Docket No. 50-316

June 18, 1984

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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. 64 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications and certain license conditions in response to your application transmitted by letter dated March 1, 1984, as supplemented by letters dated March 15, 23, and 28, April 19, May 4, 11, 17, 21, and 23, June 1, and 4 of 1984. This application was supplemeted by letters from Exxon Nuclear Company dated March 2, 13 and 16, May 7, 21 and 22, 1984. The changes to certain license conditions is supported by letters from the licensee dated September 9, 1983 and November 11, 1983.

The amendment approves the Cycle 5 reload, changes the surveillance requirements for ice condenser inlet doors, revises the containment isolation valve list, corrects reactor coolant system indicated temperature (average) to account for instrument uncertainties, changes the requirement for rod position indication during shutdown, adjusts the flow balance for the Safety Injection System, and makes several adminstrative, editorial change to update Technical Specifications.

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

Sincerely,

/s/DWigginton

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

Enclosures:
1. Amendment No. 64 to DPR-74
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
CParrish	DWigginton;ps	SVarga		GLainas
6/1/84	6/1/84	6/1/84	6/1/84	6/1/84

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated March 1, 1984, as supplemented by letters dated March 5, 23, 28, April 19, May 4, 11, 17, 21, 23, June 1 and 4, 1984, and the license condition supporting letters dated September 9, 1983 and November 11, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change in Technical Specifications is to become effective before entry into the applicable mode for the Technical Specification.

4. The license condition 2.C.3(p) is amended to read as follows:

2.C.3.(p) "Operation during and subsequent to Cycle 5 with Exxon Nuclear Company 17x17 fuel assemblies is permitted subject to the following conditions pending receipt and approval of confirmatory and other information on transients and accidents as noted in the Safety Evaluation issued for Cycle 5:

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

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

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

In addition to the conditions set forth above, the licensee is not authorized to operate in Cycle 6, modes 1 and 2, until it has satisfactorily resolved the issues identified in the Safety Evaluation issued for Cycle 5 and other Cycle 6 regulatory requirements."

5. 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: June 18, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 64 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revised Appendix A as follows:

Remove Pages	Insert Pages
B-21*	B-21*
B2-2	B2-2
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-11	3/4 2-11
3/4 2-12	3/4 2-12
3/4 2-16	3/4 2-16
3/4 4-14	3/4 4-14
3/4 5-6	3/4 5-6
3/4 6-20	3/4 6-20
3/4 6-21	3/4 6-21
3/4 6-29	3/4 6-29
3/4 6-31	3/4 6-31
3/4 6-39	3/4 6-39
3/4 6-40	3/4 6-40
3/4 10-5	3/4 10-5
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-4a	B 3/4 2-4a
B 3/4 2-4b	B 3/4 2-4b
B 3/4 2-5	B 3/4 2-5

*Included as convenience copy only.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

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

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

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

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

$F^N_{\Delta H}$ = 1.48 [1 + 0.2 (1-P)] (Westinghouse Fuel)

$F^N_{\Delta H}$ = 1.49 [1 + 0.2 (1-P)] (Exxon Nuclear Company Fuel)

where P is the fraction of RATED THERMAL POWER.

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

For Exxon Nuclear Company supplied fuel, an additional limitation on $F^N_{\Delta H}$ is applied to ensure compliance with ECCS acceptance criteria. This limitation is discussed in basis section 3/4.2.2 and 3/4.2.3 and does not affect the safety limit curve.

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.

RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

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

For: Westinghouse Fuel , for: Exxon Nuclear Company Fuel

$$R = \frac{F^N_{\Delta H}}{1.48\ [1.0 + 0.2\ (1.0-P)]}\ , \qquad R = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$$

And, $F^N_{\Delta H} \leq 1.36/P$ for Exxon Nuclear Company Fuel

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

and $F^N_{\Delta H}$ = measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ and flow, without additonal uncertainty allowance, shall be used to compare with limits.

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ above the allowable limit or with the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (as applicable):

a. Within 2 hours:

1. Either restore $F^N_{\Delta H}$ and the combination of RCS total flow rate and R to within the above limits, or

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

ACTION: (Continued)

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

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER Limit required by ACTION items a.2 and/or b above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation as defined above for $F^N_{\Delta H}$ and as shown on Figure 3.2-4 or 3.2-5 (as applicable) for RCS flow rate and R prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining $\geq$ 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

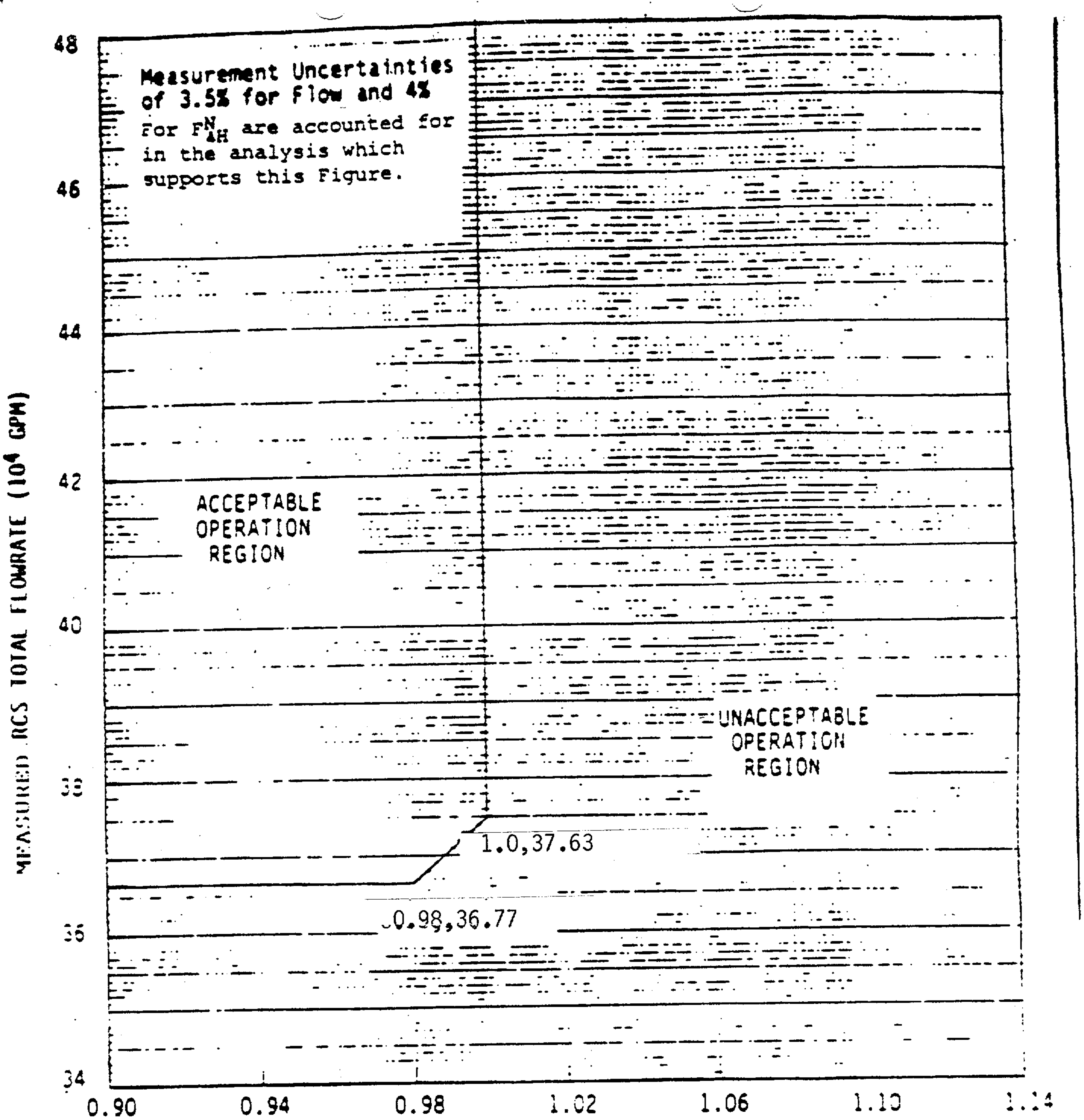
4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within the above limits and the combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-4 or 3.2-5 (as applicable):

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

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

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

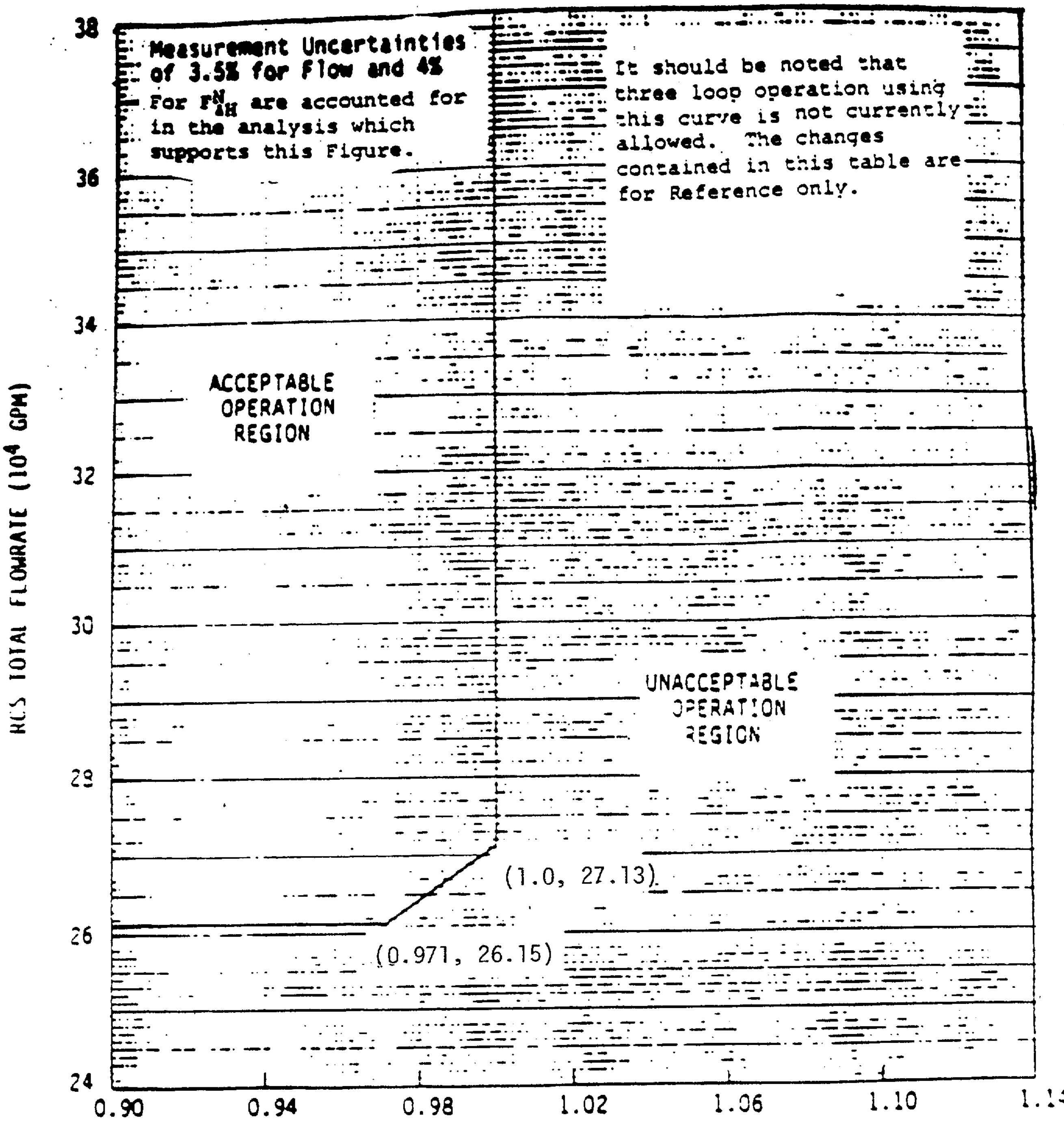
4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.



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

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

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



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

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

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation	 3 Loops in Operation***
Reactor Coolant System T_{avg}^{**}	$\leq$ 576.7°F (indicated)	$\leq$ 570°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*

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

** Indicated average of OPERABLE instrument loops.

*** It should be noted that three loop operation using this curve is not currently allowed. The changes contained in this table are for Reference only.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One of the containment atmosphere particulate radioactivity monitoring channels (ERS-2301 or ERS-2401),

b. The containment sump level and flow monitoring system, and

c. Either the containment humidity monitor or one of the containment atmosphere gaseous radioactivity monitoring channels (ERS-2305 or ERS-2405).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring channels are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Containment humidity monitor (if being used) - performance of CHANNEL CALIBRATION at least once per 18 months.

