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JAN 14 1983

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. 48 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications and imposition of a license condition in response to your application transmitted by letter dated April 7, 1982, as supplemented by letters dated June 11 and June 30, 1982, July 8, 1982, September 30, 1982, December 9 and December 22, 1982 and January 12, 1983.

This amendment approves the Cycle 4 reload, the increase in power level from 3391 to 3411 megawatts thermal, and changes the related Technical Specifications. A License Condition for Cycle 4 is imposed as is a general condition prohibiting Cycle 5 operation until further approval is obtained from the NRC.

Copies of the Safety Evaluation and Environmental Impact Appraisal and the Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

ORIGINAL SIGNED

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

Enclosures:
1. Amendment No. 48 to DPR-74
2. Safety Evaluation and Environmental Impact Appraisal
3. Notice of Issuance and Negative Declaration

cc w/enclosures:
See next page

go to form 8 FRN + AMDT only.

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USGPO: 1981—335-960

Mr. John Dolan
Indiana and Michigan Electric Company

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James G. Keppler
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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. 48
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 April 7, 1982, as supplemented by letters dated June 11 and June 30, 1982, July 8, 1982, September 30, 1982, December 9 and December 22, 1982 and January 12, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations.

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended by the addition of paragraph 2.C.3 (p) to Facility Operating License No. DPR-74 to read as follows:

    "Operation during Cycle 4 with Exxon Nuclear Company 17x17 fuel assemblies is permitted subject to:

    (1) the satisfactory completion by the licensee of the following activities on or before the timesindicated:

    i. Complete and submit an analysis within one year from the issuance of this amendment using NRC approved methodology to comply with fuel assembly structural acceptance criteria in Appendix A to SRP-4.2 for the design seismic event.

    ii. Continue to comply with the operating restrictions imposed by the rod drop accident analysis until such time as the generic review of this event has been completed and any analyses required as a result of that review are performed.

    iii. Following NRC approval of the RODEX 2 thermal analysis code, and prior to 10,000 MWD/MTU average fuel assembly burnup of the ENC 17x17 fuel assemblies during Cycle 4 operation, resubmit the cladding strain, oxidation, and pellet/cladding interaction calculations with an approved version of the RODEX 2 code, and

    (2) the following conditions pending receipt and approval of confirmatory and other information on transients and accidents as noted in the Safety Evaluation and Environmental Impact (Report) issued with Amendment No. 48 :

    i. The PTS-PWR2 model, and its adjunct thermal-hydraulic models, cannot be used by the licensee to justify changes to the set points and related uncertainties, and instrumentation response and delay time, for Reactor Protection System (RPS) and Engineered Safeguards Features (ESF) initiation and actuation functions.

ii. The maximum value of $F_Q(Z)$ for the reactor core is to be limited to a maximum value of 2.04 irrespective of any subsequent changes to this value permitted by revisions to LOCA calculations.

iii. No change is allowable to the current Technical Specifications in respect of moderator temperature coefficients.

In addition to the conditions set forth above, the licensee is not authorized to operate in Cycle 5, modes 1 and 2, until it has satisfactorily resolved the issues identified in the Safety Evaluation and Environmental Impact Appraisal (Report) issued with Amendment No. 48 and other Cycle 5 regulatory requirements."

4. Within 30 days after the effective date of this amendment, or such other time as the Commission may specify, the licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

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

FOR THE NUCLEAR REGULATORY COMMISSION

G. C. Lainas

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 14, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 1-1 | 1-1 |
| 2-7 & 2-8 | 2-7 & 2-8 |
| 2-9 | 2-9 |
| 3/4 2-5 thru 3/4 2-8a | 3/4 2-5 thru 3/4 2-8a |
| --------- | 3/4 2-8b |
| 3/4 2-9 thru 3/4 2-12 | 3/4 2-9 thru 3/4 2-12 |
| 3/4 2-17 | 3/4 2-17 |
| 3/4 2-18 | 3/4 2-18 |
| 3/4 2-19 | 3/4 2-19 |
| B2-1 & B2-2 | B2-1 & B2-2 |
| B3/4 2-1 | B3/4 2-1 |
| B3/4 2-2 | B3/4 2-2 |
| B3/4 2-4 | B3/4 2-4 |
| 2-1 & 2-2 | 2-1 & 2-2 |
| 2-3 & 2-4 | 2-3 & 2-4 |

## 1.0 DEFINITIONS

### DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

### THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

### RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

### OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

### ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

### OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

## TABLE 2.2-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

### NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+t_1 S}{1+\tau_2 S}\right) (T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: $\Delta T_o$ = Indicated $\Delta T$ at RATED THERMAL POWER

$T$ = Average temperature, °F

$T'$ = Indicated $T_{avg}$ at RATED THERMAL POWER $\leq$ 574.0°F

$P$ = Pressurizer pressure, psig

$P'$ = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for $T_{avg}$ dynamic compensation

$\tau_1$ & $\tau_2$ = Time constants utilized in the lead-lag controller for $T_{avg}$ $\tau_1$ = 33 secs, $\tau_2$ = 4 secs.

$S$ = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

| Operation with 4 Loops | Operation with 3 Loops |
|---|---|
| $K_1$ = 1.267 | $K_1$ = 1.116 |
| $K_2$ = 0.01607 | $K_2$ = 0.01607 |
| $K_3$ = 0.000926 | $K_3$ = 0.000926 |

and $f_1$ ($\Delta$I) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 40 percent and + 3 percent, $f_1$ ($\Delta$I) = 0 (where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 40 percent, the $\Delta$T trip setpoint shall be automatically reduced by 1.8 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 3 percent, the $\Delta$T trip setpoint shall be automatically reduced by 2.2 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

where: $\Delta T_0$ = Indicated $\Delta T$ at rated power

$T$ = Average temperature, °F

$T"$ = Indicated $T_{avg}$ at RATED THERMAL POWER $\leq$ 574.0°F

$K_4$ = 1.078

$K_5$ = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$K_6$ = 0.00197 for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for $T_{avg}$ dynamic compensation

$\tau_3$ = Time constant utilized in the rate lag controller for $T_{avg}$ $\tau_3$ = 10 secs.

$S$ = Laplace transform operator

$f_2(\Delta I)$ = 0 for all $\Delta I$

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 4 percent.

## POWER DISTRIBUTION LIMITS

## 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:

| Westinghouse Fuel | Exxon Nuclear Co. Fuel | |
|---|---|---|
| $F_Q(Z) \leq \frac{[1.97]}{P} [K(Z)]$ | $F_Q(Z) \leq \frac{[2.04]}{P} [K(Z)]$ | $P > 0.5$ |
| $F_Q(Z) \leq [3.94] [K(Z)]$ | $F_Q(Z) \leq [4.08] [K(Z)]$ | $P \leq 0.5$ |

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

and K(Z) is the function obtained from Figure 3.2-2 for Westinghouse fuel and Figure 3.2-2(a) for Exxon Nuclear Company fuel.

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 $\overline{R}$.

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 $F_Q(Z)$ shall be determined to be 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 $F_Q(Z)$ 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. This product defined is $F_Q^M(Z)$.

c. Satisfying the following relationships at the time of the target flux determination.

| Westinghouse Fuel | Exxon Nuclear Co. Fuel | |
|---|---|---|
| $F_Q^M(Z) \leq [\frac{1.97}{P}] [\frac{K(Z)}{V(Z)}]$ | $F_Q^M(Z) \leq [\frac{2.04}{P}] [\frac{K(Z)}{V(Z)}]$ | $P > .5$ |
| $F_Q^M(Z) \leq [3.94] [\frac{K(Z)}{V(Z)}]$ | $F_Q^M(Z) \leq [4.08] [\frac{K(Z)}{V(Z)}]$ | $P \leq .5$ |

where

$F_Q^M(Z)$ is the measured total peaking as a function of core height.

V(Z) is the function defined in Figure 3.2-3 which corresponds to the target band, K(Z) is defined in Figure 3.2-2 for Westinghouse fuel and Figure 3.2-2(a) for Exxon Nuclear Co. fuel, P is the fraction of RATED THERMAL POWER.

d. Measuring $F_Q(Z)$ in conjunction with the target flux difference and target band determination, according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 effective full power days, whichever occurs first.

*During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

e. With successive measurements indicating an increase in peak pin power, $F_{\Delta H}$, with exposure, either of the following additional actions shall be taken.

1. $F_Q^M(Z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

2. $F_Q^M(Z)$ shall be measured and a target axial flux difference reestablished at least once per 7 effective full power days until 2 successive maps indicate that the peak pin power, $F_{\Delta H}$, is not increasing.

f. With the relationship specified in 4.2.2.2.c not being satisfied either of the following actions shall be taken.

1. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied and remeasure the target axial flux difference.

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the maximum percent calculated with the following expressions with V(Z) corresponding to the target band and $P \geq .5$:

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z)}{\frac{1.97}{P} \times [K(Z)]}\right] - 1\right] \times 100 \qquad \text{Westinghouse Fuel}$$

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z)}{\frac{2.04}{P} \times [K(Z)]}\right] - 1\right] \times 100 \qquad \text{Exxon Nuclear Company Fuel}$$

g. The limits specified in 4.2.2.2.c and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 10% inclusive.
2. Upper core region 90% to 100% inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, 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.

