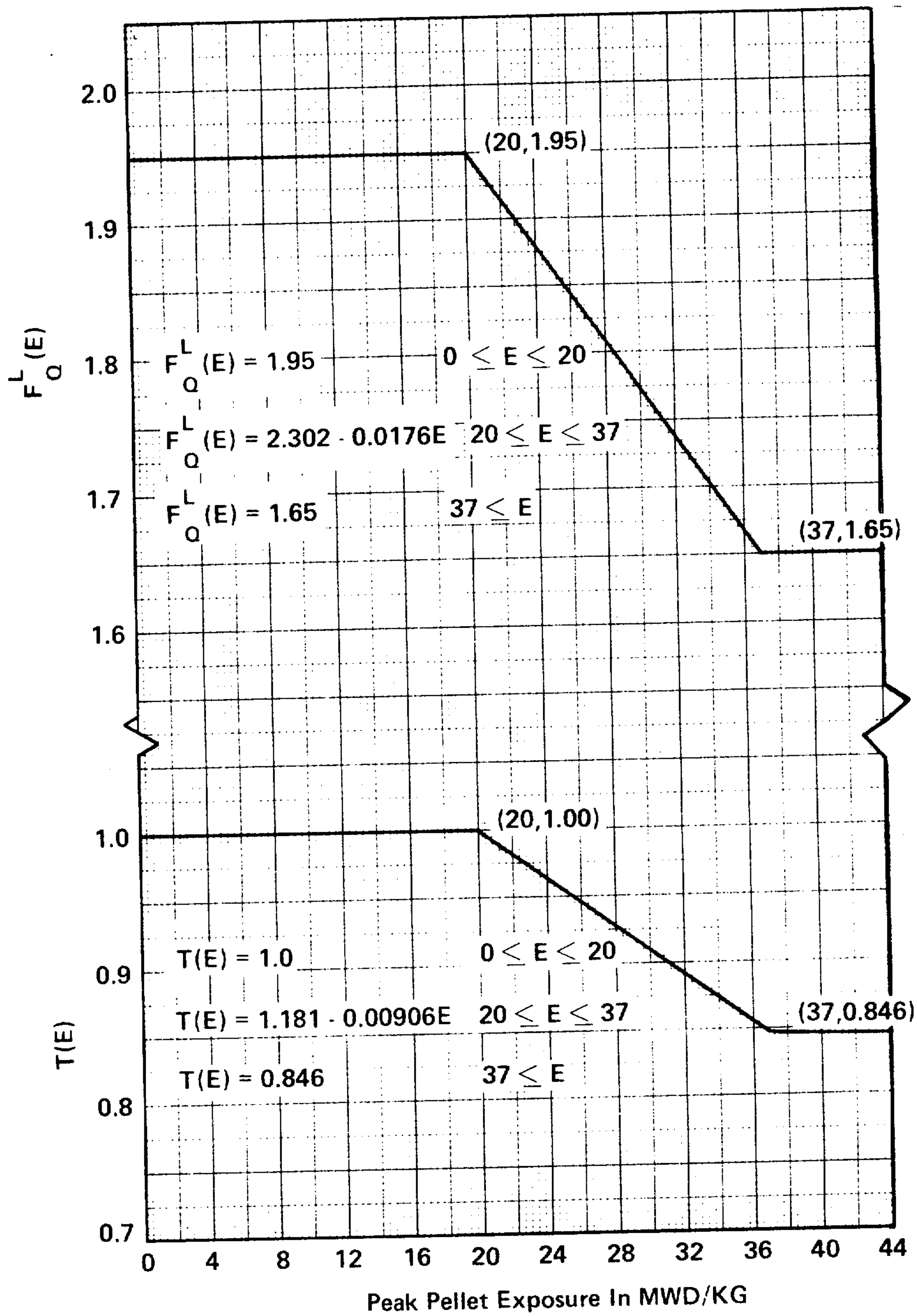

Figure 3.2-3a

Exposure Dependent F_Q Limit, $F_Q^L(E)$, and Normalized Limit T(E) as a Function of Peak Pellet Burnup for Exxon Nuclear Company Fuel.

