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Docket File
NRC PDR
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H. Denton
C. Parrish
D. Wigginton
OELD
SECY
L. J. Harmon
J. Taylor
T. Barnhart (4)
W. Jones
D. Brinkman
ACRS (10)
OPA, C. Miles
R. Diggs
CPB
ASB
AEB
D. Eisenhut

September 20, 1983

Docket No. 50-315

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 11, 1983, as supplemented by letter dated July 25, 1983.

This amendment approves the Cycle 8 reload and changes the related Technical Specifications. The Cycle 8 reload includes Westinghouse fuel of the 15 x 15 Optimized Fuel Assembly design with fuel enrichments up to 4.0 weight percent U-235, fuel burnups to 39,000 MWD/MTU, and Wet Annular Burnup Assembly (WABA) burnable poison absorbers.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/S/

David L. Wigginton, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 74 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

8310110333 830920
PDR ADOCK 05000315
P PDR

SEE PREVIOUS CONCURRENCES*

OFFICE	ORB 1*	ORB 1	ORB 1*	AD:OR:DL*	OELD*		
SURNAME	CParrish	DWigginton/ef	/rsSVarga	GLainas	WFP		
DATE	9/14/83	9/20/83	9/15/83	9/15/83	9/19/83		

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 19[illegible]

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NRC PDR
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ORB L Rdg
H Denton
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L J Harmon
E Jordan
J Taylor
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D Brinkman
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OPA,C Miles
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NSIC
GPB
ASB
AED
D Eisenhut

September 20, 1983

Docket No. 50-315

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
Post Office Box 18
Bowling Green Station
New York, New York 10004

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Sincerely,

/S/

David L. Wigginton, Project Manager
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Division of Licensing

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cc: w/enclosures
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OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD:OR:DL	OELD		
SURNAME	CParrish;	DWigginton;ef	SVarga	GLainas			
DATE	9/14/83	9/14/83	9/15/83	9/15/83	9/19/83		

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

cc: Mr. M. P. Alexich
Assistant Vice President
for Nuclear Engineering
American Electric Power
Service Corporation
2 Broadway
New York, New York 10004

Mr. William R. Rustem (2)
Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Mr. Wade Schuler, Supervisor
Lake Township
Baroda, Michigan 49101

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
P. O. Box 458
Bridgman, Michigan 49106

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Honorable James Bemenek, Mayor
City of Bridgman, Michigan 49106

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

Maurice S. Reizen, M.D.
Director
Department of Public Health
P.O. Box 30035
Lansing, Michigan 48109

The Honorable Tom Corcoran
United States House of Representatives
Washington, D. C. 20515

James G. Keppler
Regional Administrator - Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-58

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated May 11, 1983, as supplemented by letter dated July 25, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 74, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 20, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Pages
IA	IA
IV	IV
1-7	1-7
---	1-10
2-1	2-1*
2-2	2-2
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
3/4 1-1	3/4 1-1
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22*
3/4 1-23	3/4 1-23*
3/4 1-24	3/4 1-24
3/4 1-25	3/4 1-25
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7 thru 3/4 2-24	3/4 2-7 thru 3/4 2-24
3/4 3-1	3/4 3-1*
3/4 3-2 thru 3/4 3-4	3/4 3-2 thru 3/4 3-4
3/4 3-5	3/4 3-5*
3/4 3-6 thru 3/4 3-8	3/4 3-6 thru 3/4 3-8
3/4 10-1	3/4 10-1
B 2-1	B 2-1
-----	B 2-1(a)
B 2-2	B 2-2
-----	B 2-2(a)
B 2-3	B 2-3*
B 2-4	B 2-4
B 2-5	B 2-5*
B 2-6	B 2-6
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2*
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
B 3/4 3-3	B 3/4 3-3
B 3/4 4-1	B 3/4 4-1

*Included for convenience

INDEX

DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

MEMBER(S) OF THE PUBLIC

1.35 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

SITE BOUNDARY

1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

DESIGN THERMAL POWER

1.38 DESIGN THERMAL POWER shall be a design total reactor core heat transfer rate to the reactor coolant of 3411 MWt. See Table 1.3.

TABLE 1.3

Safety Analysis Basis-Power Levels

The approved maximum power operation and RATED THERMAL POWER is 3250 MWt. However, certain portions of the safety analysis provided for Donald C. Cook Nuclear Plant Unit 1 have been based on a design power of 3411 MWt. The safety analysis for which 3411 MWt has been used is as follows:

(1) Uncontrolled Rod Control Cluster Assembly (RCCA) withdrawal from a subcritical condition.

(2) Uncontrolled control rod assembly withdrawal at power.

(3) RCCA misalignment.

(4) Chemical and volume Control System malfunction.

(5) Loss of reactor flow (including locked rotor).

(6) Loss of external electrical load.

(7) Loss of normal feedwater flow.

(8) Excessive heat removal due to feedwater system malfunctions.

(9) Excessive increase in secondary steam flow.

(10) Loss of all AC power to the plant auxiliaries.

(11) Rupture of a steam pipe.

(12) Rupture of control rod drive mechanism housing (RCCA ejection).

(13) Small break Loss of Coolant Accident (LOCA).

The rated thermal power of 3250 MWt was the basis for the safety analysis used for the large break Loss Of Coolant Accident.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

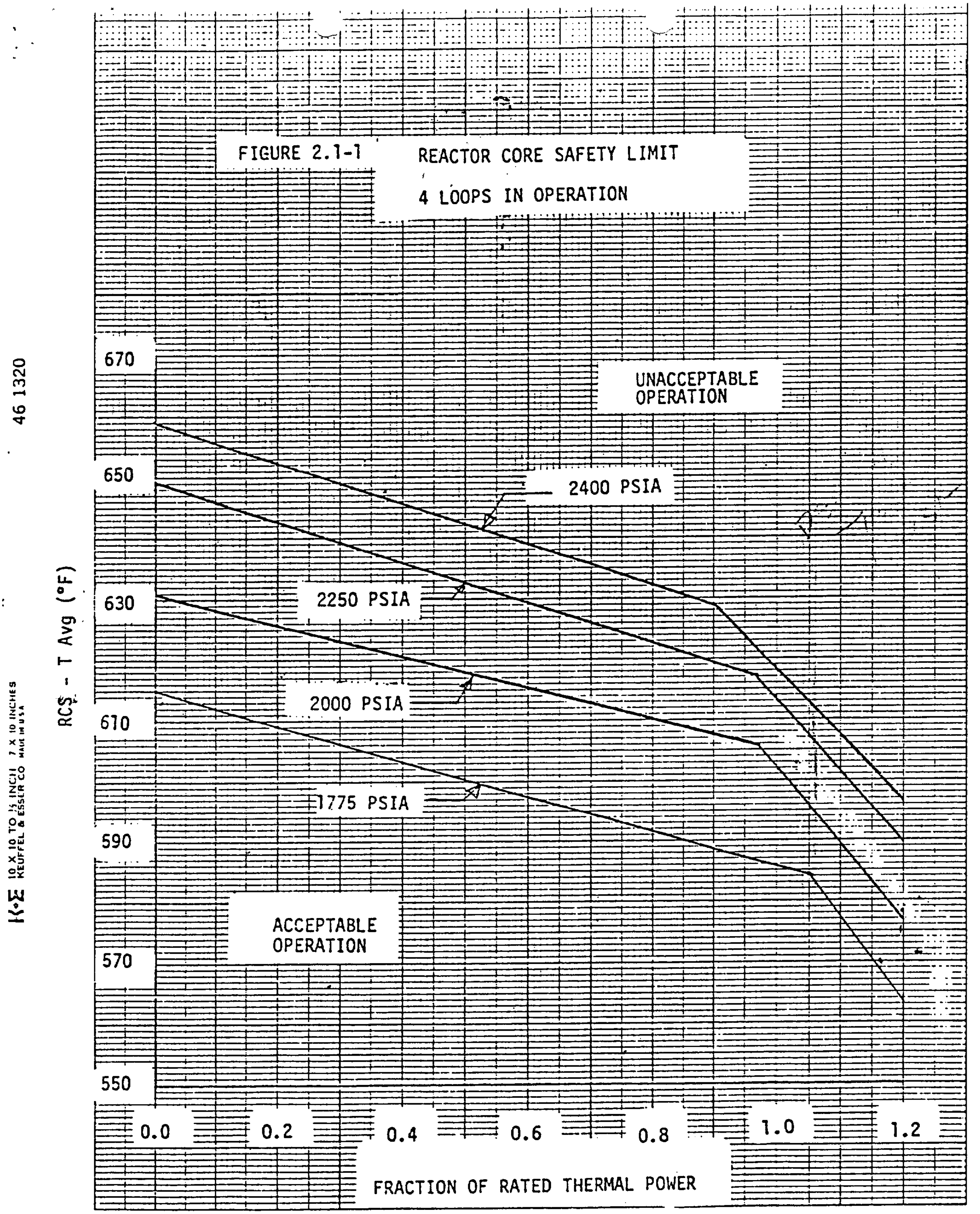

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT
4 LOOPS IN OPERATION
UNACCEPTABLE OPERATION
2400 PSIA
2250 PSIA
2000 PSIA
1775 PSIA
ACCEPTABLE OPERATION
670
650
630
610
590
570
550
RCS - T Avg (°F)
0.0
0.2
0.4
0.6
0.8
1.0
1.2
FRACTION OF RATED THERMAL POWER
46 1320
K+E 10 X 10 TO 1/2 INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where:	ΔT_o	=	Extrapolated ΔT at DESIGN THERMAL POWER
	T	=	Average temperature, °F
	T'	=	577.1°F (indicated T_{avg} at DESIGN THERMAL POWER)
	P	=	Pressurizer pressure, psig
	P'	=	2235 psig (indicated RCS nominal operating pressure)
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 33 secs, τ_2 = 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops	Operation with 3 Loops
$K_1 = 1.135$	$K_1 = 0.99$
$K_2 = 0.0130$	$K_2 = 0.01026$
$K_3 = 0.000659$	$K_3 = 0.000617$

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

(i) for $q_t - q_b$ between -37 percent and +2 percent, $f_1 (\Delta I) = 0$ (where q_t and q_b are percent DESIGN THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of DESIGN THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 2.3 percent of its value at DESIGN THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +2 percent, the ΔT trip setpoint shall be automatically reduced by 1.8 percent of its value at DESIGN THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

where:	ΔT_o	=	Extrapolated ΔT at DESIGN THERMAL POWER
	T	=	Average temperature, °F
	$T"$	=	Indicated T_{avg} at DESIGN THERMAL POWER 577.1°F
	K_4	=	1.089
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0011 for $T > T"$; $K_6 = 0$ for $T \leq T"$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
	S	=	Laplace transform operator
	$f_2(\Delta I)$	=	$f_1(\Delta I)$ as defined in Note 1 above.