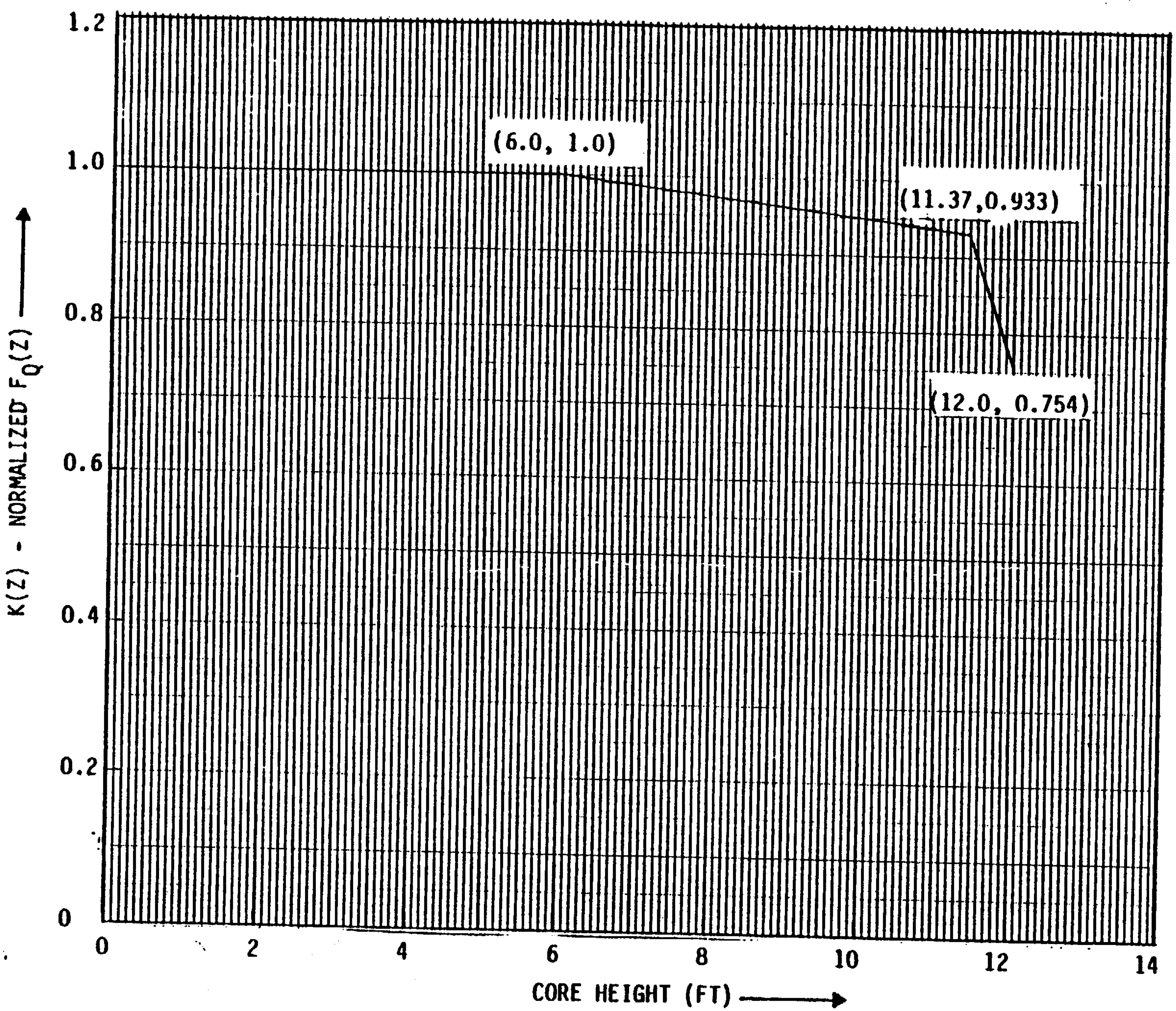



FIGURE 3.2-2 D. C. COOK UNIT 2, K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT FOR WESTINGHOUSE FUEL

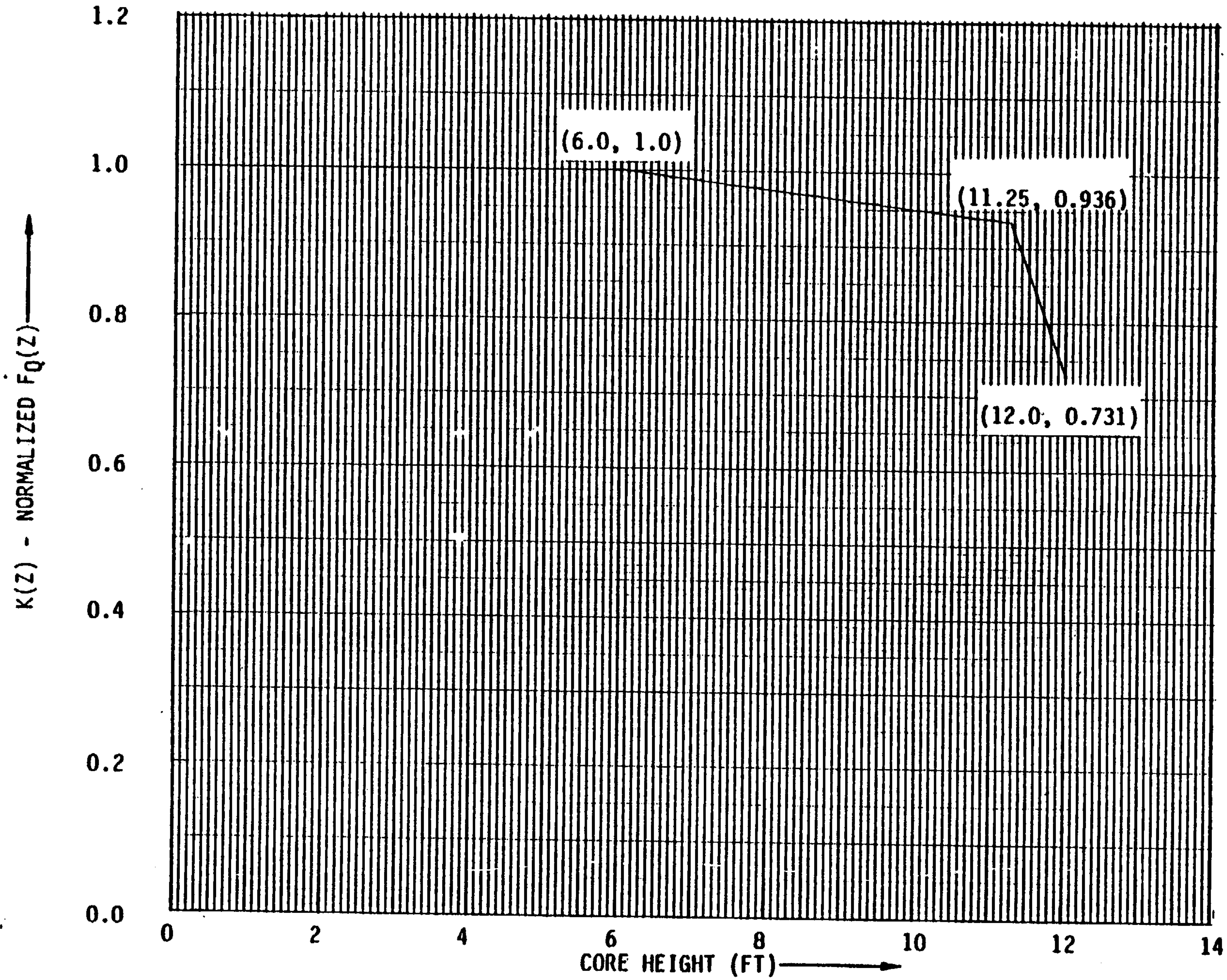



FIGURE 3.2-2(a) D. C. COOK UNIT 2, K(Z) - NORMALIZED $F_Q$ (Z) AS A FUNCTION OF CORE HEIGHT FOR EXXON NUCLEAR CO. FUEL

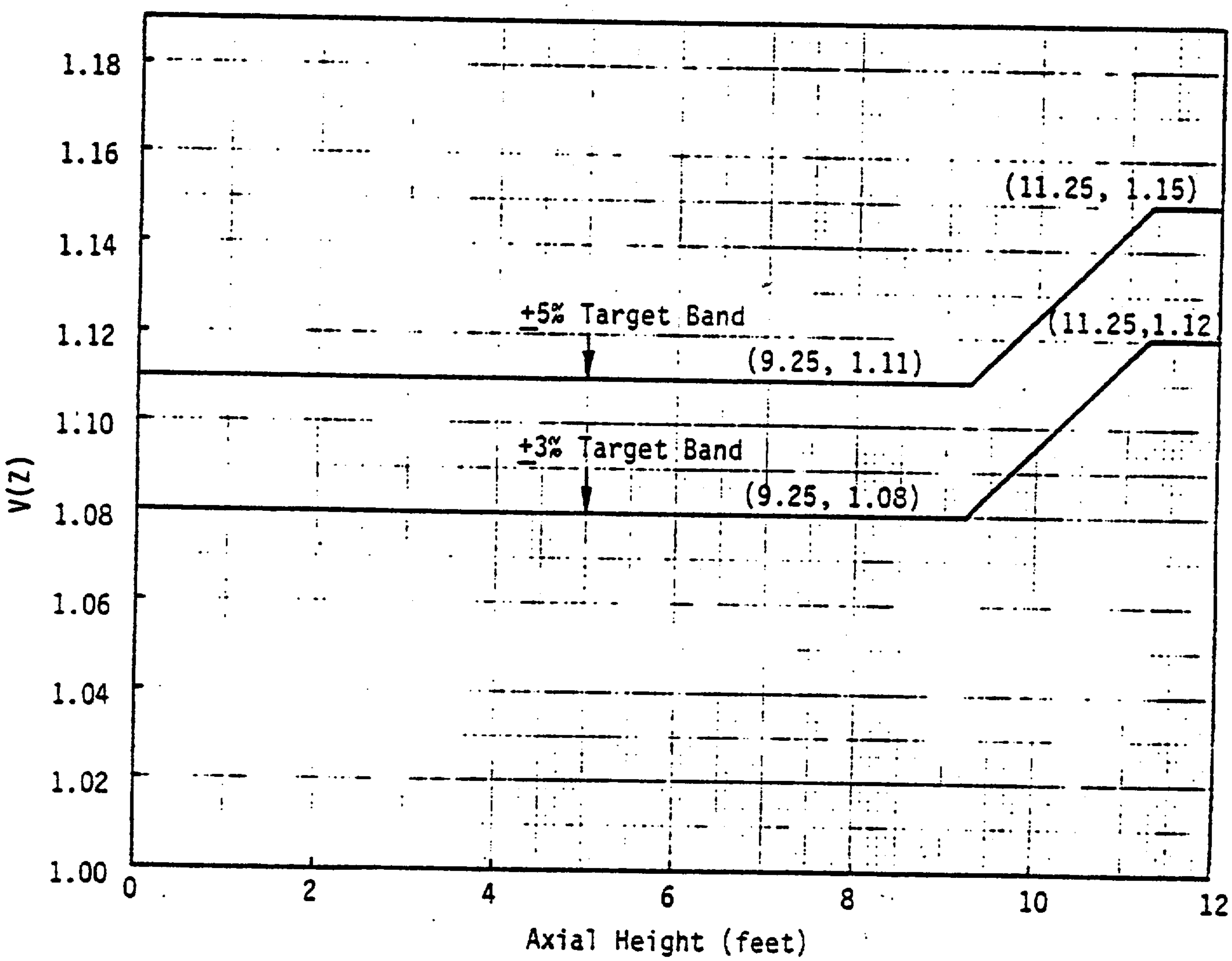



Figure 3.2-3 V(Z) As A Function of Core Height

## RCS FLOW RATE AND R

## LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figures 3.2-4 and 3.2-5 for 4 and 3 loop operation, respectively.

Where: Westinghouse Fuel / Exxon Nuclear Company Fuel

a. Westinghouse Fuel: $R = \dfrac{F^N_{\Delta H}}{1.48\,[1.0 + 0.2\,(1.0 - P)]}$ Exxon Nuclear Company Fuel: $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (as applicable):

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to $\leq$55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER Limit required by ACTION items a.2 and/or b above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total

ACTION: (Continued)

flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (as applicable) prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining $\geq$95% of RATED THERMAL POWER.

## SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-4 or 3.2-5 (as applicable):

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

Where: Westinghouse Fuel

$$R = \frac{F^N_{\Delta H}}{1.48\ [1.0 + 0.2\ (1.0 - P)]}$$

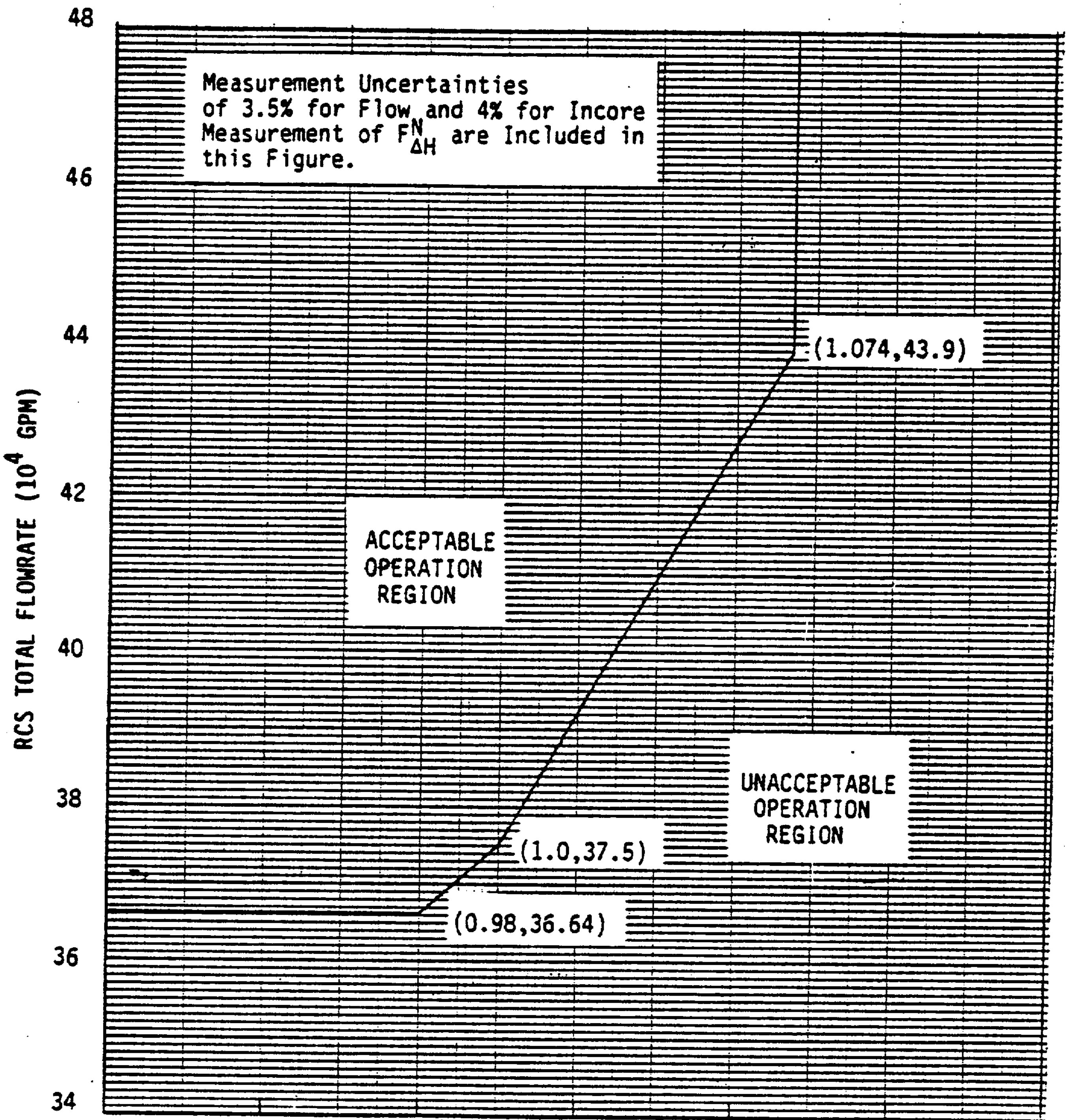
Exxon Nuclear Company Fuel

$$R = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$$

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figures 3.2-4 and 3.2-5 include measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

4.2.3.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

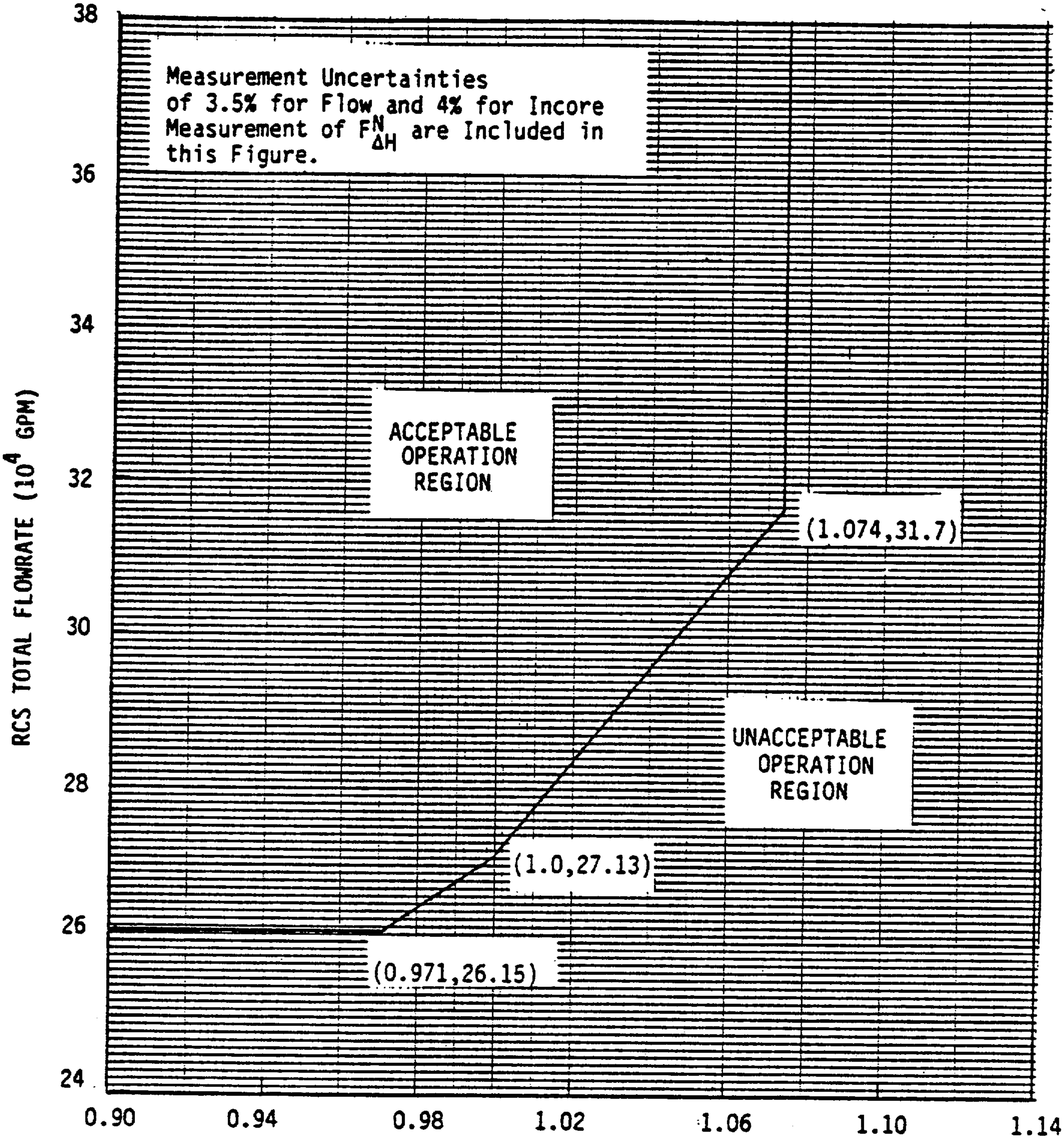
4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.




$R=F^N_{\Delta H}/1.48[1.0+0.2(1.0-P)]$ WESTINGHOUSE FUEL

$R=F^N_{\Delta H}/1.49[1.0+0.2(1.0-P)]$ EXXON NUCLEAR CO. FUEL

FIGURE 3.2-4 RCS TOTAL FLOWRATE VERSUS R - FOUR LOOPS IN OPERATION




$R=F^N_{\Delta H}/1.48[1.0+0.2(1.0-P)]$ WESTINGHOUSE FUEL

$R=F^N_{\Delta H}/1.49[1.0+0.2(1.0-P)]$ EXXON NUCLEAR CO. FUEL

FIGURE 3.2-5 RCS TOTAL FLOWRATE VERSUS R - THREE LOOPS IN OPERATION

## AXIAL POWER DISTRIBUTION

### LIMITING CONDITION FOR OPERATION

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

Westinghouse Fuel

$$[F_j(Z)]_S = \frac{[1.97]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1+\sigma_j)(1.07)}$$

Exxon Nuclear Company Fuel

$$[F_j(Z)]_S = \frac{[2.04]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1+\sigma_j)(1.07)}$$

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 Westinghouse Fuel and Figure 3.2-2(a) for Exxon Nuclear Company Fuel 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 100% or APL (whichever is less) 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 has a measured peaking factor without uncertainties or densification allowance of $F_Q^{Meas}$.