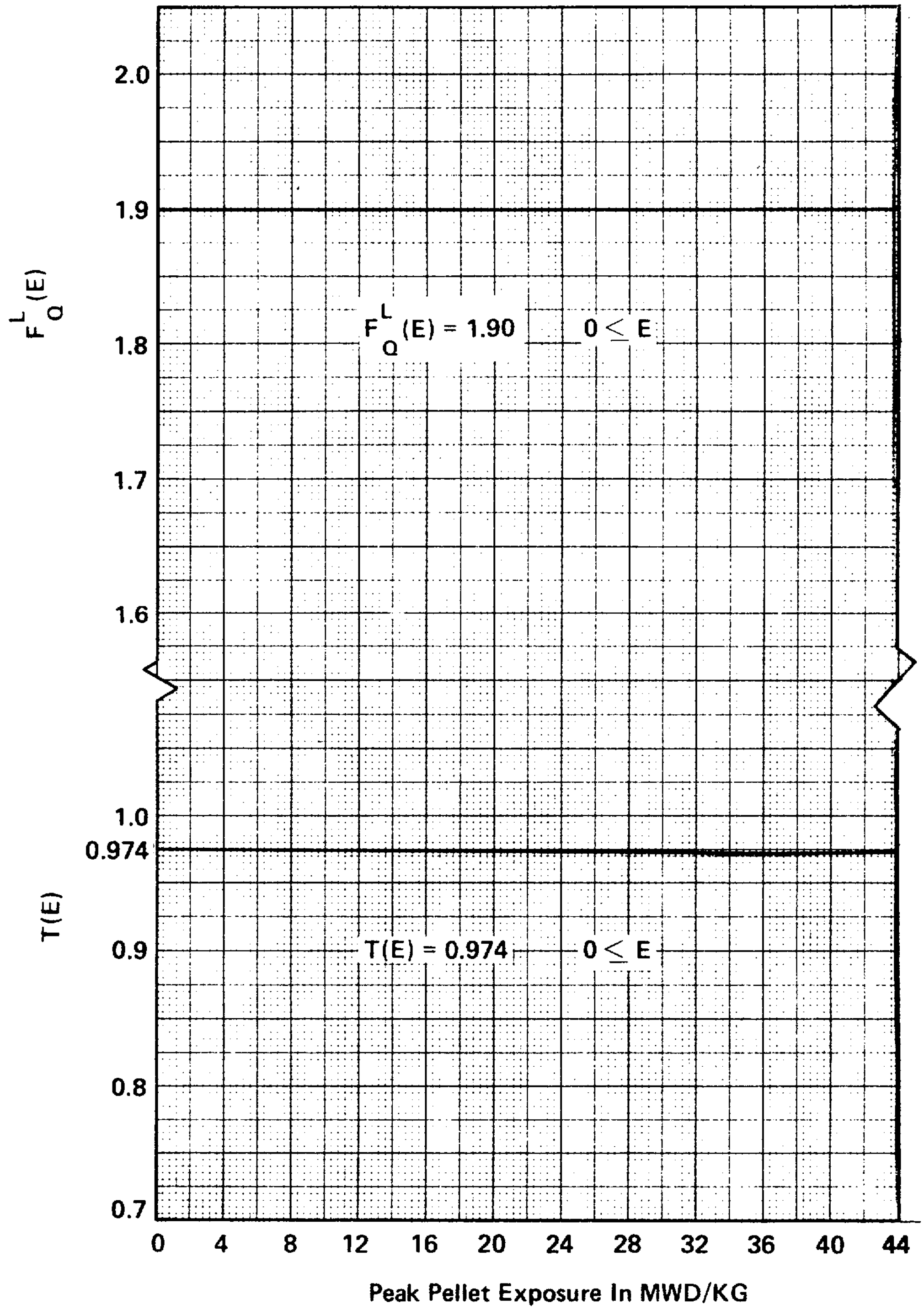

Figure 3.2 - 3b

F_Q Limit, $F_Q^L(E)$, and Normalized Limit T(E) as a Function of Peak Pellet Burnup for Westinghouse Fuel