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

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 ≥ 1.60% Δk/k.

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

ACTION:

With the SHUTDOWN MARGIN < 1.60% Δk/k, immediately initiate and continue boration at ≥ 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 ≥ 1.60% Δk/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} ≥ 1.0

##With K_{eff} <1.0

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq$ 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: Mode 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to $\leq$ 76 percent of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2and 3.10.4.
#With $K_{eff} \geq 1.0$

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figures, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

---

*See Special Test Exceptions 3.10.2 and 3.10.4

#With $K_{eff} \geq 1.0$.

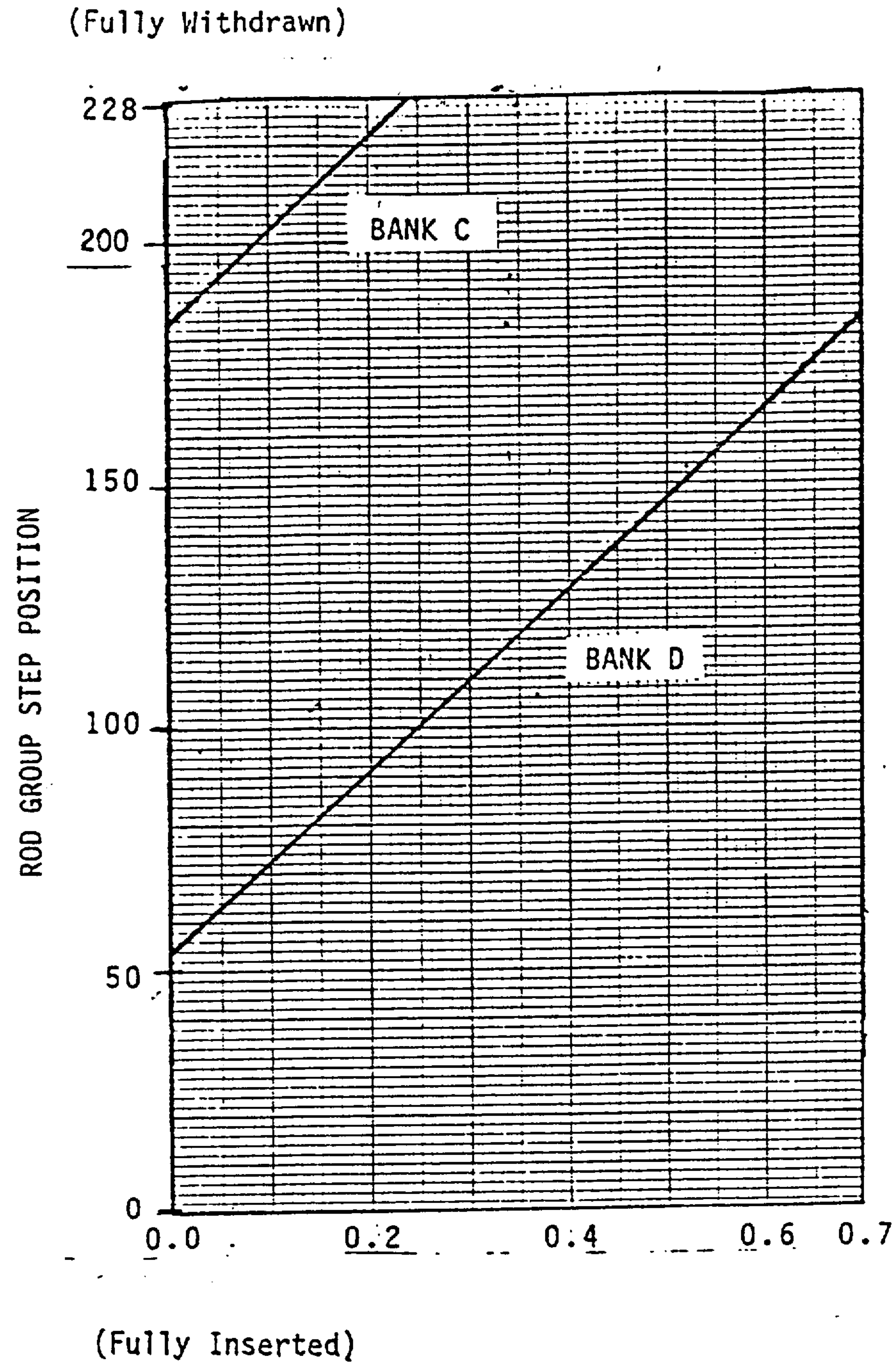

FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER
3 LOOP OPERATION

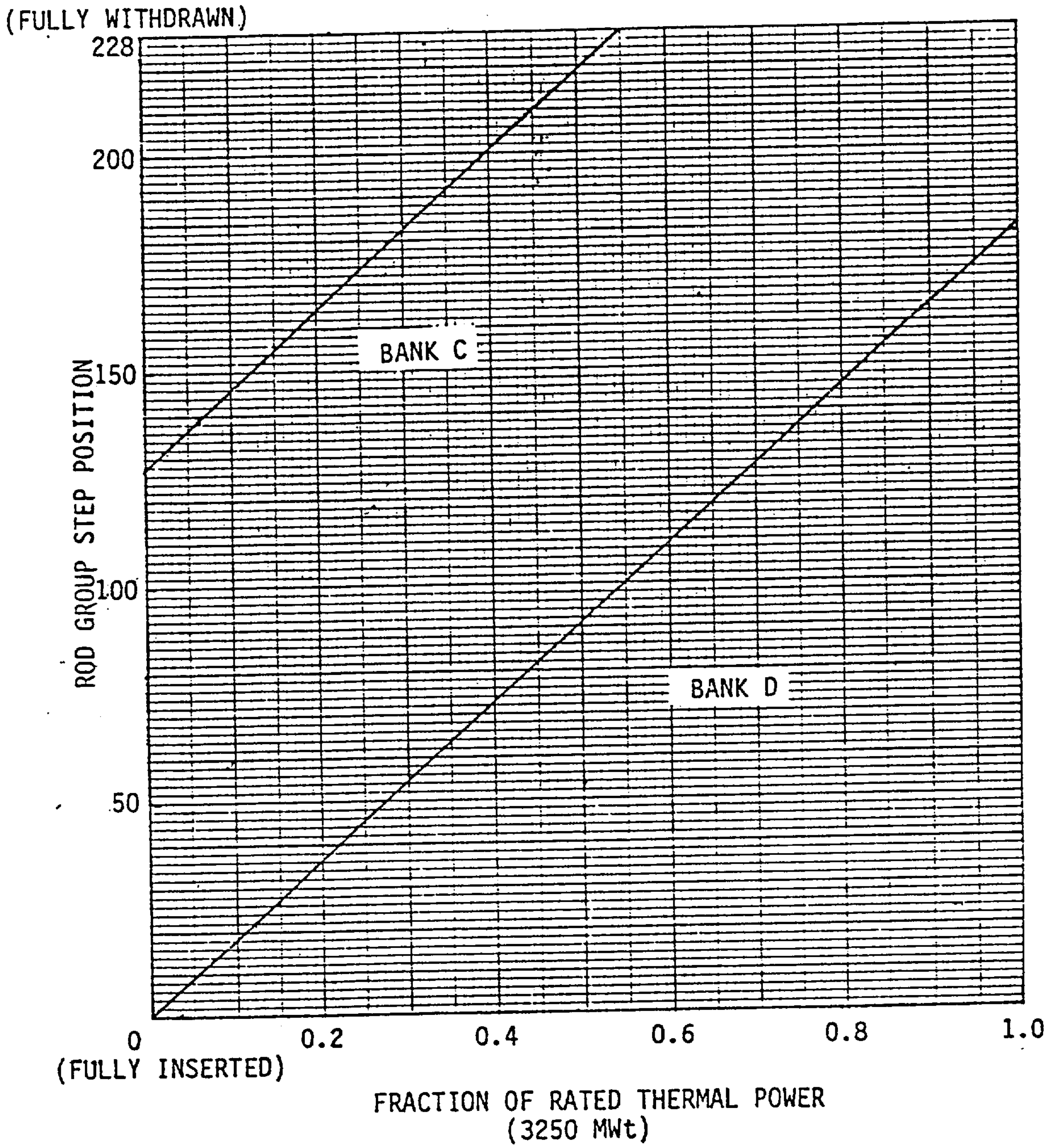

FIGURE 3.1-2 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER
4 LOOP OPERATION

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

LIMITING CONDITION FOR OPERATION

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

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q(Z,\ell) \le [\frac{1.97}{P}]\ [K(Z)]$	$F_Q(Z,\ell) \le [\frac{F_Q^L(E_\ell)}{P}]\ [K(Z)]$	$P > 0.5$
$F_Q(Z,\ell) \le [3.94]\ [K(Z)]$	$F_Q(Z,\ell) \le 2\ [F_Q^L(E_\ell)\ K(Z)]$	$P \le 0.5$

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

$F_Q^L\ (E_\ell)$ is the exposure dependent F_Q limit for rod ℓ and is defined in Figure 3.2-4 for Exxon Nuclear Co. fuel and in Figure 3.2-5 for Westinghouse fuel. E_ℓ is the maximum pellet exposure in rod ℓ. K(Z) is the function obtained from Figure 3.2-3 for Westinghouse fuel and Figure 3.2-2 for Exxon Nuclear Co. fuel. F_Q is defined as the $F_Q(Z,\ell)$ with the smallest margin or the greatest excess of the limit.