LIMITING CONDITIONS FOR OPERATION (Continued)

$\sigma_j$ 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 the minimum percent of RATED THERMAL POWER indicated by the relationships. #

$$APL = \text{min over } Z \text{ of } \frac{1.97\ K(Z)}{F_Q(Z) \times V(Z)} \times 100\% \qquad \text{Westinghouse Fuel}$$

$$APL = \text{min over } Z \text{ of } \frac{2.04\ K(Z)}{F_Q(Z) \times V(Z)} \times 100\% \qquad \text{Exxon Nuclear Company Fuel}$$

where $F_Q(Z)$ is the measured $F_Q(Z)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty, at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the function defined in Figure 3.2-3 which corresponds to the target band. The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

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 th $F_j(Z)$ 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 APL or less of RATED THERMAL POWER.

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

LIMITING CONDITIONS FOR OPERATION (Continued)

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

## 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 3.3.3.7 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 APL 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 APL 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.

## 2.1 SAFETY LIMITS

## BASES

### 2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the XNB correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the correlation DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Uncertainties in primary system pressure, core temperature, core thermal power, primary coolant flow rate, and fuel fabrication tolerances have been included in the analyses from which Figures 2.1-1 and 2.1-2 are derived.

The curves are based on a nuclear enthalpy rise hot channel factor, $F_{\Delta H}^{N}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F_{\Delta H}^{N}$ at reduced power based on the expression:

$$F_{\Delta H}^{N} = 1.48 \, [1 + 0.2 \, (1-P)] \quad \text{(Westinghouse Fuel)}$$

$$F_{\Delta H}^{N} = 1.49 \, [1 + 0.2 \, (1-P)] \quad \text{(Exxon Nuclear Company Fuel)}$$

**where P is the fraction of RATED THERMAL POWER.**

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f_1$ ($\Delta I$) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature $\Delta T$ trips will reduce the setpoints to provide protection consistent with core safety limits.

## 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

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.

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3425 MWt and are 2.04 and 1.49, respectively.

## 3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is 1.97 times the average fuel rod heat flux for Westinghouse supplied fuel and 2.04 times the average fuel rod heat flux for Exxon Nuclear Company supplied fuel.

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.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the 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 above 50% 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 the OPERABLE excore detector outputs and provides an alarm message 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 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less). During operation at THERMAL POWER levels between 50% and 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less) 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.

The basis and methodology for establishing these limits is presented in topical report XN-NF-77-57, "Exxon Nuclear Power Distribution Control for PWRs - Phase II" and Supplements 1 and 2 to that report.

## 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-4 and 3.2-5, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-4 and 3.2-5. Measurement errors of 3.5% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature ($T_{avg}$) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for 4 and 3 loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

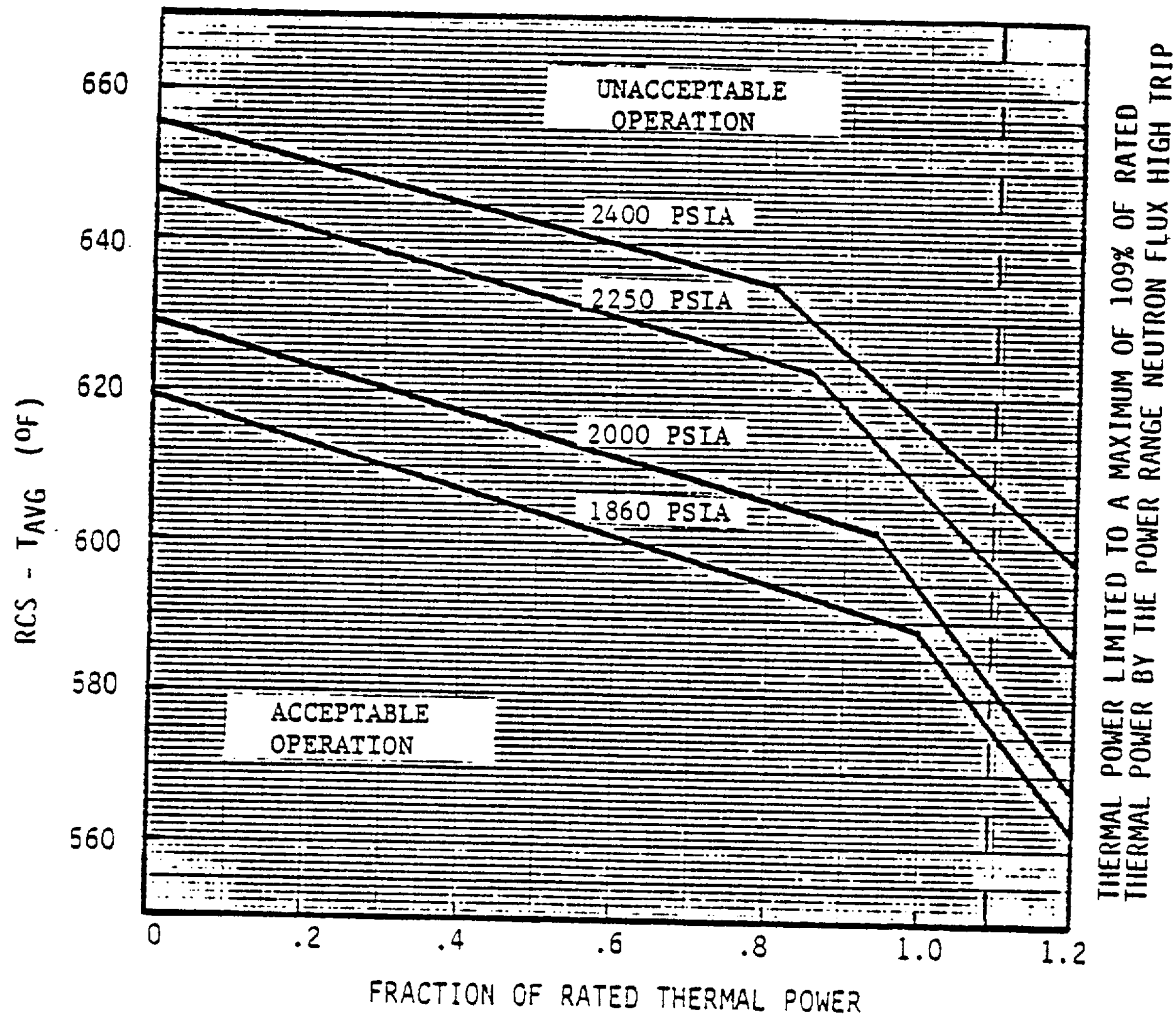



Figure 2.1-1 Reactor Core Safety Limits - Four Loops in Operation

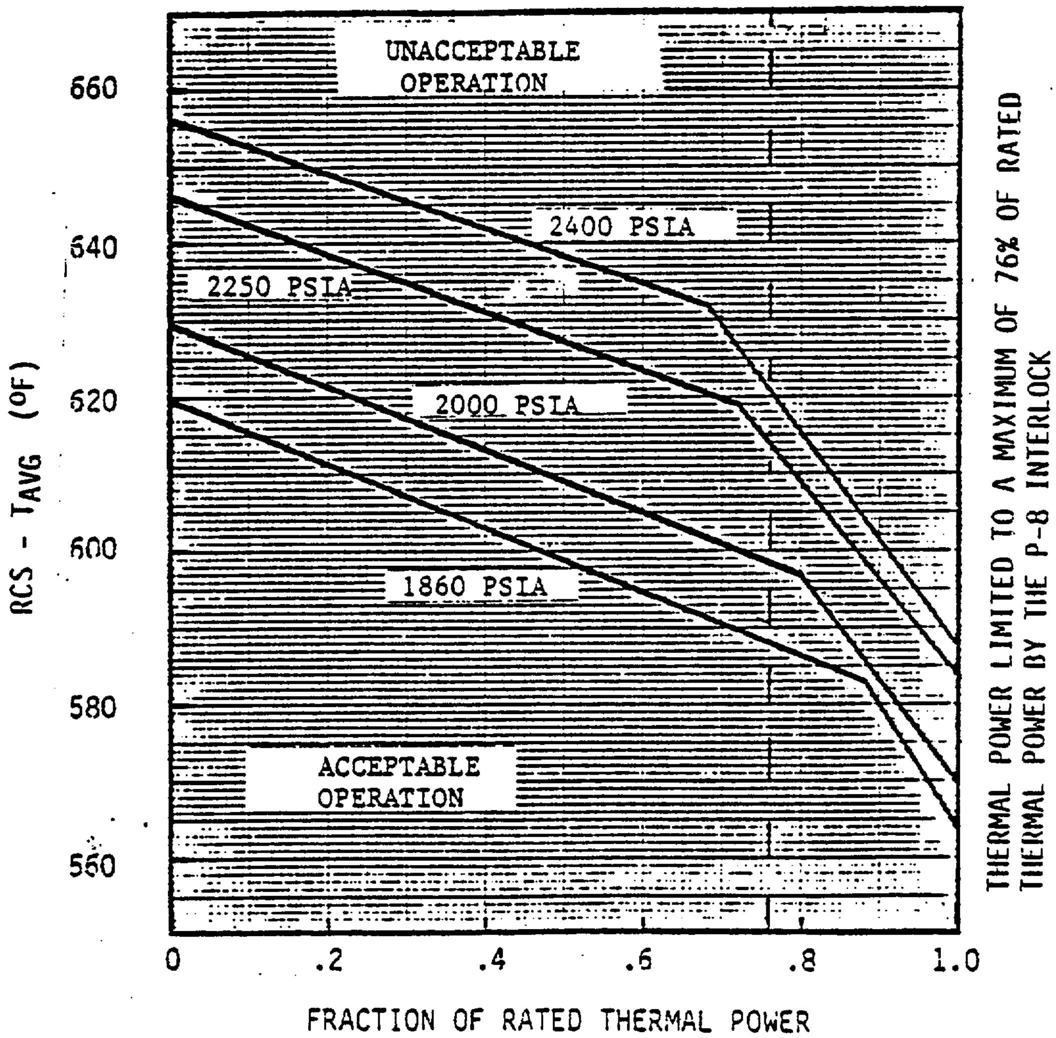



Figure 2.1-2 Reactor Core Safety Limit - Three Loops in Operation

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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

# SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL

# BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-74

# INDIANA AND MICHIGAN ELECTRIC COMPANY

# DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

# DOCKET NO. 50-316

## A. Introduction

By letter dated April 7, 1982, the Indiana and Michigan Electric Company (the licensee) submitted an application for the D. C. Cook Nuclear Plant Unit No. 2 reload for Cycle 4. The reload will include the first fuel batch fabricated by the Exxon Nuclear Company (ENC) of the 17x17 fuel assembly design. The use of this new fuel will increase the power level of the Unit 2 from its presently authorized power level of 3391 megawatts thermal to 3411 megawatts thermal and extend the plant design basis of of average fuel burnup from 33,000 MWD/MTU to over 40,000 MWD/MTU. The application on April 7, 1982, however, covers only Cycle 4 which is scheduled to begin in January 1983. The ENC fuel for Cycle 4 should achieve an average fuel assembly burnup of about 20,000 MWD/MTU; within the current plant design basis of 33,000 MWD/MTU.

On August 19, 1982, the NRC filed a "Notice of Proposed Issuance of Amendment to Facility Operating License" with the Office of the Federal Register for publication. That notice recognized the proposed increase in the maximum power level and the change in the maximum fuel enrichment from 3.5% uranium 235 to 3.8% uranium 235. On September 2, 1982, the Commission issued Amendment Nos. 57 and 41 to Facility Operating License Nos. DPR-58 and DPR-74 for the D. C. Cook Nuclear Plant Unit Nos. 1 and 2, respectively. Those amendments revised the Technical Specifications to permit storage of Exxon fuel with a uranium enrichment of less than or equal to 3.84 weight percent of U-235.

Subsequent to the April 7, 1982 letter by the licensee, a number of supplements to the original proposal have been received and were used in the evaluation of the ENC fuel for Cycle 4 operation. Each section of this evaluation includes a list of references to these supplements as well as other information used in the evaluation. The remainder of this evaluation includes:

B. Core and Fuel Performance Evaluation
C. Transients and Accident Analysis
D. Radiological Consequences
E. Environmental Impact Appraisal
F. Conclusions

Since this evaluation primarily addresses Cycle 4 and limitations of analyses and methodologies, license conditions have been recommended specifically for Cycle 4 and in general for the following cycles. Each of these is addressed in the appropriate section of the evaluation.

## B. Core and Fuel Performance Evaluation

### 1.0 Introduction

By letter dated April 7, 1982, the Indiana and Michigan Electric Company (the licensee) made application to amend the Technical Specifications for the D. C. Cook Nuclear Plant, Unit 2. The proposed amendment would increase the rated power to 3411 thermal megawatts and permit reloading and operation of the plant for Cycle 4. In support of the application the licensee submitted a reload safety analysis report, XN-NF-82-37 and a transient analysis report for the increased power, XN-NF-82-32(P) along with other documents which are referenced in the evaluation below.

### 2.0 Fuel Mechanical Design

### 2.1 Introduction

The Cycle 4 reload is the first commercial utilization of the ENC 17x17 fuel assembly design. This fuel design is described in an ENC generic topical report, XN-NF-82-25 (Ref. 1). The 17x17 assembly design is similar to the previously used ENC 14x14 design (Ref. 2), except for an increased number of guide tubes and spacers, which are intended to ensure adequate strength and stiffness. The NRC staff has reviewed XN-NF-82-25 and has approved (Ref. 3) the report as a document suitable for referencing in safety analyses.

On the grounds that the ENC 17x17 design has received generic approval, the design is approved for the D. C. Cook Cycle 4 reload, subject to the limitations on that generic approval. Those limitations and their consequences are addressed in this evaluation along with plant-specific concerns.

### 2.2 General Description

The ENC 17x17 bundle array contains 264 fuel rods, 24 guide tubes, and 1 instrument tube. The fuel rods have a slightly smaller diameter and pitch than the ENC 14x14 PWR design. The grid spacers have thicker structural members and are deeper overall for greater assembly rigidity. The design has a "quick-removable" upper tie plate to facilitate inspection and reconstitution of irradiated assemblies. The assembly design is described in Section 4.0 of XN-NF-82-25, with additional information provided in response (Ref. 4) to staff questions on that document.

The D. C. Dook-2 Cycle 4 reload will consist of 72 Exxon Nuclear Company (ENC) 17x17 fuel assemblies, which will be placed in Region 6 of the core. The rest of the fuel in Cycle 4 will be comprised of Westinghouse assemblies from fuel Regions 3, 4, and 5. The nominal Cycle 4 design burnup is 14,150 MWD/MTU.

For the ENC 17x17 fuel, the peak rod burnup will be 22,000 MWD/MTU, and the maximum assembly average burnup will be 20,000 MWD/MTU. For the Westinghouse (W) fuel that will remain in the core during Cycle 4, the peak rod burnup will be 45,600 MWD/MTU, which corresponds to a peak pellet burnup below 50,000 MWD/MTU. The W fuel design has been previously reviewed and approved for operation for its design lifetime, so we have for this reload evaluation reviewed only the ENC fuel.

## 2.3 Rod Bowing

Fuel rod bowing is a phenomenon that alters the nominal spacing between adjacent fuel rods. Bowing also affects local heat transfer to the coolant and local nuclear power peaking. Using ENC's rod bowing methodology (Ref. 5), significant rod bowing penalties (to either the permissible DNBR or total allowed peaking ($F_Q$)) are not calculated to occur until gap closures greater than 50% are obtained. The calculations show that 50% rod-to-rod gap closure does not occur until an assembly exposure of 28,000 MWD/MTU. Since the maximum burnup for ENC fuel assemblies in Cycle 4 will be much less than 28,000 MWD/MTU (viz., 20,000), a 50% gap closure will not be reached. For future cycles, the licensee has stated that the combination of rod bowing and rod power will be evaluated to determine if DNBR or peaking factor limits need to be adjusted to account for rod bowing (Ref. 6). Therefore, we conclude that bowing of ENC 17x17 fuel has been satisfactorily accounted for with respect to Cycle 4 operation. For future cycles involving burnups greater than 28,000 MWD/MTU, we will require that the licensee provide the above-described analysis and that the issue be resolved prior to operation of those cycles.