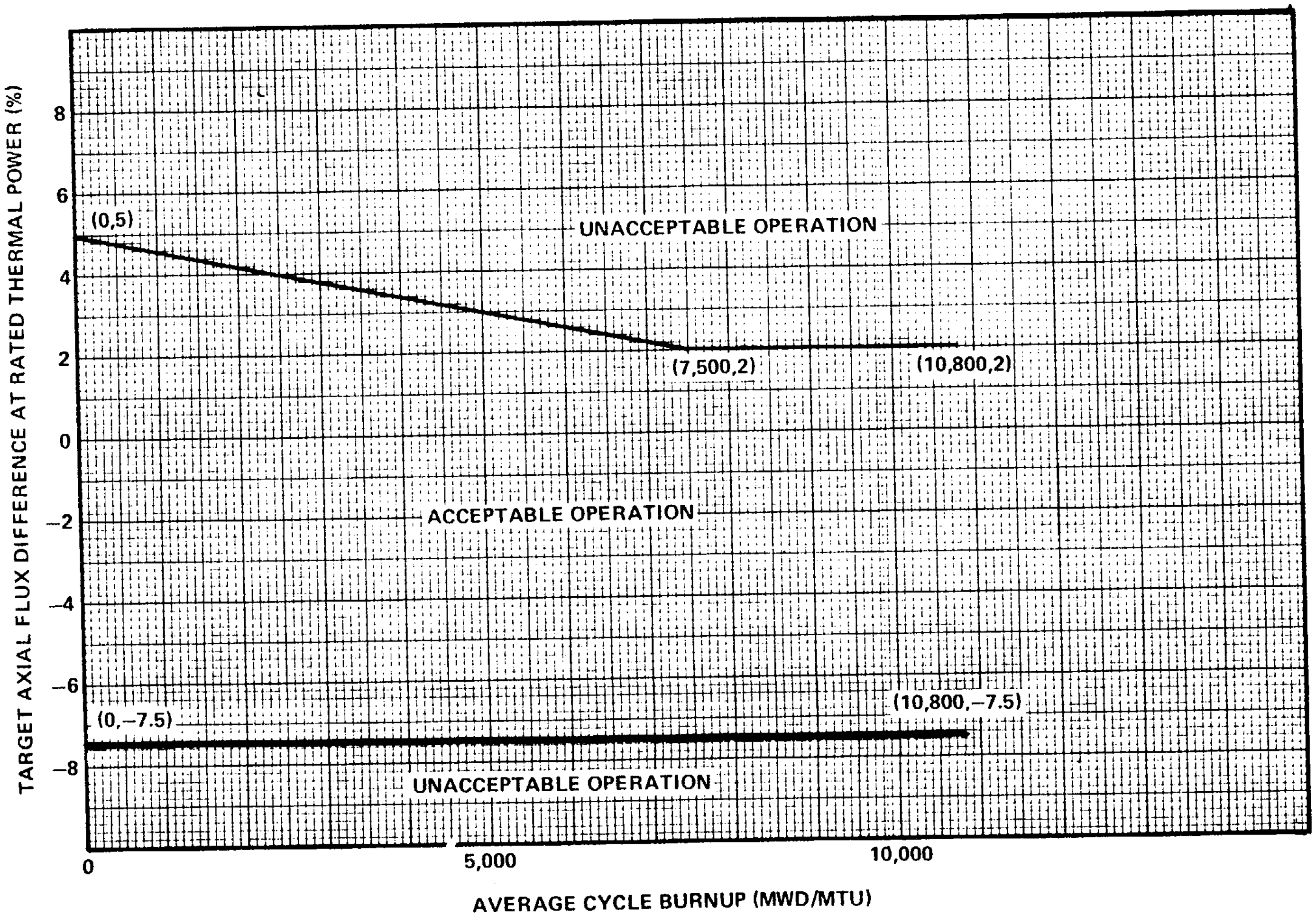

Figure 3.2-4 Target Axial Flux Difference at RATED THERMAL POWER as a Function of Cycle Burnup

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the axial flux difference detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z,\ell)$

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output to be used during its use when required for:

a. Recalibration of the excore axial flux difference detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z,\ell)$.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE seismic monitoring instruments less than required by Table 3.3-7, restore the inoperable instrument(s) to OPERABLE status within 30 days.

b. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed within 24 hours following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

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 minimum DNBR in the core $\geq$ 1.30 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 meet and the ECCS acceptance criteria limit of 2200°F is not exceeded.

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

$F_Q(Z,\ell)$ 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.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. 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.

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 75% x T(E) of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations 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 the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 75% x T(E) of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 75% x T(E) 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.

The upper bound limit (84% x T(E) of RATED THERMAL POWER) on AXIAL FLUX DIFFERENCE assures that the $F_Q(Z,\ell)$ envelope of 2.32 times K(Z) x T(E) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The lower bound limit (50% of RATED THERMAL POWER) is based on the fact that at THERMAL POWER levels below 50% of RATED THERMAL POWER, the average linear heat generation rate is half of its nominal operating value and below that value, perturbations in localized flux distributions cannot affect the results of ECCS or DNBR analyses in a manner which would adversely affect the health and safety of the public.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

A burnup dependent F_Q is specified as a result of the ECCS evaluation in accordance with 10 CFR Part 50 Appendix K and to meet the acceptance criteria of 10 CFR 50.46. The basis for this dependence is given in document XN-76-51, Supplements 1, 2, 3, and 4 for Exxon fuels and the exemption granted by the Commission on May 18, 1978 for Westinghouse fuel.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above 84% x T(E) of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA.

The unit may operate with fuel assemblies supplied by the Exxon Nuclear Company and by Westinghouse Electric Corporation. An F_Q limit has been specified for each of these two fuel types.

3/4.3.3.6 AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)

The OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to 1) monitor the core flux patterns that are representative of the power peaking factor in the limiting fuel rod. The limiting fuel rod is the fuel rod that has the least margin to the exposure dependent F_Q limit curve, and 2) limit the core average axial power profile such that the total power peaking factor F_Q in the limiting fuel rod is maintained within acceptable limits.