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

a. Comply with either of the following ACTIONS:

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

LIMITING CONDITION FOR OPERATION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z,\ell)$ shall be determined to be within its limit by:

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

b. Increasing the measured $F_Q(Z,\ell)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. This product is defined as $F_Q^M(Z)$.

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

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q^M(Z) \leq \left[\frac{1.97}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$F_Q^M(Z) \leq \left[\frac{F_Q^L(Z)}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$P > 0.5$
$F_Q^M(Z) \leq \left[\frac{3.94}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$F_Q^M(Z) \leq \left[\frac{2\ F_Q^L(Z)}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$P \leq 0.5$

LIMITING CONDITION FOR OPERATION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z,\ell)$ shall be determined to be within its limit by:

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

b. Increasing the measured $F_Q(Z,\ell)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. This product is defined as $F_Q^M(Z)$.

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

Westinghouse Fuel

$$F_Q^M(Z) \leq \left[\frac{2.0}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$$

$$F_Q^M(Z) \leq \left[\frac{4.0}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$$

Exxon Nuclear Co. Fuel

$$F_Q^M(Z) \leq \left[\frac{F_Q^L(Z)}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)} \qquad P > 0.5$$

$$F_Q^M(Z) \leq \left[\frac{2\ F_Q^L(Z)}{E_p(Z)}\right] \frac{K(Z)}{V(Z)} \qquad P \leq 0.5$$

SURVEILLANCE REQUIREMENTS (Continued)

where

$F_Q^M(Z) = F_Q(Z,\ell)$ at ℓ for which

$\frac{F_Q(Z,\ell)}{T(E_\ell)}$ is a maximum

$F_Q^L(Z) = F_Q^L(E_\ell)$ at ℓ for which

$\frac{F_Q(Z,\ell)}{T(E_\ell)}$ is a maximum

$F_Q^M(Z)$ and $F_Q^L(Z)$ are functions of core height, Z, and correspond at each Z to the rod ℓ for which $\frac{F_Q(Z,\ell)}{T(E_\ell)}$ is a maximum at that Z

V(Z) is a cycle dependent function and is provided in the Peaking Factor Limit Report. K(Z) is defined in Figure 3.2-2 for Exxon Nuclear Company fuel and in Figure 3.2-3 for Westinghouse fuel. $T(E_\ell)$ is defined in Figures 3.2-4 and 3.2-5. $E_p(Z)$ is an uncertainty factor to account for the reduction in the $F_Q^L(E_\ell)$ curve due to accumulation of exposure prior to the next flux map.

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$E_p(Z) = 1.0$	$E_p(Z) = 1.0$	$0 \leq E_\ell \leq 17.62$
$E_p(Z) = 1.0$	$E_p(Z) = 1.0 + [.0040 \times F_Q^M(Z)]$	$17.62 < E_\ell \leq 34.5$
$E_p(Z) = 1.0$	$E_p(Z) = 1.0 + [.0093 \times F_Q^M(Z)]$	$34.5 < E_\ell \leq 42.2$

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

e. With successsive measurements indicating an increase in max over Z of $[\frac{F_Q^M(Z)}{K(Z)}]$ with exposure, either of the following additional actions shall be taken:

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

2. $F_Q^M(Z)$ shall be measured and a target axial flux difference reestablished at least once per 7 effective full power days until 2 successive maps indicate that max over Z of $[\frac{F_Q^M(Z)}{K(Z)}]$ is not increasing.

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

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

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z,\ell)$ exceeding its limit by the maximum percent calculated with the following expressions with V(Z) corresponding to the target band and $P \geq 0.5$:

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z) \times E_p(Z)}{\frac{F_L^Q(E_\ell)}{P} \times [K(Z)]}\right] -1\right] \times 100$$

Exxon Nuclear Co Fuel

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

WESTINGHOUSE FUEL

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

1. Lower core region 0 to 10% inclusive.

2. Upper core region 90% to 100% inclusive.

4.2.2.3 When $F_Q(Z,\ell)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z,\ell)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

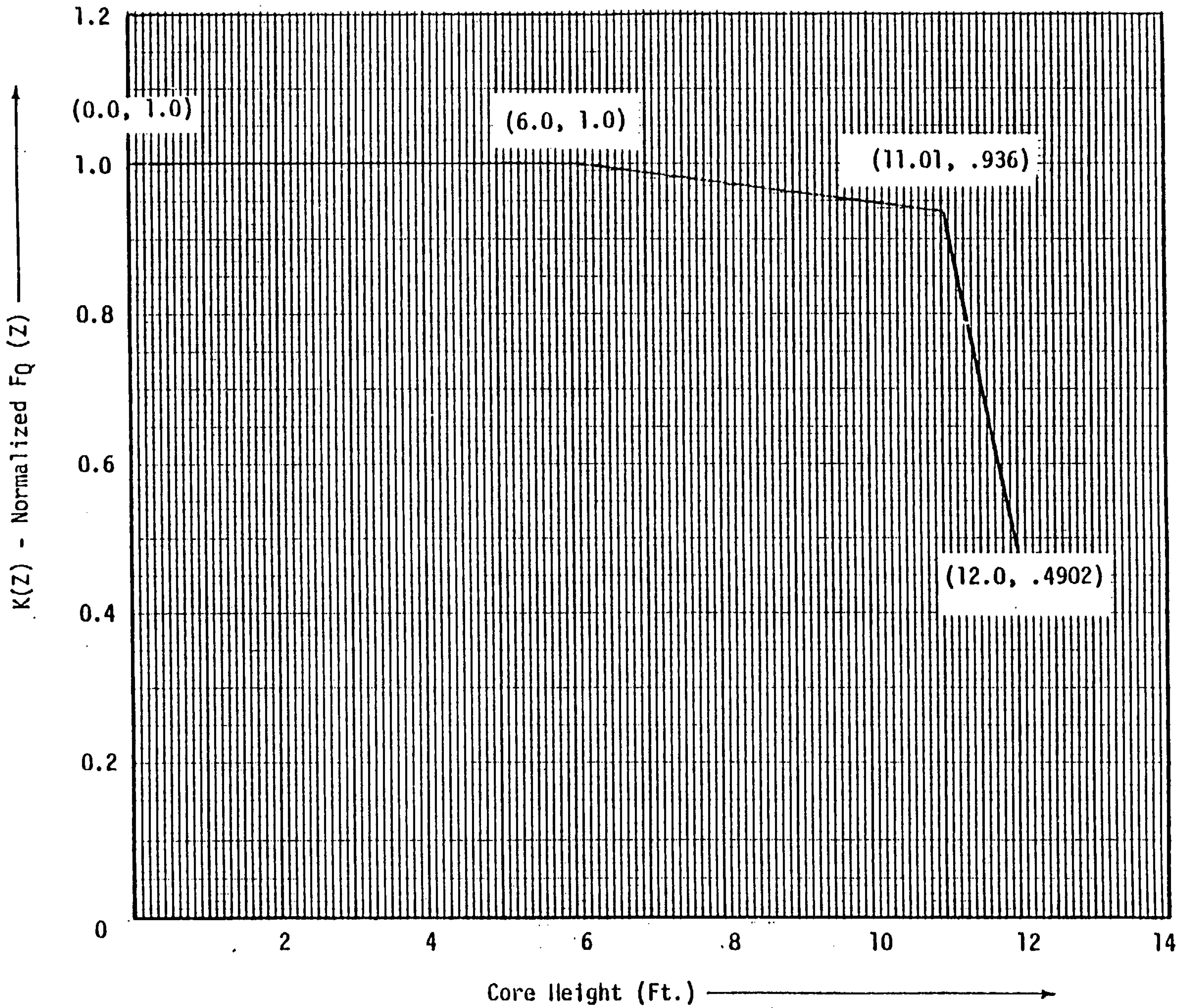

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ As A Function of Core Height for Exxon Nuclear Company Fuel