2. At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
1. 2-SI-141 L1	1. 2-SI-121 N
2. 2-SI-141 L2	2. 2-SI-121 S
3. 2-SI-141 L3	
4. 2-SI-141 L4	

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump*	Safety Injection System Single Pump**
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow $\geq$ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow $\geq$ 300 gpm
Loop 3 Boron Injection Flow 117.5 gpm	**Combined Loop 1,2,3 and 4 Cold Leg Flow (single pump) $\leq$640 gpm. Total SIS (single pump) flow, including miniflow, shall not exceed 700 gpm.
Loop 4 Boron Injection Flow 117.5 gpm	

*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow into each loop. Under these conditions there is zero mini-flow and 80 gpm simulated RCP seal injection line flow. The actual flow in each BI line may deviate from the nominal so long as the difference between the highest and lowest flow is 10 gpm or less and the total flow to the four branch lines does not exceed 470 gpm. Minimum flow (total flow) required is 345.8 gpm to the three most conservative (lowest flow) branch lines.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)		
67.	NCR-252	Primary Water to Pressurizer Relief Tank	≤ 10
68.	QCM-250	RCP Seal Water Discharge	≤ 15
69.	QCM-350	RCP Seal Water Discharge	≤ 15
70.	QCR-300	Letdown to Letdown Hx.	≤ 10
71.	QCR-301	Letdown to Letdown Hx.	≤ 10
72.	QCR-919	Demin Wtr. Supply for Refueling Cavity	≤ 10
73.	QCR-920	Demin Wtr. Supply for Refueling Cavity	≤ 10
74.	PCR-40	Containment Service Air	≤ 10
75.	RCR-100	PRZ Relief Tank to Gas Anal.	≤ 10
76.	RCR-101	PRZ Relief Tank to Gas Anal.	≤ 10
77.	VCR-10	Glycol Supply to Fan Cooler	≤ 10
78.	VCR-11	Glycol Supply to Fan Cooler	≤ 10
79.	VCR-20	Glycol Supply from Fan Cooler	≤ 10
80.	VCR-21	Glycol Supply from Fan Cooler	≤ 10
81.	XCR-100	Control Air to Containment	≤ 10
82.	XCR-101	Control Air to Containment Isolation	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)		
83.	XCR-102	Control Air to Containment Isolation	≤ 10
84.	XCR-103	Control Air to Containment	≤ 10
B.	PHASE "B" ISOLATION		
1.	CCM-451	CCW from RCP Oil Coolers	≤ 60
2.	CCM-452	CCW from RCP Oil Coolers	≤ 60
3.	CCM-453	CCW from RCP Thermal Barrier	≤ 30
4.	CCM-454	CCW from RCP Thermal Barrier	≤ 30
5.	CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	≤ 60
6.	CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	≤ 60
7.	ECR-31	Containment Airborne Rad Monitor	≤ 10
8.	ECR-32	Containment Airborne Rad Monitor	≤ 10
9.	ECR-33	Containment Airborne Rad Monitor	≤ 10
10.	ECR-35	Containment Airborne Rad Monitor	≤ 10
11.	ECR-36	Containment Airborne Rad Monitor	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
E. OTHER (Continued)		
18. PA-342	Containment Service Air	NA
19. NPX-151 VI	Dead Weight Calibrator	NA
20. N-160	N_2 to R. C. Drain Tank	NA
21. SM-1	Air Particle/Radio Gas Detect Return	NA
22. N-102	N_2 to Accumulators	NA
23. SI-171	Safety Injection Test Line	NA
24. SI-172	Safety Injection Test Line	NA
25. SI-194	Safety Injection Test Line	NA
26. PW-275	Primary Wtr. to Pre. Relief Tank	NA
27. CS-321	R.C.S. Charging	NA

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
E. OTHER (Continued)		
40. PPP-300	Instrument Penetration	NA
41. PPP-301	Instrument Penetration	NA
42. PPP-302	Instrument Penetration	NA
43. PPP-303	Instrument Penetration	NA
44. PPA-310 and PPA-311	Instrument Penetration	NA
45. PPA-312 and PPA-313	Instrument Penetration	NA
46. Blind Flange	Fuel Transfer Penetration	NA
47. Blind Flange	Ice Condenser Ice Supply	NA
48. Blind Flange	Ice Condenser Ice Return	NA
49. Blind Flange	In-Core Flux Thimble Access	NA

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

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

b. Demonstrated OPERABLE during shutdown (MODES 5 and 6) at least once per 9 months by:

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

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

3. Testing a sample of at least 50% of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component. The doors selected for determination of the "door opening torque" shall be selected to ensure that all doors are tested at least once during two test intervals.

SURVEILLANCE REQUIREMENTS (Continued)

4. Testing a sample of at least 50% of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component. The doors selected for determination of the "door closing torque" shall be selected to ensure that all doors are tested at least once during two test intervals.

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

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

a. Verified closed and that opening of each door is not impaired by ice, frost or debris by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 18 months by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

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

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

SPECIAL TEST EXCEPTION

POSITION INDICATOR CHANNELS SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The demand position indication system is OPERABLE* during the withdrawal of the rods, and

c. The rod position indicator is OPERABLE* during the withdrawal of the rods.

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the rod position indicator channels or the demand position indication system not OPERABLE*, or more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The demand position indication system shall be determined to be OPERABLE* by verifying the demand position indication system is responsive to a rod movement demand signal during withdrawal.

4.10.5.2 The rod position indicator channels shall be determined to be OPERABLE* by verifying the rod position indicator channels indicate rod movement during withdrawal.

*OPERABILITY for this Technical Specification is defined by the above Surveillance Requirements.

BASES

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

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

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

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

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3411 MWt. The limit on $F_Q(Z)$ is 2.04. The limit on $F^N_{\Delta H}$ is 1.36 for LOCA/ECCS analysis and 1.49 for DNB analyses. The analyses supporting the Exxon Nuclear Company limits are valid for an average steam generator tube plugging of up to 5% and a maximum plugging of one or more steam generators of up to 10%. In establishing the limits, a plant system description with improved accuracy was employed during the reflood portion of the LOCA Transient. With respect to the Westinghouse supplied fuel the minimum projected excess margin of at least 10% to ECCS limits will more than offset the impact of increase steam generator tube plugging.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

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

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the

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

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

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

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

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

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

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

$F_{\Delta H}^{N}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-4 and 3.2-5, RCS flow rate and $F_{\Delta H}^{N}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F_{\Delta H}^{N}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}^{N}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. The form of this relaxation for DNBR limits is discussed in Section 2.1.1 of the basis.

An additional limitation on $F_{\Delta H}^{N}$ applies to Exxon Nuclear Company fuel. This $F_{\Delta H}^{N}$ limit, in combination with the $F_Q(Z)$ limit, ensures compliance with the ECCS acceptance criteria. An allowance is included for an increase in $F_{\Delta H}^{N}$ at reduced power based on the following expression:

$F_{\Delta H}^{N} \leq 1.36/P$ (Exxon Nuclear Company Fuel)

where: P is the fraction of RATED THERMAL POWER.
The power dependence of this allowance is 1/P because the associated $F_{\Delta H}^{N}$ limit of 1.36 results from the LOCA analysis.

The more restrictive of the flow dependent DNBR $F_{\Delta H}^{N}$ limit and the LOCA $F_{\Delta H}^{N}$ limit for Exxon Nuclear Fuel Company fuel must be applied.

BASES: (Continued)

Figure B 3/4 2-2 illustrates the implementation of the limits as a function of power. A measured flow will result in a limiting value for R which must be obtained from Figure 3.2-4 or Figure 3.2-5. From this limiting R, a limiting $F^N_{\Delta H}$ can be obtained because:

Westinghouse Fuel

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

Exxon Nuclear Company Fuel

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

$$\text{Where: } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

Figure B 3/4 2-2 displays two limiting DNBR $F^N_{\Delta H}$ curves for Exxon Nuclear Company fuel for flows of 36.77×10^4 gpm, and 37.63×10^4 gpm. Also displayed on Figure B 3/4 2-2 is the limit on $F^N_{\Delta H}$ which results from the LOCA analysis for Exxon Nuclear Company fuel. $F^N_{\Delta H}$ must be maintained below and to the left of both the applicable DNBR $F^N_{\Delta H}$ limit and the LOCA $F^N_{\Delta H}$ limit.

For Westinghouse fuel there is only one $F^N_{\Delta H}$ limit. It must be obtained from the applicable relationships among R, $F^N_{\Delta H}$, P, and flow.

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

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of 3.5% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

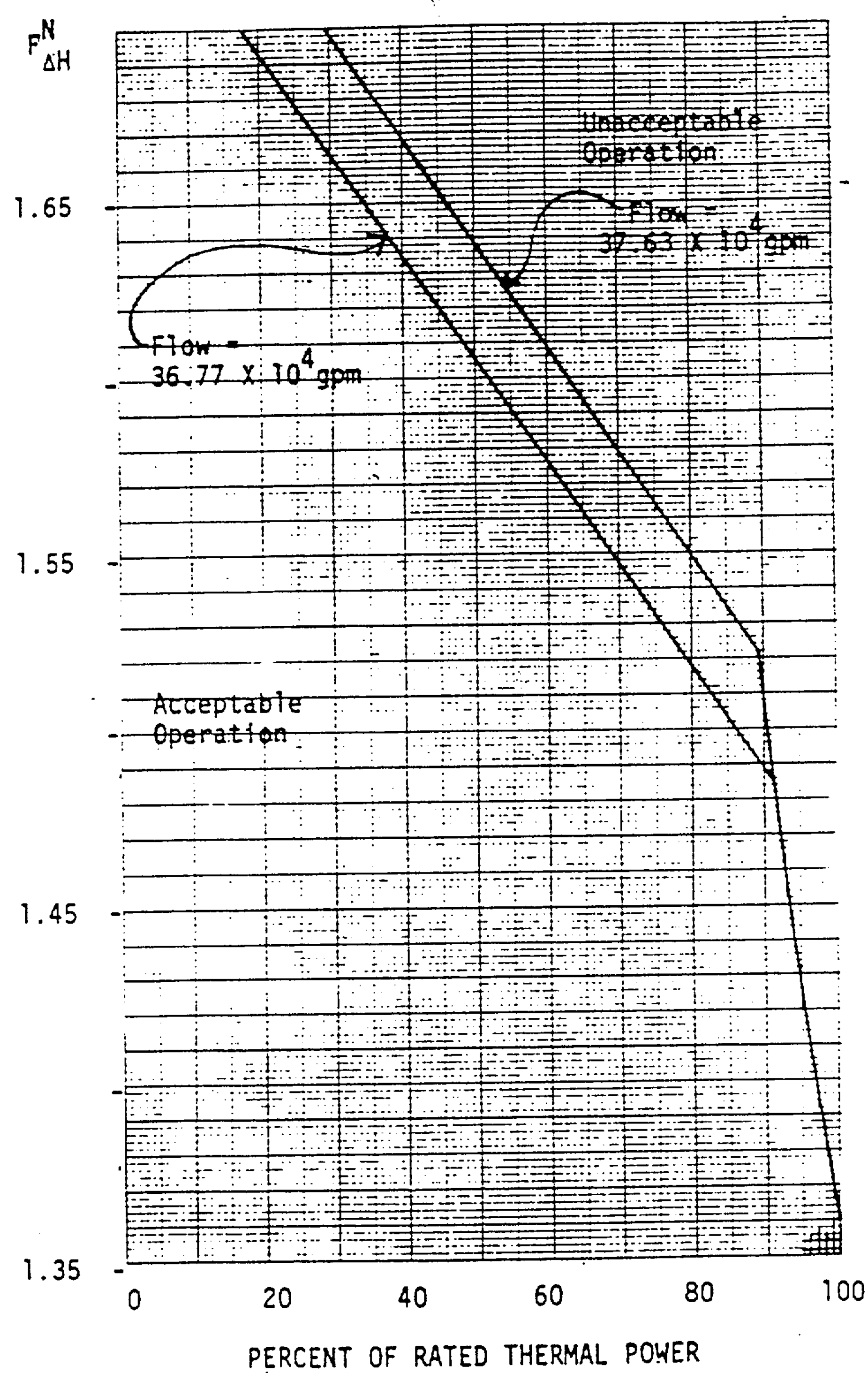


FIGURE B 3/4 2-2 ILLUSTRATIVE EXAMPLE OF

$F^N_{\Delta H}$ LIMIT VERSUS PERCENT THERMAL POWER FOR EXXON 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.

