Docket No. 50-315

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

FEB 16 1977

attach to dis amdt #18
in a Rpts jacket

Gentlemen:

In response to your requests dated July 20, and December 7, 1976, and February 4 and 9, 1977, supplemented by letters dated July 19, October 1, November 5, 17, 23 and 30, December 7, 9 and 13, 1976, and February 8 and 9, 1977, the Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant Unit No. 1.

The amendment consists of license and Technical Specification changes which authorize operation with (1) reactor power levels not in excess of 3250 megawatts (thermal) for core cycle 2 with 65 Exxon Nuclear Company reload fuel assemblies and an Exxon Nuclear Company emergency core cooling system analysis, (2) revised technical specification requirements for the ice condenser system, and (3) modifications to certain electrically operated valves to preclude single failures that would result in loss of emergency core cooling system capacity and to eliminate the need for actuation of these valves by personnel outside the control room. The amendment also corrects minor errors and inconsistencies in the technical specification requirements for containment air recirculation fan response time, containment penetration and valve leakage rates, the audit responsibility of the Nuclear Safety and Design Review Committee, and safety related hydraulic snubbers.

In addition, this amendment updates paragraph 2.D. of the Facility Operating License to reference the latest approved revisions of the security plan for the Donald C. Cook Nuclear Plant.

Some of your proposed Technical Specification changes have been modified to meet our requirements. These modifications have been discussed with and accepted by your staff.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

NRC FORM 318 (9-76) NRCM 0240

☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

FEB 16 1977

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

Sincerely,
Original signed by
Richard D. Silver
Dennis L. Ziemann, Chief for
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 18 to
   License No. DPR-58
2. Safety Evaluation
3. Notice

c w/enclosures:
See next page

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ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan
RBenedict
MMWnczalc
RBaer

Licensee and I&E:HQ were informed on February 16, 1977, that the amendment had been approved. MHFletcher 2/16/77

DOR:STS	DOR:AD/OT	DOR:RS/OT	
JMMcGough	DEisenhut	RBaer	14Feb77 PSC CPB/DSS
2/15/77	2/15/77	2/15/77	

OFFICE ➤	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:AD/ORs	DSS:AD/RS
SURNAME ➤	MHFletcher:ah	RMDiggs	K.Blaels	DLZiemann for	KRGoller	DRoss
DATE ➤	2/14/77	2/14/77	2/16/77	2/16/77	2/16/77	2/15/77

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U. S. Environmental Protection Agency
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U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
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Lake Township
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Honorable W. Mabry, Mayor
City of Bridgman, Michigan 49106

cc w/enclosures and cy of I&M
filings dtd. 10/1/76; 11/5, 17
& 23/76; 12/7, 9 & 13/76 and
2/4, 8 & 9/77:
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Room 1 - Capitol Building
Lansing, Michigan 48913

Office of Intergovernmental
Relations
Department of Management
and Budget
2nd Floor, Lewis Cass Building
P. O. Box 30026
Lansing, Michigan 48909

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

INDIANA & MICHIGAN ELECTRIC COMPANY
INDIANA & MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. DPR-58

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

   A. The applications for amendment by Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensees) dated July 20 and December 7, 1976 and February 4 and 9, 1977, supplemented by letters dated July 19, October 1, November 5, 17, 23 and 30, December 7, 9 and 13, 1976, and February 8 and 9, 1977, comply 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 paragraphs 2.C(1), 2.C(2) and 2.D of Facility Operating License No. DPR-58 are hereby amended in their entirety to read as follows:

"2.C(1) Maximum Power Level

The licensees are authorized to operate the Donald C. Cook Nuclear Plant, Unit No. 1, at steady state reactor core power levels not to exceed 3250 megawatts (thermal).

2.C(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 18, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

2.D The licensees shall maintain in effect and fully implement all provisions of the NRC Staff-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of proprietary documents, collectively titled, "Donald C. Cook Nuclear Plant Industrial Security Plan," as follows:

Original, submitted with letter dated August 15, 1972, with revisions dated September 21, 1972, January 22, 1973, November 27, 1973, May 24, 1974, November 13, 1974, November 14, 1975, April 5, 1976, October 4, 1976, and December 20, 1976."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 16, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-315

INDIANA & MICHIGAN ELECTRIC COMPANY
INDIANA & MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. DPR-58, issued to Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensees), which revised the license and Technical Specifications for operation of the Donald C. Cook Nuclear Plant Unit No. 1 (the facility) located in Berrien County, Michigan. The amendment is effective as of its date of issuance.

The amendment revised the facility license and its appended Technical Specifications to authorize continued full power operation with (1) 65 Exxon Nuclear Company (ENC) 15 x 15 reload fuel assemblies, (2) operating limits based on an ENC ECCS evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50, (3) modifications to certain electrically operated ECCS related valves, (4) revised surveillance requirements for ice condenser portion of the containment systems, and (5) corrections to the Technical Specification requirements for containment air recirculation fan response time, containment penetration and valve leakage rates, the audit responsibilities of the off-site review committee, and safety related hydraulic snubbers.

The applications for the amendment comply 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 items (1) and (2) above was published in the Federal Register on September 7, 1976 (41 F.R. 37679). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on items (1) and (2) above. Prior public notice of items (3), (4) and (5) above was not required since these items do not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated July 20 and December 7, 1976, and February 4 and 9, 1977, and supplements dated July 19, October 1, November 5, 17, 23 and 30, and December 7, 9, and 13, 1976, and February 8 and 9, 1977, (2) Amendment No. 18 to License No. DPR-58, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Maude Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16th day of February, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard D. Silver

Richard D. Silver, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

ATTACHMENT TO LICENSE AMENDMENT NO. 18

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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

Pages

3/4 1-1
3/4 2-1
3/4 2-2
3/4 2-3
3/4 2-4
3/4 2-5
3/4 2-15
3/4 2-16
3/4 2-17
3/4 2-18 (added)
3/4 2-19 (added)
3/4 3-29
3/4 5-4
3/4 5-5
3/4 6-2
3/4 6-22
3/4 6-26
3/4 6-27
3/4 6-30
3/4 6-31
3/4 7-33
B 3/4 1-1
B 3/4 2-1
B 3/4 2-2
B 3/4 2-5
B 3/4 2-6
6-11

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1.75% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 1.75% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1.75% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2##, at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

---

*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

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

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

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

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

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

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

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

   a) POWER OPERATION may continue provided:

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

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

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

*See Special Test Exception 3.10.2
#E(t) is defined on Figure 3.2-3

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

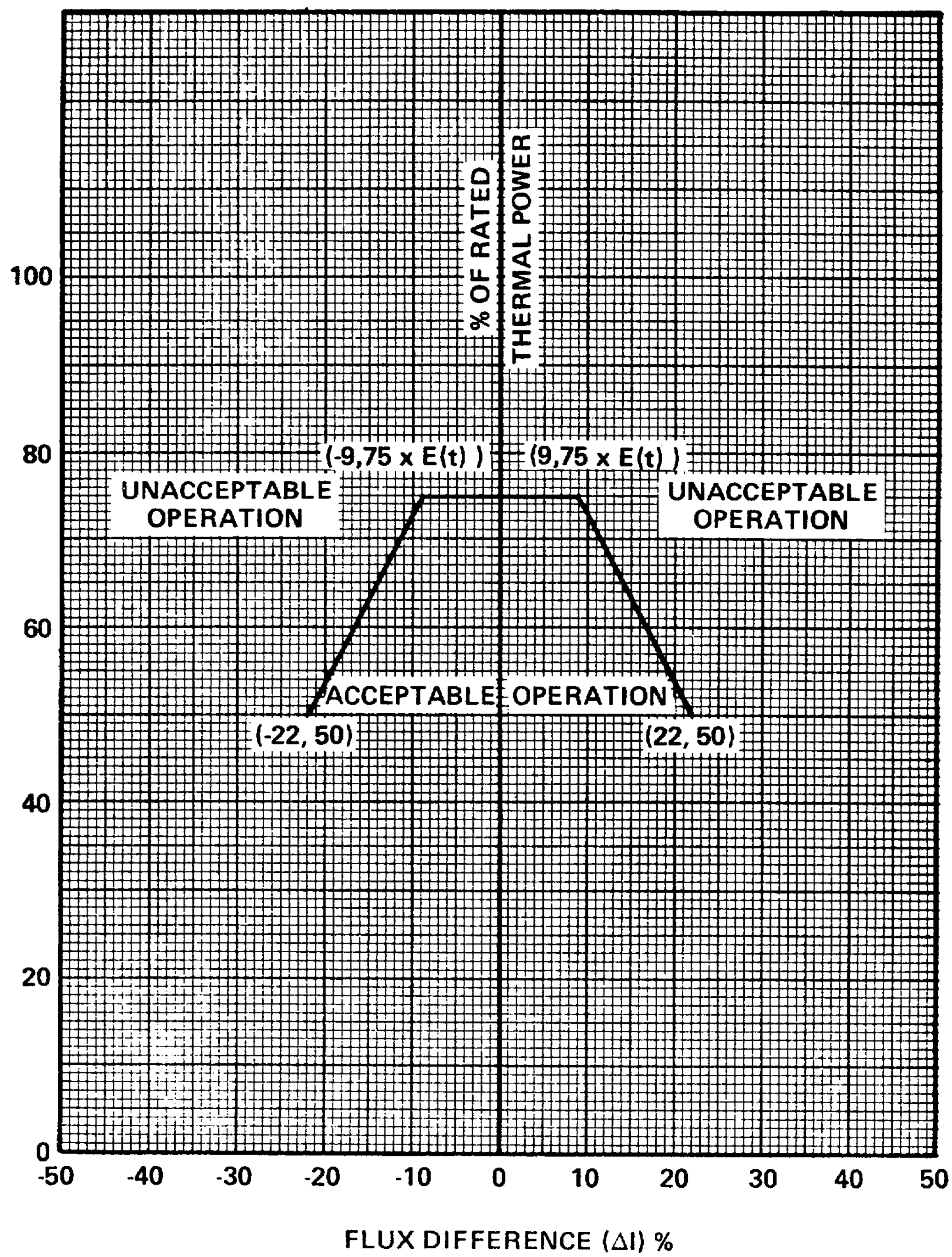

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \left[\frac{1.95}{P}\right] [E(t)] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [(3.90)] [E(t)] [(K(Z)] \text{ for } P \leq 0.5$$

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

E(t) is defined on Figure 3.2-3,

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

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

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

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

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

2. The relationship:

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

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

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

Where:

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

b. P_L is the fraction of RATED THERMAL POWER.

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

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns above 84% x E(t) of RATED THERMAL POWER in accordance with:

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

Where:

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

e. E(t) is defined on Figure 3.2-3.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

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

The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 above 84% x E(t) OF RATED THERMAL POWER#.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER one percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to 84% x E(t) or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER to 84% x E(t) or less of RATED THERMAL POWER within 15 minutes.

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

SURVEILLANCE REQUIREMENTS

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

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

1. At least once per 8 hours, and

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

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

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

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

1. At least once per 8 hours, and

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

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

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

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

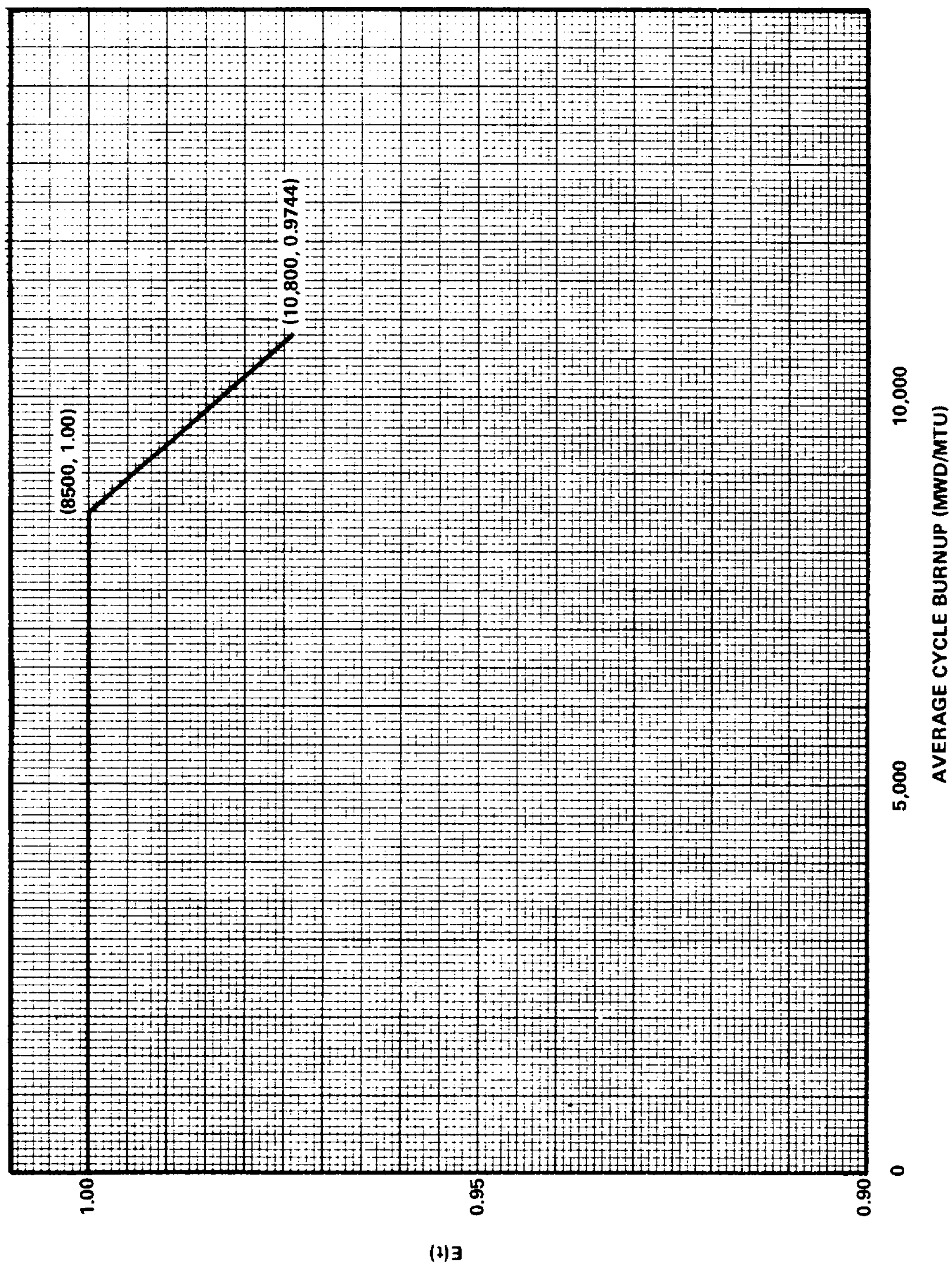

Figure 3.2-3 E(t) Normalized Core Power as a Function of Cycle Burnup

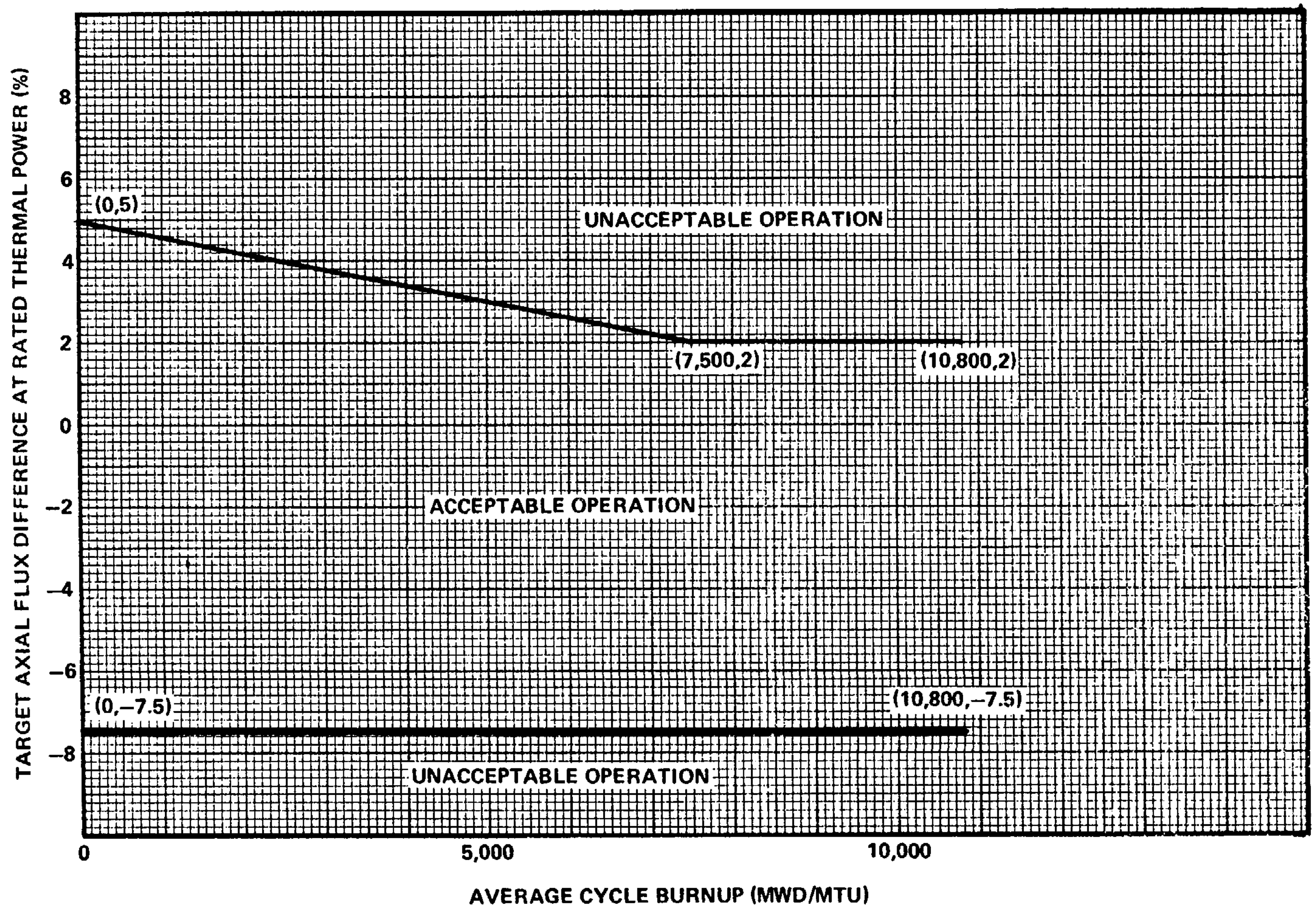

Figure 3.2-4 Target Axial Flux Difference at RATED THERMAL POWER as a Function of Cycle Burnup

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq$ 14.0#/48.0##
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Recirculation Fan	$\leq$ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

\# Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\# Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each centrifugal charging pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 2405 psig on recirculation flow.

   c) Operates for at least 15 minutes.