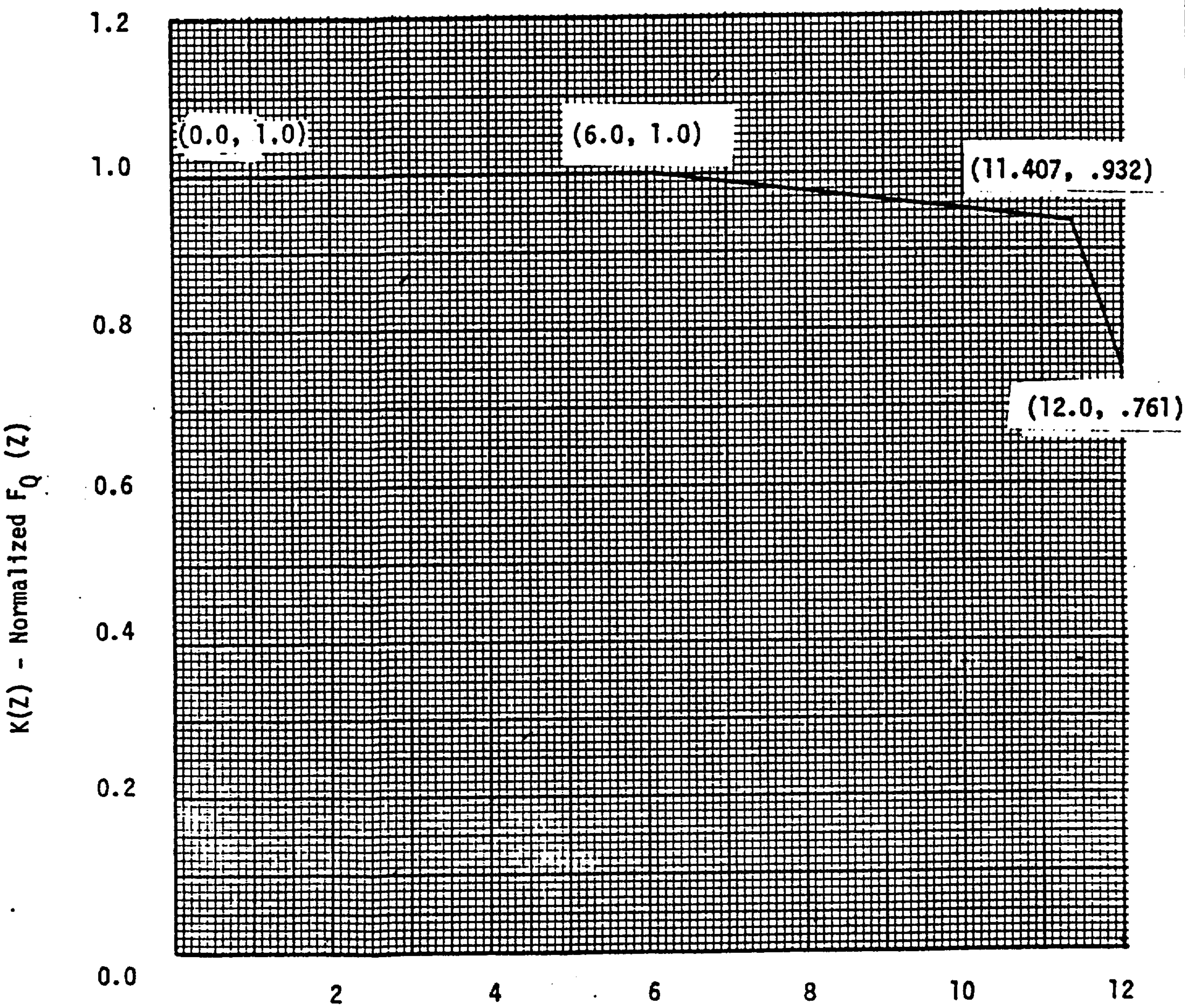

Figure 3.2-3 K(Z) - Normalized F_Q (Z) As A Function of Core Height For Westinghouse Fuel

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationships:

$F^N_{\Delta H} \leq 1.49\ [1 + 0.3\ (1-P)]$ (for Westinghouse fuel)

and $F^N_{\Delta H} \leq 1.45\ [1 + 0.2\ (1-P)]$ (for Exxon Nuclear Co. fuel)

where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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

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

c. The provisions of Specification 4.0.4 are not applicable.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but $\leq$ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per month.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 4 Loops In Operation at RATED THERMAL POWER	LIMITS: 4 Loops In Operation at DESIGN THERMAL POWER	LIMITS: 3 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System T_{avg}	$\leq$ 570.5°F	$\leq$ 579.8°F	$\leq$ 570.5°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq 1.386 \times 10^8$ lbs/hr	$\geq 1.386 \times 10^8$ lbs/hr	$\geq 0.9917 \times 10^8$ lbs/hr

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

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

Westinghouse Fuel

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

Exxon Nuclear Co. Fuel

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

where:

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

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained for a given core height location from Figure 3.2-2 for Exxon Nuclear Company fuel and from Figure 3.2-3 for Westinghouse fuel.

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 at 100% or APL (whichever is less) of RATED THERMAL POWER in accordance with:

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

where:

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

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

e. $F_{Qi\ell}^{Meas}$ is the limiting total peaking factor in flux map i. The limiting total peaking factor is that factor with least margin to the $F_Q^L(E\ell)$ curve defined in Figure 3.2-4 for Exxon Nuclear Company fuel and in Figure 3.2-5 for Westinghouse fuel.

For Exxon Nuclear Company fuel, $T(E\ell)$ is the ratio of the exposure dependent $F_Q^L(E)$ to 2.04 and is defined in Figure 3.2-4. $T(E\ell)$ is equal to 1.0 for fuel supplied by Westinghouse Electric Corporation as given in Figure 3.2-5.

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

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

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

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

The factor 1.03 is the engineering uncertainty factor.

g. F_p is an uncertainty factor for Exxon fuel to account for the reduction in the $F_Q^L(E_\ell)$ curve due to an accumulation of exposure prior to the next flux map. The following F_p factor shall apply:

LIMITING CONDITION FOR OPERATION (Continued)

Westinghouse Fuel	ENC Fuel	
$F_p = 1.0$	$F_p = 1.0$	$0 \leq E_\ell \leq 17.62$
$F_p = 1.0$	$F_p = 1.0 + [.0015 \times W]$	$17.62 < E_\ell \leq 34.5$
$F_p = 1.0$	$F_p = 1.0 + [0.0033 \times W]$	$34.5 < E_\ell \leq 42.2$

where W is the number of effective full power weeks (rounded up to the next highest integer) since the last full core flux map.

APPLICABILITY: Mode 1 above the minimum percent of RATED THERMAL POWER indicated by the relationships.*

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

$$APL = \text{min over Z of } \frac{F_Q^L(E_\ell) \times K(Z)}{F_Q(Z,\ell) \times V(Z) \times F_p(Z)} \times 100\,\%$$ Exxon Nuclear Co. Fuel

where $F_Q(Z,\ell)$ is the measured $F_Q(Z,\ell)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty, at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the function given in the Peaking Factor Limit Report. The above limit is not applicable in the following core plane regions.

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

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER 1 percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next 2 hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to APL 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 APL or less of RATED THERMAL POWER within 15 minutes.

SURVEILLANCE REQUIREMENTS

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

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.6 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 APL of RATED THERMAL POWER, or