## 2.4 RODEX 2 -- Strain, Oxidation, Pellet/Cladding Interaction (PCI) Analyses

As pointed out in our generic Safety Evaluation (Ref. 3) of Exxon's 17x17 fuel assembly analysis report (Ref. 1), the RODEX 2 thermal analysis code (Ref. 7) which is currently under review, was used in the design analysis of several important fuel performance phenomena including cladding strain, external corrosion (oxidation), fuel rod internal pressure, fuel pellet temperature, and pellet/cladding interaction. We, therefore, have required applicants and licensees intending to use this fuel to confirm or resubmit the analyses of those fuel performance issues with an approved code.

As a recent meeting with the staff, the D. C. Cook 2 licensee indicated (Ref. 6) that in the cladding strain, oxidation, and PCI analyses the pertinent features of RODEX 2 are either identical to previous calculations (i.e., oxidation) or have been benchmarked to the mechanical performance of irradiated fuel (i.e., strain, PCI). Thus, it was stated, that since RODEX 2 is calibrated to actual strain and PCI rod behavior, any subsequent code modifications to other features such as temperature or gas release, would not significantly affect the strain or PCI results for D. C. Cook 2. We cannot agree with that position for the following reasons.

While our review of RODEX 2 has progressed to a point where we can conclude that certain features (e.g., the oxidation correlation) are acceptable, RODEX 2 does not appear to predict temperatures very well when compared with experimental data. Since RODEX 2 is used to provide input into other models and codes (such as RAMPEX, (Ref. 8) which were used to calculate cladding stresses and strains), we believe that the effect of the temperature input to those calculcations still requires confirmation. From our review of the ENC 17x17 fuel design report (Ref. 1), we have determined that the current calculations(using RODEX 2) for cladding strain, oxidation and PCI easily satisfy the acceptance criteria with considerable margin. For that reason, therefore, we consider this issue to be confirmatory in nature. Thus, while we will not require further calculations prior to Cycle 4 startup, the licensee is required to submit, and the amendment is conditioned upon the submittal of, the above described calculations during Cycle 4 operation and prior to 10,000 MWD/MTU burnup on the ENC fuel. The licensee should resubmit the results of the cladding strain, oxidation and PCI calculations with the then-approved version of the RODEX 2 code.

2.5 Cladding Collapse-Review Criterion

For the ENC's 17x17 fuel design a revised cladding collapse criterion and calculation procedure has been developed. That revised approach to calculating cladding collapse is described in an ENC generic topical report on high burnup fuel (Ref. 9) which is under review. Cladding collapse is a phenomenon that is not a concern until rather late in the fuel assemblies life, and therefore, it is not expected to impact the operation of ENC 17x17 fuel during Cycle 4 operation (where the peak rod burnup is projected to be 22,000 MWD/MTU). This view is supported by calculations using the previously-approved COLAPX (Ref. 10) procedure, which showed (Ref. 6) that the criterion of maintaining a free standing unsupported tube was met for the highest burnup Exxon 17x17 rod in Cycle 4. Accordingly, we conclude that there is reasonable assurance that cladding collapse will not occur in ENC 17x17 fuel rods during Cycle 4 operation. However, prior to authorization of Cycle 5 operation we will require the licensee to reaffirm, with an approved model, that creep collapse will not occur in ENC 17x17 fuel operated to the target burnup.

2.6 Fuel Centerline Temperature

According to information presented in Reference 6, the peak $UO_2$ centerline temperature was calculated to be 3500°F, using the Exxon GAPEX thermal analysis code (Ref. 11). Since this temperature was calculated by an approved code and is well below the $UO_2$ melting temperature of about 5000°F, we conclude that the "no-centerline-melting" criterion is satisfied for ENC 17x17 fuel for D. C. Cook 2 Cycle 4 operation.

## 2.7 Rod Pressure

As indicated in Exxon's generic report, XN-NF-82-25, (Ref. 1), the ENC 17x17 fuel rods are designed such that the internal gas pressure of the fuel rods does not exceed coolant pressure. Although RODEX 2 (Ref. 7), which is under review, was used for the thermal design analysis described on ENC's generic report, information received (Ref. 6) as part of the Cycle 4 reload submittal, indicates that fission gas release was also calculated with the approved GAPEX code (Ref. 11), to a rod exposure of 22,000 MWD/MTU. Since the peak rod exposure for ENC 17x17 fuel during Cycle 4 operation will be ≤22,000 MWD/MTU (Ref. 12), we conclude that there is reasonable assurance that the rod internal pressure will not be exceeded during this cycle. However, prior to Cycle 5 operation which will achieve rod burnups greater than 22,000 MWD/MTU, we will require the licensee to provide an analysis of rod internal pressure with an approved code (GAPEXX or RODEX 2 with modifications) that shows that the rod internal pressure criterion continues to be satisfied for the most limiting rod.

## 2.8 On-Line Monitoring

Section 4.2.II.D.2 of the Standard Review Plan indicates that the on-line fuel rod failure detection methods (instrumentation and procedures) should be reviewed. Because of the newness of the ENC 17x17 fuel design that will be used in D. C. Cook 2 during Cycle 4, there is a need to assure that any unexpected failures of that fuel (as well as the older, W fuel) would be readily detected. The instrumentation (failed fuel detection system) is described in the D. C. Cook 2 FSAR and is not at issue here. The issue is the capability and commitment of the licensee to use appropriate systems as needed to assure that fuel failures would be detected. The introduction of ENC fuel does not present any unique fuel failure detection problems.

As indicated in D. C. Cook 2 Technical Specification 4.4.8 Surveillance, a beta-gamma analysis of the primary coolant is required every 72 hours. Moreover, the licensee has a procedure (Ref. 6) that actually results in the performance of such an analysis every 48 hours, instead of the 72 hours required by the Technical Specification. Inasmuch as a description of the failed fuel detector is provided in the plant FSAR, while the coolant sampling procedure is described in the cited Technical Specification, and further, since the change in the type of fuel presents no fuel failure detection problems previously unanalyzed, we conclude that the issue of on-line monitoring has been adequately addressed for D. C. Cook 2 Cycle 4 operation.

## 2.9 Post-Irradiation Examination (PIE)

As indicated in SRP Section 4.2.II.D.3, a post-irradiation fuel surveillance program should be established to detect anomalies or confirm expected fuel performance. For a new fuel design, such as the ENC 17x17 fuel, such as program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. In a recent submittal (Ref. 13), the D. C. Cook 2 licensee stated that visual examinations would be performed on the ENC 17x17 after its 1st cycle of operation (Cycle 4). The examination would include binocular inspections of 50% of the assemblies as they are being transferred to the spent fuel pool following Cycle 4 operation (all the assemblies are to be off-loaded, even those that will be reinserted for Cycle 5). In addition, a more detailed underwater television or periscope examination will be performed on each face of four Exxon assemblies from this batch at the end of Cycle 4. During subsequent refuelings AEP plans to visually inspect those assemblies from the first batch of ENC 17x17 fuel that will be permanently discharged. We conclude that the proposed PIE program satisfies the intent of the Standard Review Plan and is, therefore, acceptable.

2.10 Seismic-and-LOCA Loadings

An analysis of the structural adequacy of the fuel assemblies in D. C. Cook Unit 2 in response to seismic-and-LOCA loadings was an initial plant requirement (see FSAR Section 3.2.1.3.2). Such an analysis was provided for the Westinghouse fuel (WCAP-8236, December 1973) in the FSAR.

In 1975 an additional loading due to asymmetric blowdown forces on PWRs during LOCA was identified. As a result, NRC issued NUREG-0609 (Asymmetric Blowdown Loads on PWR Primary Systems) to address this concern and required all PWRs to submit such an analysis for evaluating fuel assembly structural adequacy.

Westinghouse A-2 Owners Group including D. C. Cook Units 1 and 2 submitted two reports, WCAP-9558, Revision 2 and WCAP-9787, for staff review in response to NUREG-0609. They claimed that a rapid blowdown is very unlikely because the stainless steel primary piping would leak before it breaks during a LOCA; therefore, the reports argue that the requirements of NUREG-0609 can be waived.

Although the review of Westinghouse A-2 Owners Group reports has not yet been completed, no structural response analysis is presently being required. However, there still remains the original FSAR requirements of analyzing seismic effects on fuel assemblies for D. C. Cook Unit 2. The coming Cycle 4 core (mixed Westinghouse and Exxon fuels) and future cores (mixed and pure Exxon fuels) of D. C. Cook 2 must, therefore, be shown to be structurally adequate regarding the seismic effect because the original analysis did not cover Exxon fuel.

The licensee in a letter dated January 12, 1983, submitted information about the structural adequacy of the ENC 17x17 fuel assemblies to respond to this requirement. In that submittal, the licensee cited an Exxon analysis which stated that the resulting loads on 17x17 fuel assemblies due to the increased number of grid spacers, and tests of grid spacers show greater strength for ENC 17x17 fuel assembly is adequately designed to withstand earthquakes and LOCA as compared to the 15x15 fuel assembly, which was analyzed in the report XN-NF-76-47. Although the staff reviewed that generic report, only the analytical methods were approved; the calculated results presented in the report were not found to be generically bounding. Therefore, plant-specific analyses must be performed to account for Cook 2 core accelerations and to determine loads on fuel rods, guide tubes, and other fuel assembly components.

Based on the information submitted which indicates favorable results, we conclude that the seismic effect on the structural adequacy of the Cook 2 Cycle 4 core has been adequately addressed for Cycle 4. However, to assure that a complete and thorough analysis has been performed, documented, and found completely satisfactory, the licensee must submit a revised plant-specific analysis to the NRC; such an analysis can be completed within a reasonable time period of a year. The analysis should also address future cores (mixed W and ENC and pure ENC). The analysis should use the approved methodology (XN-NF-76-47) and demonstrate compliance with fuel assembly structural acceptance criteria (SRP-4.2 Appendix A) for the design seismic event applicable to D. C. Cook 2. Cycle 4 operation is approved and a license condition is imposed to require the revised plant specific analysis within one year from the date of the license amendment.

## 2.11 Fuel Mechanical Design Summary

The NRC staff has reviewed the ENC 17x17 fuel design analysis for D. C. Cook 2 Cycle 4 operation. The staff reviewed both the information provided in the generic topical report (XN-NF-82-25) for this design and recently-submitted plant-specific analyses and information. Based on that information we conclude that D. C. Cook 2 Cycle 4 operation with the ENC 17x17 fuel is acceptable to the target burnups (22,000 MWD/MTU peak rod, 20,200 MWD/MTU maximum assembly) with the following understandings and conditions:

1. For future cycles involving burnups greater than 28,000 MWD/MTU (prior to Cycle 5), the licensee must provide a rod bowing analysis to determine whether DNBR or peaking factor limits require adjustment.

2. The licensee must resubmit the cladding strain, oxidation, and PCI calculations with the approved version of the RODEX 2 code during Cycle 4 operation and prior to 10,000 MWD/MTU burnup on the ENC fuel. This is a license condition.

3. Prior to Cycle 5 operation of ENC 17x17 fuel, involving rod burnups greater than 22,000 MWD/MTU, creep collapse calculations must be performed (and the analysis provided to the NRC) using a approved method such as COLAPX or, if it has been reviewed and approved by then, the ENC creep collapse procedure described in XN-NF-82-06(P), as revised.

4. Prior to D. C. Cook 2 Cycle 5 operation involving rod burnups greater than 22,000 MWD/MTU, a rod internal pressure analysis must be provided (using an approved code) that shows that the rod internal pressure criterion continues to be satisfied for the most limiting rod.

5. The licensee must complete a revised analysis within one year using the approved methodology to comply with fuel assembly structural acceptance criteria in Appendix A to SRP-4.2 for the design seismic event. This is a license condition.

## 3.0 Nuclear Design

In order to support the reloading and operation of D. C. Cook Unit 2 for Cycle 4 the licensee has submitted a safety analysis report prepared by Exxon Nuclear Company. We have reviewed the nuclear design of the proposed reload. The neutronics design of the core was performed with the XTG code which has been reviewed and approved by the staff as part of the Exxon nuclear design methods for pressurized water reactors. Values of the moderator, isothermal, and Doppler temperature coefficients, boron worths, total peaking factor, delayed neutron fraction and shutdown margin are presented for beginning and end of cycle at full and zero power conditions.

These values are bounded by those used in the transient and accident analyses. They are compared to similar quantities from Cycle 3. The differences may be attributed to the difference in core design.

Beginning-and end-of-cycle radial power distributions are presented. These indicate that the values for total peaking factor and maximum relative pin power should remain within limits during Cycle 4. Power distribution control during the cycle will be accomplished by following the procedures presented in the report, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II" (PDC-II). These procedures have been reviewed and found acceptable by the staff.

Based on the above analysis and referenced documents, we conclude that the nuclear design of the Cycle 4 reload is acceptable.

## 4.0 Thermal-Hydraulic Evaluation

This evaluation includes a detailed review of the thermal hydraulic design analysis for D. C. Cook Unit 2, Cycle 4. This detailed review is necessitated by the fact that Cycle 4 will contain a mixed loading of Exxon Nuclear and Westinghouse fuel assemblies. The composition of the core during Cycle 4 will be 72 Exxon assemblies and 121 Westinghouse assemblies, with the Exxon fuel rods being 3.7% smaller in rod diameter.

The objective of this review is to confirm that the thermal hydraulic design of the reload has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients. Besides a normal review of the Technical Specifications and reload safety analysis reports an expanded review was performed in the following areas:

1. Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations.

2. Review of XNB, Exxon Nuclear DNB Correlation for PWR Fuel Designs.

3. The hydraulic compatibility of ENC fuel with the existing Westinghouse fuel and the acceptability of any changes in hydraulic performance between Cycle 3 and Cycle 4 cores.

## 4.1 Mixed Core Thermal-Hydraulic Design Methodology

The thermal-hydraulic design methodology used by ENC is comprised of two steps. Initially, a core-wide calculation is performed on an assembly-by-assembly basis. In this analysis the limiting bundle is placed at its allowable-maximum radial peak while the remaining bundles are at their nominal powers. Inlet flow maldistributions are accounted for by a reduction of 5% in the hot bundle flow. The results of this calculation are the axial flow distribution for the hot assembly and the crossflow boundary conditions which will be used in the detailed subchannel model. These boundary conditions were originally stored as the average of all the boundary conditions on the hot assembly. However, during the course of our review, Exxon modified their code to properly store the corewide crossflow boundary conditions. That is, they do not average the crossflow conditions but use the actual crossflows as seen by the limiting assembly.

Next, an octant of the hot assembly is modeled on a rod-by-rod basis to determine the minimum DNBR for the core. In this model, crossflow between the limiting and adjacent fuel assemblies is accounted for by using the boundary conditions stored during the corewide calculations while flow redistribution within the limiting assembly is accounted for via crossflow between adjacent subchannels. As part of their subchannel analysis, Exxon increases the peak rod heat flux by typically 3% to account for extremes in fuel rod manufacturing tolerances and uses a flat peaking distribution within the rod array except for the limiting rod which is placed at its maximum peak.