2. Verifying that each safety injection pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 1445 psig on recirculation flow.

   c) Operates for at least 15 minutes.

3. Verifying that each residual heat removal pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure $\geq$ 195 psig on recirculation flow.

   c) Operates for at least 15 minutes.

4. Verifying that the following valves are in the specified positions with control power locked-out:

Valve Number	Valve Function	Valve Position
a. IMO-390	a. RWST to RHR	a. Open
b. IMO-315	b. Low head SI to Hot Leg	b. Closed
c. IMO-325	c. Low head SI to Hot Leg	c. Closed

SURVEILLANCE REQUIREMENTS (Continued)

Valve Number	Valve Function	Valve Position
d. IMO 262*	d. Mini flow line	d. Open
e. IMO 263*	e. Mini flow line	e. Open
f. IMO 261*	f. SI Suction	f. Open
g. ICM 305*	g. Sump line	g. Closed
h. ICM 306*	h. Sump line	h. Closed

5. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

6. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing containment integrity, and

   2. Of the areas affected within containment at the completion of each containment entry when containment integrity is established.

c. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

*These valves must change position during the switchover from injection to recirculation flow following LOCA.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection signal.

3. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

   1. All penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1, and

   2. All equipment hatches are closed and sealed,

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12.0 psig, and

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$ or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at P_a, 12.0 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

TABLE 3.6-1 (Continued)

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
C. CONTAINMENT PURGE AND EXHAUST (Continued)			
12. VCR-205	Upper Comp. Purge Air Inlet	Yes	10
13. VCR-206	Upper Comp. Purge Air Outlet	Yes	10
14. VCR-207*	Cont. Press Relief Fan Isolation	Yes	10
D. MANUAL ISOLATION VALVES(1)			
1. ICM-111	RHR to RC Cold Legs	Yes	NA
2. ICM-129	RHR Inlet to Pumps	No	NA
3. ICM-250	Boron Injection Inlet	Yes	NA
4. ICM-251	Boron Injection Inlet	Yes	NA
5. ICM-260	Safety Injection Inlet	Yes	NA
6. ICM-265	Safety Injection Inlet	Yes	NA
7. ICM-305	RHR Suction from Sump	Yes	NA
8. ICM-306	RHR Suction from Sump	Yes	NA
9. ICM-311	RHR to RC Hot Legs	Yes	NA
10. ICM-321	RHR to RC Hot Legs	Yes	NA
11. DW-209	Demineralized Water Supply for Refueling Cavity	Yes	NA
12. DW-210	Demineralized Water Supply for Refueling Cavity	Yes	NA
13. NPX 151 VI	Dead Weight Tester	Yes	NA
14. PA 145*	Containment Service Air	No	NA
15. SF-151*	Refueling Water Supply	Yes	NA
16. SF-153*	Refueling Water Supply	Yes	NA
17. SF-159	Refueling Cavity Drain to Purification System	Yes	NA
18. SF-160	Refueling Cavity Drain to Purification System	Yes	NA
19. SI-171	Safety Injection Test Line	Yes	NA
20. SI-172	Accumulator Test Line	Yes	NA

TABLE 3.6-1 (Continued)

VALVE NUMBER		FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
D.	MANUAL ISOLATION VALVES[(1)] (Continued)			
21.	CCR-440	CCW from Main Steam Penetration	Yes	NA
22.	CCR-441	CCW from Main Steam Penetration	Yes	NA
23.	MCM-221	Main Steam to Auxiliary Feed Pump	No	NA
24.	MCM-231	Main Steam to Auxiliary Feed Pump	No	NA
25.	CCM-430	CCW to East Pressure Equalization Fan	Yes	NA
26.	CCM-431	CCW from East Pressure Equalization Fan	Yes	NA
27.	CCM-432	CCW to West Pressure Equalization Fan	Yes	NA
28.	CCM-433	CCW from West Pressure Equalization Fan	Yes	NA
29.	SM-8*	Upper Containment Sample	Yes	NA
30.	SM-10*	Upper Containment Sample	Yes	NA
31.	SM-4*	Instrument Room Sample	Yes	NA
32.	SM-6*	Instrument Room Sample	Yes	NA

NA - Manual Valve-Isolation time not applicable.
(1) - Includes motor operated valves which do not isolate automatically.
* - May be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying during a recombiner system functional test that the heater sheath temperature increases to $\geq$ 1200°F within 5 hours and is maintained for at least 4 hours.

4. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test immediately following the above required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

3/4.6.5 ICE CONDENSER

ICE BED

LIMITING CONDITION FOR OPERATION

3.6.5.1 The ice bed shall be OPERABLE with:

a. The stored ice having a sodium tetraborate concentration of at least 1800 ppm boron and a pH of 9.0 to 9.5,

b. Flow channels through the ice condenser,

c. A maximum ice bed temperature of $\leq$ 27°F,

d. Each ice basket containing at least 1220 lbs of ice, and

e. 1944 ice baskets.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ice bed inoperable, restore the ice bed to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 The ice condenser shall be determined OPERABLE:

a. At least once per 12 hours by using the ice bed temperature monitoring system to verify that the maximum ice bed temperature is $\leq$ 27°F.

b. At least once per 12 months by:

1. Chemical analyses which verify that at least 9 representative samples of stored ice have a boron concentration of at least 1800 ppm as sodium tetraborate and a pH of 9.0 to 9.5.

2. Weighing a representative sample of at least 144 ice baskets and verifying that each basket contains at least 1220 lbs of ice. The representative sample shall include 6 baskets from each of the 24 ice condenser bays and

SURVEILLANCE REQUIREMENTS (Continued)

shall be constituted of one basket each from Radial Rows 1, 2, 4, 6, 8 and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1220 pounds of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1220 pounds/basket at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - bays 1 through 7, Group 2 - bays 8 through 14, and Group 3 - bays 15 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8 and 9 in each group shall not be less than 1220 pounds/basket at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,371,450 pounds.

3. Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on flow passages between ice baskets, past lattice frames, through the intermediate and top deck floor grating, or past the lower inlet plenum support structures and turning vanes is restricted to a thickness of $\leq$ 0.38 inches. If one flow passage per bay is found to have an accumulation of frost or ice with a thickness of $>$ 0.38 inches, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each 1/3 of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

ICE BED TEMPERATURE MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The ice bed temperature monitoring system shall be OPERABLE with at least 2 OPERABLE RTD channels in the ice bed at elevations 652' 2-1/4", 672' 5-1/4" and 696' 2-1/4" for each one third of the ice condenser.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the ice bed temperature monitoring system inoperable, POWER OPERATION may continue for up to 30 days provided:

    1. The ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed;

    2. The last recorded mean ice bed temperature was ≤ 20°F and steady; and

    3. The ice condenser cooling system is OPERABLE with at least:

        a) 21 OPERABLE air handling units,

        b) 2 OPERABLE glycol circulating pumps, and

        c) 3 OPERABLE 25 ton refrigeration chillers;

    otherwise, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the ice bed temperature monitoring system inoperable and with the ice condenser cooling system not satisfying the minimum components OPERABILITY requirements of a.3 above, POWER OPERATION may continue for up to 6 days provided the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed and the last recorded mean ice bed temperature was ≤ 15°F and steady; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The ice bed temperature monitoring system shall be determined OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours.

ICE CONDENSER DOORS

LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more ice condenser doors open or otherwise inoperable, POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained $\leq$ 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the inlet door position monitoring system, and

b. Demonstrated OPERABLE during shutdown at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 6 months thereafter by:

   1. Verifying that the torque required to initially open each door is $\leq$ 675 inch pounds.

   2. Verifying that opening of each door is not impaired by ice, frost or debris.

   3. Testing a sample of at least 25% of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional

SURVEILLANCE REQUIREMENTS (Continued)

torque component. The doors selected for determination of the "door opening torque" shall be selected to ensure that all doors are tested at least once during four test intervals.

4. Testing a sample of at least 25% of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component. The doors selected for determination of the "door closing torque" shall be selected to ensure that all doors are tested at least once during four test intervals.

5. Calculation of the frictional torque of each door tested in accordance with 3 and 4, above. The calculated frictional torque shall be $\leq$ 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

Door	Lifting Force
1. Adjacent to Crane Wall	$\leq$ 37.4 lbs.
2. Paired with Door Adjacent to Crane Wall	$\leq$ 33.8 lbs.
3. Adjacent to Containment Wall	$\leq$ 31.8 lbs.
4. Paired with Door Adjacent to Containment Wall	$\leq$ 31.0 lbs.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 3 months by visually verifying:

a. That the doors are in place, and

b. That no condensation, frost, or ice has formed on the doors or blankets which would restrict their lifting and opening if required.

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
(Ser. No. 11928)	RC Pump Seal Water Supply. Az 130°, Elev. 610'. Between RC Pump No. 2 and Crane Wall, immediately under grating	I	Yes	No
1-FWS-4	Feedwater. Az 160°, Elev. 648' Behind Stm. Gen. No. 2	I	Yes	No
1-FWS-4	Feedwater. Az 160°, Elev. 646' Behind Stm. Gen. No. 2	I	Yes	No
1-FWS-5	Feedwater. Az 163°, Elev. 647' Behind Stm. Gen. No. 2	I	Yes	No
1-FWS-6	Feedwater. Az 157°, Elev. 630' Behind Stm. Gen. No. 2	I	Yes	No
1-MSS-3	Main Steam. Az 175°, Elev. 648' Between Stm. Gen. No. 2 & No. 3	I	Yes	Yes
1-MSS-4	Main Steam. Az 170°, Elev. 648' Between Stm. Gen. No. 2 & No. 3	I	Yes	Yes
1-MSS-5	Main Steam. Az 185°, Elev. 648' Between Stm. Gen. No. 2 & No. 3	I	Yes	Yes
1-MSS-6	Main Steam. Az 185°, Elev. 648' Between Stm. Gen. No. 2 & No. 3	I	Yes	Yes
1-FWS-9	Feedwater, Az 194°, Elev. 634' Behind Stm. Gen. No. 3	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
1-GRC-S-582	Reactor Coolant. Az 195°, Elev. 619'. Near Reactor Cavity Wall, across from Stm. Gen. No. 3	I	Yes	No
1-FWS-8	Feedwater. Az 200°, Elev. 648' Behind Stm. Gen. No. 3	I	Yes	No
1-FWS-8	Feedwater. Az 200°, Elev. 646' Behind Stm. Gen. No. 3	I	Yes	No
1-FWS-7	Feedwater. Az 195°, Elev. 647' Behind Stm. Gen. No. 3	I	Yes	No
Ser. No 25.12620.007-1	Steam Generator No. 1, Elev. 665'	I	Yes	Yes
Ser. No. 25.12620.007-5	Steam Generator No. 1, Elev. 665'	I	Yes	Yes
Ser. No. 25.12620.007-7	Steam Generator No. 1. Elev. 665'	I	Yes	Yes
Ser. No. 25.12620.007-14	Steam Generator No. 1, Elev. 665'	I	Yes	Yes
Ser. No. 25.12620.007-3	Steam Generator No. 2, Elev. 665'	I	Yes	Yes
Ser. No. 25.12620.007-6	Steam Generator No. 2, Elev. 665'	I	Yes	Yes
Ser No. 25.1[illegible]0.007-9	Steam Generator No. 2, Elev. 665'	I	Yes	Yes

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.75%Δk/k is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. With T_{avg} <350°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Δk/k shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 ± 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, and 3) T_{avg} is above the P-12 interlock setpoint.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 5106 gallons of 20,100 ppm borated water from the boric acid storage tanks or 52,622 gallons of 1950 ppm borated water from the refueling water storage tank.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are meet and the ECCS acceptance criteria limit of 2200°F is not exceeded.

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

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

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

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

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

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 75% x E(t) of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 75% x E(t) of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 75% x E(t) and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

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

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

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

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

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

A burnup dependent F_Q is speficied for fuel Cycle 2 to account for the effects of uncertainties in the Exxon reload fuel internal pin pressure on flow blockage calculations for 10 CFR Part 50 Appendix K criteria. The internal fuel pin pressure uncertainty was calculated using the methods of "Revision 1 to the Staff Safety Evaluation Report on the Exxon Nuclear Company WREM-Based Generic PWR-ECCS Evaluation Model ENC-WREM-II," dated January 5, 1977.

3/4.2.4 QUADRANT POWER TILT RATIO

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

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

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

BASES

3/4.2.5 DNB PARAMETERS

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

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

3/4.2.6 AXIAL POWER DISTRIBUTION

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

During Cycle 2 operation, the unit will have 65 fuel assemblies supplied by the Exxon Nuclear Company with the balance remaining from the original fuel supplied by Westinghouse Electric Corporation. The specified limit of F_Q represents the Exxon fuel, the more restrictive of these two fuel types during initial Cycle 2 operation.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NSDRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. Any other area of facility operation considered appropriate by the NSDRC.

AUTHORITY

6.5.2.9 The NSDRC shall report to and advise the Senior Executive Vice President, Engineering and Construction, AEPSC, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSDRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSDRC meeting shall be prepared, approved and forwarded to the Senior Executive Vice President, Engineering and Construction, AEPSC, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Executive Vice President, Engineering and Construction, AEPSC, within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Executive Vice President, Engineering and Construction, AEPSC, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PNSRC and submitted to the NSDRC and the Vice President, Nuclear Engineering.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO LICENSE NO. DPR-58

INDIANA & MICHIGAN ELECTRIC COMPANY
INDIANA & MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

INTRODUCTION

By letters dated July 20, and December 7, 1976, and February 4 and 9, 1977, supplemented by letters dated July 19, October 1, November 5, 17, 23 and 30, and December 7, 9 and 13, 1976, and February 8 and 9, 1977, Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensee) requested amendments (hereinafter referred to as amendment) to Facility Operating License No. DPR-58 for the D. C. Cook Nuclear Plant Unit No. 1 (the facility). The amendment would authorize operation with reactor power levels not in excess of 3250 megawatts (thermal) for core cycle 2 with (1) 65 Exxon Nuclear Company reload fuel assemblies, (2) an Exxon Nuclear Company emergency core cooling system analysis, (3) revised technical specification requirements for the ice condenser system, and (4) modifications to certain electrically operated valves to preclude single failures that would result in loss of emergency core cooling system capacity and to eliminate the need for actuation of the valves by personnel outside the control room. The amendment also (5) corrects minor errors and inconsistencies in the technical specification requirements for containment air recirculation fan response time, containment penetration and valve leakage rates, the audit responsibilities of the Nuclear Safety and Design Review Committee, and safety related hydraulic snubbers.