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 from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

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

"The four loop T_{avg} (Indicated) valve of 576.7° F is the equivalent of 578°F less the instrument inaccuracies."

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-316

I. Introduction

By letter dated March 1, 1984, the Indiana and Michigan Electric Company (the licensee) submitted an application to amend Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. This application was supplemented by licensee letters dated March 15, 23, 28, April 19, May 4, 11, 17, 21, 23, June 1, and June 4, 1984. The licensee was also supported in the Cycle 5 reload review by Exxon Nuclear Company (ENC). The ENC submittals which were subsequently adopted by the licensee, were dated March 2, 13, 16, May 7, 21, and 22, 1984. The licensee had earlier submittal letters dated September 8, 1983 and November 11, 1983 in response to Cycle 4 license conditions. The following evaluation is arranged as follows:

II. Cycle 4-5 Related Technical Specification Changes

- A. Ice Condenser Inlet Doors Surveillance and Containment Isolation Valves
- B. Safety Injection Miniflow Line Modifications
- C. Control Rod Position Indication - Rod Drop Measurements
- D. T_{avg} (Indicated)
- E. Editorial and Adminstrative Changes

III. Cycle 4 License Conditions

IV. Cycle 5 Reload Review

- A. Introduction
- B. Core and Fuel Performance Evaluation
- C. Transients and Accident Analysis
- D. Radiological Consequences
- E. Environmental Considerations
- F. Final No Significant Hazards Consideration

V. Conclusions

Each section of this evaluation may include a list of references to the submittals as well as other information used in the evaluation.

On April 11, 1984, the request for amendment was initially noticed (49 FR 14458) as a "Notice of Consideration of Issuance of Amendment to Facility

Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity For Hearing." No comments were received and no request for hearing was made within the 30 days normally allowed. On May 21, 1984, the licensee proposed a change to the original submittal and on May 24, 1984, the Federal Register published a subsequent notice on the proposed changes (49 FR 22008). In that subsequent notice, only 15 days were provided for comment. The Commission will make a final determination of no significant hazards consideration (see Section IV. F) on this subsequent change and a notice will be published in the Federal Register for opportunity of a hearing on that month. No comments were received in the 15 days period and no request for hearing has been made on the subsequent change.

II. Cycle 4-5 Related Technical Specification Changes

A. Ice Condenser Inlet Door Surveillance and Containment Isolation Valves

By letter dated March 1 and April 19, 1984, the licensee proposed certain changes to the facility Technical Specifications. This section addresses two of the proposed changes, concerning 1) ice condenser inlet door surveillance; and 2) containment isolation provisions for the containment air service penetration. The staff's evaluation of these proposed changes follows:

1) Ice condenser inlet door surveillance (Technical Specification 4.6.5.3.1)

   The proposed change would increase the surveillance interval for verifying that ice condenser inlet door opening/closing torque is within prescribed limits. The proposed change would also increase the size of the sample (i.e., number of ice condenser inlet doors) required to be tested during each surveillance test.

   The surveillance interval would be changed from 6 months (3 months during the first year) to 9 months. Since this testing cannot be performed during unit operation, the existing specification requires a unit outage every 6 months to perform the surveillance. Changing the interval to 9 months would allow this testing to coincide with outage to weigh ice baskets per Technical Specification 4.6.5.1.

It is also proposed that the sample size for verifying the "door opening torque" and "door closing torque" be increased from 25% to 50%. By testing a larger sample of doors, the change would result in each door being tested more frequently, i.e., at least once per 18 months rather than 24 months under the existing specification, despite the increased surveillance interval.

The surveillance history of the ice condenser inlet doors at D. C. Cook, Unit 2 was also reviewed. Inlet door surveillance has been performed 17 times over a six year period. During this time six reports were submitted to the NRC describing various inlet door deficiencies that were observed during the surveillance testing. These deficiencies included a door status annunciator failure, missing spring cotter pins and, in a few cases, door opening torque exceeding the 675 inch-pound acceptance criterion; only minor maintenance was required to restore the doors to an operable status. There was only one instance where a door could not be opened, and two instances where a door could only partially open. These occurrences, however, did not impair the safety function of the ice condenser. Since each test sample included 25% of the doors (48 doors total), the potential for a reduced ice condenser flow area is low. Also, the proposed change in surveillance testing will actually increase the test frequency of each door, which will provide additional assurance of ice condenser availability.

We have also considered the exposure to the individuals performing the tests and find that there should be no increase in the individual or cumulative exposures as a result of two examinations in an 18 month period as opposed to the current 3 examinations in the same period of time. With fewer entires, these should also be fewer releases from the containmet to permit entry.

2) **Containment air service penetration isolation barriers.**

The licensee proposes to use an automatic isolation valve (PCR-40), in lieu of a blind flange, outside containment, and a check valve (PA-343), in lieu of the manual valve (PA-243), inside containment as the containment isolation barriers for the containment service air line. The automatic isolation valve is actuated upon receipt of a Phase A isolation signal. The purpose of this change is to permit the use of the containment air service penetration above MODE 5. We find that the change in isolation barriers meets the isolation requirements of General Design Criterion 56, and, therefore, is acceptable. Accordingly, the proposed revision of Table 3.6-1 of the Technical Specifications to reflect the above design change is acceptable.

In summary, we conclude that the proposed changes to the Technical Specifications concerning 1) ice condenser inlet door surveillance; and 2) containment isolation barriers in the containment air service penetration, are acceptable.

B. Safety Injection Miniflow Line Modification

Background

The Indiana & Michigan Electric Company (IMECO) submitted a request to modify the piping geometry of the miniflow line for the D.C. Cook Unit No. 2 safety injection pumps by letters dated March 1 and 15, 1984. This modification has been previously performed for D.C. Cook Unit No. 1. As presently configured, the miniflow line for Unit 2 is comprised of both 1.5 inch and 0.75 inch diameter piping. The licensee has requested that it be allowed to replace the 0.75 inch diameter piping with 1.5 inch piping, thereby making the entire piping in system of one diameter. The purpose of this modification is based on economic and maintenance considerations. By maintaining both Units 1 and 2 as similar as possible, the licensee is able, in many cases, to apply one analysis to both units.

Increasing the miniflow line piping diameter doubles its flow rate from 30 gpm to 60 gpm. The increased flow is beneficial to the SI pump when operating in the shut-off configuration in that it reduces the temperature rise through the pump. This provides an added benefit of increased pump reliability by allowing smoother operation at reduced temperatures.

Increasing the miniflow coolant rate has a negative influence on ECCS performance in that it reduces the injected flow to the reactor coolant system. At runout conditions, the ECCS injection rate is decreased from 63.0 lbm/sec to 61.6 lbm/sec. At the other extreme, the ECC injection rate at 1314.7 psia is reduced from 19.0 lbm/sec to 16.1 lbm/sec. Since only the SI is influenced by the proposed hardware modification, the impact on large break LOCAs is insignificant (total ECCS flow, not including accumulator injection, is reduced from 463.0 lbm/sec to 461.6 lbm/sec). This would have negligible impact on the calculated peak clad temperature for the large breaks.

For the limiting small break LOCA, however, IMECO has determined that the peak clad temperature would increase by about 87°F. This analysis was conservatively calculated for Unit 1, and was submitted by IMECO as applicable to Unit 2.

To demonstrate that the temperature increase for Unit 1 was applicable to Unit 2, IMECO had the reactor vendor (Westinghouse) confirm that the ECCS pump characteristics for both Unit 1 and Unit 2 are identical. Having anticipated the desirability to modify the geometry of the miniflow line for Unit 2 as well, the limiting small break LOCA for Unit 1 was analyzed at the Unit 2 power rating (3411 MWt versus 3250 MWt). In addition, the linear peak heat generation rate was analyzed at 16.67 kw/ft for Unit 1 (Unit 2 is rated at 12.88 kw/ft). Since the linear heat generation rate for Unit 1 is significantly greater than that for Unit 2, the calculated heat up rate would be conservative when applied to Unit 2.

The applicability of the Unit 1 calculation to Unit 2 was also based on comparison of the volumetric fuel heat generation rate for the total core. The volumetric heat generation rate for Unit 1 was calculated at 9887 kw/ft^3 of fuel and for Unit 2 at 9835 kw/ft^3 of fuel. The total volume of coolant in the core was also calculated to be nearly identical (614.8 ft^3 and 613.0 ft^3 for Units 1 and 2, respectively). With respect to the remaining primary system coolant volume, both plants are identical.

The reduction of ECC injection by the SI pump resulted in an additional 6 inches of calculated core uncovery (5.5 ft versus 5.0 ft). This corresponded to a 10 second delay in coolant recovery of the core (838 versus 848 seconds). The consequential increase in peak clad temperature was 87°F. With the present calculated small break peak clad temperature of 1668°F, the Unit 2 core response for the limiting small break LOCA is expected to be less than 1750°F. This is well below the 2200°F licensing limit.

CONCLUSION OF THE MINIFLOW LINE REVIEW

We have reviewed the submittal by the Indiana & Michigan Electric Company to increase the pipe diameter of the miniflow line to the injection pumps. The acceptability of the miniflow line modification is based on the Unit 1 LOCA analysis and it's applicability to Unit 2. We find the analysis and applicability acceptable, and therefore find acceptable, the requested modification of increasing the cross sectional diameter of the miniflow line from 0.75 inch to 1.5 inch. We requested, however, that the SI pump flow characteristic be confirmed to be consistent with the analysis assumptions prior to full power operation. The licensee has agreed to perform this test prior to startup.

II.

C. Control Rod Position Indication - Rod Drop Measurements

In a letter from M. P. Alexich to H. R. Denton, dated March 15, 1984, the licensee requested changes to Technical Specification 3.10.5 which defines control rod position indication requirements during rod drop time measurements. Basically, the objective of the licensee's request was to remove the requirement that the rod position and demand position indicators be in agreement within 12 steps during withdrawal of the rods for the rod drop test. In the cold shutdown modes this specification cannot readily be met because the calibration of the rods is normally performed hot. Since the accident analysis from which the 12 step requirement stems is at power, it is not necessary to impose stringent requirements on rod position indication during the rod drop test, when the reactor is not critical.

At our request, the licensee modified his proposed changes substantially to include use of both position indication systems during withdrawal of the control rods. The Specification was submitted in a letter from R. F. Hering to H. H. Denton, dated June 4, 1984. This modification requires the position indication system to be operable but only so far as to indicate rod movement during rod withdrawal. We find the revised Technical Specification to be acceptable.

II.

D. Tavg (Indicated)

By letter dated March 1, 1984, the licensee proposed to change the Reactor Coolant System T_{avg} value given in Technical Specification Table 3.2-1 for four loop operation from 578°F to 576.7°F (Indicated) to account for instrument uncertainties. This change would provide a Technical Specification number consistent with plant operations. This change is also consistent with the analysis performed for the reload review as stated by the licensee. We agree that the Technical Specification valve for T_{avg} should account for instrument inaccuracies so that this valve is of direct use for plant operations. This proposed Technical Specification is acceptable, however, we will add a statement to the Bases Section to reflect the 578°F value and the new 576.7°F (Indicated) value.

E. Editorial and Administrative Changes

In the March 1, 1984 letter, the licensee proposed several changes to Technical Specifications to delete obsolete statements and clarify others. Technical Specification 3.6.5.3 on Ice Condenser Doors has been changed to delete reference to surveillance required once per three months after the ice bed is loaded. This requirement has been met and is no longer required. The footnote on this Technical Specification is no longer applicable and we agree it should be removed. Other changes to this Technical Specification were addressed in Section A above.

II. Cont.

The surveillance requirement for this Technical Specification has also been clarified to show the intent of examination of the intermediate deck doors. The requirement to verify the doors are free of frost accumulation has been changed to verify that opening of each door is not impaired by ice, frost, or debris. This change is more specific in its requirement and is acceptable.

The licensee also proposes an editorial change to Technical Specification 3.4.6 on Reactor Coolant System Leakage Detention System. In Amendment 43, the radioactive monitors were incorrectly numbered for Unit 2. The change would correct this and would delete the footnote which is no longer applicable. We find these changes accepable.

III. Cycle 4 License Conditions

A. Introduction

As a result of the Cycle 4 reload review and as addressed in Amendment No. 48 dated January 14, 1983, the Donald C. Cook Nuclear Plant, Unit No. 2 Facility Operating License was amended with, among others, the following conditions:

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

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

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

By letter dated September 8 and November 11, 1983, the licensee provided information to address these concerns.