The analytical tools which comprise the design methodology are the XCOBRA-IIIC computer code (XN-NF-75-21(P), Revision 2) and the XNB critical heat flux (CHF) correlation (XN-NF-621, Revision 1) or the W-3 correlation depending on the core being analyzed.

The methodology detailed in XN-NF-82-21(P), is the subject of a separate staff review which is described in the memorandum from L. Rubenstein to T. Novak dated December 13, 1982, "Review of XN-NF-82-21(P); Revision 1." The staff position transmitted in this memorandum is that the thermal-hydraulic design methodology presented in XN-NF-82-21(P), Revision 1 is acceptable for performing steady-state core thermal-hydraulic calculations when the proper method of storing crossflow boundary conditions is used. In addition, an adjustment of 2% on the minimum DNBR must be included for mixed cores containing hydraulically different fuel assemblies.

As a result of our review, the staff has found XN-NF-82-21(P), Revision 1 an acceptable and referential report with the limitation noted in the above paragraph.

### 4.2 Exxon Nuclear DNB Correlation for PWR Fuel Designs (XNB)

The generic review of the XNB correlation is currently in progress and is nearing completion. Based on the review of the correlation to date, the staff has determined that it is acceptable for use in licensing the D. C. Cook Unit 2 reload. With this correlation a minimum departure from nucleate boiling ratio (MDNBR) of 1.17 provides 95% probability against boiling transition with 95% confidence.

### 4.3 Thermal Hydraulic Compatibility and Cycle to Cycle Comparisons

Hydraulic performance differences between Westinghouse and Exxon fuel were tested with pressure drop tests performed in Exxon Nuclear Company's Hydraulic Loop Test Facility. Using the loss coefficients from these tests Exxon determined that the overall hydraulic resistance of Exxon reload fuel was within 0.3% of the resistance of existing Westinghouse fuel. Thus insertion of Exxon fuel into D. C. Cook Unit 2 reactor will not significantly impact primary coolant flow.

The licensee also was asked to compare the major Thermal Hydraulic Parameters of Cycle 3 and Cycle 4 and to justify the differences in the principal parameters. These parameters are given in Table B-1 with explanations given in the notes to the table.

The staff finds, assuming the adjustment of 2% on the minimum DNBR imposed by the generic review of the mixed core methodology, the hydraulic differences between the Exxon Nuclear assemblies and Westinghouse assemblies and their effect on the major hydraulic performance parameters for Cycle 4 are acceptable.

### 4.4 Thermal-Hydraulic Evaluation Summary

We have reviewed the D. C. Cook Unit 2 Cycle 4 reload thermal design and find it acceptable provided:

1. The XNB correlation is used with a 1.17 MDNBR.

2. An adjustment of 2% on the minimum DNBR is imposed to conservatively bound any uncertainties in the mixed core methodology.

TABLE B-1

D. C. COOK UNIT 2 THERMAL-HYDRAULIC PARAMETERS AT FULL POWER

| General Characteristics | Unit | Reference Cycle 3 | Cycle 4 |
|---|---|---|---|
| Total Heat Output (core only) | MWt<br>$10^6$ Btu/hr | 3391<br>11573.5 | 3425(1)<br>11689.5 |
| Fraction of Heat Generated in Fuel Rod | | .974 | .974 |
| Primary System Pressure | | | |
| Nominal | psia | 2250 | 2250 |
| Minimum in steady state | psia | 2220 | 2220 |
| Maximum in steady state | psia | 2280 | 2280 |
| Inlet Temperature | °F | 541.3 | 543.1(2) |
| Total Reactor Coolant Flow (steady state) | gpm<br>$10^6$ lb/hr | 375,000<br>142.7 | 375,000<br>142.7 |
| Coolant Flow through Core | $10^6$ lb/hr | 136.3 | 136.3 |
| Hydraulic Diameter (nominal channel) | ft | .438 (W) | .479 (ENC)(3) |
| Average Mass Velocity | $10^6$ lb/hr-ft$^2$ | 2.72 | 2.613(4) |
| Pressure Drop across Core | psi | - | 24.8 |
| Total Pressure Drop across Vessel (based on nominal dimensions and minimum steady state flow) | psi | 51. | 51. |
| Core Average Heat Flux (accounts for above fraction of heat generated in fuel rod and axial densification factor) | Btu/hr-ft$^2$ | 188700. | 197560.(5) |
| Total Heat Transfer Area | ft$^2$ | 59866. | 57625. |
| Film Coefficient at Average Conditions | Btu/hr-ft$^2$ | - | ~6000 |
| Average Film Temperature Difference | °F | - | ~35°F |
| Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod) | kW/ft | 5.41 | 5.46(6) |
| Average Core Enthalpy Rise | Btu/lb | 84.92 | 85.78 |
| Maximum Clad Surface Temperature | °F | - | ≤850 |

D. C. COK UNIT 2 THERMAL-HYDRAULIC PARAMETERS AT FULL POWER (Cont'd)

| General Characteristics | Unit | Reference Cycle 3 | Cycle 4 |
|---|---|---|---|
| **Calculational Factors** | | | |
| Engineering Heat Flux Factor | | - | 1.03 |
| Engineering Factor on Heat Channel Heat Input | | - | - |
| Rod Pitch and Clad Diameter Factor | | - | - |
| Fuel Densification Factor (axial) | | - | 1.01 |
| **Total Planar Radial Peaking Factors** | | | |
| For DNB Margin Analyses ($F_{\Delta H}$) | | 1.55 | 1.60 |
| $F_Q$, Transient Analyses | | - | 2.55 |
| $F_Q$, ECCS | | - | 2.04 |
| Limiting Transient Minimum DNBR | | >1.8 (CEA Drop) | 1.35 (Loss of Feedwater Heater) |
| Minimum Allowable DNBR | | 1.17 (WRB-1) | 1.17 (XNB) |

NOTES

1. The 3425 MWt core power level is analyzed in the Cycle 4 thermal-hydraulic analyses to bound the new plant operating point.

2. The Cycle 4 reactor coolant inlet temperature of 543.1 °F reflects the thermal design flow rate of $142.7 \times 10^6$ lb/hr and the vessel average temperature and power associated with the new operating points.

3. The hydraulic diameter cited for Cycle 4 represents ENC fuel and reflects the ENC fuel's decreased rod diameter and increased flow area.

4. The core average mass velocity for Cycle 4 is decreased from the Cycle 3 value to account for the increased cross sectional flow area of the ENC fuel bundle.

5. The Cycle 4 core average heat flux is larger than the Cycle 3 value due to the decrease rod surface area of the ENC assembly and to the 1% increase in core power level assumed for the thermal-hydraulic analyses.

6. The 1% larger linear heat rate for Cycle 4 reflects the 1% increase in core thermal power assumed for the thermal-hydraulic analyses.

## 5.0 Technical Specification Changes

The Technical Specification changes which implement the Exxon Power Distribution Control Procedure have been previously approved (memorandum from L. Rubenstein to G. Lainas dated August 30, 1982). For Cycle 4 these procedures are used to enforce an $F_Q$ value of 2.04 for the Exxon fuel and 1.97 for the Westinghouse fuel. Based on approved methods being employed to determine the parameters involved, we conclude the Technical Specifications 3/4.2.1, 3/4.2.3, and 3/4.2.6 are acceptable.

B.6.0 References

1. R. A. Pugh, "Generic Mechanical Design Report -- Exxon 17x17 Fuel Assembly," Exxon Report XN-NF-82-25, April 1982.

2. C. A. Brown, R. B. Macduff, and P. D. Wimpy, "Generic Mechanical, Thermal Hydraulic and Neutronic Design for Exxon Nuclear TOPROD Reload Fuel Assemblies for Pressurized Water Reactors," Exxon Report XN-NF-80-56, November 19, 1980.

3. C. O. Thomas (NRC) letter to R. B. Stout (ENC) "Acceptance for Referencing of Licensing Topical Report XN-NF-82-25(P)," dated January 11, 1983.

4. R. B. Stout (ENC) letter to C. O. Thomas with response to staff questions on XN-NF-82-25(P), November 24, 1982.

5. (a) "Computational Procedure for Evaluating Fuel Rod Bowing, "XN-NF-75-32, Supplements 1-4, July 1979, January 1980, and May 1982.

   (b) L. S. Rubenstein (NRC) memorandum to T. M. Novak, "SERs for Westinghouse, Combustion Engineering, Babcock and Wilcox, and Exxon Fuel Rod Bowing Topical Reports," October 25, 1982.

6. Responses to Staff Questions on D. C. Cook 2 Cycle 4 Safety Analysis Report, XN-NF-82-37, at meeting in Bethesda, Maryland, December 2, 1984 (see Meeting Summary Report by D. Wigginton, December 7, 1982).

7. K. R. Merckx, "RODEX 2: Fuel Rod Thermal Mechanical Response Evaluation Model," XN-NF-81-58(P), August 1981.

8. K. R. Merckx, "RAMPEX: Pellet-Clad Interaction Evaluation Code for Power Ramps, "Exxon Report XN-70-22 (undated).

9. M. J. Ades, "Qualification of Exxon Nuclear Fuel for Extended Burnup," ENC Report XN-NF-82-06(P), Revision 1, June 6, 1982.

10. K. R. Merckx, "Cladding Collapse Calculational Procedure, "ENC Report JN-72-23, November 1972.

11. "GAPEXX: A Computer Program for Predicting Pellet-to-Cladding Transfer Coefficients," Exxon Report XN-72-25, August 23, 1973.

12. M. Tokar (NRC) Telecommunication with A. Lobel (AEP), December 1982.

13. R. S. Hunter (AEP) letter to H. R. Denton (NRC) letter number 637H, December 1982.

## C. TRANSIENT AND ACCIDENT ANALYSES

### 1. Introduction

The licensee submitted copies of a report entitled "Plant Transient Analyses for the Donald C. Cook Unit 2 Reactor at 3425 mwt "under Reference (8), and a revision 1 under reference 18. Further, the licensee has submitted additional information and revisions in references 29, 30, 34, and 38.

Plant transients have been submitted for rupture of a CRDM Housing (RCCA Ejection), uncontrolled rod withdrawal (from full power), loss of main steam line break. The results of these events were reviewed to assess which were the most limiting in respect of thermal margins; these were: locked rotor, transient events caused by feedwater system malfunctions, excessive load increase and main steam line break. These events were reviewed in substantive detail. Remaining plant transients for which reanalyses have not been submitted include major rupture of a main feedwater pipe, small break loss of coolant accident, RCCS misalignement, uncontrolled boron dilution, start up of an inactive reactor coolant loop, turbine trip, loss of normal feedwater, loss of offsite power to the station auxiliaries (blackout), turbine generator accident, steam generator tube rupture and the Uncontrolled Rod Withdrawal from Subcritical event. For these events which have not been re-analyzed, the licensee has concluded that the reference analyses remains valid for cycle 4, or that other events which have been reanalyzed for cycle 4 have been shown in the reference analysis to be more limiting.

2. Methodologies For Calculating The Thermal Hydraulics of the Reactor Coolant System and the Reactor Core

The thermal-hydraulic transients in the reactor coolant system (RCS) of the D.C. Cook Unit 2 were calculated using an EXXON analysis model known at PTS- PWR-2, These transients identify the thermal-hydraulic conditions for the reactor core, and also the circumstances in the remainder of the reactor coolant primary system up to and beyond the point of minimum DNBR. Adjunct thermal hydraulic models and correlations are used both to provide "biased" core data into PTS-PWR2, and also to receive core data from PTS-PWR2, to calculate MDNBR. The PTS-PWR2 model was originally used in a less developed form for a very limited number of anticipated operational occurrences (AOOs). This particular application for the D.C. Cook 2 Cycle 4 reload uses a substantially updated version of the original model and extends the application to an increased number of AOOs, and postulated accidents. This updated model and its application to the broader range of accidents has not been subject to generic review to verify and validate its methodology, nor has it received approval on any plant specific application. This generic model has only recently been received (October 1982) on the docket for the D.C. Cook 2 cycle 4 reload.

This fuel reload for Cycle 4 of the D.C. Cook Unit 2 is the first use of Exxon fuel in this reactor, and also the first of a batch which has been specifically designed for extended burn-up life-times. In this specially designed fuel by EXXON, the number of fuel pins and their general arrangement remains unchanged from the Westinghouse assemblies;

however pin diameters have been reduced and the pressure drop of the assembly increased requiring new XN-DNBR correlations to be validated. In addition, placement of this fuel in a Westinghouse matrix results in a "mixed core" which also requires new methodologies for its evaluation. The Donald C. Cook Unit 2 was the first facility to be licensed on the basis of using the Westinghouse Improved Thermal Design Basis (ITDB) procedure. This procedure was adopted for Cycle 2 for a limited number of events, excluding accidents. The ITDB methodology is not applicable to EXXON fuel. The transient analyses for Cycle 4 are performed assuming worst case values of each input parameter.

3. <u>REVIEW OF TRANSIENTS</u>

a. <u>Rod Withdrawal Events</u>

We have reviewed the analyses of the uncontrolled rod withdrawal events, the rod drop event, and the rod ejection accident. The zero-power rod withdrawal event (start-up accident) is not affected by the rated power of the reactor. The rated power event was determined to be limiting in the final safety analysis report. Increasing the rated power will not alter that conclusion. It is, however, necessary to reanalyze the event at the higher power. The licensee submitted analyses performed by Exxon for both fast and slow rod withdrawals. In each case the nuclear overpower trip terminated the excursion before departure from nucleate boiling occurred. For the slow rod withdrawal event an additional anaysis was performed in which the nuclear overpower trip was assumed not to occur in order to verify the adequacy of the overtemperature-delta temperature trip setpoint. For this case, departure from nuclear boiling did not occur. For the reasons stated above, we conclude that the consequences of rod withdrawal events are acceptable for the higher rated power.

The reanalysis of the rod drop event consisted of calculating the DNB ratio at the higher rated power assuming a radial power distribution caused by the presence of a dropped rod. This is consistent with the analysis in the FSAR. However, this analysis has been shown by Westinghouse to be deficient and consequently Westinghouse supplied reactors now have certain Interim Operating Procedures pending resolution of the issue. Accordingly, it is a condition of this amendment that D. C. Cook continue to use these interim procedures until such time as the licensee supplies an analysis which supports operation without them to the satisfaction of the staff.

The rod ejection accident has been reanalyzed for Cycle 4. Analyses were performed at beginning and end of cycle for both zero power and full power conditions. Conservative values of the Dopple coefficient and nominal values of the delayed neutron fraction were used. Results were obtained by using the methods presented in XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors." These methods have been previously used for this purpose in licensing actions and have been found to be acceptable for purposes of obtaining maximum fuel enthalpies. The maximum full pellet enthalpy was 168 calories per gram for the hot full power end-of-cycle case. This meets our criterion of 280 calories per gram for this quantity and is acceptable.

Based on the discussion presented above we conclude that, with respect to the transients and accidents described, operation of D. C. Cook Unit for Cycle 4 at 3411 megawatts of thermal power is acceptable.

b. <u>Loss of Single Reactor Coolant Pump - Locked Rotor (and Broken Pump Shaft)</u>

<u>Licensing Basis</u>

The loss of forced reactor coolant flow arising from a single locked rotor was analyzed in detail for Cycle 2 (see Reference 1, Section 14.1.6). In addition, the single reactor coolant pump shaft break with a free spinning rotor was also calculated. These calculations were performed at a rated core output of 3391 Mwt, a zero moderator coefficient and least negative values of doppler coefficient. For these calculations, an evaluation of the consequences with respect to fuel rod thermal transients was performed. The results obtained represented the upper limits with respect to clad temperature and zirconium-water reaction. In the evaluation, the total peaking coefficient was conservatively assumed to be at value of 2.5. DNB was assumed to occur at the beginning of the event.