BACKGROUND

Operation of D. C. Cook Unit No. 1 at steady state reactor core power levels not in excess of 3250 megawatts thermal (100 percent of rated power) was authorized by Amendment No. 14 to Facility Operating License DPR-58 issued by the Commission on May 28, 1976. This authorization is effective only until the reactor is shutdown for refueling at which time, unless the Commission takes further licensing action, the authorized power level would be 2632.5 megawatts thermal (81 percent of rated power). This restriction on maximum power level after the first cycle, was made by the Nuclear Regulatory Commission in accordance with the advice of the Advisory Committee on Reactor Safeguards (ACRS) in the March 11, 1976 letter from Dade W. Moeller, ACRS Chairman, to the Honorable William A. Anders, Chairman of the Nuclear Regulatory Commission.

For the first refueling of the D. C. Cook Unit No. 1 reactor (scheduled for December 1976 - January 1977), the licensee has proposed to replace 65 of the original Westinghouse Electric Corporation fuel assemblies with Exxon Nuclear Company (ENC) assemblies and to demonstrate conformance of the facility's ECCS with the requirements of 10 CFR Part 50.46 by using an ENC ECCS analysis. The Core Reload Evaluation section of this report addresses the licensee's proposal.

In the ACRS letter dated October 17, 1973, regarding D. C. Cook and in the Regulatory staff evaluation of "Tests Conducted to Demonstrate the Functional Adequacy of the Ice Condenser Design" dated April 25, 1974, the need for a program to periodically measure the weights of selected ice baskets in the ice condenser was recognized. In support of this program, the licensee has submitted, for our review, the results of the ice weighings since January 1976. We have combined this data with previous weighing results and our evaluation is given in the section of this report entitled Ice Condenser Evaluation.

In Section 5.4 of Supplement No. 5 to the D. C. Cook Unit No. 1 Safety Evaluation Report, dated January 1976, we identified certain valves whose spurious actuation could adversely affect the performance of the ECCS following a postulated loss of coolant accident (LOCA). The staff concluded that removing AC power from the valves would be an acceptable method to prevent such spurious actuation. However, several of the valves must be repositioned about 25 minutes after the postulated LOCA when the reactor cooling mode is shifted from injection to recirculation. To operate these valves after the LOCA, operator action would be required to restore electrical power to the valves at the motor control centers outside of the control room. The licensee committed to modify the control circuits of these valves to eliminate the need for operator action outside the control room and preclude single failures that would result in spurious valve operation. By letters dated February 27 and November 23, 1976, the licensee submitted proposed control circuit modifications for our review. The acceptability of these modifications is discussed in this report.

CORE RELOAD EVALUATION

Discussion

By Reference 1, Indiana and Michigan Electric Company and Indiana and Michigan Power Company (I&M) requested that the operating license of the Donald C. Cook Nuclear Plant Unit No. 1 (License No. DPR-58) be amended to permit continued operation at steady state core power levels up to 3250 MWt (100% power).

By Reference 2, the licensee proposed changes to the Technical Specifications based upon an Exxon Nuclear Company (ENC) ECCS evaluation model which conforms to the requirements of the Commission's regulations in 10 CFR 50.46.

The D. C. Cook Unit No. 1 core consists of 193 fuel assemblies, each having a 15x15 array of fuel rods. Each fuel assembly contains 204 fuel rods, 20 rod control cluster (RCC) guide tubes, and one instrumentation tube. Cycle 1 fuel was designed and fabricated by Westinghouse Electric Company. For Cycle 2, 65 original fuel assemblies will be replaced by fuel assemblies which were designed and fabricated by the Exxon Nuclear Company (ENC). The Westinghouse fuel remaining in the core during Cycle 2 (64 assemblies with an enrichment of 2.8% U-235, and 64 assemblies with an enrichment of 3.3% U-235) will be scatter loaded throughout the interior of the core.

One Exxon fuel assembly will be loaded in the center of the core, and the remaining 64 Exxon fuel assemblies will be loaded in the core periphery. The Exxon fuel has an enrichment of 2.95% U-235; 16 of the new fuel assemblies have burnable poison rods - B_4C pellets (8 assemblies having 8 burnable poison rods, and 8 assemblies having 4 burnable poison rods).

Mechanical Design

The Cycle 2 core will consist of 65 ENC assemblies and 128 Westinghouse assemblies. The fuel assembly design parameters are shown in Table 1.

The ENC reload fuel is clad with Zircaloy-4 and prepressurized with helium. One significant difference between the ENC and Westinghouse fuel is its clad thickness. The ENC fuel cladding is 23% thicker than the Westinghouse fuel cladding. The ENC fuel also has shorter pellets than the Westinghouse fuel pellets. The staff believes that

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS*

	Westinghouse Low Enrichment Fuel	Westinghouse High Enrichment Fuel	Exxon Fuel
Enrichment (wt % U-235)	2.80	3.30	2.95
Number of Assemblies	64	64	65
Pellet Density, (%)	95	95	94
Pellet-to-Clad Gap (mils)	7.5	7.5	7.5
Pellet Diameter (inches)	0.3659	0.3659	0.3565
Fuel Stack Height (inches)	143.4	142.8	144
Number of Fuel Rods/Assembly	204	204	204
Region Average Burnup at BOC2, (MWD/T)	18,100	13,900	0
Cladding Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Cladding OD (inches)	0.422	0.422	0.424
Cladding Thickness (inches)	0.0243	0.0243	0.030
Instrument Tube Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Instrument Tube OD (inches)	0.546	0.546	0.544
Spacer Grid Material	Inconel	Inconel	Zircaloy-4 structural members with Inconel springs
Number of Spacer Grids	7	7	7

*initial unirradiated conditions

these design differences, thicker cladding and shorter fuel pellets, are improvements with regard to pellet-cladding-interaction (PCI) (Reference 27). Hence the ENC fuel is expected to be more resistant to PCI than the original fuel.

The total weight of the ENC fuel bundles and the Westinghouse high enrichment fuel bundles does not differ by more than 2%.

The ENC fuel design for D. C. Cook Unit 1 Cycle 2 is similar to that supplied by ENC for other facilities. The cladding material, Zircaloy-4, was used in previous ENC fuel supplied for Palisades Core II, Yankee-Rowe Core XII, and H. B. Robinson Core IV. 136 assemblies were loaded into Palisades Core II, 40 assemblies were loaded into Yankee Rowe XII, and 52 assemblies were loaded into H. B. Robinson Core IV. The enrichment of the fuel for D. C. Cook is in the range of that used in the above cores. The general dimensions of the fuel rod (including diametral gap which is of importance for stored energy) are within the range of PWR fuel designs previously irradiated successfully.

In response to our question regarding compatibility between the D. C. Cook Unit 1 fuel handling equipment, and the Exxon reload fuel, the licensee performed fit-up tests at the D. C. Cook plant (Reference 4). These tests indicated that there should be no difficulties in handling the Exxon fuel at the D. C. Cook Nuclear Plant.

Approximately 1000 bundles manufactured by ENC are in-core, in PWRs and BWRs, with burnups ranging from first cycle to 25,000 MWD/MTU. Approximately 10% of these have exposures between 15-20,000 MWD/MTU. Based on sipping results and surveillance of representative assemblies, no failures have been observed or detected.

The design of the ENC 15 x 15 reload fuel assemblies is described in Reference 3 which is a generic report giving a detailed description of fuel assembly design methods and bases. Portions of this report regarding the effects of fuel densification have been reviewed by the NRC staff and found acceptable. Other sections of the report are currently under review on a generic basis; and, therefore, have not been considered in our review of the use of ENC fuel in D. C. Cook Unit No. 1. Our conclusions concerning the acceptability of the use of ENC fuel in D. C. Cook Unit No. 1 are based on (1) the fuel design and analytical methods which have previously been reviewed by the staff, (2) the similarity of the reload fuel to that used in Cycle 1 which was previously found acceptable, and (3) the successful operating performance of ENC fuel. Based on these factors, we conclude that there is reasonable assurance that the performance of the ENC reload fuel will be acceptable.

In Reference 1, the licensee indicated that one or two fingers had broken off a control rod during drop timing tests which were performed during an April, 1976 outage. The finger(s) which had detached from the rod control cluster assembly are presently in the fully inserted position. Subsequent to the control rod finger failure the licensee performed analyses of the core to correlate the incore flux measurements with the known failures. The licensee's study and measurements showed a slight skew in the burnup of the fuel assemblies surrounding the the failed rod. The licensee concluded that the effect is not significant enough to have a restrictive impact on the shuffling scheme for Cycle 2.

In Reference 4, the licensee stated that he has reviewed this control rod problem with Westinghouse. Based upon Westinghouse's extensive testing and evaluation program that was conducted prior to commercial use of these control rods, and based upon over 2700 rod-years of operation in commercial nuclear power plants, Westinghouse and the licensee believe that the control rod failure is an isolated incident not indicative of generic failures. Westinghouse has examined whether control rod scram capabilities would be compromised by failure of additional rodlet/fingers. After considering the possible rod failure modes Westinghouse concluded that such failures would not affect the reactor scram times.

As indicated in Reference 5, the licensee, by letter dated February 8, 1977, provided additional information concerning the failed control rod. At the end of Cycle 1, rod drop timing and drag tests were performed on each control rod and all test results were within acceptable limits. The results of the licensee's visual examinations of the affected control rod including the rod drive rod, guide tube removable insert, and the inside of the upper guide tube were provided. The examination revealed that a two-rodlet vane had separated from the control rod hub at the vane-hub interface (at a tack welded and brazed joint); however, no cause for the failure could be determined. The fuel assembly containing the broken rod has been removed from the core. The licensee is continuing to investigate the cause of the failure and will provide additional data 60 days after startup.

Based on the above information, we have concluded that, although the specific cause of the failure has not been established, there is no evidence to indicate that this is a generic problem and, therefore, Cycle 2 operation of the D. C. Cook Unit No. 1 need not be delayed. We will, however, evaluate future information provided by the licensee on this subject.

Nuclear Design and Technical Specification Changes

Technical Specification changes required as a result of the nuclear design for D. C. Cook Cycle 2 are discussed in the following sections.

Shutdown Margin

In the analysis of the steam line break accident in Reference 6 for end of cycle 2 with the reactor at no load operating temperature, a minimum shutdown margin of 1.75% $\Delta k/k$ is initially required to control the reactivity transient. The corresponding shutdown margin for Cycle 1 was 1.6% $\Delta k/k$. Accordingly, the licensee has proposed to change the Technical Specification end of cycle shutdown margin requirement to 1.75% $\Delta k/k$. We find this acceptable because it will prevent return to criticality in the event of a small steam line break and also provide an acceptable margin to DNB in the unlikely event of a large steam line break accident.

Power Distribution Control and Monitoring

The ECCS analysis, Reference 7, was performed with an assumed heat flux hot channel factor, $F_Q(Z)$ of 1.95. The maximum $F_Q(Z)$ at full power for Cycle 1 was 1.98. By letter dated February 9, 1977, the licensee reported the results of an analysis of the effect of burnup on $F_Q(Z)$ using the NRC staff assumptions in Reference 15 for flow blockage calculations. The licensee determined that the value of 1.95 would hold until a burnup of 8500 MWD/MTU in Cycle 2 and then would decrease linearly to 1.90 at an expected end-of-life burnup of 10,800 MWD/MTU. The D. C. Cook Unit No. 1 Technical Specifications will be changed to reflect this behavior of the $F_Q(Z)$ limit. We find this to be acceptable.

The licensee will continue to use the Axial Power Distribution Monitoring System, APDMS, during cycle 2 to ensure that $F_Q(Z) \times P$ (P = fraction of full power) does not exceed the $F_Q(Z)$ limit during normal operation. The APDMS essentially performs direct measurements of the core peaking factor with in-core movable detectors and requires, by Technical Specification, power reduction and other appropriate actions if the peaking factor exceeds its limit. Experience with the APDMS during Cycle 1 operation in D. C. Cook, and in other reactors employing an APDMS indicates this system provides an adequate indication of F_Q violations. Data accumulated thus far support at least a 95% probability with a 95% confidence level that the $F_Q(Z)$ will not exceed the APDMS measured value with the uncertainties assigned to the APDMS and failure probabilities taken into account.

The power level, above which APDMS monitoring is required, is determined by the $F_Q(Z)$ which can be justified by monitoring with ex-core detectors. For the latter part of Cycle 1 operation, the APDMS was required to be in operation above 90% of full rated power. This was a result of a plant specific analysis of ex-core detector monitoring using constant axial offset control (CAOC) which indicated an $F_Q(Z)$ of 2.18 would not be exceeded using these procedures. Thus 90% $\leq$ 1.98/2.18 X 100.

For Cycle 2 operation, Exxon has provided, in Reference 8, an analysis of CAOC procedures which they term power distribution control (PDC) which indicates $F_Q(Z)$ will not exceed 2.30 during normal operation. Reference 8 has been reviewed and approved by the staff in Reference 23. Based on the methods of Reference 8, a new APDMS monitoring threshold of 1.95/2.30 x 100 = 84% is required from 0-8500 MWD/MTU with a linear decrease to 1.90/2.30 x 100 = 82% at 10,800 MWD/MWU to reflect the burnup dependence of $F_Q(Z)$. The technical specifications will be modified to reflect these requirements.

At power levels up to the APDMS monitoring threshold, PDC procedures will ensure that the $F_Q(Z)$ limit assumed for the LOCA analysis will be maintained. Above this level, the APDMS will provide the same assurance. The PDC procedures are also required to be observed above the monitoring threshold to ensure that axial power shapes not allowed by PDC do not occur and thus potentially violate DNB analyses. This is presently required by the Technical Specifications. We conclude the above provisions will adequately ensure that initial conditions assumed for the LOCA and DNB analyses will be maintained during normal operation at power levels up to and including 100% rated power during Cycle 2.

The PDC study in Reference 8 addressed target offsets in the range -7.5% to 0.0%. To increase plant operating flexibility using PDC, ENC, by letter dated Feburary 11, 1977, submitted an addendum to Reference 8 which provided additional analysis with regard to positive target offsets. Based on this addendum, we have determined that target offsets up to +5% at beginning of Cycle 2 decreasing linearly to +2% for burnup of 7500 MWD/MTU and greater will continue to protect the PDC F_Q limit and are, therefore, acceptable for Cycle 2 operation of D. C. Cook Unit No. 1. These target axial offset values will be incorporated into the facility Technical Specifications.

Physics Test Program

The physics start-up test program proposed for Cycle 2 (Reference 5) is acceptable if the following guidelines are used in verifying predicted control rod bank reactivity worths and the shutdown margin. Control rod bank worths must be measured for banks D, C, B, and A, individually. If any one bank worth differs from the predicted value by more than 15%, or the sum of the worths of the four banks differs from the predicted value by more than 10%, the first shutdown bank should be measured. If the sum of the worths of the control banks and the shutdown bank differs from the predicted value by more than 10%, additional shutdown bank measurements should be performed to verify technical specification shutdown margin. The licensee will be required to include this test in the startup test program.

Analytical Methods

The analytical methods used by the licensee and ENC in the calculation of operational parameters for Core II are described in References 9 and 10.

These documents present the ENC neutronic design calculational methods along with the results obtained when these methods are compared to experimental measurements. We have reviewed and approved these documents. Therefore, we conclude that the analytical methods used to calculate the operational parameters for D. C. Cook Unit No. 1 Core II are acceptable.

Thermal and Hydraulic Design

The thermal-hydraulic analyses of the Cycle 2 core (Reference 1) shows the following:

a. The ENC and Westinghouse assemblies are thermally and hydraulically compatible.

b. The minimum departure from nucleate boiling ratios (MDNBR) for both fuel types are always greater than 1.30 for normal operation and anticipated transients.

The analyses include both experimental measurements and theoretical calculations. ENC has performed hydraulic flow tests to evaluate the compatibility between the ENC and the Westinghouse 15x15 fuel assemblies. The results of these tests show that even though the Westinghouse and ENC fuel assemblies exhibited some differences in the plenum-to-plenum pressure drops and the pressure drops between the tie plates, the difference in flow through the ENC and Westinghouse assemblies is small (average flow difference of 1.4% between the two types of fuel). This difference of coolant flow has been considered in the analyses. It has been determined that it has a negligible effect upon the margin to DNB.

The adequacy of the ENC fuel for meeting MDNBR requirements has been verified with transient analyses performed at 102% power. The results of the transient calculations are discussed later in this evaluation.

DNB calculations show that the MDNBR is greater than the minimum acceptable limit of 1.30 for both ENC and Westinghouse fuel assemblies under the operating conditions of Cycle 2. Additional margin is provided by the fact that the steady state DNB calculations were performed at a stretch power level of 3640 MWt although D. C. Cook Unit 1 will be licensed for only 3250 MWt for Cycle 2.

Based on the above, we have concluded that the thermal and hydraulic design of the Cycle 2 core is acceptable.

On August 9, 1976 Westinghouse Electric Corporation presented data to the staff from recent experiments which showed that fuel rod bowing could have a significant effect on the departure from nucleate boiling ratio (DNBR). In particular, these experiments showed that if a heated fuel rod was bowed to contact with an unheated rod (thimble rod), a reduction in DNBR significantly greater than that expected would occur.