B. Evaluation of License Conditions

1. Seismic Analysis

As described in the Condition 1, the licensee was required to complete a seismic analysis for a mixed core of ENC and Westinghouse fuels to comply with the fuel assembly structural acceptance criteria in Appendix A to SRP 4.2. Instead of a full-blown computer code analysis, we agreed that a comparative method may be used for demonstrating that ENC fuel assemblies are similar in strength to the Westinghouse 17x17 fuel assemblies already in the core and are capable of withstanding the design-basis seismic events.

By letter dated November 11, 1983, the licensee submitted a comparative analysis with a report (XN-NF-739) entitled "Seismic Evaluation of Exxon Nuclear 17x17 Assemblies in Westinghouse PWR's". The report, in particular, addresses two assembly components, the spacer grid and the guide tube. Our evaluation is thus based on the findings of these two assembly components.

1.1 Physical Properties Comparison

The comparison of physical properties including geometry between ENC 17x17 and Westinghouse 17x17 show little variation. Most characteristics are identical or nearly identical for the two designs. Thus, the physical parameters input for the seismic analysis have no major differences in the two different fuel designs. Based on our review, we agree with this conclusion.

1.2 Mechanical Properties Comparison

Two important mechanical properties, natural frequency and through-grid stiffness, need to be addressed for structural response under seismic load.

The fundamental fréquency of an ENC fuel assembly is very close to that of a Westinghouse assembly, based on room temperature measurements. ENC concluded that the natural frequencies of the two designs are sufficiently close that dynamic response would not be significantly affected. Based on our review, we agree with the finding.

As for through-grid stiffness, the ENC assembly stiffness is much less than the Westinghouse assembly stiffness. However, ENC demonstrated by calculation (XN-NF-739) that a smaller through-grid stiffness resulted in lower calculated loads on fuel assemblies for either an entirely ENC-fueled core or mixed core of ENC and Westinghouse fuel assemblies. A core of all Westinghouse assemblies had the highest maximum loading. Therefore, ENC concluded that the Westinghouse assemblies' structural response formed a conservative basis for establishing design margins for ENC fuel assemblies. We conclude that this is an acceptable way of considering design margins for ENC assemblies.

1.3 Spacer Grid

The ENC spacer grid strength was obtained at room temperature conditions using an approved method. After correcting for reactor temperature conditions and minimum spacer thickness, the minimum seismic strength of an ENC spacer grid was compared to the maximum through grid seismic loading calculated by Westinghouse for a conservative comparison as discussed in the preceding section.

The result shows that the ENC spacer grid has a conservative margin for a seismic design event calculated by a bounding Westinghouse analysis. We, therefore, conclude that the ENC spacer grid has adequate strength for design-basis seismic loading.

1.4 Guide Tube

ENC uses a finite-element model to calculate the loading on guide tubes. The maximum guide tube stress occurs near the center of the assembly. The guide tube stresses are all significantly below the allowable stress limits, which are derived according to Section III of the ASME code. Therefore, the licensee concludes that the guide tubes have adequate strength during seismic loading. We find the results acceptable.

1.5 Conclusion

Based on adequate seismic strength for the ENC fuel assemblies including components of the spacer grids and guide tubes, we conclude that the license condition requiring compliance with structural acceptance criteria in Appendix A to SRP 4.2 has been satisfied and can be removed for future ENC fuel reloads to D. C. Cook Unit 2.

2 Rod Drop Accident Analysis

The license condition requiring compliance with the operating restrictions imposed by the rod drop accident analysis is a restriction which existed prior to Cycle 4 or 5 and with other operation reactors as well. The inclusion as a license condition here is not due to unique requirements on D.C. Cook Unit No. 2 and therefore the license condition can be removed without effecting the restrictions on operation. The general restriction imposed by the rod drop accident analysis will continue until such time as the generic review of this event has been completed and any analyses required as a result of that review are performed. Removal of this license condition is acceptable.

3 RODEX2-related Analysis

With the approval of the RODEX2 code (XN-NF-81-58, Revision 2), the licensee submitted new results regarding to cladding strain, oxidation, and PCI (Alexich, September 8, 1983). The results show that new analyses still conform to SRP 4.2. We thus conclude that the license condition on RODEX2-related analysis can be removed for future ENC fuel reloads.

3.0 Summary

We have reviewed the licensee's submittals to resolve the Cycle 4 license conditions. We conclude that license conditions on seismic analysis, rod drop, and RODEX2 analysis can be removed.

IV. Cycle Reload Review

A. Introduction

In License Amendment No. 48 issued January 1, 1983, the Cycle 4 reload review was approved for the initial core loading with Exxon Nuclear Company (ENC) fuel in Cook Unit No. 2. In support of the Cycle 5 reload review, the licensee has submitted a reload safety analysis report, a transient analysis report for operation with 5% steam generator tube plugging, and other documents which are referenced in the following evaluation.

B. Core and Fuel Performance Evaluation

The D. C. Cook-2 reactor contains 193 fuel assemblies each having a 17x17 fuel rod array. Each assembly contains 264 fuel rods, 24 RCC guide tubes and one instrumentation tube. The Cycle 5 core will consist of 164 Exxon Nuclear Company (ENC) assemblies (of which 92 will be fresh) and 29 Westinghouse assemblies. The Cycle 5 burnup has been projected to be 17,900 MWD/MT at a core power of 3411 MWT.

The fuel and nuclear core design, the thermal-hydraulic and transient analyses, and Technical Specifications for the D. C. Cook Unit 2 Cycle 5 have been reviewed. Specific aspects of the safety analysis are discussed in the following sections.

2.0 EVALUATION OF FUEL MECHANICAL DESIGN

2.1 Introduction

The D. C. Cook Unit 2 Cycle 5 reload is composed of both Westinghouse and ENC 17x17 fuel. While the commercial utilization of the Westinghouse fuel has been extensive, the ENC 17x17 fuel was used for the first time in Cycle 4 of D. C. Cook-2. The ENC 17x17 assembly design is similar to the previously used ENC 14x14 design (Ref. 5) except for an increased number of guide tubes and spacers, which are intended to provide additional strength and stiffness. The general topical report describing the ENC 17x17 design, XN-NF-82-25 (Ref. 6), has been approved (Ref. 7) as a document suitable for referencing in safety analyses. Where the methods used in the Cycle 5 analysis are unchanged from previously approved methods, it is concluded that no additional review is required for Cycle 5 operation.

2.2 General Description

The ENC 17x17 bundle array contains 264 fuel rods, 24 guide tubes, and 1 instrument tube. The fuel rod has a slightly smaller diameter and pitch than the ENC 14x14 PWR design. The grid spacers have thicker structural members and are deeper overall for greater assembly rigidity. The design has a "quick-removable" upper tie plate design to facilitate inspection and reconstruction of irradiated assemblies. The assembly design is described in Section 4.0 of Reference 6 with additional information provided in response (Ref. 8) to staff questions on that document.

On the grounds that the ENC 17x17 design has received generic approval, the design is approved for the D. C. Cook 2 Cycle 5 reload, subject to the limitations on that generic approval. Those limitations and their consequences are addressed below along with plant-specific concerns.

2.3 Rod Bowing

Fuel rod bowing is a phenomenon that alters a nominal spacing between adjacent fuel rods and as a result perturbs local heat transfer to the coolant and local nuclear power peaking. Exxon has submitted a topical report (Ref. 9) describing the methods to be used for estimating the magnitude of fuel rod bowing and the resulting effects on DNBR and power peaking. These methods have been approved (Ref. 10) for application to the ENC 17x17 fuel design. The licensee has also stated that the Cycle 5 Westinghouse fuel has adequate margin in FQ and DNBR to off-set the rod bowing penalty (Ref. 11).

A rod bowing evaluation was performed for Cycle 5 using these generically approved methods. The results indicated that there exists sufficient margin between the DNBR limit and the minimum DNBR even with the calculated rod bow penalty. Also, the calculations indicated that the allowance for total power peaking uncertainty was sufficient to account for rod bowing. We find this analysis acceptable.

2.4 RODEX--Strain Oxidation, PCI Analyses

As pointed out in the generic safety evaluation (Ref. 7) of Exxon's 17x17 fuel assembly analysis report (Ref. 6), the RODEX 2 thermal analysis code (Ref. 12) was used in the design analysis of several important fuel performance phenomena including cladding strain, external corrosion (oxidation), fuel rod internal pressure, fuel pellet temperature, and pellet/cladding interaction. During the review of the D. C. Cook-2 Cycle 4 safety analysis, it was noted that the licensee was required to redo the cladding strain, oxidation and PCI calculations for the ENC 17x17 fuel design with the approved version of RODEX 2 prior to the end of Cycle 4 (Ref. 13). This analysis has been completed and indicates that the cladding strain, oxidation and PCI satisfy the acceptance criteria.

2.5 Cladding Collapse-Review Criterion

The licensee has performed an analysis using a new calculational procedure developed by Exxon Nuclear Company (ENC) to predict the occurrence of cladding collapse in the Exxon 17x17 fuel. The licensee, using this new analysis method, has demonstrated that Exxon fuel in D. C. Cook Unit 2 will not collapse. We have reviewed the licensee's analysis and the proposed model on which it is based and find both to be acceptable. Our evaluation of the proposed model is given below.

The analysis of the creep collapse of the Westinghouse fuel in D. C. Cook Unit 2 is based on Westinghouse analytical methods which have been previously approved. The licensee has stated that there will be no collapse of either the Exxon or Westinghouse fuel.

Evaluation of Proposed ENC Creep Collapse Criterion

Part of any safety analysis for fuel rod operation in a commercial reactor is the concern that the cladding, as it creeps inward due to the difference between the reactor coolant pressure and the internal fuel rod pressure, will collapse into a pre-existing axial gap along the fuel column. The axial gap can be formed as the fuel column length decreases due to fuel densification at the same time as a pellet somewhere in the fuel pellet column "hangs up", i.e., becomes stuck in one axial position and cannot move downward as the fuel column below densifies and shortens.

Past creep collapse analyses have required that no collapse occurs during the irradiation life of a fuel rod with the assumption that the fuel rod is a tube of infinite length with no fuel pellets to prevent its collapse. Past creep collapse analyses have also assumed that an axial gap exists in all fuel rods within a reactor core that is large enough to allow cladding to collapse. Obviously, these are conservative assumptions. They were initiated by NRC at a time when fuel densification, pellet hang up and creep collapse were not understood (Ref. 33).

The licensee has proposed a new criterion for analyzing creep collapse which relies on the elimination of axial gap formation as a viable mechanism (Ref. 34). This is accomplished by demonstrating that pellet hang-up will not occur early in life due to fuel-clad gap closure when fuel densification is still active. Fuel-clad gap closure later in life is not a concern since fuel densification is complete and thus no mechanism exists for axial gap formation in ENC designed fuel. This proposed criterion will be referred to as the "Proposed" method.

The two areas of review of the "Proposed" method were:

(1) the likelihood of pellet hang up due to mechanisms other than fuel-clad gap closure, e.g., pellet chips and pellet cocking, and

(2) the adequacy of the "Proposed" method of predicting pellet hang-up and the margin of conservatism in this method.

The first area of concern, pellet hang-up due to pellet chips and pellet cocking, has been addressed by ENC through the examination of several hundred fuel rods (Ref. 35). ENC has examined 434 BWR fuel rods by axial gamma scanning and found only nine rods with relatively small axial gaps (<0.145 inches). These gaps were attributed to thermal differences between the hot and cold conditions and were not believed to exist in the hot condition, nor were these gaps found to be permanent. ENC has also examined 4,690 PWR fuel rods visually for crud pattern irregularities with assembly burnups ranging from 7 MWd/kgM to 46 MWd/kgM. ENC has demonstrated that irregularities in crud patterns can be associated with axial gap formations with a detection limit of at least 0.4 inches. No such crud pattern irregularities were observed in the 4,690 PWR rods examined. It should be noted that definitive crud patterns in PWR fuel rods do not form until two cycles of operation or more. However, even if it is assumed that only one-third of the rods examined have definitive crud patterns, it can be concluded that approximately 1500 fuel rods have not shown axial gap formation, by examination of crud patterns, at a detection limit significantly below that necessary for cladding collapse. Consequently, through the examination of at least 2,000 ENC fuel rods, ENC has found no axial gaps near the size necessary for creep collapse and no evidence of permanent pellet hang-up and axial gap formation due to pellet chips and/or cocking. The lack of axial gap formation in ENC designed fuel can be attributed to the relatively stable fuel and prepressurized design used by ENC for PWR rods.