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

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

1. At least once per 8 hours, and

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

a) Increasing the THERMAL POWER above APL of RATED THERMAL POWER, or

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

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

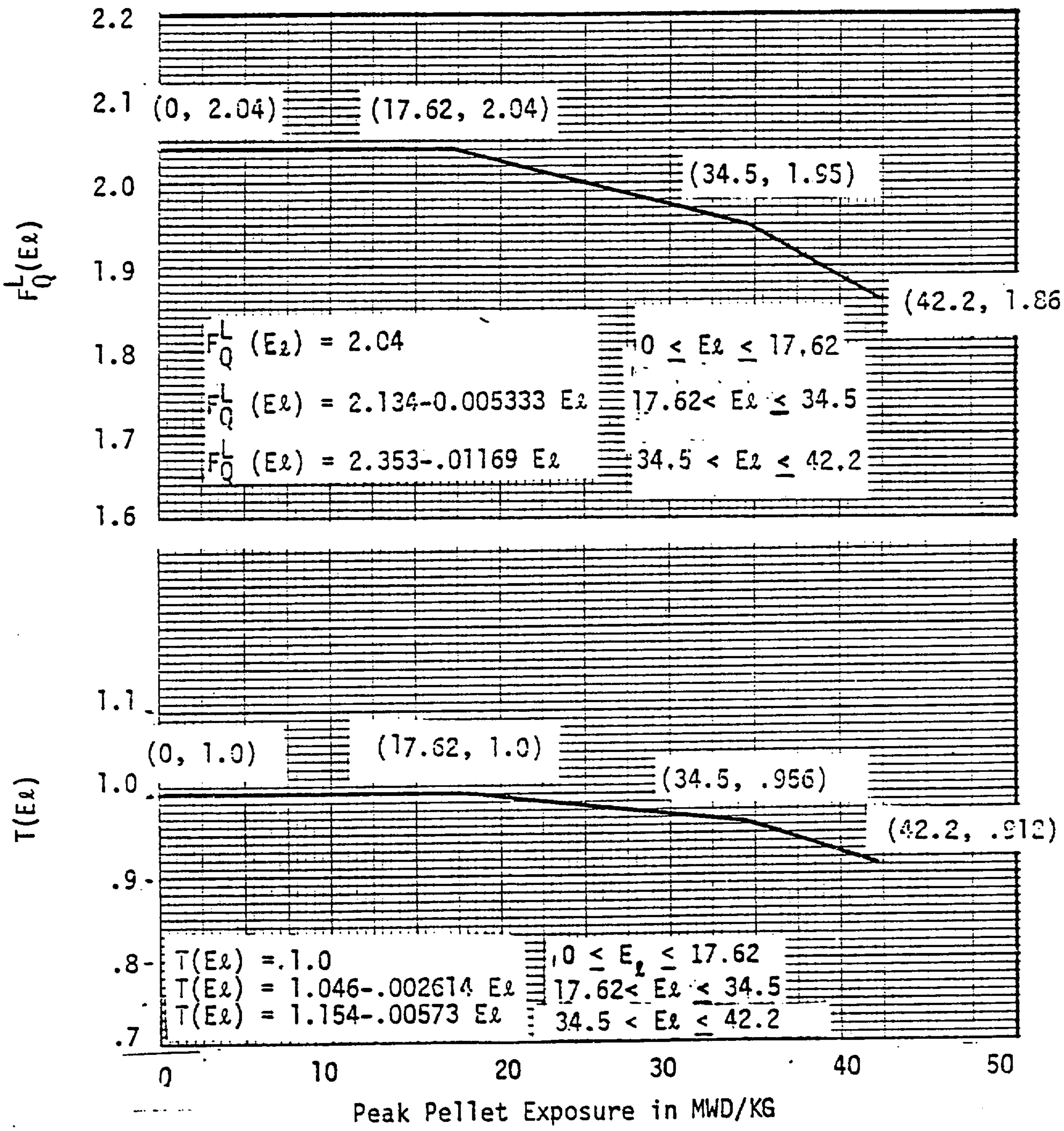

FIGURE 3.2-4

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

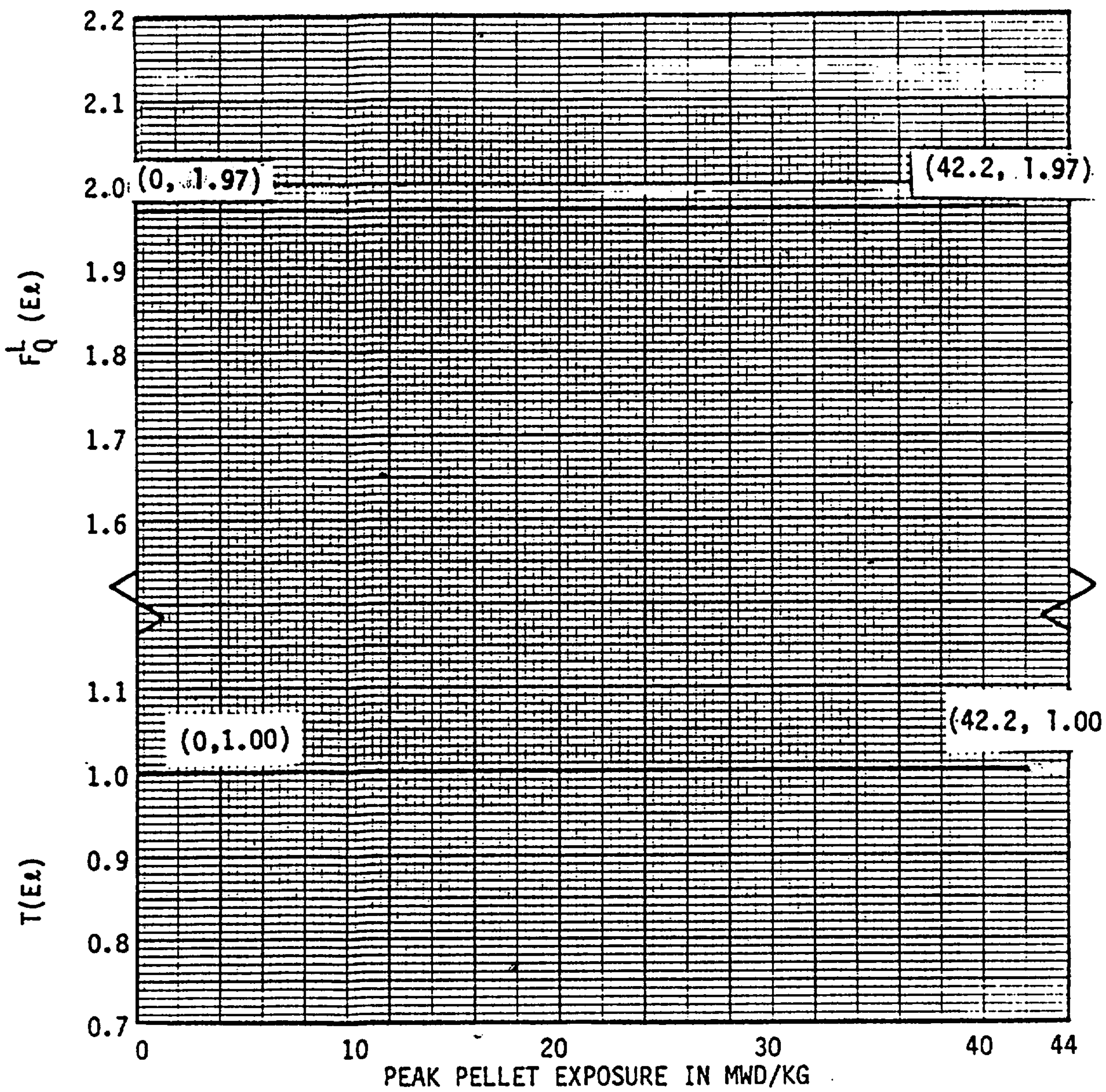

FIGURE 3.2-5

Exposure Dependent F_Q Limit, F_Q^L (E_ℓ), and Normalized Limit $T(E_\ell)$ as a Function of Peak Pellet Burnup for Westinghouse Fuel

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	4	1**	3	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	4	1**	3	1, 2	9
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#
16. Undervoltage-Reactor Coolant Pumps	4/1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	4	1	7#
19. Safety Injection Input from ESF	2	1	2	1, 2	1

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10
B. Above P-7	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1, 2*	1
22. Automatic Trip Logic	2	1	2	1. 2*	1

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied.

a. The inoperable channel is placed in tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met: however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.c.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

TABLE 3.3-1 (Continued)

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above P-6 but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above P-6, operation may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of the other channels per Specification 4.3.1.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-1 (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Total Numbers of Channels and with the THERMAL POWER level above P-7, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels $< 6 \times 10^{-11}$ amps.	P-6 prevents or defeats the manual block of source range reactor trip.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated OPERABLE by verifying its rod drop time to be $\leq$ 2.4 seconds within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

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

The DNB design basis is as follows: there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent confidence that the minimum DNBR for the limiting rod is greater than or equal to the applicable design DNBR limit for each fuel type (as defined below). For 4 loop operation, the improved thermal design procedure is used. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit (as defined below), establishes a design DNBR limit value, which must be met in plant safety analyses, using values of input parameters without uncertainties. For 3 loop operation, a conservative set of uncertainties are used in the safety analyses.

The table below indicates the relationship between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design.

BASES

	4 Loop Operation				3 Loop Operation	
	(WRB-1 Correlation)		(W-3 Correlation)		(W-3 Correlation)	
	Westinghouse Fuel (15x15 OFA)		Exxon Nuclear Co. Fuel (15x15)		W and ENC Fuels	
	Typical	Thimble	Typical	Thimble	Typical	Thimble
Correlation Limit	1.17	1.17	1.30	1.30	1.30	1.30
Design Limit DNBR	1.32	1.31	1.58	1.50	1.30	1.30
Safety Analysis Limit DNBR	1.69	1.69	1.58	1.50	1.30	1.30

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor $F^N_{\Delta H}$, of 1.49 for Westinghouse fuel and an $F_{\Delta H}$ of 1.45 for Exxon Nuclear Co. fuel and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power, based on the expressions:

$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$ (for Westinghouse fuel)

and $F^N_{\Delta H} = 1.45\ [1 + 0.2\ (1-P)]$ (for Exxon Nuclear Co. fuel)

where P is the fraction of RATED THERMAL POWER

Note, do not include a 4% uncertainty value, since this measurement uncertainty has been included in the design DNBR limit values, which are listed in the bases for Section 2.1.1.

Although the N-loop operation curves are calculated for operation at DESIGN THERMAL POWER, $F^N_{\Delta H}$ values for RATED THERMAL POWER are reported here in order to be consistent with Section 3.2.3. The $F^N_{\Delta H}$ values of Section 3.2.3 are limited by the LOCA analyses which were performed at RATED THERMAL POWER.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 9 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless, manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

BASES

Operation with a reactor coolant loop out of service below the 4 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature ΔT set point. Three loop operation above the 4 loop P-8 set point is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature ΔT channels and raising the P-8 set point to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above 51% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the applicable safety analysis design limit DNBR value for each fuel type, (as listed in the bases for Section 2.1.1) during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the applicable safety analysis design limit DNBR value for each fuel type, (as listed in the bases for Section 2.1.1) during normal operational transients and anticipated transients when 3 loops are in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip, to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses, but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

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 1.60%$\Delta k/k$ 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% $\Delta k/k$ 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.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS $F_Q(Z)$ and $F^N_{\Delta H}$

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

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

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

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

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

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits, provided conditions (a) through (d) above are maintained.

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for, and 4% is the appropriate allowance for a full core map taken with the incore detection system. This 4% measurement uncertainty has been included in the design DNBR limit value. The specified limit for $F^N_{\Delta H}$ also contains an additional 4% allowance for uncertainties. The total allowance is based on the following considerations:

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

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

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

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

3/4.2.4 QUADRANT POWER TILT RATIO

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

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

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

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated to be adequate to maintain the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The monthly periodic RCS elbow tap flow measurement is adequate to detect flow degradation and to ensure the correlation of the flow indication channels with measured flow, as determined at the beginning of each cycle using a power balance around the steam generators, such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis. Measurement uncertainties have been accounted for in determining the DNB parameters limit values.

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 APL of RATED THERMAL POWER. This additional limitation on F_Q is necessary, in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA.

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

3/4.3.3.6 AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)

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

R_j factors are used to determine the APDMS setpoint limits $[F_j(Z)]_S$. On a full core basis the R_j and σ_j factors are calculated in accordance with the equations on Pages 3/4 2-18 and 3/4 2-19.

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

In order to cope with this difficulty, it is permissible to calculate as many σ_j's and $\bar{R}_j$'s for each thimble as there are fuel types or regions in the core. Each $\bar{R}_j$ and σ_j for a thimble j is to be calculated from the equations on Pages 3/4 2-18 and 3/4 2-19 with the following exception. For each $\bar{R}_j$ and σ_j for thimble j, a different $F_{Qi\ell}$ and $\bar{T}(E)$ shall be used. The different σ_j's and R_j's for thimble j shall be calculated substituting for $F_{Qi\ell}^{meas}$ and T(E) the values pertaining to the limiting peak relative power from each fuel region. Obviously for one of these calculations the limiting peak relative power from one region will be the core limiting total peaking factor.

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

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the applicable design limit DNBR value during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER, until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above the applicable design limit DNBR values for each fuel type. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (51 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing an RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 188°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any over-pressure conditions which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-315

A. INTRODUCTION

By letter dated May 11, 1983, the Indiana and Michigan Electric Company (the licensee) submitted an application for the Donald C. Cook Nuclear Plant, Unit No. 1 reload for Cycle 8. The reload will include the first fuel batch fabricated by Westinghouse (W) of the 15X15 optimized fuel assembly design and the first use in the Donald C. Cook Nuclear Plant of the W Wet Annular Burnable Absorber (WABA) burnable poison rods. The reload fuel will have an enrichment up to 4.0 weight percent U 235 and may achieve extended burnup in future cycles to 39,000 MWD/MTU (average region discharge). The application has also defined a new term "design basis power level" of 3411 MWt at which a number of accidents and transients have been analyzed. However, no request has been made to increase the approved power level for operation and some of the more significant evaluations, i.e., large break loss of coolant accident (LOCA), have not been submitted at this higher power level.
The approved maximum power level for the Donald C. Cook Nuclear Plant, Unit No. 1 remains at 3250 Mwt.

On June 22, 1983, the NRC issued a "Monthly Notice: Amendments to Operating Licenses Involving No Significant Hazards Considerations; Duquesne Light Company et al." with the Office of the Federal Register for publication. That notice recognized the proposed core reload for Cycle 8 and the related changes to the Technical Specifications. In a related licensing action, on May 4, 1983, the NRC issued Amendments 73 and 55 to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively. Those amendments revised the Technical Specification to permit storage of the W fuel with a uranium enrichment of less than or equal to 4.00 weight percent U-235.