The staff has developed a model based on this data to calculate the DNBR reduction to be expected in operating reactors. This model consists of three components. First, a method of calculating the clearance reduction between adjacent rods due to rod bowing is used to estimate the extent of fuel rod bowing for a given burnup. Second, using the Westinghouse data for DNBR reduction, the DNBR reduction for the calculated extent of rod bow is determined. Finally, the calculated DNBR reduction may be offset by available margin. D. C. Cook has margin available to offset the calculated reduction in DNBR, as discussed below.

For Cycle 2, D. C. Cook will operate with a combination of ENC and Westinghouse fuel. ENC has presented no data on the extent of fuel rod bowing in ENC fuel; however, an analytical method of predicting fuel rod bowing has been presented to the staff (Reference 11). This analytical method has not been accepted because we do not believe a mechanism for fuel rod bowing has been satisfactorily identified. Thus a mechanistic calculation should not be employed.

We therefore have assumed, as an interim position, that the amount of fuel rod bowing expected for ENC fuel will be equal to that expected for Westinghouse fuel. This assumption is considered conservative because the thicker fuel rod cladding and slightly larger rod diameter of the ENC fuel provide a larger moment of inertia to resist bowing forces.

Since the Westinghouse and ENC fuel are of similar design, as described in the Mechanical Design section, the Westinghouse calculational model was used to determine the DNBR reduction for both fuel types (See Attachment A). The maximum calculated reduction in DNBR for D. C. Cook is 27.6%. The NRC staff has permitted licensees to offset calculated DNBR reductions by accounting for certain parameters which affect DNBR calculations for their plants (Attachment A). For D. C. Cook Unit No. 1, the licensee has utilized the minimum DNBR of 2.01 which was calculated for the most limiting anticipated transient. The difference between 2.01 and the current DNBR safety limit of 1.3 results in a 54.6% credit which more than offsets the DNBR reduction of 27.6%. Therefore, no changes in Technical Specifications are necessary to offset the effects of fuel rod bowing on DNBR during Cycle 2.

Transient and Accident Analyses

The licensee provided results of their ECCS analysis in References 2, 7, and 12, and descriptions and results of other transient analyses in References 2, 6 and 13.

ECCS Cooling Performance (LOCA) Analysis

Evaluation Model

The licensee provided Exxon's analysis of the ECCS cooling system performance. (References 2, 7, and 12). The model (Reference 14) addressed hot channel performance for the reload fuel, and the overall reactor response to the composite fueled core. The calculational model used by Exxon for D. C. Cook Unit 1 Cycle 2 was reviewed by the staff, and approved by the staff's Safety Evaluation (Reference 15).

The NRC staff, in Reference 15, specified assumptions to be used to determine the effects of fuel rod internal pressure on flow blockage calculations to demonstrate conformance to 10 CFR Part 50 Appendix K criteria. Using these assumptions, the licensee, by letter dated February 9, 1977, reported that the value of the $F_Q(Z)$ limit would be 1.95 until a Cycle 2 burnup of 8500 MWD/MTU at which time the value would decrease linearly to 1.90 at 10,800 MWD/MTU. The decrease in $F_Q(Z)$ is required to compensate for the assumed increase in fuel pin internal pressure as a function of burnup.

Break Spectrum

The worst break location was identified as the cold leg at the pump discharge. For the first cycle, Westinghouse's analysis identified a double-ended guillotine break of the pump discharge line as the worst break.

ENC performed a series of break size calculations at the pump discharge line, assuming the worst single failure (loss of a low pressure ECCS pump, Reference 16). The calculations were performed for double ended guillotine breaks with discharge coefficients of 1.0, 0.8, and 0.6. The split break configurations were calculated with the break area equal to twice the cross sectional pipe area (8.25 ft^2) and also for the cases of the flow area reduced to 0.8 and 0.6 times that area (6.6 ft^2 and 4.95 ft^2 respectively). As shown in References 2 and 7, it was determined that the 8.25 ft^2 split break is most limiting. The maximum peak clad temperature was shown to be 2196°F, which is below the acceptable upper limit of 2200°F as specified in 10 CFR 50.46(b). In addition, the maximum local metal/water reaction of less than 8% and the total core metal/water reaction of less than 0.8% were within the allowable limits of 17% and 1% respectively. These calculations were done using a total peaking factor of 1.95. Based on this analysis and the analysis of the effect of fuel pin internal pressure on $F_Q(Z)$, the peak linear heat generation rate for the ENC fuel for Cycle 2 is 13.41 kw/ft until 8500 MWD/MTU and then decreases linearly to a value of 13.06 kw/ft at 10,800 MWD/MTU, end-of-life.

With regard to small breaks, in Reference 2 the licensee indicated that the small breaks would result in conditions substantially below the limiting large break results and clearly within the requirements of 10 CFR 50.46.

We have reviewed the above results and agree that the break spectrum has been defined sufficiently to assure that the worst break size and location for D. C. Cook Unit 1/Cycle 2 has been identified and analyzed. We find the break spectrum calculations acceptable. Therefore, we have concluded that operation with the reload core consisting of Westinghouse and ENC fuel assemblies meets the requirements of 10 CFR 50.46 and is acceptable.

Post LOCA Long Term Cooling

In Reference 2, the licensee informed the staff that the existing analyses which demonstrated the Emergency Core Cooling System's capability to meet the long term cooling requirements for Cycle 1 operations are valid and applicable for Cycle 2 operation. We find this to be acceptable.

Upper Head Temperature Analysis for ENC Fuel

In Reference 12, ENC reported the results of studies performed to determine the sensitivity of LOCA calculations to upper head temperature. These studies verified that the use of hot leg temperature for the upper head is conservative for ENC fuel. Consequently, the ENC full break spectrum analysis which was performed with the hot leg temperature assumed for the upper head is conservative and the results are acceptable.

Upper Head Temperature Analysis for the Remaining Westinghouse Fuel

The licensee has submitted a reevaluation of ECCS performance for D. C. Cook Unit No. 1 (Reference 24) in response to our Order for Modification of License issued on August 27, 1976. The reevaluation was made using the October, 1975 version of the Westinghouse ECCS Evaluation Model assuming the upper head fluid temperature equal to the fluid outlet (hot leg) temperature. This analysis supersedes the previously performed ECCS evaluation which used the same October, 1975 version of the evaluation model but which was based on the assumption that the upper head temperature was equal to the cold leg temperature. The reevaluation of the ECCS performance in Westinghouse plants was required because recent experimental data had indicated that the actual temperature in the upper reactor vessel head was in the range of 50-75 percent of the difference between vessel inlet and outlet temperatures (Reference 25).

The reanalysis consisted of the evaluation of ECCS performance for double ended cold leg guillotine breaks (DECLG) with a discharge coefficient C_D of 0.8. The licensee claimed that this break size was representative of the limiting value of peak clad temperature and $Zr-H_2O$ reaction. To justify limiting the ECCS analysis to only one break size, the licensee referenced the previously approved ECCS analysis and the Westinghouse topical report WCAP-8855 which provided sensitivity studies for four loop (15 x 15) plants and which also had been reviewed and approved by the staff (Reference 26). The previous ECCS analysis was performed for a spectrum of four breaks specific for D. C. Cook using the October, 1975 evaluation model with the assumption of upper head temperature equal to the cold leg temperature. This analysis identified the worst break size as the DECLG with $C_D = 0.8$. In addition, the sensitivity studies performed by Westinghouse and reported in Reference 25 indicated that, for a specific plant, the change of upper head temperature from cold to hot leg temperature did not affect the critical break type or size. Based on these references, the licensee has concluded that the break size analyzed is the critical break for the D. C. Cook Plant resulting in the peak clad temperature of 2164°F and the maximum local $Zr-H_2O$ reaction of 6.39 percent.

Based on our review of the submitted documents, we conclude that the results of the ECCS reanalysis, using the October, 1975 version of the Westinghouse ECCS Evaluation Model with upper head temperature equal to the outlet (hot leg) fluid temperature, are conservative relative to the 10 CFR 50.46 criteria, and are acceptable.

CONTAINMENT LOCA ANALYSES

We have evaluated the effects of the D. C. Cook Unit No. 1. Cycle 2 core on containment pressure response following a postulated LOCA. Since the ENC reload core has been designed to the same thermal power rating (3250 megawatts) as the original core, only the core stored energy could alter the blowdown used for the original containment analysis. The thicker clad of the ENC reload fuel results in an increase of 1.5% in the core stored energy. Because core stored energy released to containment constitutes only about 2.5% of the total energy released, the ENC fuel will result in approximately a .04% increase in the integrated energy released to the containment at the time of ice melt. This increase is negligible in comparison to the conservatisms in the currently approved Westinghouse containment analysis mass and energy release model for D. C. Cook Unit No. 1. Therefore, we conclude that there is a negligible change in the LOCA containment analysis as a result of the Cycle 2 core.

Conclusions

Based upon the above information we have concluded that:

The ECCS cooling performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46(b). In addition the plant will conform with the criteria to maintain a coolable geometry and provide satisfactory long term cooling.

The LOCA analyses assumed that there was a coincident loss of offsite power at the initiation of the accident, which would result in pump coastdown. Since these analyses were performed for only four loop operation, and since other modes of operation have not been demonstrated to meet paragraph 50.46, reactor operation will not be permitted with one or more idle loops.

Rod Ejection Accident

In Reference 2, the licensee provided an analysis for the rod ejection incident for fuel Cycle 2. The licensee determined that for the worst case event the fuel limits would not be exceeded. We find this acceptable.

Rod Drop Transient

In Reference 2, the licensee provided information on analysis of the Rod Drop Incident for fuel Cycle 2. The analysis showed that the results for the dropped rod incident and for the dropped bank incident for Cycle 2 are more favorable than those for Cycle 1. We find this acceptable.

Rod Withdrawal Transient

The licensee has provided the results of a reanalysis of the rod withdrawal transient from full power using the Exxon PTSPWR2 Code (Reference 17). We previously reviewed this code and found its results to be acceptable. The rod withdrawal transient was analyzed from an initial power of 3315 MWt (102% power) for both slow and fast rod withdrawal as shown on Table 2. The slow rod withdrawal results in the more severe conditions, but still within the Technical Specification limits (MDNBR = 2.15 vs. lower limit of 1.30, maximum pressurizer pressure of 2279 psia vs. 2750 psia); therefore we find these results to be acceptable.

TABLE 2

SUMMARY OF RESULTS FOR ENC FUEL

Transient	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (W-3)
Initial Conditions For Transients	3315.	210,500.	2220.	2.43
Uncontrolled Rod Withdrawal @8.0 x 10-4$\Delta\rho$/sec	4230.	222,100.	2230.	2.24
Uncontrolled Rod Withdrawal @ 2.0 x $10^{-5}\Delta\rho$/sec	3633.	228,200.	2279.	2.15 (1.93)+
Loss of Flow - 4 Pump Coastdown	3315.	210,500	2256.	2.01 (1.86)+
Loss of Flow - Locked Rotor	3315.	210,500	2242.	1.98
Loss of Load	3321.	210,500	2538.	2.43
Large Steam Line Break	406.	23,140.	*	2.90
Small Steam Line Break	**	**	*	**

* Pressure decreases from initial value.
** The core does not return to criticality.
\+ With rod bow penalty

Loss of Coolant Flow Transient

The analysis of the initial reference cycle showed the loss of coolant flow incidents, pump coastdown and locked rotor, to be the most limiting with respect to DNB (MDNBR's of 1.40 and 1.07 for the pump coastdown and the locked rotor respectively). ENC's reanalysis of these incidents resulted in MDNBR's of 2.01 and 1.98 for the pump coast-down and locked rotor cases respectively. ENC's analysis shows that the maximum pressurizer pressure for these events was 2256 psia (for the pump coastdown). The MDNBR's and maximum pressurizer pressures for these events are within the Technical Specification limits (DNBR > 1.30 and pressurizer pressure < 2750 psia). We find this to be acceptable.

Loss of Load Transient

The loss of load transient was analyzed for the second cycle. This transient was limiting with respect to system pressure. For Cycle 2 the maximum pressurizer pressure calculated for this event was 2538 psia whereas the Technical Specification limit is 2750 psia. The MDNBR and maximum pressurizer pressure are well within the Technical Specification limits and therefore are acceptable.

Other Transients and Accidents

The kinetics parameters for the remaining transients and accidents are within the envelope of parameters analyzed for the reference cycle. Therefore, the results of the reload cycle will be bounded by those for the reference cycle. We find this to be acceptable.

ICE CONDENSER EVALUATION

Since our January 1976 report on the status of the ice weight surveillance program of the D. C. Cook Nuclear Plant Unit No. 1 presented in Supplement 5 to the Safety Evaluation Report (SER), the licensee has performed four additional ice weighings. The results of all ice basket weighing programs and the licensee's conclusions and recommendations have been documented in five reports (References 18 through 22). Our evaluation and conclusions are based upon review of the information presented in the referenced documents.

In January, April, July, and September of 1976 the licensee weighed a sample population of ice baskets as a part of the continuing long-term evaluation of the ice condenser system. The sample populations were composed of 166 baskets in January, 172 baskets in April, 177 baskets in July, and 179 baskets in September. Analysis of these data and comparison with prior data indicate that the average ice loss rate continues to be about 2%/yr, (28 #/yr/basket) with a statistical maximum of about 2.5%/yr, (35 #/yr/basket).

Data from the March, July, and October 1975 basket weighings revealed that ice loss is not uniformly distributed over the ice condenser. A pattern of preferential ice loss was evident, with the baskets in rows closest to the containment wall having the lowest ice loss and the losses becoming progressively greater as the basket positions approach the crane wall. The additional data taken in April, July and September of 1976 have confirmed the existence of the preferential loss pattern within the ice condenser. Analysis of the data for the period from April 1976 through September 1976 indicates that ice baskets adjacent to the containment wall (radial row 1) have an average ice loss rate of about 1/2%/yr while those adjacent to the crane wall (radial row 9) have ice loss rates averaging about 5-3/4%/yr.

In Supplement No. 5 to the SER, we reported that the licensee had developed special weighing equipment which permitted successful weighing of baskets in radial rows 1 and 9 (wall baskets adjacent to the containment wall and crane wall, respectively) during the July and October, 1975 weighing programs. In April of 1976, the licensee weighed the wall baskets with an improved model of the wall basket weighing device. The improvements to the wall basket weighing device have resulted in a signficant reduction in repeatability error associated with the weighing of wall baskets, such that the repeatability error is now comparable with the error associated with

TABLE 3

Average Ice Loss Rate/yr - April 1976 to September 1976

Radial Row	Average Loss Rate
1	7 #/yr (1/2%/yr)
2	9.6 #/yr (3/4%/yr)
4	17 #/yr (1/4%/yr)
6	32 #/yr (2-1/2%/yr)
8	66 #/yr (4-1/2%/yr)
9	82 #/yr (5-3/4%/yr)

NOTE: Row 1 is adjacent to the containment wall
Row 9 is adjacent to the crane wall