The second area of concern, the adequacy of the "Proposed" method and the margin of conservatism, has been addressed by ENC in two ways. The first is by comparison against data from fuel rods irradiated in the Ginna reactor, some of which experienced in-reactor creep collapse. The "Proposed" method for creep collapse (Ref. 34) predicted gap closure early-in-life, indicating that creep collapse was likely for these rods. This comparison has indicated that the "Proposed" method is at least best estimate in nature but it has not provided a measure of the conservatism that exists in this methodology. The measure of conservatism is also not apparent by close examination of the elements that exist in the "Proposed" method, because it cannot be easily related to the hot fuel-clad gap conditions that exist in reactors. In order to provide a measure of conservatism, ENC has provided a mechanistic method of predicting in-reactor hot gap closure to serve as a standard for comparison against the "Proposed" method (Ref. 36). ENC has labeled the mechanistic method as "Best Estimate"; however, this is misleading, because conservatism has been introduced in the input values and the calculational models to provide a conservative bound on the calculated gap closure. This "Best Estimate" method has been reviewed and found to have an appropriate margin of conservatism for calculating gap closure and thus pellet hang-up. The "Best Estimate" method has predicted pellet hang-up and axial gap formation for the Ginna Rods. Also, comparison of the "Best Estimate" and the "Proposed" methods to a variety of ENC designs has indicated that the latter will predict pellet hang-up and thus creep collapse before the "Best Estimate" method. Consequently, the "Proposed" method is the more conservative of the two methods and because the "Best Estimate" method has been judged to have an appropriate margin of conservatism, the "Proposed" method is also judged to have an appropriate margin of conservatism.

In summary, the use of the above ENC methodology for D. C. Cook Unit 2 for determining creep collapse is acceptable based on (1) ENC's examination of several hundred ENC fuel rods with no evidence of permanent axial gaps in the fuel column stacks due to early-in-life pellet hang up, and (2) an appropriate margin of conservatism in the "Proposed" method.

2.6 Fuel Centerline Temperature

According to information presented in Reference 16, the peak UO_2 centerline temperature was calculated to be 3500°F using the Exxon GAPEX thermal analysis code (Ref. 17). Since this temperature was calculated by an approved code and is well below the UO_2 melting temperature of about 5050°F, we conclude that the no-centerline-melting criterion is satisfied for ENC 17x17 fuel for D. C. Cook 2 Cycle 5 operation.

2.7 Rod Pressure

As indicated in Exxon's generic analysis, XN-NF-82-25, (Ref. 6), the ENC 17x17 fuel rods are designed such that the internal gas pressure of the fuel rods does not exceed coolant pressure. The thermal design analysis, described in ENC's generic report, was performed with the approved RODEX 2 method (Ref. 12) and demonstrates that the rod internal pressure criterion is satisfied for the most limiting rod during Cycle 5.

2.8 On-Line Monitoring

Section 4.2.II.D.2 of the Standard Review Plan indicates that the on-line fuel rod failure detection methods (instrumentation and procedures) should be reviewed. Because of the newness of the ENC 17x17 fuel design that will be used in D. C. Cook 2 during Cycle 5, there is a need to assure that any unexpected failures of that fuel (as well as the older W fuel) would be readily detected. The instrumentation (failed fuel detection system) is described in the D. C. Cook 2 FSAR. The D. C. Cook 2 Technical Specification 4.4.8 Surveillance requires a beta-gamma analysis of the primary coolant every 72 hours. Moreover, the licensee has a procedure (Ref. 16) that results in the performance of such an analysis every 48 hours. We find this acceptable.

2.9 Post-Irradiation Examination (PIE)

A post-irradiation fuel surveillance program should be established, as stated in SRP Section 4.2.II.D.3, to detect anomalies or confirm expected fuel performance. For a new fuel design, such as the ENC 17x17 fuel, such a program should include appropriate qualitative and quantitative inspection to be carried out at interim and end-of-life refueling outages. Similar inspections of the ENC 17x17 fuel were recommended in the approval of the Exxon Rod Bowing Methodology (Ref. 10). In a recent submittal (Ref. 18) the licensee stated that visual examinations would be made on the ENC 17x17 fuel after its first cycle of operation (Cycle 4). The examination would include binocular inspections of 50% of the assemblies as they are being transferred to the spent fuel pool following Cycle 4 operation (all the assemblies are to be off-loaded, even those that will be reinserted for Cycle 5). In addition, a more detailed underwater television or periscope examination will be performed on each face of four Exxon assemblies from this batch at EOC 4. The results of these inspections were not available for evaluation for the present Cycle 5 reload application review. We will review the results of these inspections when they become available. During subsequent refuelings AEP plans to visually inspect those assemblies from the first batch of ENC 17x17 fuel that will be permanently discharged. We conclude that the proposed PIE program satisfies the intent of the Standard Review Plan and is, therefore, acceptable.

2.10 Seismic-and LOCA Loadings

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

In 1975 an additional loading due to asymmetric blowdown forces on PWRs during LOCA was identified. As a result, NRC issued NUREG-0609 (Asymmetric Blowdown Loads on PWR Primary Systems) to address this concern and required all PWRs to submit such an analysis for evaluating fuel assembly structural adequacy. Westinghouse A-2 Owners Group including D. C. Cook Units 1 and 2 submitted two reports, WCAP-9558, Revision 2 and WCAP-9787, for staff review in response to NUREG-0609. They claimed that a rapid blowdown is very unlikely because the stainless steel primary piping would leak before it breaks during a LOCA; therefore, the reports argue that the requirements of NUREG-0609 can be waived.

Although the review of Westinghouse A-2 Owners Group reports has not yet been completed, no structural response analysis is presently being required. However, there still remain the original FSAR requirements of analyzing seismic effects on fuel assemblies for D. C. Cook Unit 2. The coming Cycle 5 core (mixed Westinghouse and Exxon fuels) and future cores (mixed and pure Exxon fuels) of D. C. Cook 2 must, therefore, be shown to be structurally adequate regarding the seismic effect because the original analysis did not cover Exxon fuel.

On behalf of the licensee, Exxon submitted information about the structural adequacy of the ENC 17x17 fuel assemblies to respond to this requirement in a letter dated December 20, 1982 (G. F. Owsley to S. L. Wu). In that submittal, Exxon stated that the resulting loads on 17x17 fuel assemblies are expected to be lower than those on 15x15 fuel assemblies due to the increased number of grid spacers, and tests of grid spacers show greater strength for ENC 17x17 fuel than for ENC 15x15 fuel. Exxon thus concluded that the 17x17 fuel assembly is adequately designed to withstand earthquakes and LOCA as compared to the 15x15 fuel assembly, which was analyzed in the report XN-NF-76-47. Although the staff reviewed that report, only the analytical methods were approved because the results presented were not found to be generically bounding. Therefore, plant-specific analyses must be performed to account for Cook 2 core accelerations and to determine loads on fuel rods, guide tubes, and other fuel assembly components.

Consequently, as a license condition for Cycle 4, the licensee was required to submit a plant specific analysis within one year to account for Cook 2 core accelerations and to determine loads on fuel rods, guide tubes, and other fuel assembly components (Ref. 13). The licensee submitted a plant specific structural analysis (Ref. 19) using the approved methods described in XN-NF-76-47. This analysis considers both single fuel Type (ENC) and multiple fuel type (both W and ENC) cores. The staff review of this analysis has been completed and an SER approving the analysis has been prepared (Ref. 37). This satisfied the requirements discussed in this section.

2.11 Fuel Mechanical Design Summary

The Exxon fuel design analysis for D. C. Cook 2 Cycle 5 operation described in References 1 and 2 and the supporting documents have been reviewed. On the basis of the information provided in the generic topical report (XN-NF-82-25) and recently-submitted plant-specific analyses and information, we conclude that D. C. Cook 2 Cycle 5 operation with the ENC 17x17 fuel is acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

In support of the reload and operation of D. C. Cook Unit 2 Cycle 5, Indiana and Michigan Electric Company has submitted a safety analysis report prepared by Exxon Nuclear Company. The nuclear design of the proposed reload has been reviewed. The neutronic calculations have been performed using the Exxon nuclear design methodology for pressurized water reactors (Refs. 20-22). The D. C. Cook-2 Cycle 5 reload will consist of 164 Exxon Nuclear Company 17x17 fuel assemblies, which will constitute Regions 6 and 7 of the core. The remaining fuel in Cycle 5 will consist of 29 Westinghouse assemblies and will be located in Region 5. The 92 fresh assemblies have an average enrichment of 3.64 w/o U235 and are scattered, in octant symmetry, in the outer core regions. These fuel assemblies contain Al_2O_3-B_4C burnable absorber pins whose number per assembly vary from 0 to 20. The scatter-loading of the fresh fuel throughout the core results in a low radial leakage fuel management plan. The expected BOC-5 HZP, ARO, xenon-free critical boron concentration is 1569 ppm. Power distributions have been obtained with the three-dimensional quarter core XTG code (Ref. 23). The expected total peaking factor, along with values of the moderator, isothermal, and Doppler temperature coefficients, boron worths, delayed neutron fraction and shutdown margin, presented for beginning and end of cycle at full and zero power conditions, are conservative with respect to those used in the transient and accident analyses. They are compared to similar quantities from Cycle 4, and the differences may be attributed to the difference in core design.

Beginning and end-of-cycle radial power distributions are also presented. These indicate that the values for total peaking factor and maximum relative pin power should remain within limits during Cycle 5. Power distribution control during the cycle will be accomplished by following the procedures presented in References 24-26. These procedures have been reviewed and approved by the staff.

We conclude that the nuclear design of the Cycle 5 reload is acceptable.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Cycle 5 of D. C. Cook Unit 2 will consist of a mixed loading of 29 Westinghouse and 164 Exxon Nuclear fuel assemblies. Cycle 4 was the first mixed core and included 121 Westinghouse and 72 Exxon fuel assemblies. In anticipation of steam generator tube plugging, the licensee has requested Exxon Nuclear to provide the analysis needed to support D. C. Cook Unit 2 operation in Cycle 5 with up to 5% of the steam generator tubes plugged.

A detailed review of the D. C. Cook Unit 2 Cycle 4 thermal-hydraulic design analysis, necessitated by the mixed loading, was performed by the staff as part of the Cycle 4 reload review (Ref. 13). This review concluded that (i) the Exxon mixed core thermal-hydraulic design methodology (Refs. 27, 28, 29) is acceptable with the inclusion of a conservative adjustment of 2% on the minimum DNBR, (ii) the XNB correlation (Refs. 30, 31) is acceptable with a MDNBR of 1.17, and (iii) the hydraulic differences between Exxon Nuclear and Westinghouse fuel assemblies and their effect on the major hydraulic performance parameters for Cycle 4 are acceptable. The D. C. Cook-2 Cycle 5 thermal-hydraulic analysis is discussed in the following sections.

4.1 Thermal-Hydraulic Compatibility and Comparison to Cycle 4

The hydraulic compatability between Exxon and Westinghouse fuel assemblies was found acceptable for Cycle 4 on the basis that the overall hydraulic resistance of the Exxon Nuclear fuel is within 0.3% of that of the Westinghouse fuel. The close match in hydraulic resistances also implies that loading Exxon fuel in the D. C. Cook Unit 2 core does not significantly affect the primary coolant flow rate. MDNBR's for Exxon Nuclear and Westinghouse fuel were evaluated for Cycle 4 at 118% reactor overpower to be 1.42 and 1.68. respectively.

The Cycle 4 safety analysis contains an evaluation of future cycles (with larger fractions of Exxon Nuclear fuel) indicating small increases ($\Delta 1\%$) in the limiting assembly flow for both Exxon and Westinghouse fuel. We conclude that the increase in the number of Exxon fuel assemblies by itself will have an insignificant effect on the MDNBR calculation.

As noted in the approval of XN-NF-82-21(P), Revision 1, Reference 29, an adjustment of 2% on the minimum DNBR must be included for mixed cores containing hydraulically different fuel assemblies. This has been provided for the thermal-hydraulic analysis (Ref. 32).

4.2 Effect of 5% Steam Generator Tube Plugging

Exxon Nuclear has determined (Ref. 3) that in D. C. Cook Unit 2 up to 5% steam generator tube plugging results in a 1.1% reduction in primary coolant flow. The plant transient analyses for D. C. Cook Unit 2 were redone with reduced primary coolant flow rate and reduced steam generator heat transfer area characteristic of a 5% steam generator tube plugging level.

4.3 Impact of Rod Bow on MDNBR

The Exxon Nuclear methodology for computing a rod bow penalty to DNBR (Ref. 9) has been reviewed and approved by the staff (Ref. 10). Use of this methodology requires that for the limiting anticipated operational occurrence the MDNBR be reduced by 13.2% at a peak D. C. Cook Unit 2 Cycle 5 assembly exposure of 43,000 MWD/MTU. The plant transient analysis with 5% steam generator tube plugging shows that the limiting transient (slow control rod withdrawal event) results in a MDNBR greater than 1.35. The lowest acceptable MDNBR using the XNB correlation is 1.17, and the criterion that the MDNBR for the limiting anticipated operational occurrence (1.35) reduced by 13.2% should exceed 1.17 is met.