Subsequent to the May 11, 1983 letter by the licensee, a number of supplements to the original proposal have been received and were used in the evaluation of the W fuel for Cycle 8 operation. The evaluation section includes a list of references to these supplements as well as other information used in the evaluation.

B. EVALUATION

1. Introduction:

By letter dated May 11, 1983, the Indiana and Michigan Electric Company (the licensee) made application to amend the Technical Specifications of the Donald C. Cook Nuclear Plant, Unit No. 1, in order to reload and operate the plant for Cycle 8. In support of the application, attachments A through G were appended to the letter. The Core Performance Branch has reviewed the application and prepared the following evaluation.

For Cycle 8 the licensee is switching fuel vendors from EXXON (ENC) to Westinghouse who performed the analyses for this reload. In addition, in anticipation of an application for a power increase from the currently licensed 3250 MWt to 3411 MWt, all analyses were performed at the higher power with the exception of the LOCA analysis.

2. Fuel Mechanical Design

The D. C. Cook, Unit 1, Cycle 8 reload core will consist of 80 Westinghouse 15x15 optimized fuel assemblies (OFAs) and 113 Exxon Nuclear (ENC) 15x15 fuel assemblies. Although the Westinghouse 15x15 OFA fuel is a new design, it is very similar to the Westinghouse 15x15 standard low parasitic (LOPAR) fuel design, which previously operated in Cook Unit 1 and has substantial commercial operating experience. The major change introduced by the 15x15 OFA design is the use of five intermediate Zircaloy grids replacing five intermediate Inconel grids in the LOPAR fuel. The Zircaloy grids have thicker and wider straps than the Inconel grids in order to closely match the Inconel grid strength. Furthermore, the 15x15 OFA Zircaloy grid design is similar to the Westinghouse 17x17 OFA grid design, which was decribed in WCAP-9500-A (Ref. 1), which has been reviewed and approved by the NRC.

In performing our review of the 15x15 OFA fuel for Cook, Unit 1, we asked the licensee to verify that the design criteria and evaluation methods used for 17x17 OFA in WCAP-9500-A were also used for Cook's 15x15 OFA. The licensee verified that both criteria and methods were exactly the same (Ref. 2). The balance of our review thus focused on those plant-specific issues identified in the SER for WCAP-9500-A insofar as they are applicable to Cook, Unit 1, Cycle 8. Our evaluation of those issues follows.

2.1 Cladding Collapse

The licensee uses an approved method described in WCAP-8377 (Ref. 3) to analyze cladding collapse. The result for Cook, Unit 1 shows that no cladding collapse is expected up to 40,000 EFPH (in excess of 50,000

MWd/MTU peak-rod average burnup) for the new Westinghouse fuel design. The ENC fuel remains bounded by the previously accepted analysis. We conclude, therefore, that no cladding collapse is expected during Cycle 8 operation.

2.2 Rod Bowing

The rod bow magnitude for the Westinghouse OFA fuel was calculated with an approved method described in WCAP-8691, Revision 1 (Ref. 4). The rod bow magnitude for the ENC fuel was calculated in an earlier Cook, Unit 1 reload safety analysis and found to be acceptable by the NRC staff. Penalties associated with these adequately calculated bow magnitudes are discussed in Section 4.0 of this evaluation.

2.3 Fuel Thermal Conditions

The D. C. Cook, Unit 1, Cycle 8 reload submittal (Ref. 5) is based, in part, upon fuel thermal analyses generated with a revised (Ref. 6) version of a previously approved Westinghouse code called PAD (Ref. 7). The single revision to the PAD code is currently under staff review. A request for additional information was issued (Ref. 8) and responses (Ref. 9) have been obtained from the fuel vendor (Westinghouse).

Due to unexpected computational difficulties, the responses obtained from Westinghouse have not shown that certain analytical assumptions (e.g., worst time in life) continue to be met with the revised version of PAD. Pending resolution of this problem, and to avoid impacting the Cycle 8 reload schedule, the licensee submitted an addendum (Ref. 10) to the Cycle 8 reload report which (partially) reverts back to the previously approved version of PAD. The reanalysis results in a slightly lower LOCA Fq limit of 1.97, compared to an Fq of 2.00 using the revised thermal safety model (Ref. 6). The lower Fq limit and its associated K(Z) envelope have been incorporated into the revised Technical Specifications for D. C. Cook, Unit 1.

The revised Fq limit is based on an updated large break LOCA analysis described in Attachment C to Reference 10. The worst break was reanalyzed (at 3250 MWt) using previously approved methods, including the approved version of PAD. Results show that the D. C. Cook, Unit 1 emergency core cooling system will meet the acceptance criteria in 10 CFR 50.46 for Cycle 8 conditions. We find this result, and the manner in which it was obtained, acceptable. The manner in which the revised Fq limit and associated K(Z) envelope have been incorporated into the plant Technical Specifications has also been examined (see Section 3.0 of this SER) and found acceptable.

Other non-LOCA analyses in the Cycle 8 submittal continue to rely on the unapproved version of PAD. However, Westinghouse has performed (Ref. 10) an evaluation to determine if the use of the revised PAD model impacts other core operating limits. The initial fuel conditions used in non-LOCA transients were re-examined and it was found that the revised PAD code has only a slight impact on the safety analysis. In all cases, the appropriate design bases are still met. The small break LOCA ECCS analysis was not reanalyzed because the event is not limiting. In addition, cladding heatup occurs after core uncovery for this event and is not sensitive to changes in initial stored energy.

We conclude that the methods used to determine fuel thermal conditions, including limited use of the unapproved, revised version of PAD, are acceptable in support of the D. C. Cook, Unit 1, Cycle 8 reload safety analysis and the resulting modifications to the plant Technical Specifications.

2.4 Cladding Swelling and Rupture

For large break loss-of-coolant accident analysis, the licensee used the approved 1981 large break ECCS evaluation model (Ref. 11), which includes an approved cladding swelling and rupture model. The use of this ECCS model obviates the need for supplemental ECCS calculations mentioned in the SER for WCAP-9500-A. We thus find that cladding swelling and rupture have been adequately treated in the Cycle 8 reload analysis.

2.5 Seismic and LOCA Loads

Three major fuel types have been recently analyzed for seismic-and-LOCA loads in Cook Unit 1. These fuel types are: (1) LOPAR (standard Westinghouse Inconel-grid 15x15 fuel, now completely discharged from Cook Unit 1), (2) ENC (Exxon Nuclear Zircaloy-grid 15x15 fuel that constitutes the entire Cycle-7 core), and (3) OFA (new Westinghouse 15x15 Optimized Fuel Assemblies to be loaded in one region of the core for Cycle 8). Exxon Nuclear previously performed a seismic (only) loads analysis for a mixed-core configuration of LOPAR and ENC fuel; that analysis demonstrated that fuel rod and guide tube integrity and core coolable gemoetry would be maintained (Ref. 14). As part of the present reload safety analysis, Westinghouse performed a seismic-and-LOCA loads analysis for a mixed-core configuration of ENC and OFA fuel; that analysis demonstrated that fuel rods and guide tubes (thimbles) have ample margin (almost a factor of 2) even when seismic-and-LOCA loads were combined (Ref. 2). In the Westinghouse analysis, spacer grids had adequate margin to withstand seismic-and-LOCA loads separately, but grid deformation in core-peripheral fuel assemblies would be expected if seismic-and-LOCA loads were combined.

Several circumstances are noteworthy. First, Cook Unit 1 is one of the plants covered by a Westinghouse Owners' Group analysis that shows that pipe cracks will leak before they break so that the large LOCA load will not be present (Ref. 15). In light of that analysis, Cook Unit 1 does not presently have an obligation to address LOCA loads in the conservative manner analyzed by Westinghouse. Second, Westinghouse has shown in other cases (Ref. 16) that grid deformation has small consequences even when it is assumed to occur (less than 20°F increase in LOCA peak cladding temperature). Third, both the Exxon Nuclear and Westinghouse analyses mentioned above involved assumptions about the competitor's fuel design since neither Westinghouse nor Exxon Nuclear possesses complete details of each other's fuel design.

In light of the above circumstances and results -- particularly the large margin on the important guide tubes (thimbles) -- we conclude that all combinations of LOPAR, ENC, and OFA in Cook Unit 1 meet the appropriate mechanical loads requirements.

2.6 Wet Annular Burnable Absorbers

Cycle 8 will utilize a new burnable poison design, the Wet Annular Burnable Absorber (WABA), in 68 of the OFA's. The WABA rod design consists of annular pellets of aluminum oxide and boron carbide (Al_2O_3-B_4C) burnable absorber material encapsulated within two concentric Zircaloy tubings. The reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod. The topical report describing the WABA design (Ref. 12) has been recently reviewed and approved (Ref. 13), and the utilization of WABA rods in D. C. Cook 1 would thus be automatically approved subject to certain conditions described in the NRC approval of the generic topical report (those conditions concern surveillance and the analysis of core bypass flow). The WABA surveillance is discussed in Section 2.7 and the analysis of core bypass flow is discussed in Section 4.0 of this evaluation.

2.7 Post-irradiation Surveillance

As indicated in SRP* Section 4.2.II.D.3, a post-irradiation fuel surveillance program should be established to detect anomalies or confirm expected fuel performance.

The licensee states that a routine fuel inspection program will be implemented on the irradiated and discharged OFAs from the initial reload region (Ref. 2). The program involves visual examination on a representative sample of assemblies from the initial fuel region during each refueling until this fuel is discharged. Visual examination includes, but is not limited to, crud buildup, rod bowing, grid strap conditions, and missing parts. Additional fuel inspections would be performed if coolant activity or visual inspections indicate a need. We conclude that this satifies the fuel surveillance guidelines in the SRP 4.2.

As for the WABAs, the licensee agrees to have a supplementary surveillance program as described in Reference 13 if D. C. Cook Unit 1 is the first or second lead plant to discharge the WABAs. We find this acceptable.

2.8 Conclusion

We have reviewed the fuel assembly mechanical design for Cook, Unit 1, Cycle 8. We conclude that the Cycle-8 fuel mechanical design, which includes the Westinghouse 15x15 Optimized Fuel Assemblies (OFAs) and the Wet Annular Burnable Absorbers (WABAs), is acceptable.