The cycle 3 analyses incorporated a positive moderator coefficient of 5 pcm/°F up to 70% of power at beginning of cycle (BOC). Transients were re-calculated at this condition. These analyses showed that the limiting case remained the same as for Cycle 2.

this condition. These analyses showed that the limiting case remained the same as for cycle 2.

Cycle 4 was calculated on a "conservative" basis in which a rated reactor core power of 3425 +2% (i.e., 3494 Mwt), was used with a corresponding NSSS power allowing for energy input from the reactor coolant pumps (RCPs). For DNBR calculations, reactor core inlet temperature is increased by 4°F, RCS pressure reduced by 30 psi, and RCS flow reduced by 3½%, compared to rated values. Reactivity parameters for this cycle include a positive reactivity coefficent of +5pcm/F°, technical specifications (TS) limit this value to 0.0 at full power. There is no significant difference in doppler coefficients. However, there is a significant difference in the character of the reactivity insertion following reactor trip; Reference 18, Page 21, Fig 23 shows a scram curve which is significantly delayed by approximately 0.4 secs over that of Fig. 14.1-3 of reference (1) (FSAR).

The current calculation of Cycle 4 also uses an initial coolant flow of $142.7 \times 10^6$ lbs/hr. Reactor coolant flow has been measured at $145.7 \times 10^6$ lbs/hr with an uncertainty of ±3½%. Calculations should have been performed at a minimum flow value of $140.6 \times 10^6$ lbs/hr (ie. a further reduction of 1.5%). The design peaking coefficient used to calculate the DNBRs for Cycle 4 was 2.55 and includes a radial peaking factor of (1.49 x 1.04 x 103) 1.60 and an axial peaking factor of (1.55 x 1.02)= 1.58.

Models Used

The following features of the PTS-PWR2 model used, and its output, are discussed below for the Locked Rotor (and broken shaft) event.

I. While initiation time for the low flow trip in the faulted loop and reactor trip time are virtually the same as in the cycle 2 analyses, the reduction in nuclear power and thermal power (from the core) occurs approximately one and two seconds earlier, respectively, than shown in the FSAR, although the scram curve used in the PTS-PWR2 model shows a reactivity insertion worth which is delayed by about 0.4 secs relevent to that used in Cycle 2.

II. The primary system pressure increase is about one third of the magnitude calculated by the earlier models accepted for D.C. Cook 2 in Cycle 2. Cycle 2 predicts a pressure increase of 280 psi to a maximum of 2633 psia over the first three seconds, compared with an increase of 100 psi using PTS-PWR2.

III. The model calculates only average surface (i.e., clad) and average fuel temperatures. Information is not available on the capability of PTS-BWR2 and its adjunct thermal-hydraulic models to calculate the detailed response of the fuel during these fast transients including: stored energy, internal temperatures with possibly fuel melting, gap conductances, and clad surface temperatures to ensure continuing core cooling capability and to assess zirconium/water and steam reactions.

EXXON has stated that the principal function of its PTS-PWR model is to conservatively calculate DNBR, and not necessarily to determine the detailed thermal-hydraulic conditions for the rest of the loop. The staff therefore depends on the values given for Cycle 2 in the FSAR, Reference 1, figure 14.1.6-13. Peak pressure was calculated to be 2633 psia using a "conservative" initial pressurizer pressure of 2280 psia, and conservatively high pressure drops in the primary system. The pressure rise in the primary system, is calculated ignoring the pressure relieving capability of the three power operated relief valves and the pressure reducing effects of the pressurizer spray. The actual pressure rise for cycle 2 was approximately 2633-2350-283 psi at a rated reactor core power level of 3391 Mwt and NSSS Power level of 3403 Mwt. Allowing for a 1% increase in rated power for cycle 3 and a +2% margin for conservative calculations we would expect the related pressure supplement to be approximately 10 psia. There is also, an additional correction on RCS flow of -1.5% which could contribute to an additional increase.

At this time, the permissable maximum value under transient conditions is 2735 psig (Ref. Tech Specs) and we consider that the available margin of 2735 - 2633 = 100 psi, is sufficient to cover the above marginal increases to be expected, until either an improved model is developed by the licensee or additional justification for the acceptability of the present methods is submitted.

Results

EXXON has calculated a minimum DNBR for this event, using automated cross flow methodology, of 1.42. If the current proposals on thermal margins for mixed cores are valid for this event, then this would be reduced to a value of 1.39. The minimum DNBR at the 95/95 probability/confidence limit is currently 1.17.

A substantive conservative assumption in the calculations is that although a total peaking coefficient of 2.55 was used, the actual peaking coefficient during operation will be limited to 2.04 by LOCA considerations, or approximately 80% of the peak power presumed in the transient analyses.

The analysis also did not assume loss of offsite power per GDC 17. We will require that the licensee provide a confirmatory analysis which demonstrates that specified acceptable fuel design limits are not violated for the case of loss of offsite power. Justification for any delays assumed between reactor/turbine trip and loss of offsite power must be provided.

Conclusion

There is a substantive uncertainty in the validity of the cycle 4 predictions. However the hot spot in the core will be restricted by Technical Specification (LOCA limit FQ) to approximately 80% of the peak power assumed in the transient analyses.. This represents a considerable conservatism in predicted DNBR and/or anticipated transient clad and fuel temperatures. It is on this basis, (ie, the margin between the

LOCA limited core and the assumptions of the transient analysis), that operation during cycle 4 is acceptable.

## C. RUPTURE OF A MAIN STEAM LINE

Licensing Basis

The rupture of a main steam line was analyzed for Cycle 2 in Reference 1, Section 14.2.5. The event was not reanalyzed for Cycle 3.

The existing licensing basis, Cycle 2, calculated four combinations of break sizes and initial plant conditions, and concluded that three cases warranted detailed thermal-hydraulic analysis. Those were (a) complete severance of pipe downstream of the steam flow restrictor with the plant initially at no load conditions and all reactor coolant pumps running; (b) complete severance of a pipe inside containment at the steam generator, with the same plant conditions as in (a) above; and (c) Case (b above with the loss of offsite power simultaneous with the generation of the safety injection signal.

All these cases were initiated at no load equilibrium conditions with a 1.6 percent end-of-life shutdown margin and assuming the most reactive RCCA rod stuck in its fully withdrawn position. The W-3 correlation for calculating DNBR was used. The conclusion from the Cycle 2 analysis was that in all three cases examined, the minimum DNBR was maintained above 1.30. For each case, the minimum injection of high concentration boric acid (20,000 ppm) solution, corresponding to the most restrictive single failure in the ECCS, was used. No credit for boron concentration upstream of the boron injection tank (BIT) was taken.

The licensee has submitted MSLB reanalyses in References 8 and 18 for the forthcoming cycle 4. The licensee has used values of core reactivity as a function of both temperature and core power which are virtually identical with those used in the earlier analysis, and concludes that they are conservative compared to ENC-calculated best-estimate values for the reload core. As in the cycle 2 analysis, 20,000 ppm boron concentration in the BIT and no boron concentration upstream of the BIT is modelled. Safety injection discharge characteristics have not been provided for comparison with the cycle 2 analyses. The cycle 4 analyses model delayed safety injection relative to previous analyses.

Initial RCS flow has an impact on the minimum predicted DNBR. The RCS flow used by the licensee in the safety analyses was $142.7 \times 10^6$ lbs/hr. This value was appropriate for The Westinghouse Improved Thermal Design Basis (ITDB) Methodology. Because of the change from Westinghouse only, to mixed Westinghouse and EXXON fuel, the ITDB Methodology is not applicable for cycle 4. The actual measured value, less the measurement uncertainty, should be used in the safety analyses. The measured value is $145.7 \times 10^6$ lbs/hr (Reference 14) and the related uncertainty in the technical specifications is ±3½% (i.e., maximum and minimum values of $150.8 \times 10^6$ lbs and $140.6 \times 10^6$ lbs/hr respectively). The licensee has stated that the flow measurement uncertainty determined by DC Cook Unit 2 plant personnel is 2%. Until the licensee provides the basis for this reduced uncertainty, together with a proposed change to the technical specifications, for review by the NRC, the staff will retrain the 3½% uncertainty as the the licensing basis.

Models Used

The following features of the PTS-PWR2 model and its output, are discussed below for the MSLB event:

I. The model does not provide for two phase flow conditions in the loop and further provides that when the pressurizer is "empty", loop pressure is determined to be the saturation pressure corresponding to the temperature at exit from the reactor vessel. This misrepresents the potential presence of two (2) phase flow conditions in the loop and the consequential effects on (a) calculated MDBNRS, and (b) coalescence into formation of bulk voidage upon trip of the Reactor Coolant Pumps (see NUREG-0737 Action Item II.K.3.5).

II. Accumulator injection has not been modelled. As such, the model is not appropriate when primary system pressure is calculated to drop below the accumulator systems injection pressure.

III. The model does not represent main feedwater or auxiliary feedwater systems and their effect.

IV. The model assumes perfect pressure vessel lower plenum mixing.

V. Safety injection actuation and main steam isolation valve (MSIV) closure on low-low pressurizer pressure trip, and at a pressure much less than that of technical specifications, has been modelled. The ESF systems of the plant provide for much earlier MSIV and SI using the "steam line pressure-low signal;" this signal is used in the reference cycle 2.

VI. Main steam isolation valve closure is modeled at 5 secs into the event whereas the technical specifications require closure within 8 sec.

VII. General

The licensee has concluded that while the above items represent discrepancies regarding the actual values of parameters, they are conservatively biased. Based on our review, we find that many of these conclusions hold varying degrees of validity, and have been considered in a qualitative manner to offset negative consequences of modeling and input deficiencies.

Results

In References 8 and 18, the licensee submitted a calculated MDNBR of 1.32 (modified to 1.29 by mixed core methodology) for the main steam line break, based on the modified Barnett correlation by Hughes. The minimum allowable DNBR for this correlation is taken as 1.135; we have assumed that this correlation remains valid for the new EXXON fuel.

In Reference 34, Item 4.1, the licensee has used the Westinghouse information for core parameters used in the MSLB DNBR analysis provided in the FSAR Table 14..2.5-1 Reference 1. Their calculated values in Reference 34 show substantial margins to DNB, although the details of these calculations have not been submitted. Reference 18, Section 3.7 assumes a radial peaking factor of 10, but there is information to suggest that values of up to 15 may be physically realizable.

Conclusion

Although a number discrepancies exist in the licensee analyses, we have concluded that the results of the main steam line break would be within the values of 10 CFR 100 and hence acceptable.

Staff conclusions are based upon: (1) the licensee predictions that MDNBR limits will not be violated for cycle 4 and hence fuel failure is not predicted to occur, and (2) that even if MDNBR limits were violated, DNB would be restricted to a small region of the core underneath the stuck rod. Only a small fraction of the core would be affected, and 10CFR Part 100 limits would not be exceeded.

D. EXCESSIVE LOAD INCREASE INCIDENT

Licensing Basis

This event was analyzed in detail for Cycle 2 (See Reference 1, Section 14.1.10) and was not analyzed for Cycle 3.

An excessive load increase is defined as a rapid increase in the steam flow caused by a power mismatch between the reactor core power and the steam generator load demand. The accident could result from either an administrative violation such an excessive loading by the operator, or an equipment malfunction in the steam dump control or turbine speed control systems.

The existing licensing basis for cycle 2 analyzed four (4) cases at full power:

1) Reactor control in manual at beginning-of-life
2) Reactor control in manual at end-of-life
3) Reactor control in automatic at beginning-of-life
4) Reactor control in automatic at end-of-life

The Cycle 2 calculations were undertaken with ITDB methodology using a nominal rated core power level of 3391 Mwt and an NSSS power of 3403 Mwt, with a reactor coolant inlet temperature of 541.3°F and an initial RCS flow of 142.7 x $10^6$ lbs/hr.

Cycle 4 was calculated using the following values:

a) Reactor core power of 3425 Mwt + 2% uncertainty, (i.e. 3494 Mwt)
b) NSSS thermal power equal to reactor core power plus RCP power
c) Reactor coolant inlet temperature of 543.1°F + 4°F (i.e., 547.1°F)
d) Primary coolant system pressure of 2250 psia - 30 psia = 2220 psia.
e) RCS flow of 142.7 x $10^6$ lbs/hr. As previously stated, we disagree with the licensee's use of the value of 142.7 x $10^6$ lbs/hr for the calculations; a correction for -1½% RCS was considered by the staff in the evaluation of the results.

The licensee has submitted two sets of calculations for this event, with final submittals provided in References 30, 34 and 38. Each of these sets of transients are calculated only for full power at EOC.

The reactivity coefficients for the licensee's final submittals bound cycle 2 and the "calculated" values of the cycle 4 core.

Models Used

The PTS-PWR2 model has the following characteristics discussed below considered significant for this event.

1) The model does not represent the Westinghouse automatic full length rod control system.
2) Steam generator heat transfer characteristics have been revised for later submittals in References (30), (34) and (38).

Reference 34 has described, in general,the approach adopted for the revision of the PTS-PWR 2 model based on the NRC staff review , and has proposed that the submittals of revised calculations for this event in Reference 30 and their comparisons with the earlier transients of Cycle 2, validate this revised model.

Results

The calculations proposed as adequate representations of this event are given in two sets; a) in Reference (8) and (18) and b) in Reference (30), (34) and (38). Each of the calculations is for EOC at full power with undisclosed rod control methodology. The principal difference between the two calculations is a revised steam generator heat transfer characteristic described in reference (30), (34) and (38), together with revised reactivity coefficients.

References (8) and (18) gave an MDNBR based on automated cross flow methodology of 1.43. Applying the current thermal margin for mixed core methodology in Reference (36) to this value reduces it to 0.98 x 1.43 = 1.40 compared with the current value proposed for XDNBR of 1.17.

Considering the transients in Reference (30), the results need correction for automated cross-flow methodology, an adjustment to the W-3 correlation to allow for mixed flow methodology effects, and a recognition of the further correction required for -1½% RCS flow. Our estimate is 0.95 x 1.52 x 0.98 = 1.415 to be compared with a W-3 value of 1.3 x 1.02 = 1.33.

Conclusion

The hot spot in the core will be restricted by technical specifications (LOCA limit FQ) to approximately 80% of the peak power assumed in the transient analyses; this represents a considerable conservatism in the predicted DNBR and it is on this basis that operation during cycle 4 is acceptable.

E. EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

This event was analyzed for Cycle 2 (See Reference 1, Section 14.1.10) and was not reanalyzed for Cycle 3 (See Reference 1, Appendix 14-B, Accident Analysis Item 1, C).

The existing Licensing Basis, (Cycle 2), consists of two analyzed cases, namely 1) the accidental opening of one feedwater control valve with the reactor critical at zero load conditions assuming a conservatively negative moderator temperature coefficient characteristic at end of core life, and 2) accidental opening of one feedwater control valve with the reactor in automatic control at full power (at end of core life conditions).

The Cycle 2 calculations were undertaken with ITDB methodology using "nominal" values of a core power level of 3391 Mwt and an NSS power of 3403 Mwt with a reactor coolant inlet temperature of 541.3 °F and an RCS flow of 142.7 x $10^6$ lbs/hr.