4.4 Thermal-Hydraulic Evaluation Summary

The D. C. Cook Unit 2 Cycle 5 reload thermal design analysis was carried out with the approved mixed core thermal-hydraulic methodology and the approved XNB DNBR correlation. An adjustment of 2% on the minimum DNBR has been included (Ref. 32) to bound conservatively any uncertainties in the mixed core methodology. The thermal-hydraulic design is, therefore, acceptable.

5.0 TRANSIENTS AND ACCIDENTS

The Plant Transient and LOCA-ECCS analyses performed for D. C. Cook Unit 2 Cycle 5 will be reviewed separately. The only accident specifically addressed in this SER is the Rod Ejection Accident. Analyses were performed at beginning and end of cycle for both zero power and full power conditions. Conservative values of the Doppler coefficient and nominal values of the delayed neutron fraction were used. The BOC delayed neutron fraction is larger for Cycle 5 than for Cycle 4 and contributes to the lower energy deposition rates for this cycle, particularly at HZP where the ejected rod worth is considerably higher than it was for Cycle 4.

Results were obtained by using the methods presented in XN-NF-78-44(A). These methods have been used for this purpose in previous reload analyses and are acceptable for determining maximum fuel enthalpies. The calculated maximum fuel pellet enthalpy was 162 calories per gram for the hot full power beginning-of-cycle case. This meets the 280 calories per gram limit of Regulatory Guide 1.77.

6.0 TECHNICAL SPECIFICATION CHANGES

The Cycle 5 limits on F_Q (Z) and F_{H} are 1.97 and 1.48, respectively, for the Westinghouse fuel and 2.04 and 1.49, respectively, for the Exxon fuel. These values are the same as applied in Cycle 4. However, the limits are now applied for up to 5% steam generator tube plugging. In establishing the limits for the Exxon fuel, an improved plant system description was employed during the reflood portion of the LOCA treatment. For the Westinghouse fuel, the maximum F_Q predicted during Cycle 5 operation is 1.68 (Ref. 11). The available margin to a limiting $F_Q(Z)$ of 1.97 is sufficient to offset the effects of 5% steam generator tube plugging.

In a letter from M. P. Alexich to H. R. Denton, dated May 21, 1984, the licensee provided a revised LOCA/ECCS analysis as it relates to the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, and proposed changes to the $F^N_{\Delta H}$ Technical Specification, 3.2.3. As discussed in the section on LOCA analysis this revised analysis satisfies the 2200°F clad temperature criterion of 10 CFR 50.46 using an $F^N_{\Delta H}$ of 1.415 (and an unchanged total nuclear hot channel factor, F^T_Q, of 2.04). The analysis furthermore indicates that as power is reduced and $F^N_{\Delta H}$ is allowed to increase at a rate inversely proportional to the power level (as is F^T_Q) the LOCA analysis results will be less than or equal to those at full power.

The $F^N_{\Delta H}$ Technical Specification is based upon a measured value which does not contain an uncertainty allowance. The measurement uncertainty allowance for $F^N_{\Delta H}$ is 4%. The Technical Specification revision therefore contains an $F^N_{\Delta H}$ of 1.36 (at full power) which is 4% less than the 1.415 used in the revised LOCA analysis. The alone approved $F^N_{\Delta H}$ relationships as a function of power to protect against DNB remain in effect.

We reviewed the page by page implementation of the above Technical Specification change and bases as contained in the referenced submittal and find all of the changes appropriate. In view of this and the finding in the LOCA analysis section that the $F^N_{\Delta H}$ change produces acceptable clad temperatures, the proposed changes are acceptable.

We have reviewed the technical specification changes indicated in Sections 3/4.2, 3/4.2.1, 3/4.2.2 and 3/4.2.3 for Cycle 5 operation. On the basis that approved methods were used to determine the parameters involved, we find the above mentioned Technical Specifications acceptable.

7.0 SUMMARY OF EVALUATION

Based on our review we conclude that the fuel design, nuclear design, and thermal hydraulic design are acceptable. We further conclude that the analysis of the rod ejection event and proposed Technical Specifications cited above are acceptable.

8.0 REFERENCES

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3. XN-NF-82-32(P), Rev. 1, "Plant Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3425 MWt, Operation with 5% Steam Generator Tube Plugging," February 1984.

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

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

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

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18. R. S. Hunter (AEP) letter to H. R. Denton (NRC) letter number 673H, December 1982.

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20. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, June 1975.

21. XN-75-27(A), Supplement 1, September 1976.

22. XN-75-27(A), Supplement 2, December 1977.

23. XN-CC-28, Revision 5, "XTG - A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing," Exxon Nuclear Company, July 1979.

24. XN-NF-77-57(A), "Exxon Nuclear Power Distribution Control For Pressurized Water Reactors - Phase II," Exxon Nuclear Company, January 1978.

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26. XN-NF-77-57(A), Supplement 2, September 1981.

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36. Letter, J. C. Chandler (ENC) to Dr. C. O. Thomas (NRC), Subject: XN-NF-82-06 (P), "Qualification of Exxon Nuclear Fuel for Extended Burnup," dated April 16, 1984.

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IV. Cycle 5 Reload Review, cont.

C. Transients and Accident Analysis

1. Introduction

In reference 1, the licensee provided a LOCA analysis in support of the Cycle 5 reload for D.C. Cook 2. An analysis of the limiting break, a double-ended guillotine break in the pump discharge piping with a discharge coefficient of 1.0, was performed using the EXEM/PWR ECCS evaluation model, reference 2. Supplemental information in support of the LOCA analysis was provided in references 3 and 4.

This SER presents our evaluation of these submittals. We first address the compliance of the EXEM/PWR evaluation model to the requirements of Appendix K to 10 CFR 50. We then evaluate the adequacy of the LOCA analyses performed to demonstrate compliance to 10 CFR 50.46. Finally, we examine the adequacy of the proposed changes to Technical Specification 3.2.3 which were necessary as a result of the LOCA analysis.

2. Evaluation Model

The ECCS evaluation model utilized to perform the LOCA analysis for D.C. Cook 2 is the revised Exxon Nuclear Company (ENC) evaluation model. This model is called EXEM/PWR and is documented in reference 2. This model is currently under staff review and a separate, more detailed SER on EXEM/PWR will be issued separately. This section documents our review of EXEM/PWR, as utilized for the D.C. Cook 2 Cycle 5 LOCA analysis, and evaluates its conformance to the required features of Appendix K to 10 CFR 50.

EXEM/PWR contains several models updated to the currently approved ENC-WREM IIA PWR ECCS evaluation model, reference 5. The model updates for EXEM/PWR are shown on Table 1. Each of these changes is discussed separately below.

2.1 Fuel Rod Model-RODEX2 Code

The RODEX2 Code is documented in reference 6. The RODEX2 code is based upon the previously approved GAPEX code, reference 7. As part of the EXEM/PWR model, ENC uses the RODEX2 code to provide the initial fuel stored energy and fuel rod internal pressures utilized as inputs to various portions of the evaluation model.

The staff has previously reviewed and approved the RODEX2 code for LOCA applications. Our evaluation of this code is contained in reference 8. Specifically, we found that the RODEX2 code satisfies the requirements of Appendix K, section I.A.I.

2.2 Clad Swelling and Rupture Model

In reference 9, ENC proposed a revised clad swelling and rupture model. This model, which includes the data base of NUREG-0630, reference 10, is used in the RELAP4 and TOODEE2 codes.

The staff has previously reviewed this model for compliance with section I.B. of Appendix K. As documented in reference 11, we found this model to meet those requirements.

2.3 REFLEX Leakage Flow Model

The currently approved ENC REFLEX code, which is used for calculating the reflooding phase of a LOCA, does not consider leakage flow paths from the upper plenum to the downcomer. ENC has proposed a modification to the REFLEX noding to account for this leakage path. ENC has stated that this model will be utilized only when the leakage flow path can be characterized and justified. Sensitivity studies, documented in reference 2, have been performed and show that this model change results in only a small reduction, approximately 20°F, in peak cladding temperature.

Inclusion of a leakage flow path in REFLEX will result in a more representative model of the plant configuration. In fact, the leakage flow path is already included in the blowdown model. For the D.C. Cook 2 analysis, the flow holes drilled between the upper plenum and upper downcomer were simulated. Since these flow holes can be well characterized and are already used in the blowdown model, the staff find this model change acceptable.

2.4 Split Break Model

Currently the REFLEX code only simulates a guillotine break configuration with a discharge coefficient of 1.0. This assumption is conservative for split breaks and guillotine breaks with discharge coefficients less than 1.0. As part of EXEM/PWR, the REFLEX code has been modified to allow modelling of split breaks and guillotine breaks with smaller discharge coefficients.

For modelling of split breaks, the REFLEX code has been modified to allow the fluid streams from the downcomer and steam generators to mix before leaving the break. A junction is then used to simulate the break path to containment.

Double-ended guillotine breaks with smaller discharge coefficients are simulated with the current REFLEX noding scheme. However, to account for the smaller discharge coefficient, an equivalent K-factor is used to simulate the increased break resistance.

We have reviewed these model changes and find them acceptable.

2.5 REFLEX Core Outlet Enthalpy Model

The currently approved REFLEX model uses a constant value for the core exit enthalpy. The core exit enthalpy used is determined at the upper plenum pressure and the fluid temperature corresponding to the steam generator secondary side saturation temperature. The core exit enthalpy model has been upgraded such that fluid enthalpy is calculated based upon an energy balance performed for the core.

The revised core outlet enthalpy model accounts for energy added to the fluid below the quench front, stored energy release as the quench front progresses, and energy added to the fluid above the quench front. To demonstrate the appropriateness of the model, ENC performed benchmarks of FLECHT tests 34711, 34610, and 31922, reference 12. These benchmarks showed good agreement to the data.

Based upon the benchmarks performed, and a detailed review of the equations utilized, we have concluded that this model is acceptable.

2.6 Steam Cooling Model

Section I.D.5 of Appendix K to 10 CFR 50 requires that a steam cooling model be utilized to predict heat transfer coefficients when flooding rates fall below one inch per second. In addition, the steam cooling model must take into account the effect of flow blockage relative to both local steam flow and heat transfer.

Exxon developed, as part of their currently approved ENC WREM-IIA PWR ECCS evaluation model, a steam cooling model which fully complied with these requirements. However, recent experimental data in References 13 and 14 have shown that the currently approved Exxon steam cooling model is overly conservative. As a result, Exxon developed, and submitted as part of EXEM/PWR, a revised steam cooling model.

The revised steam cooling model calculates an equivalent steam flow for use in the TOODEE-2 (Reference 15) energy solution which assures that superheated steam exists the core. This flow rate includes the effect of blockage based upon the currently approved flow divergence model of the ENC WREM-IIA PWR ECCS evaluation model.

The rod surface heat transfer coefficients are calculated by the following method. First, local unblocked heat transfer coefficients are calculated using an appropriate reflood heat transfer correlation for the fuel modeled. The local heat transfer coefficients are then modified to account for the effect of blockage on mass flux and hydraulic diameter. In addition, the heat transfer coefficients are adjusted to account for the effect of increased turbulence and breakup of entrained liquid droplets downstream of the blockage. The net effect of these modifications is a decrease in heat transfer downstream of the flow blockage relative to that which would be obtained in an unblocked core. Calculations performed by Exxon with the revised steam cooling model indicate that peak cladding temperatures are approximately 15°F higher relative to that which would be obtained using the unblocked ENC-2 FLECHT coefficients.

The staff has reviewed the revised steam cooling model and finds it acceptable. Recent experimental data in Reference 13 and 14, obtained with flooding rates below one inch per second, indicate that the effect of blockage is to enhance heat transfer, relative to an unblocked fuel assembly, downstream of the blockage plane. Since the revised EXXON steam cooling model predicts decreased heat transfer, we find that the effect of flow blockage on local steam flow and heat transfer has been treated conservatively. Thus, the revised steam cooling model fully meets the requirements of Section I.D.5 of Appendix K to 10 CFR 50.

2.7 17x17 Carryout Rate Fraction (CRF) and Heat Transfer Coefficient Correlation

The FLECHT SEASET experimental program, reference 12, has expanded the core reflood heat transfer data base to included 17x17 sized fuel assemblies. ENC has used this data to develop new CRF and heat transfer correlations, for use as part of EXEM/PWR, which are applicable to 17x17 fuel rods.