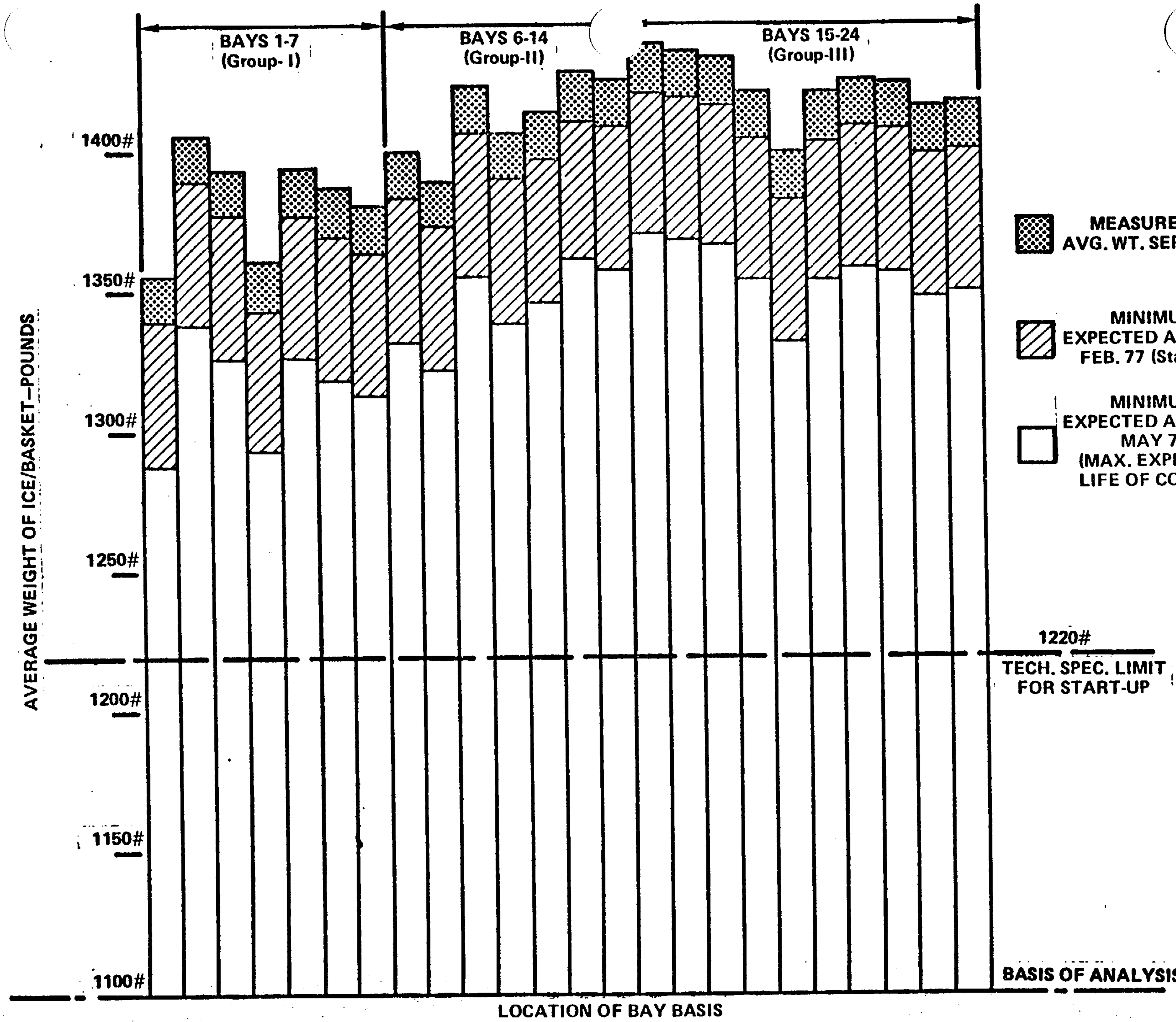
BAYS 1-7
(Group- I)
BAYS 6-14
(Group-II)
BAYS 15-24
(Group-III)
1400#
1350#
1300#
1250#
1200#
1150#
1100#
AVERAGE WEIGHT OF ICE/BASKET—POUNDS
MEASURED
AVG. WT. SEPT. 76
MINIMUM
EXPECTED AVG. WT.
FEB. 77 (Start-Up)
MINIMUM
EXPECTED AVG. WT.
MAY 78
(MAX. EXPECTED
LIFE OF CORE II)
1220#
TECH. SPEC. LIMIT
FOR START-UP
BASIS OF ANALYSIS
LOCATION OF BAY BASIS

Figure-1

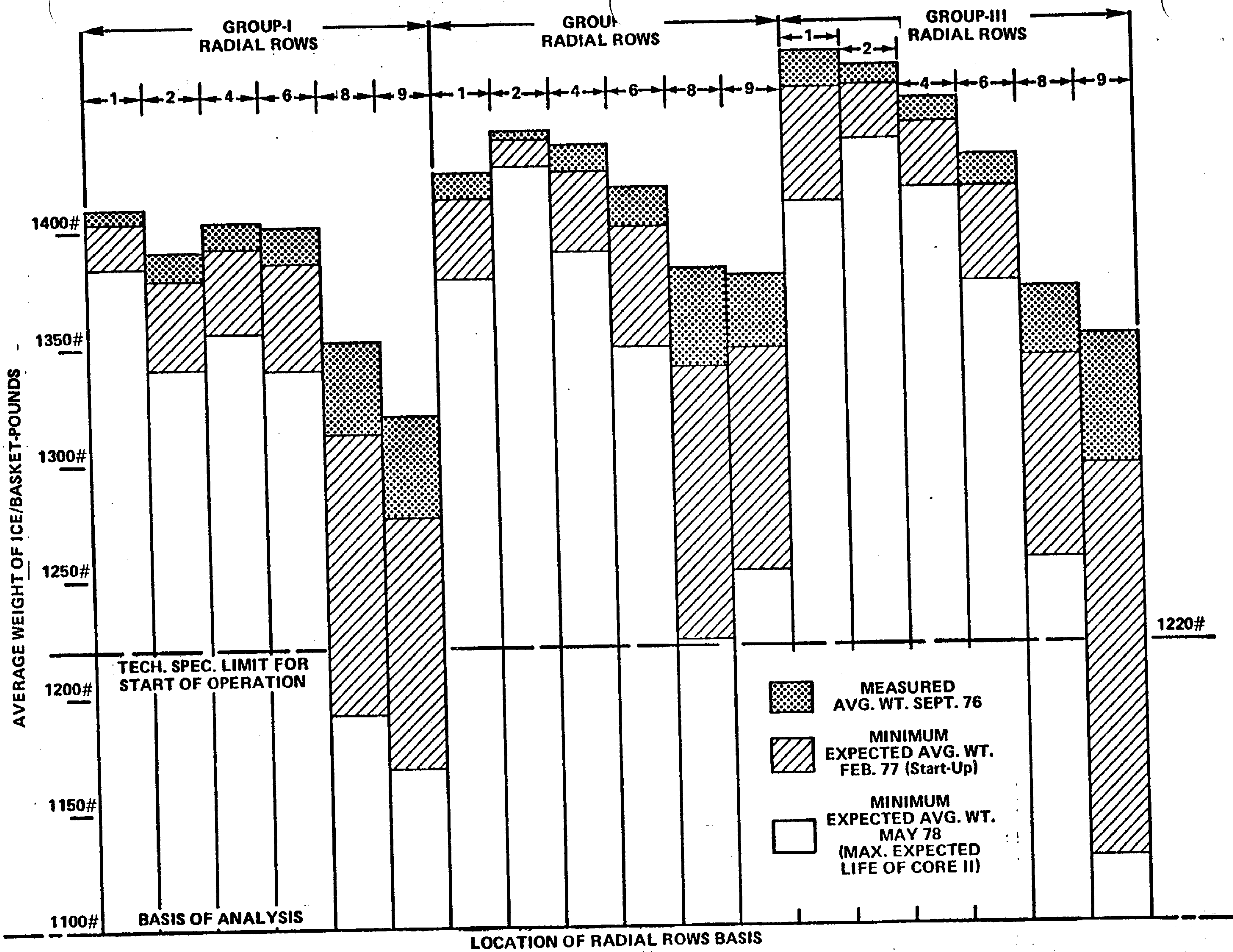
GROUP-I
RADIAL ROWS
GROUP
RADIAL ROWS
GROUP-III
RADIAL ROWS
1
2
4
6
8
9
1
2
4
6
8
9
1
2
4
6
8
9
1400#
1350#
1300#
1250#
1200#
1150#
1100#
1220#
AVERAGE WEIGHT OF ICE/BASKET-POUNDS
TECH. SPEC. LIMIT FOR
START OF OPERATION
BASIS OF ANALYSIS
MEASURED
AVG. WT. SEPT. 76
MINIMUM
EXPECTED AVG. WT.
FEB. 77 (Start-Up)
MINIMUM
EXPECTED AVG. WT.
MAY 78
(MAX. EXPECTED
LIFE OF CORE II)
LOCATION OF RADIAL ROWS BASIS

Figure-2

weighing the remainder of the ice baskets. Analysis of the data taken with the improved wall basket weighing device now indicates that the average ice loss rate in radial row 9 (wall baskets adjacent to the crane wall) is greater than the average loss rate in radial row 8. Average loss rates in radial rows 1 and 2 are significantly less than in rows 8 and 9. Also, rows 1 and 2 do not exhibit a marked difference between the average ice loss rates. The average loss rates for ice baskets in radial rows 1, 2, 8 and 9 for the period of April 1976 to September 1976 are 7 #/yr (1/2%/yr), 9.6 #/yr (3/4%/yr), 66 #/yr (4-1/2%/yr) and 82 #/yr (5-3/4%/yr) respectively as indicated in Table 3.

The distribution of ice within the ice condenser is shown in Figures 1 and 2 on an average weight per bay basis and an average weight per radial row basis, respectively. These figures indicate the distribution of the ice last measured in September 1976, and the projected distribution of the ice based on measured average loss rates and the uncertainty associated with the measurement of loss rates at a 95% level of confidence. The projected distributions, therefore, represent the minimum expected ice weights in the ice condenser for two different future times. The selected points in time are February 1977 and May 1978. We expect February 1977 to be the approximate time for the next ice condenser weighing program and the completion of the first reactor refueling and subsequent plant start-up for fuel cycle 2. May 1978 represents the maximum expected life of fuel cycle 2 based on a design fuel life of 12 months and a 25% contingency.

As we reported in Supplement No. 5 to the SER, the minimum amount of ice uniformly distributed throughout the ice condenser to prevent containment overpressurization in the event of the design basis accident is 1098 pounds/basket. With this as a basis, we have previously established a Technical Specification average weight limit of 1220 pounds/basket for initiation of an operating period (i.e., operability of the ice condenser). This limit is established to assure that during the specified operating period the average weight of any significant group of ice baskets will not be less than the minimum uniformly distributed amount of ice (1100 pounds/basket) assumed in the design basis accident analysis. As may be seen from Figures 1 and 2 using the latest measured average ice weights and the maximum projected loss rates at a 95% level of confidence we would not project any bay or radial row of baskets to fall below the 1100 pound/basket value used in the design basis accident analysis during Cycle 2.

Continued weighings of the weighable basket nearest the ice condenser lower plenum personnel access door (basket 2-8 in bay 24) indicate a local area of greatly increased loss rate. The current rate of ice loss for basket 2-8 is about 155 pounds per year (12%/yr) and the measured weight of ice in the basket in September, 1976 was 1128 pounds. Clearly basket 2-8 and the surrounding baskets (i.e., baskets 1-7, 1-8, and 1-9 and 2-7 and 2-9) in bay 24 would be expected to weigh less than the 1100 pounds assumed in the accident analysis before the start of cycle 2. As a result, the licensee has committed to add approximately 200 pounds of ice to each of the six baskets indicated above during the current refueling outage, by a technique demonstrated during the April 1976 basket weighing program. The licensee and Westinghouse have developed a method by which up to 300 pounds of ice may be added to an individual basket, by drilling a 2 inch diameter hole in the upper six feet of a basket and "trickling" a 34°F solution of borated water into the basket over an extended duration. This method of ice addition appears practical when only a few baskets are involved, but has yet to be proven as a feasible method of ice addition if entire bays or radial rows would require an ice addition. Continued development efforts by the licensee and NRC staff review of procedures and equipment are necessary to permit the large scale addition of ice to signficant groups of baskets. The alternative to large scale ice addition is the complete melt-out of the ice condenser and refilling the ice baskets, a process which would require 3-6 months to complete.

In the October report (Reference 5) the licensee submitted the results of an analysis of the plant response to the design basis accident assuming a maldistribution of the ice in the ice condenser. The analysis shows that the design pressure of the containment (12 psig) is not exceeded when the average ice weight in two bays (162 baskets) is 850 pounds per basket and the average ice weight in the remaining 22 bays (1782 baskets) is 1120 pounds/basket. An analysis of this type which recognizes the measured distribution of the ice inventory may be required in the future to demonstrate the acceptability of the ice condenser for continued operation. However, we believe that operation of the ice condenser with known groups of baskets below an average weight of 1220 pounds/basket should not be permitted solely on the basis of the licensee's analysis until the staff has a confirmatory long term containment analysis capability. We expect the CONTEMPT-4 long term containment code with ice condenser modeling will be available by September 1977.

After reviewing the five reports by the American Electric Power Service Corporation regarding the basket weight history and analysis during the first twenty (20) months of plant operations, we have reached the following conclusions regarding the future operation of D. C. Cook, Unit No. 1, and analysis of the ice condenser:

1. Sufficent data have been collected to conclude that the plant can be operated safely at the full design power level for the expected life of the second reactor core (i.e., until about May 1978).

2. Calculations of average ice weight per basket for a bay, a radial row or the total ice condenser should be biased to account for lighter ice baskets under the intermediate deck center support beams identified during the December 1974 basket weighing program. This conclusion was identified and the basis discussed in Supplement No. 5 to the SER.

3. Based on our review of the rate of ice loss and the pattern of loss in the ice condenser, we expect that the ice condenser may not have sufficient ice inventory to allow initiation of operation of the plant for fuel cycle 3. As a result, it appears that additional emphasis should be placed on the development of ice addition techniques and equipment. Analysis of the containment considering the measured and projected distribution of the ice may also be required. It should be noted that the development of a confirmatory long term ice condenser containment code for the staff will be required to confirm the continued safe operation of the plant with maldistribution of ice, without requiring a complete melt-out and refilling of the ice condenser. It appears that these developments will be required before cycle 4 operation and could possibly be required prior to cycle 3 operation.

4. We have determined that the following changes to Technical Specification 3/4.6.5 regarding the minimum ice weight for operation of D. C. Cook Nuclear Plant, Unit No. 1, are required. The changes would:

   a. increase the number of ice baskets to be weighed,

   b. increase the ice basket weighing frequency, and

   c. assure sufficient ice for continued operation on a radial row basis as well as a bay by bay basis.

The licensee has agreed with these conclusions and, by letter dated December 7, 1976, has proposed changes to Technical Specification 3/4.6.5. The proposed changes would (1) increase the minimum number of baskets to be weighed from 96 to 144 and would include baskets from radial rows 1 and 9, (2) increase the inspection frequency from 18 months to 12 months, and (3) demonstrate a sufficient ice inventory in specific groups of baskets on a radial row basis.

We have concluded that the proposed Technical Specifications are consistent with the results of our evaluation. The increase in the minimum number of baskets to be weighed will assure that sufficient data are obtained to continue to evaluate the pattern and extent of preferential sublimation losses in the ice condenser and, to the maximum extent practical, at least one basket is weighed in each bay from a location where maximum ice loss is expected to occur. The increased inspection frequency is consistent with the maximum expected life of fuel cycle 2 and the demonstration of sufficient ice inventory on a radial row basis will properly account for the observed preferential ice loss patterns. We conclude that the proposed ice weighing Technical Specifications are acceptable.

5. Additional weighing of the ice baskets in accordance with Technical Specification 3/4.6.5 will be required before the Technical Specification may be modified to include the measured ice loss rates of the ice condenser during normal plant operation. As indicated in our bases for the Technical Specifications, we believe that data used to calculate a representative ice loss rate for the ice condenser during normal plant operation should be obtained over a period of at least three years.

In a letter dated December 7, 1976, the licensee proposed changes to the ice condenser technical specification. These changes would (1) reduce the surveillance interval for measurement of ice condenser inlet door opening, closing, and frictional torques from 18 months to 6 months, (2) reduce the surveillance interval for verifying the intermediate deck doors were closed and free of frost accumulation from 3 months to 7 days, and (3) correct an inconsistency between Table 3.3-5 and Specification 4.6.5.6.a concerning the response time of the containment air recirculation fan. The first and second proposed changes are consistent with the licensee's initial ice condenser operating experience and with observations made during a review by the Office of Inspection and Enforcement of the adequacy of the ice condenser Technical Specifications.

The following information concerning ice condenser inlet and intermediate deck doors was compiled by the staff:

1. The inlet doors to the ice condenser have been found to have higher than allowable opening torques on approximately 8 occasions when inspected each 90 days during the period November 1974 to present. On each of these occasions, 1 to 2 doors have been found with higher than permissible opening torques because of seal freeze up. The technical specification limit on opening torques is less than 675 inch pounds. The inlet doors were typically found with opening torques in the range of 800 to 1200 inch pounds.

2. The original analysis of the Cook facility indicated that up to 8 of the 48 inlet doors could fail to open during LOCA conditions with acceptable consequences.

3. Westinghouse Electric Corporation has installed several new prototype seals in several inlet doors to confirm their design suitability under actual operating conditions. These seals presently have about 9 months of operating experience. One design appears to show decreased

freeze up tendencies although it also appears to have corner sealing problems which will require further modification. The presumption is that when a suitable seal is developed, it will be installed in the D. C. Cook facility and incorporated into future ice condenser plants.

4. Inspection of the inlet doors has to be performed with the plant at zero power because of the high radiation levels in their vicinity during power operation. Inspection of the intermediate deck doors can be performed during power operation without excessive personnel exposure.

5. Although the Technical Specifications require inspection of the inlet and intermediate deck doors on an 18 month interval, the licensee is presently inspecting the inlet doors each 3 months and the intermediate decks doors each 7 days.

6. The tendency for the inlet doors to exceed the specified torques is reduced by more frequent exercising of the doors.

7. Intermediate deck doors may become inoperable because of ice formation from condensation.

Based on the above considerations, we have concluded that the proposed Technical Specifications changes for surveillance of inlet and intermediate deck doors will improve the operability of the ice condenser and are acceptable. We also have determined that the proposed change to Table 3.3-5 does remove an inconsistency between that table and Specification 4.6.5.6.a and is acceptable.

CONTROL CIRCUIT MODIFICATIONS FOR CERTAIN ELECTRICALLY OPERATED VALVES

By letter dated November 23, 1976, the licensee revised the proposed control circuit design for eight electrically operated valves which was previously submitted on February 27, 1976. The modifications are designed to eliminate the need for operator action to restore power to the valves from outside the control room and precludes a single failure which could cause a loss of ECCS cooling capability.