R_j factors are used to determine the APDMS setpoint limits $[F_j(Z)]_S$. On a full core basis the R_j and σ_j factors are calculated in accordance with the equations on Pages 3/4 2-15 and 3/4 2-16.

However, near BOC, thimbles not in the region of fuel which contains the limiting total peaking factor, $F_{Qi\ell}$, may not follow the axial power distribution of the hot rod. This situation will manifest itself in the form of large σ_j for thimbles not in the same region as the total peak $F_{Qi\ell}$. In this situation, if the rod with the limiting total peaking factor were to move from one fuel region to another, the neutron flux in the thimble with the smallest σ_j would not necessarily follow the axial power distribution of the power in the new limiting rod.

In order to cope with this difficulty, it is permissible to calculate as many σ_j's and $\bar{R}_j$'s for each thimble as there are fuel types or regions in the core. Each $\bar{R}_j$ and σ_j for a thimble j is to be calculated from the equations on Pages 3/4 2-15 and 3/4 2-16 with the following exception. For each $\bar{R}_j$ and σ_j for thimble j, a different $F_{Qi\ell}$ and T(E) shall be used. The different σ_j's and R_j's for thimble j shall be calculated substituting for $F_{Qi\ell}^{Meas}$ and T(E) the values pertaining to the limiting peak relative power from each fuel region. Obviously for one of these calculations the limiting peak relative power from one region will be the core limiting total peaking factor.

If this option is chosen, the σ_j set to use for APDMS thimble selection and the R_j set to use for the calculation of $[F_J(Z)]_S$ shall be the set obtained using the limiting peak relative power from the same fuel type as the $F_{Qi\ell}$ from the most recent incore flux map.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO LICENSE NO. DPR-58

INDIANA AND MICHIGAN ELECTRIC COMPANY
INDIANA AND MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

Introduction

By application dated February 3, 1978, supplemented by letter dated April 17, 1978, the licensee proposed Technical Specification changes to Facility Operating License No. DPR-58 for the D. C. Cook Nuclear Plant, Unit No. 1. The changes would modify existing calculational techniques for the fuel exposure dependent heat flux hot channel factor (F_Q) and the allowable fuel exposure. The bases are contained in Exxon Nuclear Company (ENC) Document XN-NP-76-51, WREM-Based Generic PWR ECCS Evaluation Model ENC-WREM-II, Supplements 1 through 4.

In addition, the heat flux hot channel factor limit for Westinghouse Electric Corporation fuels would be included in the Technical Specifications. Our evaluation for this change was a part of the Commission's exemption dated May 18, 1978.

Other minor editorial changes, including one relating to initial implementation of certain fire protection surveillance activities, would be incorporated. These changes were discussed with and agreed to by the licensees' staff. These changes have no safety significance and will not be discussed further in this safety evaluation.

Discussion and Evaluation

The power distribution control and monitoring required to meet assumptions contained in the Emergency Core Cooling System analysis for D. C. Cook Nuclear Plant Unit No. 1 are in Technical Specification Section 3/4.2. The F_Q limit for Cycle 2 was based upon an allowable peak pellet exposure of 20 MWD/KG. By letter dated February 3, 1978, the licensees' proposed changes would extend the allowable fuel burnup to 48 MWD/KG. Our evaluation of the ENC Document XN-NP-76-51, contained in Attachment A, concludes that the allowable peak pellet exposure should be 44 MWD/KG. The Technical Specification has been modified accordingly and has been discussed with the licensee.

The Technical Specification affected by the proposed change is Section 3/4.2, Power Distribution Limits. Section 3.2.6 limits the axial power distribution for D. C. Cook Unit No. 1 and specifies use of the axial power distribution monitoring system (APDMS). Since the licensees desire to retain the flexibility to perform APDMS monitoring on either a corewide or fuel batch by batch basis, they proposed Technical Specifications to this end by letter of February 3, 1978. As the result of several telephone conversations and meetings held on March 2 and April 4, 1978, between ENC, the licensees and our staff, the licensees supplemented the earlier application by letter dated April 17, 1978, providing specifications which allow the flexibility desired, but which are not significantly more complicated than the existing specifications. We consider the specifications as modified to be acceptable.

In the April 17, 1978 letter, the licensees' also proposed interim limits on F_Q for Westinghouse Electric Corporation fuels still remaining in the core. These limits were necessary due to the recently discovered Zirconium-water reaction calculational error reported by Westinghouse to the NRC staff by letter dated April 7, 1978. The Commission has issued an exemption dated May 18, 1978, which permits operation with the F_Q limit which has only been added in Technical Specification Figure 3.2-3b. The safety evaluation is a part of the exemption package.