Data from 23 of the FLECHT SEASET runs were utilized to develop the new correlations. The range of applicability for these correlations is shown on Table 2. To assure conservative predictions of CRF, ENC used the mass stored in the bundle to determine CRF. This approach yields higher CRF than that which can be obtained using the mass effluent at the bundle exit. The rods chosen for heat transfer coefficient determination exclude the cooler rods in the outer two rows of the bundle and rods which were adjacent to failed rods. Thus, the hotter rods were used to develop the correlation. The staff finds the data selection used by ENC to be appropriate.

In addition to the CRF and heat transfer coefficient correlations, ENC formulated correlations for quench time and quench front velocity. The quench time correlation was developed neglecting the "top down" quenching observed in some of the FLECHT SEASET tests. The quench front velocity correlation was obtained by differentiating the quench time correlation.

To assure the adequacy of the new 17x17 correlations, we reviewed numerous comparisons to FLECHT data. These comparisons were presented in Figures 3.4 through 3.77 of Reference 2. Based on these comparisons we have concluded:

a. The quench time, and hence quench velocity, correlation is acceptable over the parameter ranges given in Table 2.

b. The CRF correlation has been determined based on applicable data and is acceptable over the parameter ranges given in Table 2. Thus, the correlation meets the requirements of Appendix K to 10 CFR 50, Section I.D.3.

c. The heat transfer coefficient correlation generally predicts conservative coefficients when reflooding rates are below 1.5 inches/second. Thus, the correlation meets the requirements of Appendix K, Section I.D.5.

d. Above 1.5 inches/seconds, the heat transfer coefficient correlation appears to be non-conservative.

Based upon the above observations, we have concluded that the heat transfer and CRF correlations are acceptable below 1.5 inches/second. The D.C. Cook 2 analyses, which are discussed later, result in flooding rates which are always less than 1.5 inches/second.

The correlations described above have been developed for constant flooding rates. During a plant simulation, reflood rates will continuously vary. To apply the correlations ENC uses an effective reflooding velocity, defined in equation 3.11 of reference 2, to account for the time varying reflooding rates.

During our review of the ENC proposed method for applying the various correlations, we asked for additional justification for the effective reflooding velocity method. Via reference 3, ENC provided comparisons of the predicted heat transfer coefficients for the variable flooding rate FLECHT test, FLECHT runs 32333 and 32335, using three different methods of defining effective reflooding rates. In addition, comparisons of the predicted heat transfer coefficients for the D.C. Cook 2 flooding rates, from reference 1, were provided for these three different approaches.

The three different methods studied by ENC were: 1) the effective flooding rate velocity as defined by EXEM/PWR; 2) the time shift method of WCAP-7665 with scaling parameters; and, 3) the time shift method of WCAP-7665 without scaling parameters. The time shift is defined such that the total amount of water injected in the bundle with variable flooding rates will equal the instantaneous flooding rate multiplied by the real time plus the time shift. WCAP-7665 further adjusts this time shift with scaling parameters. It should be noted that the second approach is that used in the currently approved ENC WREM-II model, while the third method is the same as that approved in another LOCA evaluation model.

Examining the comparisons of the EXEM/PWR methodology to the two other methods, it was seen that the EXEM/PWR methodology calculated later quench times than the two other methods. Similar heat transfer coefficients were obtained using all the methods, with the EXEM/PWR methodology yielding heat transfer coefficients which fell between the two other acceptable approaches. In addition, the EXEM/PWR methodology predicted conservative heat transfer coefficients relative to those obtained from FLECHT tests 32333 and 32335. Based on these comparisons, we find the EXEM/PWR methodology to be acceptable.

In addition to the use of an effective flooding rate velocity, the proposed EXEM methodology applies two other corrections to the heat transfer correlation in order to apply the method to a reactor simulation. These corrections account for the effect of local rod peaking and mixing vanes on the heat transfer coefficient. At this time, we have concluded that insufficient justification has been presented for the correction factors presented in reference 2. Thus, these factors can not be used in the D.C. Cook 2 LOCA analysis.

As a result of the staff's determination, ENC proposed, in reference 3, a revised method to account for the effect of local rod peaking on heat transfer coefficient. The revised ENC methodology, which was developed specifically for application for Cycle 5 operation of D.C. Cook 2, used a time and rod elevation dependent heat transfer multiplier to account for local rod peaking effects. The mixing vane correction factor was not proposed for application for D.C. Cook 2.

These heat transfer multipliers were determined by using the upgraded ENC reflood heat transfer correlations of reference 16. These correlations combine FLECHT data from both the 15x15 and 17x17 fuel rod data. The combined correlation explicitly accounts for local rod peaking effects.

Comparisons of the upgraded methodology to FLECHT data were presented in reference 16.

While we have not yet fully completed our review of the upgrade methodology, we have concluded that the correlation conservatively accounts for the effect of local rod peaking observed in the FLECHT data. That is, the change in heat transfer coefficient associated with local rod peaking is underpredicted by the methodology of reference 16. Thus, we find that the time, and rod elevation dependent heat transfer multipliers proposed by ENC for Cycle 5 of D.C. Cook 2 are acceptable and satisfy the requirements of Appendix K to 10 CFR 50.

2.8 Summary of EXEM/PWR Model Compliance

Based on the foregoing, we find that the EXEM/PWR evaluation model, as utilized to support Cycle 5 operation for D.C. Cook 2, is wholly in conformance with Appendix K to 10 CFR 50.

At the request of the Office of Nuclear Reactor Regulation, the NRC Region IV Office recently performed a review of the quality assurance (QA) procedures applied to ENC safety-related computer codes. This review made several findings of nonconformance with NRC requirements (Inspection Report No. 99900081/84-01). The inspection concluded that ENC failed to prescribe adequate definition of instructions to the analyst which are necessary for satisfactory completion of safety-related computer code activities. Thus, our conclusions relative to the adequacy of the EXEM/PWR evaluation model is contingent upon satisfactory resolution of the concerns identified in the inspection report. By letter dated June 4, 1984, the licensee submitted a response to the apparent non-compliance and concurred with ENC that the analyses were in compliance with NRC requirements. We agree that the analyses are acceptable for Cycle 5 but note that resolution of the non-compliance may be the subject of seperate NRC action.

3. LOCA Analysis

In reference 1, the licensee provided a LOCA analysis in support of Cycle 5 operation for D.C. Cook 2. The analysis was performed for the limiting break, a double-ended guillotine break in the pump discharge piping, with a C_D=1.0. Previous analyses, performed in support of Cycle 4 operation and documented in reference 17, demonstrated that this break location yields the highest peak cladding temperature.

The analysis was performed assuming the following:

a. A licensed thermal power rating of 3425 MWt was used. Core power was increased by 2% as specified by part I.A of Appendix K to 10 CFR 50.

b. 5% steam generator tubes plugged.

c. Two low pressure safety injection (LPSI) pumps operating.

d. A total peaking limit (F_Q^T) of 2.04.

e. $F_{\Delta H}^N$ of 1.55

In addition to the above, a burnup sensitivity study was performed at burnups of 2.0, 10.0 and 47.0 MWD/kg. All these analyses were performed using the original EXEM/PWR model of reference 2.

In reference 4, the licensee provided a revised analysis using the modified heat transfer multipliers, for local rod peaking effects, which were proposed in reference 3. The revised analysis was performed similar to that described above except an $F_{\Delta H}^N$ of 1.415 was used. The total peaking limit, F_Q^T of 2.04, was not changed in the revised analysis. Subsequent discussions in this section are based on this revised analysis.

As noted above, the licensee assumed that both LPSI pumps operated. This is the "no single-failure" as the worst single failure case. Analyses performed in reference 18 show that this assumption yields highest peak cladding temperatures due to the effect of the reduced containment backpressure on core reflooding rates. We find this assumption to be appropriate and that Appendix K Part I.D.1 is satisfied.

In implementing the 5% average steam generator tube plugging assumption, the licensee assumed that the broken loop steam generator was 10% plugged. The other steam generator had 3.3% of its tubes plugged. Sensitivity studies performed by ENC in reference 19 showed that an assymetric distribution yields slightly higher, approximately 14°F, peak cladding temperature. The staff finds this assumption to be appropriate.

The results of the revised analysis are summarized on Table 3. As shown, peak cladding temperature is less than 2200°F, local oxidation is less than 17%, and core wide metal-water reaction is less than 1.0%. Therefore, we find that the criteria of 10 CFR 50.46 has been satisfied.

4. Technical Specification Change

In the revised analysis, the F^N_H was reduced to 1.415 in order to satisfy to 2200°F criterion of 10 CFR 50.46. The licensee changed Technical Specification 3.2.3, entitled "Power Distribution Limits, RCS Flowrate and Nuclear Enthalpy Rise Hot Channel Factor," in order to implement the revised F^N_H. The revised Technical Specification is presented in reference 20.

The revised technical specification implements a measured F^N_H of 1.36 at 100% power. This value is 4% less than the value used in the LOCA analysis in order to account for measurement uncertainty on F^N_H. In addition, in order to protect DNBR limits for non-LOCA events, the licensee employs an F^N_H of 1.49 at 100% power. The DNBR limit is the same as that used for Cycle 4 operation. Operation of Cycle 5 will be restricted by the most limiting of the two F^N_H values as a function of power level.

The revised Technical Specification allows for relaxation of the F^N_H limit as core power level is reduced. At the higher power level, greater than approximately 95% power, F^N_H is allowed to increase via the formula:

$$F^N_H = 1.36/P$$

where: P is the fraction of rated thermal power.

This formula assures that the enthalpy rise across the hot bundle, for lower power operation, will be less than or equal to the value assumed in the LOCA analysis of reference 4. Technical Specification 3.2.2 is used to assure that the maximum local heat flux is less than that assumed in the analysis. The combined effect of these two Technical Specifications is to provide assurance that the LOCA analysis is conservative at reduced power levels.

Below approximately 95% power, the F^N_H limit is based on protecting DNBR limits. To assure protection of these DNBR limits, the previously applied formula for Cycle 4 is used.

We have reviewed the revised Technical Specification 3.2.3 and find it acceptable.

5.0 Conclusions

Based upon the foregoing discussions, we find:

a. The LOCA analysis was performed using a model wholly in conformance with Appendix K to 10 CFR 50.

b. The analysis shows the operation of Cycle 5 of D.C. Cook 2 will meet the requirements of 10 CFR 50.46.

c. The licensee has implemented appropriate Technical Specification changes consistent with the LOCA analysis which was performed.

Therefore, we conclude that the consequences of a LOCA during Cycle 5 of D.C. Cook 2, with up to 5% of the steam generator tubes plugged, will not result in undue risk to the public health and safety.

TABLE 1

ECCS Model Updates of EXEM/PWR

- Fuel Rod Model - RODEX2
  - Stored Energy
  - Fission Gas Release

- Blowdown Model - RELAP4-EM Code
  - NUREG-6030 Clad Rupture/Blockage Model

- Reflood Model - REFLEX Code
  - 17x17 FLECHT Carryover Rate Fraction Correlation
  - Leakage Flow from Upper Plenum to Downcomer
  - Split Break Model
  - Core Outlet Enthalpy Model

- Heatup Model - TOODEE2
  - 17x17 FLECHT Heat Transfer Correlation
  - Revised Steam Cooling Model
  - NUREG-0630 Clad Rupture/Blockage Model

TABLE 2

Correlation Ranges

Inlet Velocity	0.6 to 6.0 inches/Sec
Pressure	50 to 60 psia
Initial Temperature	500 to 2000°F
Subcooling	130 to 145°F
Decay Peak Power	0.4 to 1.0 kw/ft

* Heat Transfer Coefficient correlation valid only to 1.5 inches/second

TABLE 3

D. C. COOK Unit 2 LOCA/ECCS Analysis Summary

Results for the Cycle 5 Core Configuration (85% ENC Fuel)

Peak Rod Average Burnup (MWD/kg)	2.0	10.0	47.0
F	2.04	2.04	2.04
$F^T{}_H$	1.415	1.415	1.415
Peak Cladding Temperature (°F)	2198	2190	2096
Maximum Local $Zr-H_2O$ Reaction (%)	7.4	7.3	5.7
Total $Zr-H_2O$ Reaction	1.0	1.0	1.0

6.0 References

(1) XN-NF-84-21(P), "Donald C. Cook Unit 2, Cycle 5, 5% Steam Generator Tube Plugging Limiting Break LOCA/ECCS Analysis", Exxon Nuclear Company, Inc., Richland, WA 99352, February 1984.

(2) XN-NF-82-20(P), Rev. 1, August 1982; Supplement 1, March 1982; and Supplement 2, March 1982, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Inc., Richland, WA 99352.

