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:

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

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

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (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: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

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 $\leq$ 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.

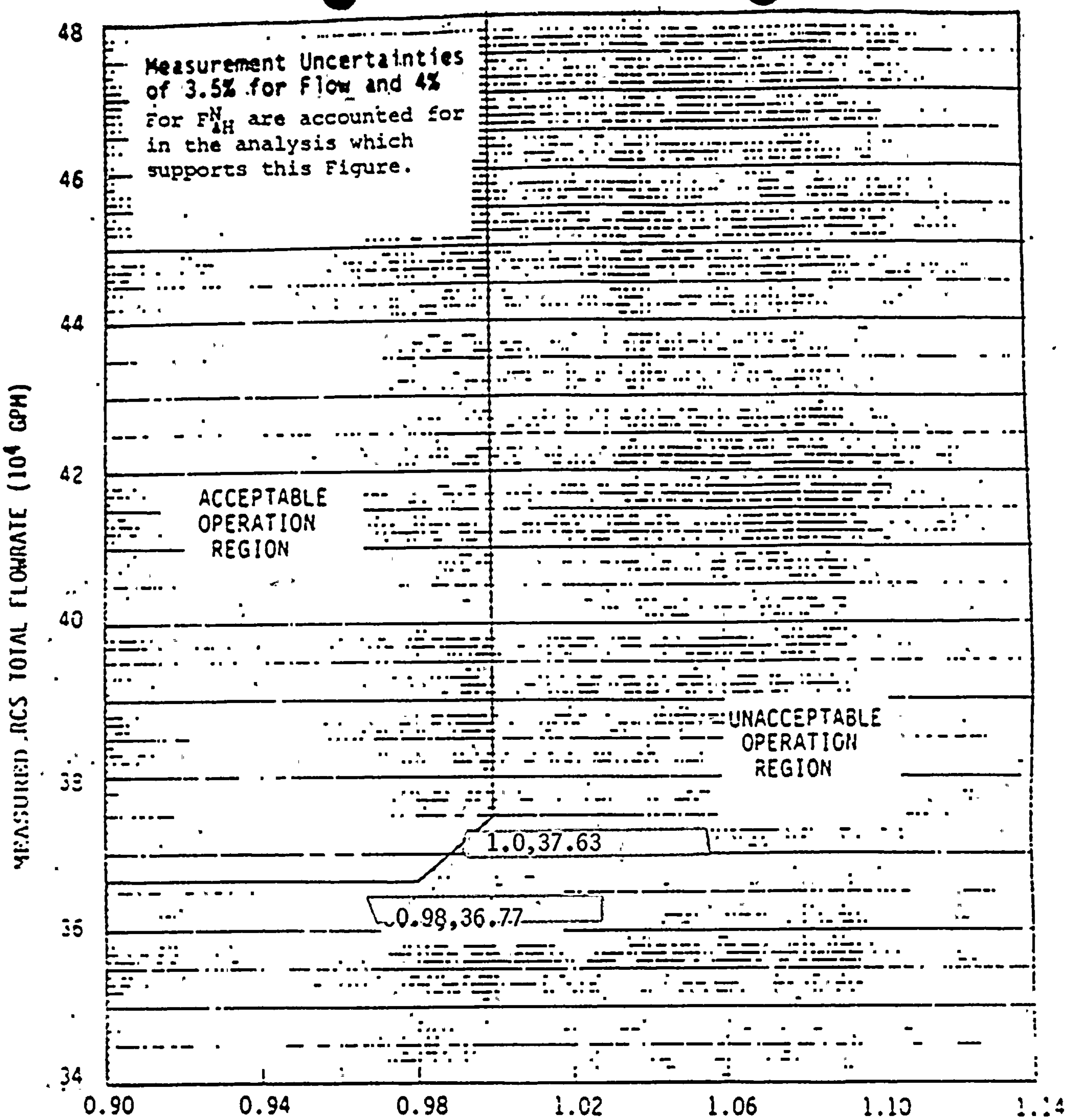
4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within the above limits and the combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-4 or 3.2-5 (as applicable):

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.