Based on the considerations discussed above, on precedents for allowance of F_Q limits on a fuel batch basis* and on our acceptance (Attachment A) of the ENC Topical Report XN-NP-76-51 through Supplement No. 4, we consider the proposed Technical Specification change to be acceptable.

Environmental Consideration

We have determined that the amendment does 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 amendment involves 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 this amendment.

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*We have approved fuel batches with different initial UO_2 densities and internal pressure which resulted in different fuel densification characteristics for H. B. Robinson Cycle 3 as early as 1974. This is a similar situation to that which exists for D. C. Cook Unit No. 1 for Cycle 3 and is acceptable.

Conclusion

We have concluded, based on the considerations discussed above, that because the portion of the amendment related to incorporation into the license of F_Q values for Westinghouse fuel, as permitted by the Commission's exemption of May 18, 1978, and the correction of inadvertant errors with respect to the Fire Protection program 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, that portion of the amendment does not involve a significant hazards consideration. As to the entire amendment, we have further concluded, based on the considerations discussed above, that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, that such activities will be conducted in compliance with the Commission's regulations, and that the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment A:
Supplement 1 to Safety Evaluation Report
on the Exxon Nuclear Company WREM-Based
Generic PWR-ECCS Evaluation Model Update
ENC-WREM-II, May 18, 1978

Date: May 30, 1978

Supplement 1

Safety Evaluation Report

On The Exxon Nuclear Company

WREM-Based Generic PWR-ECCS

Evaluation Model Update ENC-WREM-II

For Conformance to Requirements of Appendix K

To 10 CFR 50 By the Office of

Nuclear Reactor Regulation

May 18, 1978

1.0 Introduction

In the revised SER on the ENC-WREM-II ECCS model (Ref. 1), the staff identified two items to be addressed prior to subsequent ENC reload analyses. The first item was a more rigourous consideration of the uncertainties affecting the determination of rupture pressure. The second item was the use of vendor supplied containment pressure pending review of the ENC ICECON code. The first item is discussed in detail in this report. In addressing the second item, ENC has performed calculations using ICECON. The transient pressure results were nearly identical to those calculated by the NSSS supplier (Westinghouse). Since the ICECON review is not yet complete it was decided that the NSSS containment pressure should be used through the second ENC reload of D.C. Cook 1. Subsequent analyses should use an approved ENC calculation.

2.0 Rupture Pressure Uncertainty

In reference 1, the staff noted that if the calculated rupture pressure were somewhat higher or lower than the nominal value, the resultant flow blockage during reflood could be substantially greater. The staff identified several items that have a substantial impact on the determination of internal pin pressure. These items were discussed and their uncertainties estimated. ENC was requested to address these and any other related items to determine upper and lower bound uncertainties in rupture pressure. In reference 2 they presented an analysis which included several additional factors. As a result of our review it was decided to expand the analysis to account for the following additional variables:

(a) cladding elasticity, (b) cladding thermal creep, and (c) ruputure data extrapolation. Subsequently, we have audited the individual magnitudes of these and the other fuels-related variables and have concluded that the ENC sensitivity allowances (as tabulated in Table 1.1 of reference 3, 11 and 12) are acceptable. The balance of this report discusses these fuels related variables and the determination of fuel rod plenum gas temperature.

2.1 Elastic Compression

The ENC assessment of the fuel rod internal pressure change due to the elastic compression of the cladding during reactor operation was that an uncertainty of +5 psi would bound this effect. Our comparative assessment, using both thin and thick-wall approximations, yielded rod volume changes which produced rod pressure uncertainties less than those predicted by ENC.

2.2 Thermal Creep

The ENC analysis on the pressure effect resulting from cladding thermal creep was 0 + 0 psi for BOL and 0 + 5 psi for subsequent burnups. To audit these results, the staff used the total creep rate correlation as described in MATPRO (Reference 4). Our finding was that the ENC allowances were adequate.

2.3 Extrapolation of Data

The last additional concern that the staff requested ENC to include in their sensitivity study was the possible error associated with their extrapolation of experimental rupture and blockage data. Since ENC had applied the results of experimental data (References 5, 6, and 7) taken on cladding that was not dimensionally similar with their own cladding, a slight extrapolation was required. An estimation of that uncertainty was made based upon two different extrapolation techniques (thin versus thick-walled assumptions used to evaluate cladding stress). This uncertainty was found to be +5 psi in pin pressure. We concur that this allowance is sufficient.

2.4 Strain Model Changes

The model used for calculating cladding prerupture strain as described in ENC WREM-II was originally formulated by the staff (Reference 8) through use of Hardy's data (Reference 9). ENC has recently modified this formulation by using new and more prototypical PWR fuel rod behavior data from ORNL experiments (References 5, 6, and 7). The observed ORNL single rod circumferential strains are more realistic inasmuch as the ORNL experiments used internal heaters, whereas the Hardy experiment used joule heating, (self-resistance heating of the cladding). The latter method can result in large unrealistic strains due to the uniform cladding temperatures.