The modifications to the valve control circuits consist of the addition of (1) a key-lock feature for the control switch, (2) separate control power lockout switch, (3) annunciation of control power not locked out, and (4) valve position indication when valve control power is deenergized. The licensee has developed test procedures which will detect single electrical failures during periodic surveillance testing of the control circuits.

Although only five of the eight valves in question must be repositioned during the switch over from injection to recirculation cooling flow, the licensee decided to modify all the valves to eliminate the need for operator action outside the control room in the event any of the valves had to be operated.

We have reviewed the revised modifications described in the licensee's November 23, 1976 letter. Based on this review, we have concluded that the modified design for remote actuation of the valves from the control room satisfies the requirements of the single failure criteria and is acceptable.

CONTAINMENT AIR RECIRCULATION FAN AND LEAKAGE RATE TECHNICAL SPECIFICATIONS

In two letters dated December 7, 1976, the licensee requested changes to the Technical Specification requirements for containment air recirculation fan response time and containment valve and penetration leak rates. The requested changes would (1) alter the response time of the air recirculation fans in Technical Specification Table 3.3-5 from $\leq$600 seconds to $\leq$660 seconds and would (2) remove statements which indicate that Table 3.6-1 provides a list of all valves and penetrations subject to Type B or C tests, as defined in 10 CFR Part 50, Appendix J.

We have determined that requested change (1) would eliminate an inconsistency between the requirements of Table 3.3-5 and the Surveillance Requirement of specification 4.6.5.6.a which lists the response time for containment air recirculation fans as 10$\pm$1 minutes. This requested change is acceptable.

We have determined that Technical Specification Table 3.6-1 is not intended to list all containment valves and penetrations subject to Type B or C tests. Therefore the requested change to Table 3.6-1 is acceptable.

NUCLEAR SAFETY AND DESIGN REVIEW COMMITTEE (NSDRC) AUDIT RESPONSIBILITIES

By letter dated February 4, 1977, the licensee requested changes to the Technical Specification requirements for the audit responsibilities of the NSDRC. The change would remove the word "all" from the specification which now requires the NSDRC to audit:

1. The conformance of facility operation to "all" provisions contained within the Technical Specifications and applicable license conditions at least once per 12 hours.

2. The results of "all" actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

3. The performance of "all" activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

We have reviewed the proposed Technical Specification change and have determined that it is in conformance with the current NRC requirements and with the Standard Technical Specifications which the NRC currently applies in the licensing of new facilities. Therefore, the requested change is acceptable.

SAFETY RELATED HYDRAULIC SNUBBERS

By letter dated February 9, 1977, the licensee proposed to alter the location description for snubber #11928 in Table 3.7-4 of the Technical Specifications. The change in location description was found to be necessary because this snubber was not installed in the proper location. The licensee has moved the snubber to the proper location and the requested Technical Specification is necessary to identify this new location.

We find the proposed technical specification acceptable because it results from the correction of an installation error for snubber #11928 and, therefore, corrects an error in the Technical Specifications.

TECHNICAL SPECIFICATIONS

The proposed Technical Specification changes for D. C. Cook Unit No. 1 Cycle 2 operation include:

1. ENC 15x15 reload fuel limits.

2. ENC ECCS analysis limits.

3. Ice condenser surveillance requirements.

4. Specifications regarding control circuit modifications for certain electrically operated valves.

5. Correction of errors for:

    - containment air recirculation fan response time

    - containment valve and penetration leak rates

    - audit responsibility of the NSDRC

    - safety related hydraulic snubbers

Some modifications to the proposed Technical Specifications were necessary to meet NRC staff requirements. We find the proposed Technical Specifications, as modified, to be acceptable and consistent with the information submitted by the licensee.

REPORT OF THE ACRS

At its 201st meeting on January 6-8, 1977, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the proposal to refuel D. C. Cook Unit No. 1 with a partial loading of ENC fuel assemblies and to subsequently operate the facility at full rated power. A copy of the Committee's report of its review dated January 14, 1977 is enclosed as Attachment B. The ACRS had previously discussed D. C. Cook Unit No. 1 in its reports dated December 13, 1968, October 17, 1973, and March 11, 1976.

The Committee concluded that full power operation of the proposed reload core was acceptable. However, the Committee requested to be kept informed with regard to fuel pellet-clad interaction and fission gas release rate of the Westinghouse fuel for operation near the end of Cycle 2 and with regard to the control rod fingers which had broken during rod drop timing tests. The staff will keep the Committee informed regarding fuel pellet-clad interaction and fission gas release rate as the results of future analyses become available. The staff's conclusion with respect to the broken control rod fingers has been discussed in a previous section of this evaluation.

The decrease in F_Q from 1.95 to 1.90 as a function of burnup, which is described in the ECCS LOCA Analysis section of this evaluation, was not discussed at the January 6-8, 1977 ACRS meeting because this behavior of F_Q was not known at that time. The burnup dependence of F_Q resulted from the incorporation into the D. C. Cook Unit No. 1 ECCS analysis of a revised flow blockage model. The revised flow blockage model was set forth in the NRC staff's January 5, 1977 safety evaluation of the ENC ECCS model (Reference 15). Although an F_Q value of 1.90 is relatively low and the Committee has expressed concern over low peaking factors in the past, we believe, as stated in Reference 28, that the conservatisms (such as, the decay heat model) in the Appendix K criteria provide a high margin of safety and, since operation within F_Q limits as described in the ECCS LOCA Analysis section fulfills the Appendix K requirements, no additional margin is required. It bears repeating that the higher peaking factor (F_Q = 1.95) will be the limiting condition for about 80% of the fuel cycle.

Additional comments by committee members David Okrent and Milton Plesset are attached to the January 14, 1977 ACRS report. These comments deal with the staff's treatment of ECCS evaluations. These comments are all generic in nature in that they are applicable not only to the D. C. Cook Unit No. 1, but also to other licensed facilities. The staff is considering these opinions in this generic context, and will publish a report to the ACRS on its conclusions thereon in the near future.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

Based on our review of the items identified as (1) and (2) in the introduction to this evaluation, and the considerations discussed in this evaluation, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our review of the remaining items identified in the introduction to this evaluation of this evaluation and the considerations discussed in this evaluation, we have concluded that (1) because the items do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in safety margin, they do not involve a significant hazards consideration, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. We also have concluded, based on the considerations discussed in this evaluation, that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and the issuance of an amendment to the license will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 16, 1977

Attachment A:
Revision 1 to Interim Safety Evaluation Report
on Effects of Fuel Rod Bowing on Thermal
Margin Calculations, dated February 16, 1977

Attachment B:
ACRS Report dated January 14, 1977

REFERENCES

1. Indiana and Michigan Power Company letter (Tillinghast) to USNRC (Rusche) dated July 20, 1976, transmitting XN-76-25 "Donald C. Cook Unit 1 cycle 2 Reload Fuel Licensing Data Submittal."

2. Indiana and Michigan Power Company letter (Tillinghast) to USNRC (Rusche) dated November 23, 1976.

3. "Generic Fuel Design for 15x15 Reload Assemblies for Westinghouse Plants," XN-75-39, September, 1975.

4. Indiana and Michigan Power Company letter (Tillinghast) to USNRC (Rusche) dated November 5, 1976.

5. Indiana and Michigan Power Company letter (Tillinghast) to USNRC (Rusche) dated October 1, 1976.

6. "Plant Transient Analysis for the Donald C. Cook Unit 1 Nuclear Power Plant," XN-76-35, August, 1976.

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ATTACHMENT A

<u>INTERIM SAFETY EVALUATION REPORT</u>

<u>ON EFFECTS OF FUEL ROD BOWING</u>

<u>ON THERMAL MARGIN CALCULATIONS</u>

<u>FOR LIGHT WATER REACTORS</u>

(REVISION I)

February 16, 1977

CONTENTS

1.0 Introduction

Data have recently been presented (Reference 1) to the staff which show that previously developed methods for accounting for the effect of fuel rod bowing on departure from nucleate boiling in a pressurized water reactor (PWR) may not contain adequate thermal margin when unheated rods, such as instrument tubes, are present. Further experimental verification of these data is in progress. However an interim measure is required pending a final decision on the validity of these new data.

The staff has evaluated the impact of these data on the performance of all operating pressurized water reactors. Models for treating the effects of fuel rod bowing on thermal-hydraulic performance have been derived. These models are based on the propensity of the individual fuel designs to bow and on the thermal analysis methods used to predict the coolant conditions for both normal operation and anticipated transients. As a result of these evaluations the staff has concluded that in some cases sufficient thermal margin does not now exist. In these cases, additional thermal margin will be required to assure, with high confidence, that departure from nucleate boiling (DNB) does not occur during anticipated transients. This report discusses how these conclusions were reached and identifies the amount of additional margin required.

The models and the required DNBR reductions which result from these models are meant to be only an interim measure until more data are available. Because the data base is rather sparse, an attempt was made to treat this problem in a conservative way. The required DNBR reductions will be revised as more data become available.

The staff review of the amount and consequences of fuel rod bowing in a boiling water reactor is now underway. At present no conclusions have been reached. When this review reaches a stage where either an interim or final conclusion can be reached, the results of this review will be published in a separate safety evaluation report.

It should be noted that throughout the remainder of this report, all discussion and conclusions apply only to pressurized water reactors.

2.0 DNBR Reduction Due To Rod Bow

2.1 Background

In 1973 Westinghouse Electric presented to the staff the results of experiments in which a 4x4 bundle of electrically heated fuel rods was tested to determine the effect of fuel rod bowing to contact on the thermal margin(DNBR reduction) (Reference 2). The tests were done at conditions representative of PWR coolant conditions. The results of these experiments showed that, for the highest power density at the highest coolant pressure expected in a Westinghouse reactor, the DNBR reduction due to heated rods bowed to contact was approximately 8%.

Fuel bundle coolant mixing and heat transfer computer programs such as COBRA IIIC and THINC-IV were able to accurately predict the results of these experiments. Because the end point could be predicted, i.e., the DNBR reduction at contact, there was confidence that the DNBR reduction due to partial bow, that is, bow to less than contact could also be correctly predicted.

On August 9, 1976 Westinghouse met with the staff to discuss further experiments with the same configuration of fuel bundle (4x4) using electrically heated rods. However, for this set of experiments one of the center 4 fuel rods was replaced by an unheated tube of the same size as a Westinghouse thimble tube. This new test configuration was tested over the same range of power, flow and pressure as the earlier tests. However, with the unheated, larger diameter rod the reduction in DNBR was much larger than in the earlier (1973) tests.

The data consisted of points corresponding to no intentional bowing (that is, a certain amount of bowing due to tolerances cannot be prevented) and to contact. No data were taken at partial clearance reductions between rods.

The staff attempted to calculate the Westinghouse results with the COBRA IIIC computer code but could not obtain agreement with the new data. Westinghouse was also unable to obtain agreement between their experimental results and the THINCIV computer code.

On August 19, 1976 CE presented results of similar experiments to the staff. These tests were performed using a 21 rod bundle of electrically heated rods and an unheated guide tube. Results were presented for not only the case of full contact, but also the case of partial bowing.

Both sets of data (Westinghouse and CE) showed similar effects due to variations in coolant conditions. For both cases, the DNBR reduction became greater as the coolant pressure and the rod power increased.

Because both sets of data showed that plant thermal margins might be less than those intended, the staff derived an interim model to conservatively predict the DNBR reduction. Since the data with unheated rods could not be predicted by existing analytical methods, empirical models were derived. These models give the reduction in DNBR as a function of the clearance reduction between adjacent fuel rods. Two such models were derived, one based on the Westinghouse data and one based on the CE data.

2.2 Model Based on Westinghouse Data

As stated in Section 2.1, data were presented by Westinghouse for the DNBR reduction at full contact and with no bow. No data at partial gap closure were presented. Westinghouse proposed, and the staff accepted, a straight line interpolation between these two points as shown in Figure 2.1.

This approach is conservative if the DNBR reduction does not increase more rapidly than the straight line reduction shown in Figure 2.1. Although the data for DNBR reduction due to rod bowing in the presence of an unheated fuel rod cannot be predicted by existing analytical methods, one would nevertheless expect that the actual behavior would more nearly follow the curved line also shown in Figure 2.1. According to this curved line, the DNBR would be reduced gradually for small amounts of bow. As the fuel rods (or fuel rod and unheated rod) become close enough so that there is an interaction, the DNBR would decrease more rapidly. No physical mechanism has been postulated which would lead to sudden large decreases in the DNBR for small or moderate gap closures. Thus, the straight line approximation is believed to be an overestimate of the expected behavior.

Experience with critical heat flux tests also supports the assumption of a small reduction in DNBR for small amounts of fuel rod bowing. Experimental measurements of critical heat flux done on test assemblies always have some amount of rod bowing. This may be due simply to fabrication tolerances or to electromagnetic attraction forces set up between electrically resistance heated rods which simulate fuel rods.

It should be noted that this behavior (little or no reduction in DNBR for small amount of bowing) is shown by Combustion Engineering data which became available to the staff after the Westinghouse model was derived. The Combustion Engineering data is discussed in Section 2.3 and the model derived from this data is shown in Figure 2.2.

All manufacturers of reactor cores, including Westinghouse, include a factor in their initial core design to account for the reduction in DNBR that may result from pitch reduction from fabrication tolerances and initial rod bow. The amount of this pitch reduction factor varies with the fuel design and the analysis methods which are used. For any particular core this factor is not varied as a function of burnup.

In developing the interim rod bow penalties described in this report, it became apparent that the penalty should be a function of burnup since the magnitude of rod bow is a function of burnup. However, to maintain existing thermal margins early in core life when only a small amount of fuel rod bow is anticipated, the initial pitch reduction factor was included until such time as the rod bow DNBR reduction became greater. This is represented as the straight horizontal line on Figure 2.1.

2.3 Combustion Engineering Model

Combustion Engineering performed experiments to determine the effect of rod bowing on DNBR which included some cases in which the effect of partial bowing as well as bowing to contact was determined. Again, a straight line interpolation is used. However, the point of zero DNBR reduction is not at zero clearance reduction but rather, at an intermediate value of clearance reduction. This is shown schematically in Figure 2.2. The horizontal straight line, representing the initial pitch reduction factor is included as explained previously in Section 2.2

2.4 Models for Babcock and Wilcox and Exxon

On August 17, 1975 representatives of Babcock and Wilcox met with the staff to discuss this problem. Babcock and Wilcox did not present any data on the effects of rod bowing on DNBR. They had previously presented data to the staff on the amount of bowing to be expected in Babcock and Wilcox 15x15 fuel assemblies. Because Babcock and Wilcox had no data on the effect of rod bow on DNBR, the staff applied the Westinghouse model to calculate the effect of rod bowing on DNBR for Babcock and Wilcox fuel. This is acceptable since the conditions of operation are nearly the same in pressurized water reactors from both vendors and the fuel bundle designs are similar.

The amount of fuel rod bowing as a function of burnup was calculated using the Babcock and Wilcox 15x15 fuel bundle data.

Representatives of the Exxon Nuclear Corporation discussed the effects of fuel rod bowing in the presence of an unheated rod on DNBR with the staff on August 19, 1976. Exxon has not performed DNB tests with bowed rods and thus has no data pertinent to this problem. The first cycle of Exxon fuel has just been removed from H. B. Robinson and the results of measurements on the magnitude of rod bowing have not yet been presented to the staff. The effects of fuel rod bowing for Exxon fuel were evaluated on a plant by plant basis as discussed in Section 4.0

2.5 Application of the Rod Bow/DNBR Model

Using these empirical models, the staff derived DNBR reductions to be applied to both operating reactors and plants in the Operating License review stage. The procedure in applying these empirical models is as follows:

Step 1: Predict the clearance reduction due to rod bow as a function of burnup. An expression of the form

$$\frac{\Delta C}{C_o} = a+b\sqrt{BU}$$

is used where

$\frac{\Delta C}{C_o}$ = fractional clearance reduction due to rod bowing

a,b = empirical constants obtained for a given fuel design

BU = burnup (region average or bundle average, depending on the fuel designer).

Westinghouse showed in Reference 6 that an equation of the above form fit the rod bow data from 26 fuel regions. The constant a represents the initial bow of the fuel rods due to fabrication tolerance. The staff has approved the above equation (Reference 8).

Also included in the constants a and b is a factor of 1.2 to convert from the cold conditions at which the measurements were made to the hot operating conditions and a factor of 1.645 which, when multiplied by the standard deviation, gives an amount of bow greater than that expected from 95% of the fuel rods with a 95% confidence.

Step 2: Apply the previously discussed empirical models of DNBR reduction as a function of clearance reduction using the value of $\Delta C/c_o$ calculated from step 1.

Step 3: The staff has permitted the reduction in DNBR calculated in step 2 to be offset by certain available thermal margins. These may be either generic to a given fuel design or plant dependent.