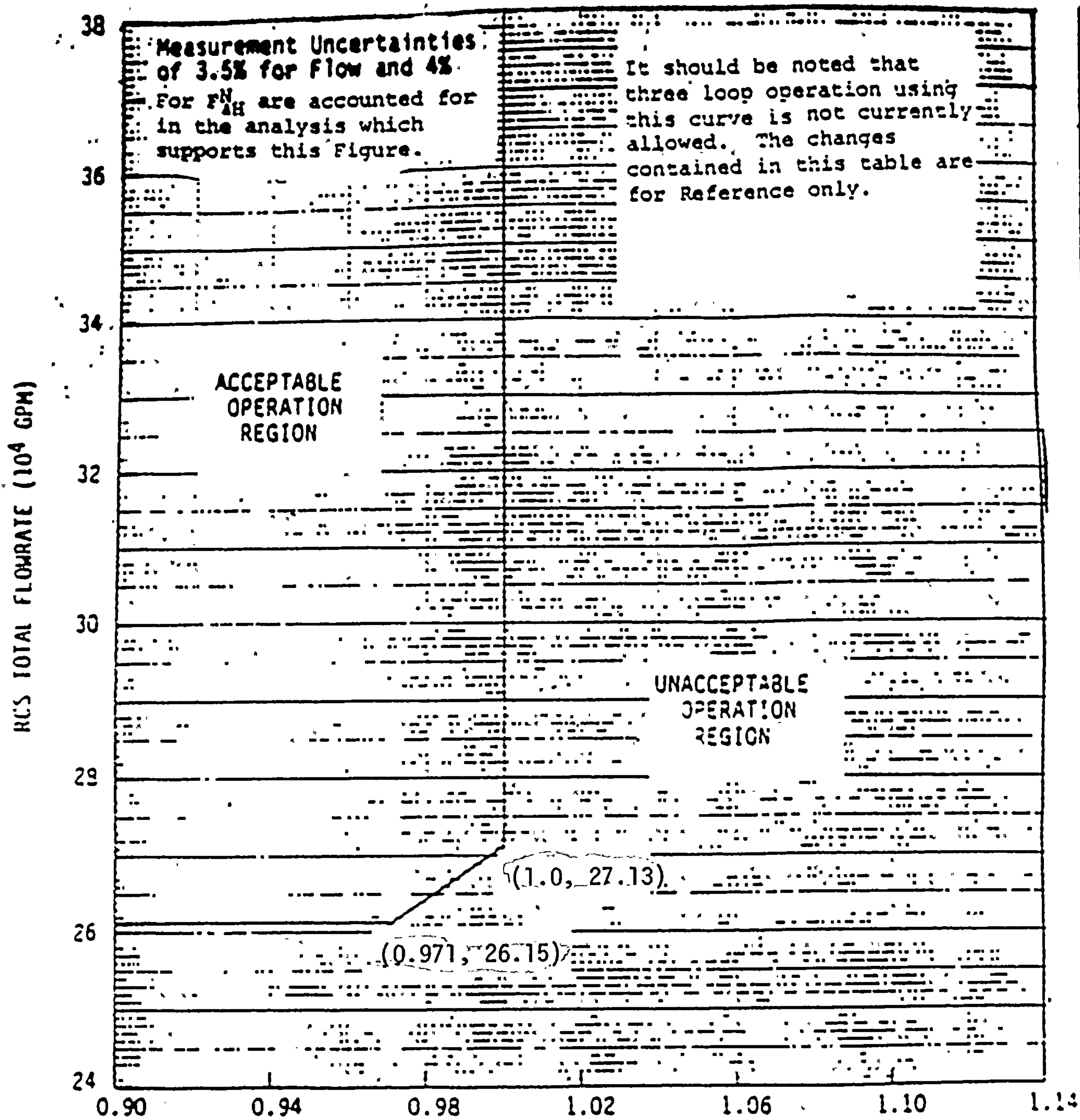
4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.



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

LIMITS

PARAMETER	4 Loops in Operation	3 Loops in Operation***
Reactor Coolant System T_{avg}**	≤ 576.7°F (indicated)	≤ 570°F
Pressurizer Pressure	≥ 2220 psia*	≥ 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

ICE CONDENSER DOORS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more ice condenser doors open or otherwise inoperable, POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained ≤ 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 ≤ 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 ≤ 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	≤ 37.4 lbs.
2.	Paired with Door Adjacent to Crane Wall	≤ 33.8 lbs.
3.	Adjacent to Containment Wall	≤ 31.8 lbs.
4.	Paired with Door Adjacent to Containment Wall	≤ 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.