Specifically, the end result of the ENC modifications is an increase in the plastic strain pre-exponential multiplier term of about 32%. This 32%-increase is based upon the 95% confidence level on the mean of the ratio between the ORNL average strain data (as calculated by ENC) to the old pre-rupture strain coefficient term from ENC WREM-II. Subsequently, the net effect of this change is a reduced internal rod pressure. (Note: the exponential temperature dependent term in the prerupture strain model has not been altered).

We find that this modified formulation is acceptable.

2.5 Volatile Gases

We have examined the volatile gas content specification imposed by ENC upon their manufactured fuel with those specifications of other fuel vendors, and have subsequently estimated and compared the sorbed gas possible for release in ENC fuel rods with those in other vendors' rods. Our conclusion is that the ENC pressure allowance of 9 + 9 psi (calculated on the assumption of a maximum sorbed gas release fraction which is equivalent to the fission product release fraction) is more than adequate to bound the effect.

2.6 Densification

ENC has examined the resinter data for the D.C. Cook reload fuel and has determined the mean density change due to densification and the upper and lower 95% confidence limits on the mean density change of the two extreme pellet lots. Consequently, the corresponding rod pressure changes are listed for various burnups. We have checked these allowances by use of the ideal gas law and find the ENC values to be acceptable.

2.7 Swelling

The ENC fuel swelling model utilizes one of the largest swelling rates found in the literature. Based upon this conservative rate, the ENC safety analysis of swelling-induced rod pressure increase are listed for various burnups. We have checked these values using the ideal gas law and agree that the ENC analysis is adequate.

2.8 Fission Gas Release

To evaluate the effect of fission gas release upon fuel rod pressure, ENC has utilized their stored energy model, GAPEX. This fuel performance code had been previously approved by the staff; however, recently we have questioned the validity of fission gas release calculations in most fuel performance codes including GAPEX for burnups greater than 20,000 MWd/tu. ENC was informed of this concern and provided with a method of correcting gas release calculations for burnups greater than 20,000 MWd/tu. This method was incorporated into GAPEX and calculations performed to 44,000 MWd/tu. This burnup results in 76% blockage, the maximum for ENC-WREM-II. The ENC calculated allowances, including the high burnup effect for fission gas release, were listed for various burnups. We have run audit calculations using GAPCON-THERMAL 2 with a high burnup correction and have concluded the ENC estimate is conservative and acceptable to 44,000 MWd/tu.

2.9 Rod Growth

The pressure effects associated with cladding rod growth were calculated by ENC. We have checked these values by comparison of measured data taken from H.B. Robinson spent fuel and observed cladding growth strains as documentation by other fuel vendors. We find the ENC values to be satisfactory.

2.10 Fuel Region Gas Temperature

The ENC-WREM I model treats the released gases adjacent to the fuel region as if they were existing at the same temperature as that of the plenum temperature. An apprarently more realistic temperature would be one which more closely represents the pellet temperature. Consequently, TOODEE2 was modified to treat this gas temperature as the average between the pellet surface and the cladding inner surface temperature and ENC has found that this revised temperature results in a 90 psi increase in pin rupture pressure. We find this evaluation to be conservative; in fact, our estimates which were based upon a gas temperature equivalent to the fuel centerline temperature and which ignored expansion effects and axial temperature distribution resulted in a pressure effect of slightly less than 90 psi.
In view of the fact that the anticipated rupture pressure is in closer proximity to the high differential pressure end of

of the ENC flow blockage trough rather than the low differential pressure end, we find the ENC treatment to be conservative.