An example of a generic thermal margin which would be used to offset the DNBR reduction due to rod bow is the fact that the DNBR limit of 1.30 is usually greater than the value of DNBR above which 95% of the data lie with a 95% confidence. The difference between 1.30 and this number may be used to offset the DNBR reduction.

For Westinghouse 15x15 fuel, the value of DNBR which is greater than 95% of the data at a 95% confidence level is 1.24 (Reference 1). For Westinghouse 17x17 fuel this number is 1.28 (Reference 1). A review of the data used to derive these numbers shows that the use of three significant figures is justified.

An example of a plant specific thermal margin would be core flow greater than the value given in the plant Technical Specifications.

A discussion of the application of this method to Construction Permit and Operating License reviews is given in Section 3.0. A discussion of the application and the results of this method to operating reactors is given in Section 4.0. The application to reactors using Exxon fuel is also discussed in Section 4.0.

3.0 Application to Plant in Construction Permit And Operating License Review Stage

3.1 CP Applications

No interim rod bow DNB penalties should be applied to CP applications. The rod bow data upon which the interim limits have been based should be considered preliminary. There is sufficient time available to review the data and assess a penalty, if any, prior to the OL stage. We will advise each CP applicant of the nature of interim penalties being applied to OL reviews and operating reactors.

As stated above, the data used to evaluate the effects of rod bow on DNBR are preliminary. They are also incomplete. In order to assess the conservatism of the straight line approximation and to obtain data on designs for which no data is now available we will require the applicant to (1) fully define the gap closure rate for prototypical bundles and (2) determine by an appropriate experiment the DNB effect that bounds the gap closure from part (1). Such requirements will be part of our CP review effort.

3.2 OL Applications

Plants which are in the operating license review stage should consider a rod bow penalty. This penalty should be as described in Section 2.2 for Westinghouse or Section 2.3 for Combustion Engineering. Babcock and Wilcox plants should use the rod bow vs. burnup curve appropriate to their fuel and the Westinghouse curve of DNBR reduction as a function of rod bow.

All applicants may propose appropriate thermal margins (as discussed in Section 2.4) to help offset the calculated DNBR reduction.

4.0 Application To Operating Reactors

This section divides the operating plants into distinct categories and lists them according to the fuel and/or reactor manufacturer. Operating plants which cannot be so categorized (such as plants with fuel supplied by more than one vendor) are placed in a separate category. The plants assigned to each category are listed in the appropriate subsection.

The conclusions reached in this section are in some cases dependent on conditions or analysis which are valid only for the present fuel cycle. Hence, the $F_{\Delta H}$ or DNBR reductions which are given (or the fact that no such reduction is concluded to be required) is valid only for the present operating cycle.

4.1 Westinghouse LOPAR Fuel

The designation LOPAR stands for low parasitic and refers to the fact that the guide tubes in the fuel bundle are made of Zircaloy. Table 4.1 gives a list of the operating plants which fall into this classification.

TABLE 4.1: PLANTS WHICH CURRENTLY USE THE WESTINGHOUSE LOPAR FUEL ASSEMBLY

15 x 15	17 x 17
Zion 1 Cycle 2	Trojan Cycle 1
Zion 2 Cycle 1	Beaver Valley 1 Cycle 1
Indian Point 3 Cycle 1	
Turkey Point 3 Cycle 4	
Turkey Point 4 Cycle 3	
Prairie Island 2 Cycle 2	
Prairie Island 1 Cycle 2	

TABLE 4.1 (cont.)

15 x 15

Surry 1 Cycle 4

Surry 2 Cycle 3

Kewaunee Cycle 2

Point Beach 1 Cycle 5

Point Beach 2 Cycle 3

The reduction in DNBR due to fuel rod bowing is assumed to vary linearly with the reduction in clearance between the fuel rods (or fuel rod and thimble rod) according to the model discussed in Section 2.2.

The maximum value of DNBR reduction (at contact), obtained from the experimental data was used to calculate the DNBR reduction vs. bow for the 15x15 LOPAR fuel. This DNBR contact reduction was adjusted for the lower heat flux in the 17x17 LOPAR fuel.

The clearance reduction is conservatively assumed to be given by the following equation for the 15x15 (and 14x14) fuel,

$$\frac{\Delta C}{Co} = a + b\sqrt{Bu}$$

where $\frac{\Delta C}{Co}$ is the reduction in clearance

Bu is the region average burnup

and a,b are empirical constants fitted to Westinghouse 15x15 rod bow data

For the 17x17 LOPAR fuel, the clearance reduction was calculated from the equation:

$$\Delta C/Co = \left(\frac{\Delta C}{Co}\right)_{15x15} \times \left(\frac{L}{I}\right)_{15x15} \times \left(\frac{I}{L}\right)_{17x17}$$

where L = the distance between grids

I = moment of inertia of fuel rod

On December 2, 1976, Westinghouse informally showed the staff new data pertaining to the magnitude of rod bow as a function of region average burnup in 17x17 fuel assemblies. This data show that the above correction is probably conservative and that the magnitude of fuel rod bowing in 17x17 fuel rods can better be represented by an empirical function. This review is now underway.

The calculated DNBR reduction is partially offset by existing thermal margins in the core design. For the Westinghouse LOPAR fuel design some or all of the following items were used in calculating the thermal margin for the operating plants:

- design pitch reduction
- conservatively chosen TDC used in design*
- Critical heat flux correlation statistics (assumed in thermal analysis safety calculations) are more conservative than required.
- Densification power spike factor included although no longer required (Reference 4)

After taking these factors into account, the reductions in $F_{\Delta H}$ shown in Table 4.2 were found necessary. All operating plants listed in Table 4.1 will be required to incorporate these reductions in $F_{\Delta H}$ into their present operating limits.

*TDC (thermal diffusion coefficient) is a measure of the amount of mixing between adjacent subchannels.

TABLE 4.2: $F_{\Delta H}$ REDUCTION FOR WESTINGHOUSE LOPAR FUEL

CYCLE	REDUCTION IN $F_{\Delta H}$ (%)		
	15x15	17x17	ZION 1&2
1st Cycle (0-15 Gwd*/MTU)	0-2 ramp	0-9.5	0-6 ramp
2nd Cycle (15-24 Gwd*/MTU)	4	12	8
3rd Cycle (24-33 Gwd*/MTU)	6	12	10

These reductions in $F_{\Delta H}$ may be treated on a region by region basis. If the licensee chooses, credit may be taken for the margin between the actual reactor coolant flow rate and the flow rate used in safety calculations. Credit may also be taken for a difference between the actual core coolant inlet temperature and that assumed in safety analyses. In taking credit for coolant flow or inlet temperature margin, the associated uncertainties in these quantities must be taken into account.

4.2 Westinghouse HIPAR and Stainless Steel Clad Fuel

The designation HIPAR stands for high parasitic and refers to the fact that the guide tubes in the fuel bundle are made of stainless steel. These two fuel types, HIPAR and Stainless Steel clad, are grouped together because the amount of bowing expected (and observed) is significantly less than that in the observed Westinghouse LOPAR fuel. The plants which fall under this classification are listed in Table 4.3.

$$* \frac{\text{Gwd}}{\text{MTU}} = 1000 \frac{\text{Mwd}}{\text{MTU}}$$

TABLE 4.3: HIPAR AND STAINLESS STEEL PLANTS

Ginna	Indian Point 2
San Onofre	Connecticut Yankee

The model for the reduction in DNBR due to fuel rod bowing is assumed to be identical to that used for the LOPAR fuel. This is acceptable since cladding material should have no effect on CHF (critical heat flux) and the same DNB correlation applies to both HIPAR and LOPAR grids.

For reactors in this category, the peak reduction in DNBR (corresponding to 100% closure) was adjusted to correspond to the peak overpower heat flux of that particular reactor.

The amount of rod bowing for the plants listed in Table 4.3 which use HIPAR and stainless steel fuel, was calculated by means of an adjustment to the 15x15 LOPAR formula. This adjustment took the form of the ratio

$$\frac{\text{amount of bow for assembly type}}{\text{amount of bow for LOPAR fuel}} = \frac{(L/IE)\ \text{assy type}}{(L/IE)\ \text{LOPAR}}$$

where L is the span length between grids

I is the moment of inertia of the fuel rod

E is the modulus of elasticity of the fuel rod cladding

Ginna Cycle 6

The Ginna plant is fueled with 121 fuel assemblies. Two of these are Exxon assemblies, and two are B&W assemblies. The remainder are Westinghouse HIPAR fuel assemblies. The experimental value of DNBR reduction was adjusted for heat flux and pressure from peak experimental to actual plant conditions. Ginna took credit for the thermal margins due to pitch reduction, design vs. analysis values of TDC and fuel densification power spike. These thermal margins offset the calculated DNBR reduction so that no reduction in $F_{\Delta H}$ is required.

San Onofre Cycle 5

San Onofre is fueled with 157 bundles of 15x15 stainless steel clad fuel. An $F_{\Delta H}$ of 1.55 was used in thermal design and in the Technical Specifications. To offset the reduction in $F_{\Delta H}$ due to rod bowing San Onofre has proposed taking credit for margin available from the assumed worst case axial power distribution used in the thermal analysis for San Onofre and that which would be possible during operation. This proposal is now being reviewed by the staff.

Indian Point 2 Cycle 2

Indian Point 2 is fueled with HIPAR fuel bundles. The experimental value of DNBR reduction was adjusted for heat flux and pressure to actual plant conditions. Indian Point Unit 2 had thermal margin to offset this DNBR reduction in pitch reduction, design vs. analysis values of TDC, fuel densification power spike and a value of $F_{\Delta H}$ of 1.65 used in the design (vs. 1.55 in the Tech Spec). Therefore, no reduction of $F_{\Delta H}$ is required for Indian Point Unit 2.

Connecticut Yankee Cycle 7

Connecticut Yankee is fueled with 157 stainless steel clad fuel assemblies. The DNBR reduction at contact was assumed to be that used for the Westinghouse LOPAR 15x15 fuel. No adjustment was made for heat flux. The value of pressure was adjusted to the overpressur trip set point value of 2300 psi. Full closure will not occur in stainless steel fuel out to the design burnup.

Connecticut Yankee has sufficient thermal margin in variable overpressure and overpower trip set points to accommodate the calculated DNBR reduction. Therefore no penalty is required.

4.3 Babcock and Wilcox 15x15

The reactors listed in Table 4.4 are fueled with B&W fuel.

TABLE 4.4: REACTOR USING B&W FUEL

Oconee 1 Cycle 3

Oconee 2 Cycle 2

Oconee 3 Cycle 1

Rancho Seco

Three Mile Island 1 Cycle 2

Arkansas 1 Cycle 1

Babcock and Wilcox met with the staff on September 8, 1975 and presented data on the amount of rod bow in B&W fuel. The staff derived a model for B&W 15x15 fuel based on this data. This model has the form:

$$\frac{\Delta C}{Co} = a + b\sqrt{Bu}$$

where $\frac{\Delta C}{Co}$ is the fractional amount of closure

Bu is the bundle average burnup

and a,b are empirical constants fitted to B&W data

The reduction in DNBR due to fuel rod bowing is assumed to vary linearly with the reduction in clearance between the fuel rods (or fuel rod and thimble rod) but can never be lower than that due to the pitch reduction factor used in thermal analysis, as explained in Section 2.2.

Babcock and Wilcox claimed and the staff approved credit for the following thermal margins:

- Flow Area (Pitch) reduction
- Available Vent Valve credit
- Densification Power Spike removal
- Excess Flow over that used in safety analyses
- Higher than licensed power used for plant safety analyses

Based on this review and the thermal margins presented by B&W to offset the new Westinghouse data, Rancho Seco is the only plant for which a reduction in DNBR is required. Table 5 gives the values for the reduction of DNBR required at this time.

TABLE 5: DNBR REDUCTIONS FOR B&W PLANTS

Burnup	DNBR Reduction
	Rancho Seco
Cycle 1 (0-15 $\frac{\text{Gwd}}{\text{MTU}}$)	0
Cycle 2 (15-24 $\frac{\text{Gwd}}{\text{MTU}}$)	1.6%
Cycle 3 (24-33 $\frac{\text{Gwd}}{\text{MTU}}$)	3%

Plans must be submitted to the staff to establish how these reductions in DNBR will be accommodated.

4.4 Combustion Engineering 14x14

Combustion Engineering has presented data to the staff on the amount of rod bowing as a function of burnup. (Reference 5) The staff used this data to derive the following model for CE 14x14 fuel (Reference 7)

$$\frac{\Delta C}{Co} = a + b \sqrt{Bu},$$

$\Delta C/Co$ = fraction of closure for CE fuel

Bu is the bundle average burnup

and a,b are empirical constants fitted to CE data

CE was given credit for thermal margin due to a multiplier of 1.065 on the hot channel enthalpy rise used to account for pitch reduction due to manufacturing tolerances. Table 4.6 presents the required reduction in DNBR using the model described above, after accounting for this thermal margin. Table 4.7 is a list of the reactors to which it applies.

A licensee planning to operate at a burnup greater than 24000 Mwd/MTU should present to the staff an acceptable method of accommodating the thermal margin reduction shown in Table 4.6. This may be done as part of the reload submittal if this burnup will not be obtained during the current cycle.

TABLE 4.6: EFFECT OF ROD BOWING ON DNBR IN REACTORS WITH COMBUSTION ENGINEERING 14x14 FUEL

BURNUP	REDUCTION IN DNBR
Cycle 1 (0-15 $\frac{\text{Gwd}}{\text{MTU}}$)	0
Cycle 2 (15-24 $\frac{\text{Gwd}}{\text{MTU}}$)	0
Cycle 3 (24-33 $\frac{\text{Gwd}}{\text{MTU}}$)	3%

TABLE 4.7: PLANTS FUELED BY CE FUEL TO WHICH VALUES OF TABLE 4.6 APPLY

St. Lucie 1	Cycle 1
Ft. Calhoun	Cycle 3
Millstone 2	Cycle 2
Maine Yankee	Cycle 2
Calvert Cliffs 1	Cycle 1

4.5 Plants Fueled Partially With Exxon Fuel

Palisades, H. B. Robinson, Yankee Rowe and D. C. Cook are partially fueled with Exxon fuel. A discussion of these reactors follows:

Palisades Cycle 2

The Palisades reactor for Cycle 2 is fueled with 136 Exxon fuel assemblies and 68 Combustion Engineering fuel assemblies.

The Combustion Engineering fuel was treated according to the Combustion Engineering model for both extent of rod bow as a function of burnup and DNBR reduction due to clearance reduction.

The Exxon fuel was assumed to bow to the same extent as the Combustion Engineering fuel. This assumption is acceptable since the Exxon fuel has a thicker cladding and other design features which should render the amount of bowing no greater than in the Combustion Engineering fuel.

The DNBR reduction was assumed to be linear with clearance reduction according to the Westinghouse type curve of Figure 2.1. The DNBR reduction at contact was based on the Westinghouse experimental data adjusted for the peak rod average heat flux in Palisades and for the coolant pressure in Palisades.

The variation of the DNBR reduction with coolant pressure is given in Reference 1. The DNBR reduction decreases as the coolant pressure decreases. The overpressure trip set point in Palisades is set at 1950 psi. At this pressure, according to the data presented in Reference 1, the penalty is greatly reduced compared to the penalty at high pressures.

The limiting anticipated transient in the Palisades reactor results in a DNBR of 1.36. The thermal margin between this value and the DNBR limit of 1.3 results in adequate thermal margin to offset the rod bow penalty.

Yankee Rowe Cycle 12

Yankee Rowe is fueled with 40 Exxon fuel assemblies and 36 Gulf United Nuclear Corporation fuel assemblies. The fuel assemblies consist of 16x16 Zircaloy clad fuel rods.

The reduction in DNBR due to fuel rod bowing was assumed to vary linearly with the reduction in clearance between fuel rods. The peak experimental conditions used in the Westinghouse test were used to fix the penalty at full closure. The calculated reduction in DNBR is still less than that which would produce a DNBR less than 1.3 for the most limiting anticipated transient (two pump out of four pump loss-of-flow). Thus, no penalty is required.

H. B. Robinson Cycle 5

H. B. Robinson is fueled with 105 Westinghouse fuel assemblies and 52 Exxon Nuclear Corporation fuel assemblies. The Westinghouse 15x15 DNBR penalty model was applied to the Westinghouse fuel with a correction for the actual heat flux rather than the peak experimental values. The Exxon fuel was considered to bow to the same extent as the Westinghouse 15x15 fuel so that the Westinghouse bow vs. burnup equation was also applied to the Exxon fuel. This assumption is conservative since the Exxon fuel has a thicker cladding and other design features which should render the amount of bowing no greater than in the Westinghouse fuel.

The DNBR reduction calculated by this method was offset by the fact that the worst anticipated transient for H. B. Robinson results in a DNBR of 1.68.