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* SRP - Standard Review Plan

3. Nuclear Design

For this cycle, 80 of the ENC assemblies will be replaced by 80 Westinghouse 15x15 Optimized Fuel Assemblies (OFA). These assemblies are identical to the Westinghouse 15x15 LOPAR (low parasitic) assemblies except that five of the interior Inconel grids have been replaced by Zircaloy grids. The LOPAR assemblies have substantial operating experience in a number of plants. The Westinghouse OFA assemblies are nearly identical from a neutronics point of view to the ENC assemblies which they replace.

The nuclear design and analysis of the D. C. Cook core was performed with the Westinghouse Reload Safety Evaluation Methodology. This methodology has been previously employed for reload design in several reactors and we find its use acceptable for the present reload. The analyses were performed for a series of cycles which proceed from Cycle 8 to a core completely loaded with the Westinghouse OFA fuel. The neutronics parameters used as input to the safety analyses were then chosen to bound the values obtained from this series. In addition the analyses were done at a power level of 3411 MWt except for the LOCA analysis as noted above.

The licensee has included a listing of the neutronics parameters used in the safety analysis to provide bounding values against which cycle dependent parameters may be compared. We conclude that the nuclear design analysis is acceptable.

4. Thermal-Hydraulic Evaluation

The D. C. Cook Unit 1 Cycle 8 core consists of 80 Westinghouse 15x15 optimized fuel assemblies (OFA) and the 113 remaining Exxon 15x15 standard fuel assemblies. Sixty-eight (68) of the 80 OFA's employ the wet annular burnable absorber (WABA) poison rods. The OFA and standard fuel assemblies have been tested and the results show that they are hydraulically compatible with the pressure drops within 0.7 percent of each other.

The thermal-hydraulic analysis of this mixed core was performed using the improved thermal design procedures (ITDP) and the THINC IV code. The WRB-1 and W-3 CHF*correlations were used for the Westinghouse OFA and the ENC fuel assemblies, respectively. The ITDP, THINC IV code, and both CHF correlations have previously been approved by the staff. However, there are areas requiring additional evaluation regarding this transitional mixed core configuration. These areas are addressed as follows:

(a) The WRB-1 correlation was approved for the 17x17 OFA, and 17x17 and 15x15 standard LOPAR fuel assemblies with DNBR limit of 1.17 for R-grid. No CHF test data is available for the 15x15 OFA and, therefore, the application of the WRB-1 correlation to the 15x15 OFA is of concern. In response to staff questions, the licensee provided W 14X14 OFA CHF test data and additional proprietary information regarding the design of the 15x15 OFA. The 15x15 OFA design is virtually identical to the 15x15 R-grid design. A scaling technique was used in the 15x15 OFA grid design to ensure that the DNB performance is not affected by the OFA grid. This scaling technique has also been used for the design of the 17x17 and 14x14 OFA grids. In order to evaluate the effect of the geometry change on the accuracy of the WRB-1 correlation, Westinghouse also performed a statistical analysis using the T-tests and F-tests for the 17x17 standard/OFA data and the 14x14 standard/OFA data. The results show that the null hypothesis that the WRB-1 correlation predicts the DNB behavior of the OFA geometry with the same accuracy as the standard R-grid geometry can not be rejected at a 5% significance level. For the case where the F-test rejects the null hypothesis, the OFA data have an appreciably lower variance which is indicative of better correlation accuracy. Therefore, even though no 15x15 OFA CHF data is available, the statistical analysis performed by Westinghouse has provided the basis for the applicability of the WRB-1 correlation on the 15x15 OFA.

(b) The use of ITDP for the analysis of a transitional mixed core has been previously reviewed by the staff and approved with a condition requiring a

* CHF - Critical Heat Flux

penalty on DNBR to account for the uncertainty associated with the inter-bundle cross-flow in the mixed core.

The licensee has performed an analysis to determine the required penalty factor in the same manner approved for the 17x17 OFA/LOPAR mixed core analysis. The result shows that a 5% penalty is required on the OFA for the Cycle 8 transitional core.

(c) The Westinghouse WABA poison rod design is described in WCAP-10021, Revision 1 which has been approved by the NRC. In order to ensure no violation of the total core bypass flow limit, the total number of WABA rods in the core should be less than the upper limit established in Table 7.2 of WCAP-10021, Revision 1. Since only 68 OFA assemblies employ WABA with a total of 864 WABA rod for Cycle 8 core, the limit is not exceeded and is therfore of no concern.

(d) The Cycle 8 projected maximum assembly burnup is 36,800 MWD/MTU for the ENC fuel. The staff audit calculation has determined that the maximum gap closure will be 40.4% for the ENC fuel by the end of Cycle 8. Therefore, no rod bow penalty is required for the ENC fuel because investigations have shown that gap closure of less than 50% has no measurable effect on DNB.

(e) The core thermal-hydraulic analysis was performed by conservatively using 3411 MWt core power and 577.1°F average coolant temperature compared to the rated values of 3250 MWt and 567.8°F, respectively for the typical and thimble cells using the ITDP. The safety analysis DNBR limit is 1.69 for both typical and thimble cells. This safety limit is 28% higher than the design limit and the margin is more than enough to account for the rod bow penalty, the transitional mixed core penalty and any uncertainty associated with the application of WRB-1 on 15x15 OFA with DNBR limit of 1.17. For the ENC fuel, the W-3 correlation with DNBR limit of 1.30 was used, and the design safety limits are 1.58 and 1.50 for the typical cell and thimble cell, respectively. We conclude that the thermal-hydraulic analysis is acceptable.

5. Transient and Accident Analyses

All of the non-LOCA transients and accidents except startup of an inactive loop were reanalyzed to include three major design changes:

1. An increased power level of 3411 MWT
2. Use of the Improved Thermal Design Procedure with both the WRB-1 and W-3 DNB correlations
3. Increase of control rod scram time from 1.8 to 2.4 seconds. This change is necessitated by the reduction in ID of the thimbles in the OFA guide assemblies.

In addition, fuel temperatures were based on the revised PAD code and a 5 pcm/degree F MTC*at full power was used for heatup events. Standard Westinghouse codes and procedures were used for these analyses.

All the transients and accidents and the LOCA were done using approved methods and acceptable initial conditions. The results presented were acceptable since they did not violate the DNBR limit nor did they exceed the maximum pressure and temperature limits.

However, it is important to clarify that this SER approves the transient and accident analysis for operation of Cycle 8 only and in no way does it approve the plant to operate at the higher power level of 3411 MWt. If Cook 1 is planning to operate at the higher power level of 3411 an independent review of the LOCA and following transient accidents, is necessary.

* MTC - Moderator Temperature Coefficient

1. malfunction of the CVCS
2. loss of reactor coolant flow
3. locked rotor event
4. loss of external load
5. loss of normal feedwater
6. excessive heat removal due to feedwater system malfunction
7. excessive load increase incident
8. loss of all AC power to station auxiliaries
9. rupture of a steam pipe

The following transients and accidents have been reviewed at the higher power level and a detailed discussion is presented. These are:

1. bank withdrawal at low power
2. bank withdrawal at power
3. rod cluster control assembly misalignments
4. rod ejection accident

5.1 Bank Withdrawal at Low Power (Startup Accident)

The consequences of the insertion of reactivity at a rate of 75 pcm/second were calculated assuming a moderator temperature coefficient of 5 pcm/°F. This insertion rate is greater than that due to the withdrawal of the two sequential banks having the greatest combined worth at maximum speed (45 inches/minute). The peak heat flux during the transient is less than 50 percent of that at full power. We conclude that fuel thermal limits are not violated and that the analysis is acceptable.

5.2 Bank Withdrawal at Power

This event is analyzed at 100 percent, 60 percent, and 10 percent of full power. Minimum and maximum reactivity feedback effects are included as well as reactivity insertion rates up to values greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed. Trip occurs on high neutron flux for the high withdrawal rates and on the overtemperatureΔT trip for the low withdrawal rates. The minimum DNBR is 1.8 at full power, 1.85 at 60 percent power and 3.96 at 10 percent power. This meets the safety analysis limit of 1.69 for OFA and 1.58 for ENC fuel.

Based on the fact that approved analysis procedures and methods are used and that the resulting minimum DNBR values meet the relevant safety limits, we conclude that the analysis of the rod withdrawal event at power is acceptable.

5.3 Rod Cluster Control Assembly Misalignments

This category includes statically misaligned rods, dropped rods and dropped rod banks. The methodology used is described in document NS-EPR-2595, "Dropped Rod Methodology for Negative Flux Rate Trip Plants" which has been reviewed and approved by the staff.

Two static misalignment cases are analyzed - Bank D inserted with one rod fully withdrawn and one rod fully inserted with Bank D withdrawn. In the first case the calculation determines the amount by which Bank D may be inserted before fuel thermal limits are violated. The result is used in establishing the Technical Specification limits on Bank D insertion (other considerations usually determine these limits). The consequences of the single rod completely inserted while the rest of Bank D is withdrawn is analyzed by computing the resulting DNBR including the effect of the increased peaking factor. Fuel thermal limits are met for this case. Inspection of peaking factors obtained when a rod from another bank is on the bottom shows that the analyzed case is limiting.

Most dropped rods or dropped banks will result in a negative flux rate trip at about 2.5 seconds. Since power is decreasing at this point no thermal limits are approached and the operator follows procedures for a reactor scram. For rods with insufficient worth to cause the trip two cases are analyzed - reactor in manual control and reactor in automatic control. In the first case the reactor reaches a new steady-state configuration at a power not higher than the initial power. This case is bounded by the case of a static rod completely inserted with the D bank withdrawn.

In the second case the automatic controller will respond to the initial reduction in power by withdrawing rods which, in the limiting case, results in a power overshoot. In a typical case a 10 percent power overshoot occurs. The range of potential dropped rod cases has been investigated and in all cases thermal limits were not violated.

On the basis that approved methods were used and the results do not show a violation of fuel thermal limits, we conclude that the analysis of the rod misoperation events is acceptable.