2.11 Evaluation of Fuel Rod Plenum Gas Temperature

In Reference 1 the staff discussed the importance of the fuel plenum temperature in calculating internal pin pressure and consequently the rupture strain and flow blockage. The current ENC fuel design and LOCA analysis for D.C. Cook 1 causes the rupture to occur in a region of minimum blockage. The current analysis also assumes that the fuel plenum temperature is within 2°F of the coolant saturation temperature. Analysis by the staff indicated the temperature might be as high as 140° F above saturation. This could increase the rupture pressure by 60 psi and put the rupture strain and blockage well outside the minimum range. The critical parameter in determining plenum gas temperature is the heat transfer from the plenum cladding to the fluid. The staff analysis assumed EM type heat transfer from the plenum cladding (e.g., adiabatic refill and FLECHT during reflood). It is ENC's contention that available data from FLECHT and FLECHT-SET experiments support the assumption that rewet of the fuel plenum would occur well before rupture (ref. 2, 3, 10). A three dimensional thermal analysis of the plenum by ENC showed that the gas temperature falls to within 10 degrees of saturation in about 30 seconds after quench (ref. 2). Thus, if the plenum quenches within 15 seconds after reflood begins (as ENC contends) the gas temperature is within 6° F of saturation at the time of postulated rupture. ENC further contends (ref. 10) that the fuel plenum will remain quenched even during refill. They argue that flashing liquid in the upper head and upper plenum will flow downward into the core. They state that the upper head "is practically full of liquid (quality of 36%) at the time of break flow reversal". It should be noted that this quality translates to a void fraction of about 97%. Their calculations indicate that the upper plenum quality is 35% ($\alpha \cong 99.8\%$). The BE/EM study showed a quality of about 80% ($\alpha \cong 99.95\%$) at end-of-bypass and 100% qualities in both plena at BOCREC. We do not believe that these conditions could guarantee fuel plenum quench during refill.

A revised detailed thermal analysis (ref. 10) ignored adiabatic heatup, reflecting the ENC arguments regarding refill heat transfer. This would not be appropriate for evaluation model analysis from the standpoint of Appendix K or for the reasons discussed above.

Non-typicality of size and structure of the FLECHT-SET upper plenum makes use of top quench data questionable.

Data from the low flooding rate FLECHT experiments appear to be inconclusive with respect to top quench. Three kinds of temperature data are available at elevations above 11 feet. They are heated rods, instrument thimbles, and steam temperatures probes. Thermocouples on heated rods at the 11 and 12 feet elevations consistantly showed quench time greater than 100 seconds for conditions applicable to the D.C. Cook analysis. For the steam probes the same conclusion generally applied with a few exceptions. The steam probe was at the 12.5 ft. elevation and therefore not subject to noticeable radiation from surrounding rods. Thimble temperatures were not available from the cosine series tests but one T/C was provided for the skewed tests at the 12 ft. elevation. This measurement generally showed little, if any, temperature rise indicating quench existed throughout the test.

It is ENC's position that this T/C provides the best indication of what the cladding temperature around the fuel plenum would be, since, like the fuel plenum, it corresponds to an unpowered hollow tube at the top of the bundle. Reference 10 states that the measurement is even conservative since it is at the bottom of the "plenum" and is subject to thermal radiation from surrounding powered rods. The staff does not believe that this can be shown definitively. The top of the last FLECHT grid spacer is exactly at the top of the heated length and would play an unknown role in radiant heat transfer from the tops of surrounding heated rods. The top grid spacer in an ENC fuel assembly is located around the fuel plenum and would have a different effect on thermal radiation and plenum quench.

While it is not known if the thimble temperatures are a conservative representation of the fuel plenum surface temperature, the heated rod T/C and the steam probes measurements would appear to be conservative. Thus, measurements not intended to represent reactor fuel rod plenum temperatures in FLECHT show widely divergent results. No data of the type required is currently available on fuel pins under reflood conditions. ENC has not provided experimental evidence of their own. NRU experiments should be able to provide applicable information in a few years. We believe that for making a best estimate evaluation of fuel plenum temperature, it is acceptable to use the thimble T/C data. However, for determining pin pressure uncertainties on the high side other factors should be considered. Typicality of the thimbles regarding thermal inertia and heat transfer cannot be readily demonstrated. Uniformity of water availability is not also a certainty (heated rod FLECHT data shows great variation in quench time at the 12 ft. elevation). Therefore the more conservative heated rod and steam probe data should be considered when evaluating fuel plenum temperatures on the high side of the pressure uncertainty band.

To address the staff's concerns, ENC has developed a new model to determine the upper bound for the fuel plenum temperature (ref. 3). To evaluate the fluid conditions adjacent to the plenum, 12.5 ft. steam probe data from the FLECHT cosine series was used. Two correlations were derived from this data: 1) the plenum clad quench time is taken to be the turnaround time of the steam temperature probe.

For a steam probe designed to exclude the pressure of liquid from the measurement, temperature turnaround is a clear indication of the presence of water in the channel outside the probe; 2) the sink temperature adjacent to the plenum is taken as maximum steam probe temperature. Multiple regression analyses were performed on the data for both correlations as functions of pressure, flooding rate, initial wall superheat, and linear heat rate. The data sets chosen by ENC for their correlations are appropriate.

The correlations are applied conservatively in their new model. Adiabatic conditions are applied during refill. The plenum clad temperature at BOCREC (482 °F) is taken as the steam temperature at that time. Bounding values of maximum reflood fluid temperature and quench time are determined from the correlations. The fluid temperature is ramped from 482 °F at BOCREC to the maximum temperature at the plenum quench time. From that time, the fluid adjacent to the plenum is assumed to be at saturation. The HEATING code model described in reference 2 is then used to determine the plenum gas temperature. This calculation showed that the gas was 40 °F above saturation. This resulted in an increase of 16 psi in the upper bound pressure uncertainty. The staff finds this model and the uncertainty acceptable.