(3) Letter, J. C. Chandler (ENC) to H. R. Denton (NRC), Subject: Additional Information Regarding Unit 2 Cycle 5 LOCA ECCS Analysis, May 7, 1984.

(4) XN-NF-84-21(P), Revision "Donald C. Cook Unit 2, Cycle 5, 5% Steam Generator Tube Plugging Limiting Break LOCA/ECCS Analysis," Exxon Nuclear Company, Inc., Richland, WA 99352, May 1984.

(5) XN-NF-78-30(A), "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA," Exxon Nuclear Company, Inc., Richland, WA 99352, May 1979.

(6) XN-NF-81-58(P), Rev. 2, "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA 99352, February 1983.

(7) XN-73-25, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," Exxon Nuclear Company, Inc., Richland, WA, August 13, 1973.

(8) Letter, C. O. Thomas (NRC) to J. C. Chandler (ENC), Subject: "Acceptance for Referencing of Licensing Report XN-NF-81-58(P)," November 16, 1983.

(9) XN-NF-82-07(A), Rev. 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, "Exxon Nuclear Company, Inc., Richland, WA 99352, March 1982.

(10) D. A. Power and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis", NRC Report NUREG-0630, April 1980.

(11) Letter, C. O. Thomas (NRC) to G. F. Owsley (ENC), Subject: Acceptance for Referencing of Topical Report XN-NF-82-07(P), Revision 1, October 14, 1982.

(12) "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," EPRI NP-1459, September 1981.

(13) M. J. Loftus, et al, "PWR FLECHT SEASET 21-Rod Bundle Flow Blockage Task Data and Analysis Report", EPRI Report NP-2014, September 1982.

(14) M. J. Loftus, et al, "PWR FLECHT SEASET 163-Rod Bundle Flow Blockage", NRC Report NUREG/CR-3314, October 1983.

(15) G. N. Lauben, NRC Report NUREG-75/057, "TOODEE2: A Two-Dimensional Time Dependent Fuel Element Thermal Analyis Program", May 1975.

(16) XN-NF-82-20(P), Supp. 1 Rev. 1, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates: Revised FLECHT Bases Reflood Carryover and Heat Transfer Correlations", June 1983.

(17) XN-NF-82-35, "Donald C. Cook Unit 2 LOCA ECCS Analysis Using EXEM/PWR Large Break Results," Exxon Nuclear Company, Inc., Richland, WA 99352, April 1982.

(18) XN-NF-82-35, Supplement 1, "Donald C. Cook Unit 2 Cycle 4 Limiting Break LOCA-ECCS Analysis Using EXEM/PWR," Exxon Nuclear Company, Inc., Richland, WA 99352, November 1982.

(19) XN-76-4, Supplement 1, "Palisades LOCA Analysis Using the ENC WREM-Based PWR ECCS Evaluation Model, February 1976.

(20) Letter, M. P. Alexich (IMECO) to H. R. Denton (NRC), Subject: "Revision to the Application for Changes to the Unit 2 Technical Specification for the Cycle 5 Reload," Docket No. 50-316, AEP:NRC:0860K, May 21, 1984.

Non-LOCA Transient and Accident Analysis

1.1 Background

Exxon Nuclear Company (ENC), the fuel vendor for the Indiana & Michigan Electric Company (IMECO), performed the Cycle 5 thermal-hydraulic reload analysis for D.C. Cook Unit No. 2. The method of analysis was based on what Exxon has termed "incremental assessment". This method involves a duplication of the original Fuel design analysis results, as documented in Chapter 15 of the Cook 2 FSAR, and an assessment of the changes to these analyses that result from the use of the new ENC fuel. This incremental analysis is complicated by the fact that neither Exxon nor IMECO have access to the reactor vendor's code, method of analysis, or fuel design. In addition, the influence of steam generator tube plugging was not factored into the incremental assessment.

2. Evaluation

We have reviewed Revision 2 to XN-NF-82-32(P) which summarizes the transient and non-LOCA accident analyses performed in support of the D.C. Cook Unit 2, Cycle 5 reload. Our review has determined that the information provided to support incremental differences from the reactor vendor's original FSAR analyses is insufficient from the standpoint that the licensee did not account for incremental differences due to changes in fuel design, steam generator tube plugging, computer code design, and initial operating conditions. We have determined that the analyses performed by the licensee's fuel vendor (Exxon Nuclear Company) did not comply with our previous SER for Cycle 4. Our SER for Cycle 4 specifically stated that the Westinghouse Improved Thermal Design Basis (ITDB) method of analyzing transient and non-LOCA events is not applicable to Exxon fuel. If the licensee intends to apply such methods in future submittals, the licensee must submit its methods for staff review and approval.

The transient and accident calculations for Cycle 5 were performed using the PTSPWR2 computer code. The limitations of using PTSPWR2 for licensing analysis, as documented in our SER for Cycle 4, remain valid. Since neither the PTSPWR2 computer code nor its method of application have been approved by the staff and since the licensee has not fully complied with our conclusions in the Cycle 4 SER (ITDB analysis), future licensing submittals should be analyzed with codes and methods which have previously been found acceptable by the NRC. Should the licensee be unable to provide analyses which use approved codes and methods, then documentation of the analyses, methods and computer programs should be submitted at least 6 months prior to the requested licensing date.

Based on our review of XN-NF-82-32(P), we require Indiana and Michigan Electric Company to provide the necessary information needed for the staff to complete its review of PTSPWR2 by September 1, 1984, and to resubmit the transient and accident analyses of the events analyzed for Cycle-5 within 90 days after staff approval of PTSPWR2. By letter dated June 1, 1984, the licensee committed to these actions.

3. Conclusions

Although the licensee must resubmit the Cycle 5 analyses, the analyses submitted to date are acceptable on an interim basis. This conclusion is based on the fact that the non-LOCA events were analyzed with a fuel peaking factor of 2.47 and the design basis LOCAs were evaluated with a peaking factor of 2.04. This represents a considerable margin of conservatism in predicting the minimum DNBR. On the basis that: (1) the Technical Specifications were not modified, (2) IMECO committed to provide by September 1, 1984, the necessary information needed for staff review of PTSPWR2, and (3) IMECO committed to perform a reanalysis of the Chapter 15 events (as outlined above) within 90 days of staff approval of the code and analysis methodology, we find operation for Cycle 5 acceptable.

D. Radiological Consequencies

1. Background

By letter dated March 1, 1984, Indiana & Michigan Electric Company, the licensee for D. C. Cook Unit 2, requested approval for Cycle 5 operation. This cycle will be at a power level of 3425 MW_t and includes burnup beyond the traditional value to 30,000 MWd/MTU core average with a peak batch discharge exposure of 40,000 MWd/MTU.

By letter dated March 13, 1984, report number XN-NF-82-90(NP), Supplement 1, "D. C. Cook Unit 2 Potential Radiological Consequences of Incidents Involving High Exposure Fuel" was submitted on the D. C. Cook Unit 2 docket. This report covers calculations by Exxon Nuclear Corporation of the radiological consequences of accidents at the stated power level for the above burnup limit.

2. Evaluation

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

The licensee did not explicitly evaluate the radiological consequences of the locked rotor, steamline break or rod ejection accidents since analyses show no fuel failures. This is acceptable since our review has accepted the licensees position on no fuel failures.

The evaluation of the fuel handling accident inside containment was performed by Exxon in accordance with the assumptions of Regulatory Guide 1.25, even though the conditions at the end of Cycle 5 will be beyond the basis stated in the Guide. Since no justification for continued conservatism of these assumptions was provided by the licensee, the staff independently evaluated this accident.

The missing justification concerns the fraction of noble gas and iodine assumed to be in the pellet-clad gap of the highest power module. Report number XN-NF-83-85, "D. C. Cook Unit 2, Cycle 5 Safety Analysis Report," shows that the highest power module is a freshly exposed first cycle module. Therefore, the case to be considered is a module at about 22,500 MWd/MTU at the highest allowable linear heat generation rate, about 13 kW/ft. For this case, calculations based on the fission gas release model in the ANS 5.4 standard shows gap fractions less than 30% of ^{85}Kr, about 10% of ^{131}I and less than 10% of all other radionoble gases and radioiodines. Therefore, it is not necessary to consider up to 30% of these nuclides within the gap, as the licensee did for the fuel handling accident outside containment. The assumptions used by the staff and the results of the calculation are given in Table 1. The results show that the delay to 100 hours from shutdown and site related parameters are adequate to mitigate the consequences of this accident.

3. Conclusion

The licensee and the staff have considered the factors dependent upon power level (to 3425 MW_t) and burnup (to 30,000 MWd/MTU core average for peak batch discharge exposure of 40,000 MWd/MTU) that impact the radiological consequences of accidents. On the basis of our acception of the licensee's evaluation of the absence of fuel failures, there are no identified issues that would preclude the extended burnup.

Radiological Consequences

Table D 1

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

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

Atmospheric Diffusion and Transport Relative Concentration, X/Q* (sec/m^3)

Exclusion Area Boundary	0-2 hours	2.1×10^{-4}
Low Population Zone	0-8 hours	1.8×10^{-5}

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

*Memorandum Hulman to Knighton, September 4, 1979

E. Environmental Consideration

In the Environmental Impact Appraisal which accompanied Amendment 48 issued on January 14, 1983, we reviewed the radiological and non-radiological impacts for the equilibrium cycle operating up to 3411 megawatts thermal. In that appraisal we concluded that there will be no environmental non-radiological impact attributable to the proposed action than has already been predicted and described in the Commission FES for D. C. Cook Nuclear Plant. That appraisal applies to Cycle 5 and the equilibrium Cycle 6. For the radiological impacts, the estimated releases of radioactive materials in liquid and gaseous effluents have been previously calculated using the PWR GALE Code described in NUREG-0017," Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressuirzed Water Reactor (PWR GALE Code)," April 1976. These releases were reported in the Donald C. Cook Nuclear Plant, Unit No. 2 SER, Supplement 7, dated December 1977. The fuel burnup is not among the principal parameters and conditions used in calculating releases of radioactive materials using the GALE Code methodolody. Therefore, no increase will occur in the estimated releases of radioactive materials in liquid and gaseous effluents as a result of the requested amendments to the Technical Specifications. In Amendment 48 issued January 14, 1983, the effects of increased power level were addressed and found acceptable.

F. Final Determination of No Significant Hazards Consideration

On May 24, 1984, the Commission published in the Federal Register (49 FR 22008) a Notice of Consideration of Issuance of Amendment To Facility Operating License And Proposed No Significant Hazards Consideration Determination And Opportunity For Hearing. That notice specifically addressed a change requested by the licensee in their letter dated May 21, 1984. Because the Commission determined there was insufficient time for its usual 30-day notice of the proposed action for public comment, that notice established a period until June 7, 1984 for comment, state that a final determination on no significant hazards would be made before issuance of the license amendment, and provided provided that if no significant hazards are involved, a subseqsuent notice of opportunity for a hearing would be published. The proposed change as requested by letter dated May 21, 1984, involves changes to the Technical Specifications on nuclear enthalpy rise hot channel factor ($F^N_\Delta H$) and power level as a result of emergency core cooling system/loss of coolant accident analysis with up to 5% of the steam generator tubes plugged. The proposed change from the original request, will include an $F^N_\Delta H$ which is flow dependent at various power levels and is limited by both loss of coolant accident (LOCA) and departure from nucleate boiling (DNB) considerations; the $F^N_\Delta H$ was previously limited by DNB considerations only.

In our evaluation of the LOCA and fuel performance analyses we determined that the revised analyses were appropariate and that 10 CFR 50.46 and Appendix K was satisfied. This, however, required that $F^N_\Delta H$ be reduced at high power levels to satisfy the 2200°F criterion for LOCA events and $F^N_\Delta H$ be maintained to protect DNBR limits for non LOCA events. The Technical Specification 3.2.3. must assure that operation in Cycle 5 will be restricted by the most limiting of the two $F^N_{\Delta H}$ valves as a function of power level. Our evaluation found the proposed Technical Specification for Cycle 5 acceptable on the basis that the models are wholly in conformance with Appendix K, the analysis show operation of Cycle 5 will meet the requirements of 10 CFR 50.46, and the previously acceptable analysis to protect against DNB remain in effect. We have determined that the proposed change does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possiblity or a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

V. Conclusions

Environmental Conclusion

This amendment involves a change in the installation or use of facility components located within the restricted area. The staff has determined that the change involves no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cummulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. A portion of the amendment proposed was subsequently changed; the Commission has also made a final no significant hazards consideration finding with respect to the changed portion of this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR Sec. 51.22(b) no envirionmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Safety 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 18, 1984

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