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

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

$F^N_{\Delta H} \leq 1.36/P$ (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 $F^N_{\Delta H}$ limit of 1.36 results from the LOCA analysis.

The more restrictive of the flow dependent DNBR $F^N_{\Delta H}$ limit and the LOCA $F^N_{\Delta H}$ 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

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

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

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.

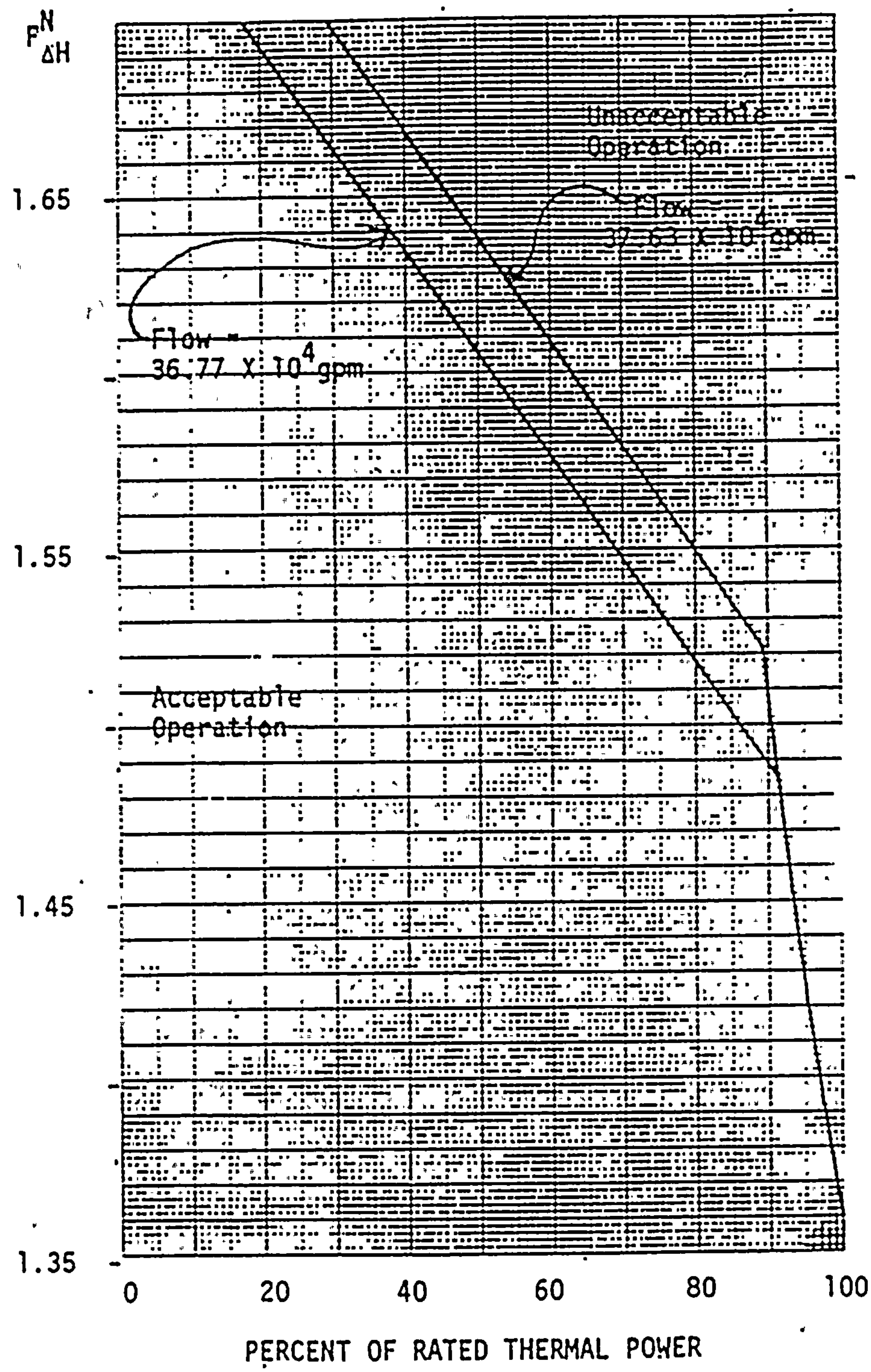


FIGURE B 3/4 2-2 ILLUSTRATIVE EXAMPLE OF

$F^N_{\Delta H}$ 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.