3.0 Conclusions

The staff has completed its review of the ENC-WREM-II topical report supplements. The rupture pressure uncertainty analysis is satisfactory. The NSSS containment pressure calculation should be used for the second ENC reload of D.C. Cook 1. Subsequent ice condenser containment analyses should use an approved ENC calculation.

The updates to the ENC ECCS model as described in the ENC-WREM-II topical reports (ref. 3, 11, 12, 13) are acceptable for use in analyzing LOCA's in low back pressure containment plants such as ice condenser plants. The reports may thus be referenced in licensing applications as an accepted ECCS evaluation model for these plants.

4.0 References

1. Revision 1 to Staff SER on ENC WREM-Based Generic PWR ECCS Evaluation Model ENC-WREM-II.

2. XN-76-51 (Supp. 1) Flow Blockage and Exposure Sensitivity Study for D.C. Cook Unit 1 Reload Fuel Using ENC-WREM-II Model.

3. XN-NF-76-51 (P) (Supp. 3) Flow Blockage and Exposure Sensitivity Study for ENC D.C. Cook Unit 1 Reload Fuel Using ENC-WREM-II Model.

4. "MATPRO - Version 09: A Handbook of Materials Properties For Use In The Analysis of Light Water Reactor Fuel Rod Behavior, "INEL Technical Report, TREE-NUREG-1005, December 1976.

5. R.H. Chapman, "Multirod Burst Test Program Quarterly Progress Report for July - September 1975," ORNL Technical Report, ORNL-TM-5154, December 1975.

6. R.H. Chapman, "Multirod Burst Test Program Quarterly Progress Report for October - December 1975," ORNL Technical Report, ORNL/NUREG/TM-10, May 1976.

7. R.H. Chapman, "Multirod Burst Test Program Quarterly Progress Report for January - March 1976," ORNL Technical Report, ORNL/NUREG/TM-36, September 1976.

8. "WREM: Water Reactor Evaluation Model, Revision 1," USNRC Division of Technical Review, NUREG-75-056, May 1975.

9. D.G. Hardy, "High Temperature Expansion of Zircaloy Tubing," paper from Topical Meeting on Water Reactor Safety, March 26-28, 1973, USAEC Report, CONF-730304, pg. 254.

10. XN-NR-76-51 (P) Supp. 2 Flow Blockage and Exposure Sensitivity Study for D.C. Cook Unit 1 Reload Fuel Using ENC WREM-2 Model.

11. XN-76-51 Donald C. Cook Unit 1 LOCA Analyses Using The ENC WREM-Based PWR ECCS Evaluation Model (ENC-WREM-II) Oct. 1976.

12. XN-76-27 "Exxon Nuclear Company WREM-Based PWR ECCS Evaluation Model Update ENC-WREM-II" and Supplement 1, Sept. 1976.

13. XN-NF-76-51, Supp. 4 Flow Blockage and Exposure Sensitivity Study for D.C. Cook Unit 1 Reload Fuel Using ENC-WREM-II Model.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-315

INDIANA & MICHIGAN ELECTRIC COMPANY
INDIANA & MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT OF FACILITY
OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 to Facility Operating License No. DPR-58, issued to Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensees), which revised the license and Technical Specifications for operation of the Donald C. Cook Nuclear Plant, Unit No. 1 (the facility) located in Berrien County, Michigan. The amendment is effective as of its date of issuance.

The amendment involves Technical Specification changes to (1) extend heat flux hot channel factor limits to greater fuel burnup values based on modified calculational techniques. The bases for the new calculational techniques are contained in Exxon Nuclear Company Document XN-NP-76-51, WREM-Based Generic PWR ECCS Evaluation Model ENC-WREM-II, Supplements 1 through 4, which the Commission finds acceptable. The amendment also (2) formally incorporates into the license heat flux hot channel factors for Westinghouse fuel as permitted by the Commission's exemption issued on May 18, 1978, and (3) corrects an inadvertant error in the Technical Specifications related to Fire Protection.

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 Proposed Issuance of Amendment to Facility Operating License in connection with item (1) above was published in the Federal Register on February 24, 1978 (43 FR 7748). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on that portion of the amendment. Prior public notice of the amendment with respect to items (2) and (3) above, was not required since those portions of 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 February 3, 1978, as supplemented by letter dated April 17, 1978, (2) Amendment No. 25 to License No. DPR-58, (3) the Commission's related Safety Evaluation, and (4) the Commission's exemption dated May 18, 1978. 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 Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A single copy of items (2), (3) and (4) 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 30th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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