<u>D. C. Cook</u> Cycle 2

D. C. Cook contains 128 Westinghouse fuel assemblies and 65 Exxon fuel assemblies. The limiting transient for D. C. Cook is the Loss of Flow (4 pump coastdown) which has a minimum DNBR of 2.01. This value of DNBR is sufficiently high to accommodate the rod bow penalty for Cycle 2 without reducing the DNBR below the safety limit value of 1.3.

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4. Letter to R. Salvatori, Manager, Nuclear Safety Department, Westinghouse Electric Corporation from D. Vassallo, Chief, Light Water Reactors Project Branch 1-1, Directorate of Licensing, December 4, 1974.
5. Letter to V. Stello, Director, Division of Operating Reactors, USNRC, from P. L. McGill, Combustion Engineering Company, December 15, 1975.
6. Reavis, J. R., et. al., "Fuel Rod Bowing" WCAP 8691 (Proprietary) Westinghouse Electric Corporation, December, 1975.
7. Letter to Mr. Ed Sherer, Combustion Engineering from D. F. Ross, Assistant Director, Reactor Safety, May 14, 1976.
8. Interim Safety Evaluation Report on Westinghouse Fuel Rod Bowing Division of System Safety, USNRC, April, 1976.

FIGURE 2.1

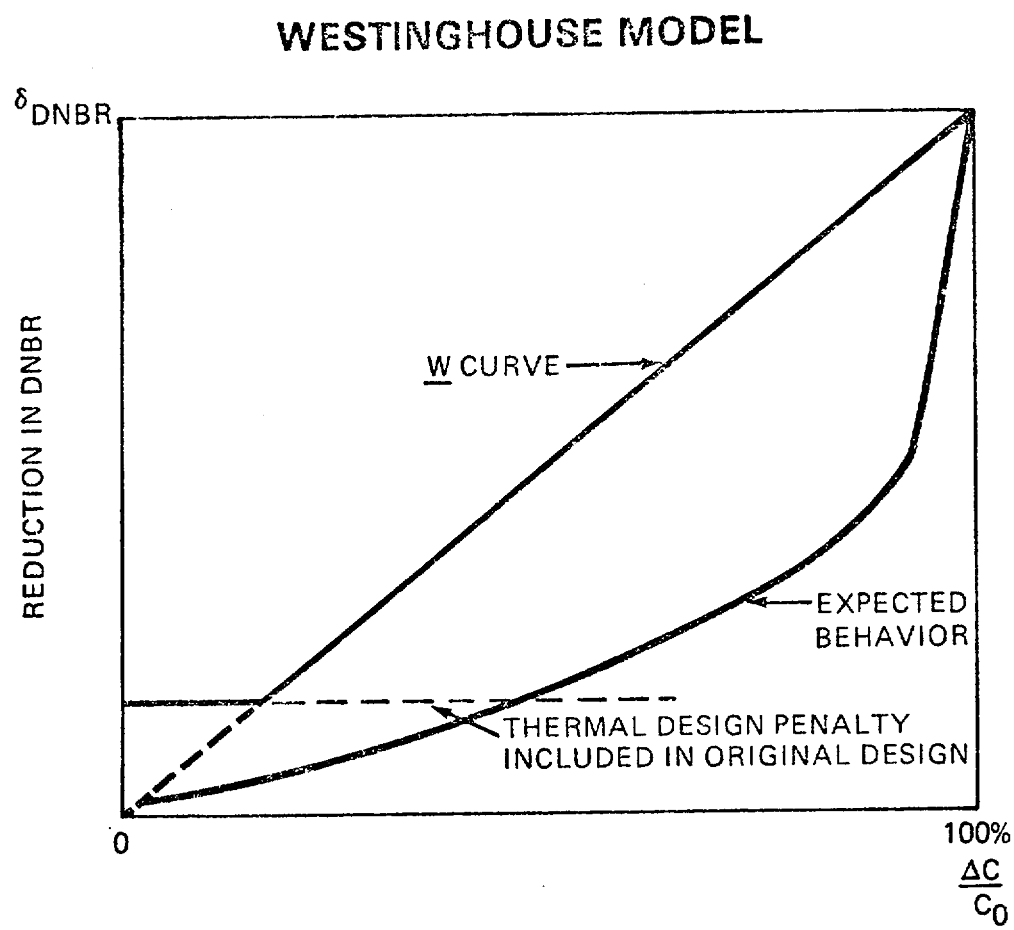
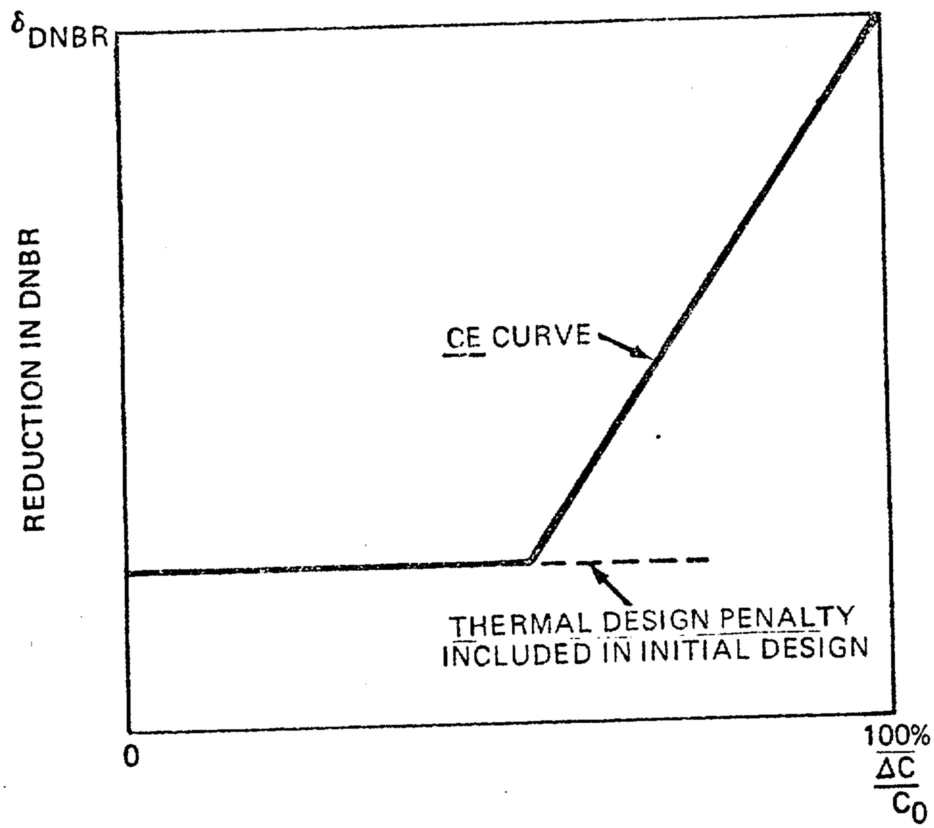

WESTINGHOUSE MODEL
δ_{DNBR}
REDUCTION IN DNBR
W CURVE
EXPECTED BEHAVIOR
THERMAL DESIGN PENALTY INCLUDED IN ORIGINAL DESIGN
0
100%
$\frac{\Delta C}{C_0}$

FIGURE 2.2

COMBUSTION ENGINEERING MODEL



ATTACHMENT B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 14, 1977

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

Dear Mr. Rowden:

During its 201st meeting, January 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the proposal to replace, during the first refueling of the Donald C. Cook Nuclear Plant Unit No. 1, 65 of the original Westinghouse Electric Corporation fuel assemblies with Exxon Nuclear Company (ENC) fuel assemblies and to operate the resulting core to produce rated reactor power of 3250 MWt. The Committee has previously discussed this plant in its reports of December 13, 1968, October 17, 1973, and March 11, 1976. A Subcommittee meeting to consider the current proposal was held in Washington, D. C., on December 22, 1976. During its review, the Committee had the benefit of discussions with representatives of Indiana and Michigan Power Company, American Electric Power Service Corporation, ENC, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The NRC Staff has concluded that the design of the ENC fuel assemblies proposed for Donald C. Cook Nuclear Plant Unit No. 1 Cycle 2 is similar to that supplied by ENC for other pressurized water reactors (PWRs). The NRC Staff has indicated that its review of ENC fuel design analytical methods is not yet complete but that the review has progressed sufficiently to indicate that the methods are adequate for application to Donald C. Cook Nuclear Plant Unit No. 1 Core 2. Approximately 1000 fuel bundles manufactured by ENC are in PWRs and in boiling water reactors with burnups ranging from first cycle to 25,000 megawatt-days per metric ton of uranium. Performance of these assemblies has been good.

Primarily because of the low back pressure produced by the ice-condenser type containment following a loss-of-coolant accident, the peaking factor required to satisfy the emergency core cooling system (ECCS) Acceptance Criteria of 10 CFR 50.46 is unusually low. The ENC analysis satisfied the ECCS Acceptance Criteria of 10 CFR 50.46 with an assumed peaking factor of 1.95 at rated power. The Licensee proposes a peaking factor

Technical Specification limit of 1.95 at rated power for Cycle 2. The Licensee proposes continued use of the axial power distribution monitoring system (APDMS) for determining conformance. Experience with APDMS during Cycle 1 operation at Donald C. Cook Nuclear Plant Unit No. 1 and in other reactors indicates this system can provide an appropriate measurement of the core power distribution.

Although sufficient information and analyses exist to predict the performance of the Westinghouse fuel at the beginning of Cycle 2, further analyses may be appropriate with regard to both fuel pellet-clad interaction and fission gas release rate, with operation near the end of the cycle. The Committee wishes to be kept informed.

During Cycle 1 operation, one or two fingers broke off a control rod during rod drop timing tests. The Licensee and Westinghouse Electric Corporation have concluded that the observed failure is not indicative of generic failures and will not adversely affect reactor control rod scram times. The NRC Staff is requiring further examination and analyses by the Licensee. The ACRS wishes to be kept informed.

The ACRS believes that, subject to the foregoing and to matters discussed in its report of March 11, 1976, the Donald C. Cook Nuclear Plant Unit No. 1 can be operated with the proposed reload core up to the design power of 3250 MWt, under the proposed operating and monitoring conditions, without undue risk to the health and safety of the public.

Sincerely yours,

M. Bender

M. Bender
Chairman

Additional Comments by Members David Okrent and Milton Plesset

In connection with the March 11, 1976 report on Donald C. Cook Nuclear Plant Unit No. 1, we made additional comments which included the following:

"First, while there may be merit in the proposed changes in the Westinghouse evaluation model, we believe further examination is warranted of several factors, including the scaling of experiments, the scatter in data, and the possible influence of super-plasticity on clad behavior during postulated loss-of-coolant accidents. Our reluctance to endorse these changes is also due, in large part, to signs of a continued process of cutting into the conservatisms built into the original evaluation models, without a concomitant build-up in our basic understanding or predictive ability for the overall LOCA-ECCS process. In this situation there are limits beyond which the use of best estimate heat transfer coefficients, etc., is no longer appropriate.

"Second, even with application of the revised Westinghouse evaluation model which has been judged acceptable by the NRC Staff, Donald C. Cook Nuclear Plant Unit No. 1 requires a LOCA - limited maximum peaking factor (F_Q) of 1.98 (plus the margin for bowing) at rated power. While this is somewhat higher than the F_Q which can be expected at steady operation for the rest of the first fuel cycle for Donald C. Cook Nuclear Plant Unit No. 1, it still represents a very large reduction in the margin that has been available for most plants between LOCA - limited F_Q and that value which would be present most of the time. This margin has been eroded until it is a small fraction of its earlier values. Furthermore, if we accept this low F_Q value for Donald C. Cook Nuclear Plant Unit No. 1, a precedent will be set by means of which all PWR's will be able to reduce what was a substantial safety margin only a few years ago. This previously available substantial safety margin could cover many of the existing uncertainties in the analysis of LOCA-ECCS. The uncertainty aspect is highlighted by the less than perfect record obtained by the experts in their pre-prediction of various separate effects experiments, by the recognized difficulties in a calculation from first principles, by the current unavailability of experiments to test all relevant effects, and by the lack of a meaningful test of Westinghouse predictive capability with experiment.

"Third, the ACRS has in the past been reluctant to accept proposed operation of reactors with F_Q's less than 2.2. In part, such caution arose from the knowledge that, with a more flattened power distribution, a much larger fraction of the fuel elements would be at or near peak temperatures, given a LOCA, and therefore potentially vulnerable to an "anomaly" in ECCS function (such as some three-dimensional flow effect or excessive steam generator leakage)."

We find that these comments apply equally to the proposed operation with Exxon Nuclear Company (ENC) fuel. We believe that the proposed new ENC ECCS evaluation model is subject to considerable uncertainty, particularly with regard to flow blockage effects, the choice of FLECHT heat transfer coefficents, and steam cooling.

More importantly, as we suggested on March 11, 1976, the NRC Staff has continued to follow a legalistic approach in its interpretation of 10 CFR 50, Appendix K, accepting so-called best-estimate parameters and models in areas where conservatism is not explicitly required. Since March 1976, a significant number of operating PWRs have been granted authority to operate with peaking factors even less than 1.98; for example, Surry Units 1 and 2 were granted approved peaking factors of 1.80 and 1.82, respectively, on August 27, 1976.

In view of the current state of knowledge, we do not believe that the path currently being followed by the NRC Staff is prudent, and we recommend that the Nuclear Regulatory Commission reexamine 10 CFR 50, Appendix K, including its actual implementation in evaluation models.

For Donald C. Cook Nuclear Plant Unit No. 1, we still believe that operation with the present design of fuel assembies and ECCS, should be limited to about 92% of rated power.

References:

1. Revision 1 to Nuclear Reactor Regulation (NRR) Safety Evaluation Report on the Exxon Nuclear Company (ENC) WREM-Based Generic PWR-ECCS Evaluation Model Update ENC-WREM-II, dated January 5, 1977
2. Letter, Indiana and Michigan Power Company (I and M) to NRR, dated December 17, 1976, concerning reactor vessel overpressurization events
3. NRR Report to the ACRS on Donald C. Cook Nuclear Plant Unit No. 1, dated December 14, 1976
4. Letter, I and M to NRR, dated December 13, 1976, concerning proposed changes to Technical Specifications on power distribution limits and surveillance requirements
5. Letter, ENC to NRR, dated November 30, 1976, forwarding information concerning the ECCS analyses
6. Letter, I and M to NRR, dated November 23, 1976, concerning modifications being made to valve control circuits and procedures
7. Letter, I and M to NRR, dated November 23, 1976, forwarding responses to NRR questions concerning a permit to operate at full power during Cycle 2
8. Letter, ENC to NRR, dated November 19, 1976, forwarding XN-76-35 Supplement 1, "Assumptions Used in the Plant Transient Analysis for the Donald C. Cook Unit 1 Nuclear Plant"
9. Letter, I and M to NRR, dated November 17, 1976, forwarding XN-76-35, "Donald C. Cook Unit 1 LOCA Analyses Using the ENC WREM-Based PWR ECCS Evaluation Model (ENC-WREM-II)"
10. Letter, I and M to NRR, dated November 17, 1976, forwarding the results of analyses of the effect of degraded grid voltage on the operability of safety-related equipment
11. Letter, I and M to NRR, dated November 11, 1976, concerning the loose-parts monitoring system
12. Letter, I and M to NRR, dated November 5, 1976, forwarding answers to NRR questions on the reload license application
13. Letter, American Electric Power Service Corporation to the Office of Inspection and Enforcement, dated October 29, 1976, forwarding a supplement to the Startup Test Report
14. Letter, I and M to NRR, dated October 27, 1976, concerning fire protection considerations

References Cont'd

15. Letter, I and M to NRR, dated October 19, 1976, concerning susceptibility to reactor vessel overpressurization events
16. Letter, I and M to NRR, dated October 1, 1976, forwarding answers to NRR questions on ENC reports XN-76-25 and XN-75-39
17. Letter, I and M to NRR, dated October 1, 1976, forwarding the report, "Long Term Evaluation of the Ice Condenser System Results of the July 1976 and September 1976 Ice Weighing Programs"
18. Letter, I and M to NRR, dated August 27, 1976, concerning the evaluation of the adequacy of the reactor pressure vessel supports
19. Letter, I and M to NRR, dated August 26, 1976, forwarding XN-76-36, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model (ENC-WREM-II)"
20. Letter, ENC to NRR, dated August 20, 1976, forwarding XN-76-35, "Plant Transient Analysis for the Donald C. Cook Unit 1 Nuclear Power Plant"
21. Letter, I and M to NRR, dated July 30, 1976, forwarding the report, "Long Term Evaluation of the Ice Condenser System - Results of the January 1976 and April 1976 Ice Weighing Programs"
22. Letter, I and M to NRR, dated July 20, 1976, concerning request to operate at full power during Cycle 2
23. Letter, ENC to NRR, dated July 19, 1976, forwarding XN-76-25, "Donald C. Cook Unit 1 Cycle 2 Reload Fuel Licensing Data Submittal"