5.4 Rod Ejection Accident

This accident postulates the rupture of a control rod drive mechanism housing and the consequent rapid ejection of the control rod from the core. This event has been analyzed by standard Westinghouse methods which have been shown to be conservative with respect to the three-dimensional calculations.

Four cases were analyzed-full power at beginning-and end-of-life and zero power at beginning-and end-of-life. Conservative values of ejected rod worth were used along with conservatively low values of delayed neutron fractions. The calculated maximum fuel enthalpy values ranged from 147 to 186 calories per gram. These values meet the acceptance criterion for this quantity of 280 calories per gram as given in Regulatory Guide 1.77. Less than 10 percent of the hot pellet melts in the two full power cases.

Less than 10 percent of the rods in the core experience departure from nucleate boiling during the event. No significant pressure surge occurs and the maximum pressure does not exceed that for emergency conditions as required by Regulatory Guide 1.77. We conclude that the analysis of the rod ejection event is acceptable.

6. Technical Specification

Changes have been proposed to the Cook Unit 1 Technical Specifications in order to account for the use of the Improved Thermal Design Procedure (ITDP), the analysis of non-LOCA events at 3411 MWt, and the introduction of Westinghouse OFA Fuel into the core. Each proposed change from Ref. 5 and 17 is discussed below.

Definition 1.27

A new power term, DESIGN THERMAL POWER (3411 MWt) is introduced in order to take advantage of the fact that safety analyses were done at 3411 MWt. In particular, the OvertemperatureΔT and OverpowerΔT trips have been recalculated for the increased power. The RATED THERMAL POWER, appearing in most specifications, is still 3250 MWt. We find this definition acceptable.

Figure 2.1-1

This figure provides the low points of the thermal power, RCS pressure and average temperature as reactor core safety limit for 4-loop operation to avoid violation of the design DNBR limit using the improved thermal design procedure. This figure is identical to Figure 3 of the Attachment C to AEP:NRC: 07450 in which the "fraction of design thermal power" is used in the abscissa and a conversion factor of (design thermal power/rated thermal power) is needed to convert the abscissa to "fraction of rated thermal power".

Table 2.2-1 (Items 7 and 8)

The algorithms for the Overtemperature ΔT and Overpower ΔT trips have been altered to reflect the use of the ITDP, the use of two different DNB correlations (WRB-1 for the Westinghouse fuel and W-3 for the ENC fuel) and the analyses at 3411 MWt. On the basis that these algorithms have been constructed by the methods which have been successfully employed on other Westinghouse reactors, we find them to be acceptable.

Bases for Specification 2.1.1 and 2.2.1

These bases have been changed to reflect the fact that two different fuel types having different DNBR limits and values of $F^N_{\Delta H}$ are present in the core and that the ITDP is used. In addition, values of the design and safety analysis values of DNBR for the two correlations are given. These changes are acceptable.

Specification 3/4.1.1.1

This specification has been modified to change the required shutdown margin from 1.75% to 1.60% reactivity change. The new value is consistent with the new steamline break analysis and is acceptable.

Specification 3/4.1.3.3

The rod drop time in this specification has been increased to $<$2.4 seconds. The change is necessary to account for the smaller diameter of the guide tubes in the optimized fuel assemblies. Since the safety analyses performed for D. C. Cook Unit 1 used the new value we find the proposed Technical Specification change acceptable.

Figures 3.1.-1 and 3.1-2

These figures show the rod group insertion limits for three-loop and four-loop operation respectively. Since these were obtained by using standard Westinghouse methodology we conclude that they are acceptable.

Specification 3/4.2

This specification has been expanded to include both Westinghouse OFA and Exxon (ENC) fuel. The format of this - the $F_q(z)$ specification - has been retained from the current specification and the OFA fuel specification has been cast in the same format with appropriate curves for the various parameters. The peaking factor of 1.97 for the Westinghouse fuel is consistent with the Cycle 8 LOCA analysis and is acceptable.

Specification 4/3.2.3

The specification is revised to include the $F^N_{\Delta H}$ value for the Westinghouse Fuel. The limiting values reflect the use of the ITDP. This is acceptable.

Specification 4/3.2.4

The editorial changes made here for clarity are acceptable.

Specification 4/3.2.6

The changes in this specification consisted in adding the Westinghouse OFA specifications and inserting a reference to the peaking factor limit report which contains the V(z) function. These changes are acceptable under the condition that the peaking factor limit report is transmitted to NRC for review 60 days prior to the scheduled startup date for the new cycle.

Table 3.3-1

A footnote has been added to certain of the FUNCTIONAL UNITS in this table to indicate that the provisions of Specification 3.0.4, dealing with entry into another operational mode is not applicable. This is consistent with Westinghouse Standard Technical Specifications and is acceptable. An addition to Action Statement 1 permits the bypassing of one channel for up to 3 hours to permit surveillance. This time is required because of the increased complexity of the surveillance procedures and is acceptable. Other changes in the table are editorial in nature and are acceptable.

Specification 4.10.1.2

This specification has been altered to make it consistent with Specification 3/4.1.3.3 (see above) and is acceptable.

7. Radiological Consequences

The licensee does not propose to increase the operating power level of the Unit 1 and does not propose to increase burnup for Cycle-8 beyond the 37,000 MWD/MTU batch average at discharge which we have previously considered and found acceptable generically. Therefore, the conclusions stemming from accident radiological analyses of record at 3250 MWt for fuel at 37,000 MWD/MTU (or the existing average burnup in Cook Unit 1, whichever is higher) are still valid. A complete radiological consequence analysis will be required for any proposed increase in the operating power level.

8. Spent Fuel Pool Cooling

The proposed reload involves fuel enriched to 4.00 weight percent U 235 This will result in increased burnup and thus decay heat production in the spent fuel pool when the fuel is eventually removed from the core, i.e., at the end of Cycle 10. We have reviewed the licensee submittal from the standpoint of decay heat load and spent fuel pool cooling capability and conclude that the increased enrichment of the fuel produces a negligible addition to the total decay heat production profile. Thus we conclude that the existing spent fuel pool cooling system is capable of handling the increased heat load.

9. Summary

We have reviewed the information submitted on Cycle 8 reload for D. C. Cook Unit 1. We find the Cycle 8 operation acceptable for the fuel system mechanical design, nuclear design, thermal hydraulic, transients and accidents, the Technical Specification proposed, and radiological consequences. In addition, we find the enriched fuel to have insignificant effect on the spent fuel pool cooling capability when the fuel is eventually discharged.

However, as stated in Section 5, the transient and accident and LOCA design are acceptable for the Cycle 8 only and operation at the higher power level of 3411 MWt will require that additional review be performed independent of this evaluation.

10. References

1. R. L. Tedesco (NRC) letter to T. M. Anderson (Westinghouse), "Reference Core Report 17x17 Optimized Fuel Assembly", May 22, 1981.

2. R. F. Hering (AEP) letter to H. R. Denton (NRC), August 31, 1983.

3. V. Stello (NRC) memorandum to R. DeYoung (NRC), "Evaluation of Westinghouse Report WCAP-8377, Revised Clad Flattening Model", January 14, 1975.

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

5. R. S. Hunter (I&MEC) letter AEP:NRC:0745C to H. R. Denton (NRC) on "Application for Reload License Amendment Using Westinghouse Optimized Fuel Assemblies", dated May 11, 1983.

6. W. J. Leech, D. D. Davis and M. S. Benzvi, "Revised PAD Code Thermal Safety Model", Westinghouse Electric Corporation Report WCAP-8720, Addendum 2 (Proprietary), October 1982. Submitted by E. P. Rahe, Jr. (Westinghouse) letter NS-EPR-2673 to C. O. Thomas (NRC) dated October 27, 1982.

7. J. V. Miller et al., "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations", Westinghouse Electric Corporation Reports WCAP-8720 (Proprietary), October 1976, and WCAP-8720, Addendum 1 (Proprietary), September 1979.

8. C. O. Thomas (NRC) letter to E. P. Rahe, Jr. (Westinghouse) on "Request Number 1 for Additional Information on WCAP-8720, Addendum 2", dated May 3, 1983.

9. E. P. Rahe, Jr. (Westinghouse) letter NS-EPR-2773 to C. O. Thomas (NRC) on "Response to Request for Additional Information on WCAP-8720, Addendum 2", dated June 2, 1983.

10. M. P. Alexich (I&MEC) letter AEP:NRC:0745F to H. R. Denton (NRC) on "Application for Unit 1 Cycle 8 Reload License Amendment: Addendums and Answers to NRC Questions", dated July 29, 1983.

11. E. P. Rahe, "Westinghouse ECCS Evaluation Model, 1981 Version", WCAP-9220-P-A (Proprietary), Revision 1, 1981.

12. E. P. Rahe, Jr. (W), letter to C. O. Thomas (NRC), "W WABA Evaluation Report", ECAP-10021, Revision 1, (Proprietary), October 18, 1982.

13. L. S. Rubenstein (NRC), memorandum for F. J. Miraglia, "SER of Westinghouse WABA Design", June 1, 1983.

14. "Lateral Core Seismic Analysis for ENC's 15x15 Reload Fuel Westinghouse Plants", XN-NF-52, September 1975.

15. "Mechanical Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack", WCAP-9558, Revision 2, May 1982; "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation", WCAP-9787, May 1981.

16. L. S. Rubenstein (NRC) memorandum, "SER Input for Millstone, Unit 2 Cycle 5 Reload", to T. M. Novak (NRC), February 18, 1982.

17. M. P. Alexich (IMEC) letter AEP:NRC:0745G to H. R. Denton (NRC) on "Additional Technical Specification Changes for Unit 1 Cycle 8," dated July 25, 1983.

18. R. F. Hering (IMEC) letter AEP:NRC:0745H to H. R. Denton (NRC) on "Further Answers to NRC Questions Concerning the Unit 1, Cycle 8 Reload" dated August 31, 1983.

C. ENVIRONMENTAL CONSIDERATION

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

D. CONCLUSION

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

Dated: September 20, 1983

Principal Contributors

A. Gill

W. Brooks

T. Chan

J. Mitchell

S. L. Wu

Y. Hsii

F. Burrows

J. Voglewede