Cycle 4 was calculated on a "conservative" design basis with increased power and inlet core temperature. The "conservative" parameters are:

a) Reactor core power 3425 + 2% uncertainty, (i.e., 3494 Mwt)
b) NSSS thermal power was not specified but assumed to be reactor core power plus RCP powers
c) Reactor coolant inlet temperature of 543.°1F + 4°F, (i.e., 547.1°F)
d) Primary coolant system pressure of 2250 - 30 psia = 2220 psia
e) RCS flow 142.7 x $10^6$ lbs/hr. As we have previously concluded, we disagree with the inital RCS flow value assumed by the licensee, and correction for -1½% RCS flow will need to be made to the results.

Reactivity parameters for this cycle calculation (from References 8 and 18) included a positive reactivity coefficient of +5pcm/°F at HFP, BOC compared with Opcm/F° for Cycle 2.

Models Used

The PTS-PWR 2 model for this event has the following characteristics considered significant for this event

1) The model does not represent the Westinghouse automatic full length rod control system
2) Steam generator heat transfer characteristics which have been revised for later submittals in References (30), (34) and (38).

Results

The accidental opening of one feedwater control valve with the reactor at zero power at EOC, which is a current licensing basis event (Cycle 3) has not been submitted by the licensee for cycle 4.

The Cycle 2 calculation proposed that the maximum reactivity insertion rate associated with the uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition bounded this event, and therefore the results of the analyses were not presented. The current licensing basis for this particular rod withdrawal event is Cycle 3, in FSAR, Reference 1, Appendix 14 B, Section II.B.1.

The accidental opening of one feedwater control valve with the reactor at 100% power at BOC

This event was proposed as a limiting event by the licensee in References 8 and 18 because of the assumption of a positive moderator coefficient of +5pcm/°F/at BOC over that of Cycle 2 which was taken as 0.0 pcm/°F. We question this because Cycle 3 was recalculated on the bases of +5 pcm, and this particular event was not included for consideration. (Reference FSAR Reference 1, Appendix 14 B, I.C).

NRC Staff review did show that the licensee, in their PTS-PWR 2 transient model, had introduced a steam generator heat transfer characteristic which behaved in a significantly different manner from earlier characteristics. That is, RCS temperature response characteristics were reversed, (i.e., on a instantaneous reduction in feedwater temperature, RCS temperature increased instead of decreased. This resulted in markedly different response characteristics for the primary system.

The licensee was asked in reference 42 to identify these different steam generator heat transfer characteristics and to discuss their application to, and validation for, the PTS-PWR 2 computer code. ENC has advised in reference (34) that (in relation to the excessive load increase) it has been found necessary to "upgrade" the PTS PWR 2 code for computation of steam generator heat transfer. We do not find this conclusion acceptable. The licensee is required to submit the necessary information to justify the acceptability of the steam generator heat transfer characteristic used in the analyses.

<u>Excessive Heat Removal Due to a Feedwater System Malfunction Causing a Bypass of the Feedwater Heating System Leading to a Reduction in Feedwater Temperature</u> (End of cycle, Full Power, and Automatic Rod Control)

The FSAR, Reference 1, identifies this fault, but does not provide any results nor does it describe it as being bound by another event. In Reference 42 we requested additional information for this event but the licensee has responded in Reference (29) by advising that since no analytical results for the event are discussed in the FSAR to Reference 1, or its predecessor, that it is a non-DNB limiting event. We cannot concur in this conclusion and require the licensee to provide supporing information to confirm this conclusion.

Excessive Heat Removal Due to a Feedwater System Malfunction Causing a Bypass of the Feedwater Heating System Causing a Reduction in Feedwater Temperature at BOC and Full Power

Licensee submittals were initially made in References (8) and (18).

Additional submittals were made in References (30) and (34) with the SG upgrade and with reactivity coefficients modified as in Reference (38); these were made for a "Constant Steam Flow Control" and a "Constant Turbine Throttle Pressure"; Westinghouse automatic rod control has not been used.

In Reference (34) the licensee proposes this transient as a bounding case for the reduced feedwater temperature event because the calculated DNBRs are considered limiting.

The calculated MDNBR, using automated cross flow methodology is 1.35.

If we correct for -2% as per the recent SER to Reference 37, this calculated value becomes 1.32 and must be compared withthe allowable XNDNBR correlation MDNBR which is 1.17.

Excessive Heat Removal Due to a Feedwater System Malfunction at Zero Load Condition Causing Feedwater Temperature to be Reduced to 70°F

This case is presented for consideration in the FSAR (Reference 1), but no analytical results or conclusions are drawn. The licensee has not submitted an analysis for this event on the basis that the preceding event (100% power, BOL and +5 pcm/°F) is the bounding case.

We require additional information to establish the acceptability of this event at this time.

Conclusions

Transient calculations for cycle 4 were performed using unapproved analytic models. The licensee asserts that the margin to DNBR limits have been demonstrated using these models. Based upon the limited staff review to date, the staff concludes that these predicted margins to DNBR can be over-estimated and that a detailed review could substantively erode these margins. However, the hot spot in the cycle 4 core will be restricted by technical specifications (LOCA Limit FQ) to approximately 80% of the peak power assumed in the transient analysis; this represents a considerable conservatism in the predicted DNBR and it is on this basis that operation during cycle 4 will be acceptable.

F. LOSS OF COOLANT ACCIDENT (LOCA)

In a series of submittals (Reference 45) through 50 the licensee has provided analyses and discussions to show conformance with 10 CFR 50.46(b) and 10 CFR 50, Appendix K.

The licensee referenced previous small break analyses submitted for Cook Unit 2 (staff review reported in Cook Unit 2 SER, Supplement No. 7) by the NSSS vendor and general break spectrum experience for Westinghouse designs to show that small breaks are not limiting for Cook Unit 2 (Reference 46).

Other Generic studies (Reference 51) were cited by the licensee (Reference 46) to indicate that cold leg breaks are the most limiting location for large breaks at Cook Unit 2.

The licensee submitted a topical report (Reference 45) containing analyses of large cold leg guillotine and split breaks which identified the double-ended cold leg guillotine break with a coefficient of discharge equal to 1.0 (DECLG, Cd=1.0) to be limiting. These analyses were performed with a newly submitted EXXON Nuclear Co. Evaluation model described in XN-NF-82-20 and its revisions and supplements. Staff review of differences between this model and the previously approved EXXON model is being reported under separate cover. That model is acceptable for the ECCS analyses for this reload.

In addition to the modified thermal-hydraulic LOCA methodology, the analyses also utilized RODEX2 to calculate stored energy input to the evaluation model calculation. RODEX2 has not been reviewed by the staff.

To confirm the identification of the worst break, hot-rod calculations were rerun for the guillotine breaks analyzed in XN-82-35. These calculations were rerun using GAPEX (staff approved; Reference 49) to calculate hot rod stored energy. Results from these reanalyses agreed with the trends shown in XN-82-35 and also confirmed the DECLG, Cd=1.0 as the limiting break.

The worst break (DECLG, Cd=1.0) was reanalyzed (Reference 48) using GAPEX to calculate stored energy for both the hot channel and the average channel. The resultant calculated peak cladding temperature for this case assuming the traditional "worst" single failure of loss of one low pressure injection pump was 2091°F.

Responding to the staff concern that, for this design, the most limiting case may be with ECCS at maximum performance ("no failure-single failure") rather than with the traditional "worst" single failure, the "no failure-single failure" case analysis was presented (Reference 48) for the DECLG, Cd=1.0 case. The principal effect of the "no failure" is on containment backpressure which influences the magnitude of calculated peak cladding temperature, but not the shape of the break spectrum. Therefore, only the previously identified worst case was reanalyzed with maximum ECCS performance. ECCS inputs were verified to have been maximized (Reference 50) to produce the greatest temperature effect. Analysis of this "no failure-single failure" case with a total peaking factor of 2.04 resulted in a calculated peak cladding temperature of 2198°F, a calculated maximum local metal/water reaction of 7.62 percent, and a total core-wide metal/water reaction of less than one percent. These are below the limits specified in 10 CFR 50.46(b) of 2200°F, 17 percent, and one percent, respectively. The calculated cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling and previous staff review (Cook Unit 2 SER, Supplement No. 7, December 1977) concluded that the ECCS design for Cook Unit 2 is adequate to maintain this condition, satisfying the long-term cooling requirement of 10 CFR 50.46(b).

All of the analyses presented by the licensee considered the clad swelling and rupture concerns expressed in NUREG-0630.

The change in total peaking factor to 2.04 has been reflected in a proposed technical specification change.

Based on the above we conclude that the requirements of 10 CFR 50.46(B), and of Appendix K have been met, and that the LOCA analyses for Cook Unit 2 are acceptable.

4. GENERAL CONCLUSIONS

I. Our review concludes that the information provided by the licensee, and other information available to the NRC, provides an acceptable level of safety for all the related licensing basis events for D.C.Cook Unit 2 for operation of Cycle 4 at a reactor core output of 3411 mwt, providing the constraints described in the following paragraphs, and which are part of the existing technical specifications, are confirmed.

II. The current state of development of the generic & plant specific application of the plant transient model PTS-PWR2 is unsatisfactory for its use as a reliable assessment of MDNBR for Cycle 4 of the Donald C. Cook 2 Nuclear Unit. This applies to both abnormal operating occurences (Class II Transients) as well as Accidents (Class IV events).

III. To ensure the restoration and maintenance of acceptable thermal margins under Transient and Accident conditions for Cycle 4 of D. C. Cook Nuclear Plant Unit 2, we are applying the Following constraints which are conditions attaching to the issuance of this SER:

a. The PTS-PWR2 model, and its adjunct thermal-hydraulic models, cannot be used by the licensee to justify changes to the set points and related uncertainties, and instrumentation response and delay time, for Reactor Protection System (RPS) and Engineered Safeguards Features (ESF) initiation and actuation functions.

b. The maximum value of $F_Q(Z)$ for the reactor core is to be limited to a maximum value of 2.04 irrespective of any subsequent changes to this value permitted by revisions to LOCA calculations. Since DNBR margins in the current calculation for Cycle 4 were calculated assuming $F_Q(Z)$ of 2.55, this represents a 20% reduction in peak power over that assumed in the transient analyses with a considerable improvement in resulting thermal margins.

c. No change is allowable to the current technical specification in respect of moderator temperature coefficients . The current T.S. are based on the Cycle 2 calculations, which basis provides additional margins over the Cycle 4 calculations and ensures maintenance of acceptable Cycle 2 references for all events.

IV. The NRC will reconsider the above constraints for D.C. Cook Nuclear Plant Unit 2, when the licensee submits plant transient and adjunct core thermal-hydraulic calculations, based on plant specific models which have been validated and verified to a level acceptable to the NRC.

V. The licensee for D. C. Cook Nuclear Plant Unit 2 cannot use the submittals for plant transient analysis for cycle 4, either for models, or results, as reference documents.

VI. The licensee must submit, within 90 days after receipt of this safety evaluation report (SER), the specific additional information identified in this SER. This 90 days represents a reasonable period of time for preparation of the information to be submitted. The additional information needed is as follows:

1. Reference: "Loss of Single Reactor Coolant Pump-Locked Rotor (and Broken Pump Shaft).

1.1 Improved model to represent this event, or additional justification for acceptability of the present method.

1.2 Provide a confirmatory analysis which demonstrates that specified acceptable fuel design limits are not violated for the case of loss of off-site power. Justification for any delays assumed between reactor/turbine trip and loss of offsite power must be provided.

2. Reference: The accidental opening of one feedwater control valve with the reactor at 100% power at BOC. Provide the information necessary to justify the acceptability of the steam generator heat transfer characteristics used in these analyses.

3. Reference: Excessive heat removal due to a feedwater system malfunction causing a bypass of the feedwater heating system leading to a reduction in feedwater temperature (EOC, full power and Westinghouse automatic rod control). Provide the additional information requested in the SER.

4. Reference: Excessive heat removal due to a feedwater system malfunction at zero load causing feedwater temperature to be reduced to 70°F. Provide the additional information requested.

# REFERENCES


1. Donald C. Cook Nuclear Plant Updated Final Safety Analysis Report, July 1982, Indiana & Michigan Electric Company, American Electric Power System. Transmitted by letter from R. S. Hunter (IMEC) to H. R. Denton (NRC) dated July 21, 1982 on subject of "First FSAR Update, Compliance with 10 CFR 50.71(e). NRC Accession No. 8207270152.

2. Letter from R. F. Hering (I&MECo) to Harold R. Denton (NRC) dated April 7, 1982 on the subject of "Application For Cycle 4 Reload and Uprate License Amendment." NRC Accession No. 8204140288.

3. Letter from R. F. Hering (I&MECo) to Harold R. Denton dated April 7, 1982, on the subject of ECCS Analysis and Power Distribution Limits Technical Specifications. NRC Accession No. 8204140366.

4. Letter from G. F. Owsley (EXXON) to D. G. Eisenhut (NRC) dated June 30, 1982 on the subject of EXXON Report Identified as XN-NF-82-32(P), XN-NF-82-35, and XN-NF-82-37. NRC Accession No. 8207080420.

5. Letter from R. S. Hunter (I&MEPCo) to Harold R. Denton (NRC) dated July 8, 1982 on the subject of Unit 2 Cycle 4 Safety Analysis. NRC Accession No. 8207140257.

6. XN-NF-82-35, Donald C. Cook Unit 2 LOCA ECCS Analysis Using EXEM/PWR Large Break Results, April 1982. Received July 27, 1982.

7. XN-NF-82-37 D. C. Cook Unit 2, Cycle 4 Safety Analysis Report, April 1982. Received by letter to Reference 5. above and NRC Accession No. 8207080420.

8. XN-NF-82-32(P), "Plant Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3425 MWt," dated April 1982. Received by letter to Reference 4. above and NRC Accession No. 8207080420.

9. XN-NF-82-35, "Donald C. Cook Unit 2 LOCA ECCS Analysis Using EXEM/PWR Large Break Results, dated April 1982. Received by letter to Reference 3. above and NRC Accession No. 8207080420.

10. XN-74-5, Rev. 1, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)".

11. XN-CC-38, "Users Manual for PTSPWR2 - A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients"

12. XN-CC-38, Supp. 1, "Users Manual for PTSPWR2 - A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients"

13. XN-NF-75-21(P) Revision 1. X COBRA - 111C: A Computer Code to Determine Distribution of Coolant During Steady State and Transient Core Operation.

14. Letter from J. E. Dolan (AEP) to J. G. Keppler (NRC) dated October 2, 1978, on subject of Donald C. Cook Nuclear Plant Unit No. 2, Start Up Test Report.

15. Letter from G. . Owsley (EXXON) to Robert B. Licciardo (NRC) dated September 8, 1982, on the subject of EXXON Reports identified as: (a) (b) and (c)

    (a) XN-74-5, Rev. 1, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)"

    (b) XN-CC-38, "Users Manual for PTSPWR2 - A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients"

    (c) XN-CC-38, Supp. 1, "Users Manual for PTSPWR2 - A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients"

16. XN-NF-75-21(P): X COBRA - 111C Revisions 2: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, July 1982.

17. Letter from G. F. Owsley (EXXON) to J. A. Mitchell (NRC) dated October 1, 1982 concerning "Potential Radiological Consequences of Incidents Involving High Exposure Fuel.

18. Letter from G. F. Owsley (EXXON) to D. G. Eisenhut (NRC) dated October 8, 1982, transmitting 25 copies of Exxon Report XN-NF-83-82(P) Revision 1. Plant Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3425 MWT, dated October 1982.

19. Letter from J. C. Chandler (EXXON) to Dr. Cecil Thomas (NRC) dated September 15, 1982, transmitting 25 (copies) of EXXON Report XN-NF-75-21(P), Revision 2. X COBRA - 111C: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, dated September 1982. (Received October 13, 1982)

20. Letter from J. C. Chandler (EXXON) to Dr. Cecil Thomas (NRC) dated September 15, 1982 transmitting (25) copies of EXXON Report XN-NF-82-21(P), Revision 1, "Application of EXXON Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations" dated September 1982. (Received October 7, 1982)

21. Letter from H. G. Shaw (EXXON) to Milton P. Alexich (AEP) dated September 15, 1982, on the subject of XN-NF-82-37(P), Supplement 1 "D. C. Cook Unit 2, Cycle 4 Safety Analysis Report, dated September 15, 1982. Advanced copy for "Information only". (Received October 16, 1982)

22. Letter from R. S. Hunter (I&MECo) to Harold R. Denton (NRC) dated September 8, 1982 on the subject of Unit 2, Cycle 4, EXXON Nuclear Company (ENC) Submittal Scheduling. NRC Accession No. 8209130053.

23. Letter from R. S. Hunter (I&MECo) to Harold R. Denton, Unit 2 dated September 30, 1982 on the subject of Unit 2 Cycle 4, Technical Specification Change Request.

24. Vacant

25. Vacant

26. Letter from G. F. Owsley (EXXON) to D. G. Eisenhut (NRC) dated November 24, 1982 on the subject of D. C. Cook Unit 2, Potential Radiological Consequence of Transients Involving High Exposure Fuel, November 1982. NRC Accession No. 8211290377 821124.

27. XN-NF-78-44 "A Generic Analyses of the Control Rod Ejection Transient for Pressurized Water Reactors, EXXON Nuclear Company, Inc., January 1979.

28. J. N. Morgan, XTRA-PWR: A Computer Code for the Calculation of rapid Transients in Pressurized Water Reactors with Moderator and Fuel Temperature Feedback, XN-CC-32, September, 1985

29. Information received by hand from the Licensee (I&MECo) through G. Mazetis (NRC) at 9:15am, December 8, 1982 and variously described as:

    a) Attachment No. 1 to AEP: NRC: 0637G. D. C. Cook N P, Unit No. 2. Description of Unit 2 Cycle 4 Technical Specification Change Requests

    b) Attachment No. 2 to AEP: NRC: 0637G. D. C. Cook N P, Unit No. 2. Revised Technical Specification Pages.

    c) Attachment No. 3 to DEP: NRC: 0637G. Presentation of Additional Information Concerning EXXON Report XN-NF-82-37.

    d) Attachment No. 4 to AEP: NRC: 0637E. Response to NRC Staff Questions.

    e) Attachment No. 5 to AEP: NRC: 0637G. EXXON Break Spectrum Analysis.

30. Letter from F. T. Adams (EXXON) to R. B. A. Licciardo (NRC), dated December 10, 1982, on the subject of D. C. Cook Unit 2 Plant Transient Simulation Cases.

31. Letter from G. F. Owsley (EXXON) to D. G. Eisenhut (NRC) dated, November 24, 1982, Transmitting (35) copies to XN-N1-82-35. Supp. 1, "Donald C. Cook Unit 2 Cycle 4 Limiting Break LOCA-ECCS Analyses During EXEM/PWR. NRC Accession No. 8211290369.

32. XN-NF-77-57(P) Supp. 2, "EXXON Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II.

33. Letter from D. C. Cook (EXXON) to J. J. Holonich (NRC) dated December 9, 1982 on the Subject of ENC Thermal-Hydaulic Analyses for PWR Mixed Cores. NRC Accession No. NBR: 8212140411.

34. Copy of Draft from H. G. Shaw (EXXON) to M. P. Alexich (I&MEC) dated December 15, 1982, on the subject of Draft Responses to Issues raised at December 2, 1982. [Proprietary]

35. Letter from R. S. Hunter (I&MEP) to Harold R. Denton (NRC) on the Letter Subject of ECCS Analysis Technical Specification and Formally Docketing: XN-NF-82-35 Supplement 1, Donald C. Cook Unit 2 Cycle 4 Limiting Break LOCA-ECCS Analyses Using EXEM-PWR; No. XN-NF-82-32, Revision 1, ("Plant Transient Analysis in the Donald C. Cook Unit 2 Reactor at 3425 MWT) and No. XN-NF-82-90 ("Donald C. Cook Unit 2 Potential Radiological Consequences of Incidents Involving High Exposure Fuel)".

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37. Memo from L. S. Rubenstein NRC/NRR/DSI/CPB to T. M. Novak NRC/NRR/DL, dated December 13, 1982 on the subject of "Review of XN-NF-82-21(P), Revision 1.

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39. EXXON Nuclear DNB Correlation for PWR Fuel Design, XN-NF-621(P) Revision 1, EXXON Nuclear Company, Inc. Richland Washington 99352, April 1982.

40. E. D. Hughes, "A correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia IN-142.

41. Letter from G. F. Owsley (EXXON) to Cecil O. Thomas (NRC) dated October 8, 1982 on the Subject of:

1) XN-NF-CC-38, "Users Manual for PTSPWR2- A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients, dated December 1976.

2) XN-NF-CC-38 Supp. 1, "Users Manual for PTSPWR2- A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transient, dated October 1979. NRC Accession No. 8210130123

42. Memo from Themis Speis (NRR/DSI/RSB) to Gus Lainas (NRR/DL) dated November 3, 1982 on the subject of Donald C. Cook Nuclear Plant Unit 2 - RSB Review of Cycle 4 Reload Applications: Plant Transient Analysis, Request for Additional Information.

43. Letter from G. C. Cooke (EXXON) to D. L. Wigginton (NRC) dated January 10, 1983 on the subject of D. C. Cook Unit 2 Reactor Coolant Flow Rate.

44. Letter from R. S. Hunter (I&MECO) to H. R. Denton (NRC) on subject of Unit 2 Cycle 4 Additional Reload and Uprate information.

45. "Donald C. Cook Unit 2 LOCA Analysis Using EXEM/PWR Large Break Results," XN-NF-82-35, EXXON Nuclear Co. Inc,. Richland, Washington, April 1982.

46. "Response to Informal NRC Concerns Related to the D.C. Cook Unit 2 ECCS Analysis," G. Owsley (ENC) to F. Orr (NRC), August 20, 1982.

47. "Response to NRC Verbal Questions Regarding D.C. Cook Unit 2 Break Spectrum Calculations," G. Owsley (ENC) to F. Orr (NRC), October 29, 1982.

48. "Donald C. Cook Unit 2 Cycle 4 Limiting Break LOCA -ECCS Analysis Using EXEM/PWR," XN-NF-82-35, Supplement 1, EXXON Nuclear Co, Inc., Richland, Washington, November 1982.

49. Attachment No. 5 to AEP:NRC;0637G, "EXXON Break Spectrum Analysis," received by hand December 2, 1982.

50. "NRC Question Regarding Full ECCS Flow," S.E. Jensen (ENC) to F. Orr (NRC), December 9, 1982.

51. "Water Reactor Evaluation Model (WREM): PWR Nodalization and Sensitivity Studies," Regulatory Staff - Technical Review USAEC, October 1977.

## D. Radiological Consequences

### Background

By letter dated April 7, 1982, as supplemented, Indiana & Michigan Electric Company, the licensee for D.C. Cook Unit 2, requested approval for Cycle 4 operation. This cycle will be at an uprated power of 3425 $MW_t$ and includes burnup beyond the traditional value to 30,000 MWd/MTU core average with a peak module burnup of 43,000 MWd/MTU.

By letter dated November 24, 1982, report number XN-NF-82-90, "D. C. Cook Unit 2 Potential Radiological Consequences of Incidents Involving High Exposure Fuel" was submitted on the D. C. Cook Unit 2 docket. This letter was subsequently referenced by the licensee in their letter dated December 9, 1982. This report covers calculations by Exxon Nuclear Corporation of the radiological consequences of accidents at the higher level for the above burnup limit.

### Evaluation

The licensee's submittal was reviewed to assure that all the requested effects were considered. That is, changes in isotopic mix of nuclides available for release following accidents, the potential for failure of fuel following accidents, pool decontamination factor changes due to rod internal pressure charges, and release of volatile fission products into the pellet-clad gap. With the exception noted below, all the factors were considered in the submittal in a manner to show that the mitigation features and the design of the plant are adequate to control the radiological consequences of accidents.

The licensee did not evaluate the radiological consequences of the locked rotor, steamline break or rod ejection accidents since the calculations show no fuel failures. We concur with the conclusion that there is no need to calculate the consequences of these accidents if there are no fuel failures anticipated for these events. In addition, it is the staff's judgment that a very small number of failed fuel rods (e.g. less than 1 percent) would not result in dose estimates exceeding the regulatory guidelines.

The evaluation of the fuel handling accident inside containment was performed by Exxon in accordance with the assumptions of Regulatory Guide 1.25, even though the conditions at the end of cycle 4 will be beyond the basis stated in the Guide. Since no justification for continued conservatism of these assumptions was provided by the licensee, the staff independently evaluated this accident.

The missing justification concerns the fraction of noble gas and iodine assumed to be in the pellet-clad gap of the highest power module. Report number XN-NF-82-37(P) Supplement 1, "D.C. Cook Unit 2, Cycle 4 Safety Analysis Report," shows that the highest power module is a first cycle module. Therefore, the case to be considered is a module at about 15,000 MWd/MTU at the highest allowable linear heat generation rate, about 13 kW/ft. For this case, calculations based on the fission gas release model in the proposed ANS 5.4 standard shows gap fractions less than 30% of of $^{85}$Kr, about 10% of $^{131}$I and less than 10% of all other radionoble gases and radioiodines. Therefore, it is not necessary to consider up to 30% of these nuclides within the gap, as the licensee did for the fuel handling accident outside containment. The assumptions used by the staff and the results of the calculation are given in Table D-1. The results show that the delay to 100 hours from shutdown and site related parameters are adequate to mitigate the consequences of this accident.

It should also be noted that the estimates of the consequences of fuel handling accidents in the spent fuel pool would similarly be affected by burnup and gap fraction changes. However, the staff's experience indicates that the fuel handling accident in the containment is the more limiting event (for off-site dose considerations), resulting from a shorter decay time, and a lack of any filtration of the effluents from the containment. Because the estimated consequences for the accident in the containment are below the staff's guidelines of 75 rem, we conclude that fuel handling accidents in the spent fuel pool would also meet the regulatory dose guidelines.

Conclusion

The licensee and the staff have considered the factors dependent upon power level (to 3425 $MW_t$) and burnup (to 30,000 MWd/MTU core average for peak module 43,000 MWd/MTU) that impact the radiological consequences of accidents. Assuming that the licensee's evaluation of the level of fuel failures (or absence of fuel failures) is confirmed, there are no identified issues that would preclude the higher power level or the extended burnup. We have further concluded that very small number of fuel failures (less than 1%) would not result in dose estimates exceeding the regulatory guidelines.

Table D-1

Assumptions for and Results of Calculation of the Fuel Handling Accident Inside Containment

| | | |
|---|---|---|
| Power level | 3425 $MW_t$ | |
| Peaking factor | 2.1 | |
| Fuel failures | 1 module of 193 | |
| No filtration | | |
| Shutdown time | 100 hrs | |

Meteorological factors* (sec/$m^3$)

| | | |
|---|---|---|
| Exclusion Area Boundary | 0-2 hours | $2.1 \times 10^{-4}$ |
| Low Population Zone | 0-8 hours | $1.8 \times 10^{-5}$ |

| Doses (Rem) | Thyroid | Whole Body |
|---|---|---|
| EAB | 73 | .3 |
| LPZ | 6 | <.1 |

*Memorandum Hulman to Knighton, September 4, 1979

## E. Environmental Impact Appraisal

### 1. Radiological

We have evaluated the potential environmental impact associated with this proposed license amendment as required by the NEPA and Section 51.7 of 10 CFR Part 51.

We have reviewed the Final Environmental Statement (FES) dated August 1973 and Supplement No. 1, dated November 1977 related to the operation of D. C. Cook Nuclear Power Plant, Unit Nos. 1 and 2.

The evaluation of the radioactive waste treatment systems was performed for a thermal power level of 3391 Mwt not for 3411 Mwt. Increasing the thermal power level of D. C. Cook to 3411 Mwt is not expected to increase the estimated releases of radioactive materials and the estimated radiological impact given in the FES. We expect the increase will be less than the percentage increase in the thermal power level (0.6 percent).

Increasing the thermal power rating to 3411 Mwt may cause an increase in radiological consequences by the ratio of the power levels (0.6 percent). This slight increase in power will not change the conclusion in the FES that the environmental risk due to postulated radiological accidents is exceedingly small.

The use of ENC 17x17 fuel assemblies also raises the equilibrium cycle average core burnup. However, for Cycle 4, the ENC fuel average burnup is 20,000 MWD/MTU which is below the design basis average burnup of 33,000 MWD/MTU. Therefore, for Cycle 4 there will be no change in potential effluent types or amounts due to the increased average burnup.

Implementation of the proposed amendment will, therefore, not significantly increase normal radiological effluents from the plant. Implementation will also not allow the licensee to discharge concentrations greater than the maximum allowed nor to discharge more activity in a year than the maximum allowed. Compliance with the present Technical Specifications will adequately control releases such that there will be no appreciable effect on the environment due to operation under these proposed changes, and the conclusion reached in the FES remains valid.

### 2. Non-Radiological Impacts

We have also performed an environmental review of the D. C. Cook proposed amendment to allow higher thermal power limits as a result of a reload of extended life fuel. Our analysis indicates that the potential nonradiological environmental effects from higher power will be confined to the aquatic environment as the plant is cooled by water from Lake Michigan. No change in chemical effluents is anticipated by this action.

The reactor heat production rate of 3411 megawatts thermal will be increased by twenty megawatts with the new core reload. This represents an increase of 0.6% on the heat rate. About 2/3 of the twenty megawatts will result in waste heat to be discharged to the environment, mainly to the condenser cooling water, again roughly a 0.6% increase. The percent increase in the condenser temperature rise would be roughly the same assuming the circulating flow is unchanged. This increase, about 0.07°C, will not result in a measurable increase in temperature in the cooling water discharge plume and, therefore, will have negligible effect on the aquatic biota in the receiving water.

The NPDES permit (administered by the State of Michigan) contains restrictions on the extent of the mixing zone (aeral plume size) in Lake Michigan and a limit on the daily maximum amount of heat discharged to the lake. The licensee believes that the 0.6% increase can be made within the existing limit and, therefore, has not requested an amendment to the permit.

We conclude that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for D. C. Cook Nuclear Plant.

3. Environmental Considerations

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Donald C. Cook Nuclear Plant Unit 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

F. Safety Conclusion

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

Date: January 14, 1983

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 to Facility Operating License No. DPR-74, issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of the Donald C. Cook Nuclear Plant, Unit No. 2 (the facility) located in Berrien County, Michigan. The amendment is effective as of its date of issuance.

This amendment approves the Cycle 4 reload, the increase in power level from 3391 to 3411 megawatts thermal, and changes the related Technical Specifications. A License Condition for Cycle 4 is imposed as is a general condition prohibiting Cycle 5 operation until further approval is obtained from the NRC.

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 this action was published in the FEDERAL REGISTER on August 27, 1982 (47 FR 37983). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility.

For further details with respect to this action, see (1) the application for amendment dated April 7, 1982, as supplemented by letters dated June 11 and June 30, 1982, July 8, 1982, September 30, 1982, December 9 and December 22, 1982 and January 12, 1983, (2) Amendment No. 48 to License No. DPR-74, and (3) the Commission's related Safety Evaluation and Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., 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 Licensing.

Dated at Bethesda, Maryland this 14th day of January, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing