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

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. DPR-70

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

   A. The application for amendment by Public Service Electric and Gas Company, et al. (the licensee) dated June 29, September 25, 1978, February 16, March 1, April 19 and 24, 1979, 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.

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2. Accordingly, the license is amendment 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-70 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 1, 1979

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ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

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

	Pages
Appendix "A"	III
	IX
	1-3
	3/4 1-5
	3/4 1-5a (added)
	3/4 1-18
	3/4 1-20
	3/4 1-26
	3/4 2-3
	3/4 2-7
	3/4 2-11
	3/4 2-12
	3/4 6-8
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	3/4 6-16
	3/4 6-17
	3/4 6-17a (added)
	3/4 10-1
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	3/4 10-2a (delete)
	3/4 10-3
	B 3/4 1-1
	B 3/4 1-2
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Appendix "B"	2.1-1
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.3 INSTRUMENTATION

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BASES

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CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

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PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.

b. Less negative than -3.8×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a, above, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b, above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

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MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.4.a, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -2.9×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -2.9×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.4.b, at least once per 14 EFPD during the remainder of the fuel cycle.

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REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 541°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F with the T_{avg} - T_{ref} Deviation Alarm not reset.

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

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SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the water level of each water source, and

   3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is < 35°F.

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REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, which are inserted in the core, shall be OPERABLE and positioned within $\pm$ 12 steps (indicated position) of their bank demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the bank demand position by more than $\pm$ 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than $\pm$ 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) An analysis of the potential ejected rod worth is performed within 72 hours and the rod worth is determined to be $<$ 0.95% Δk at zero power and $<$ 0.21% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

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* See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) The THERMAL POWER level is reduced to $\leq$ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

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POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within $\pm$ 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determ.ne the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

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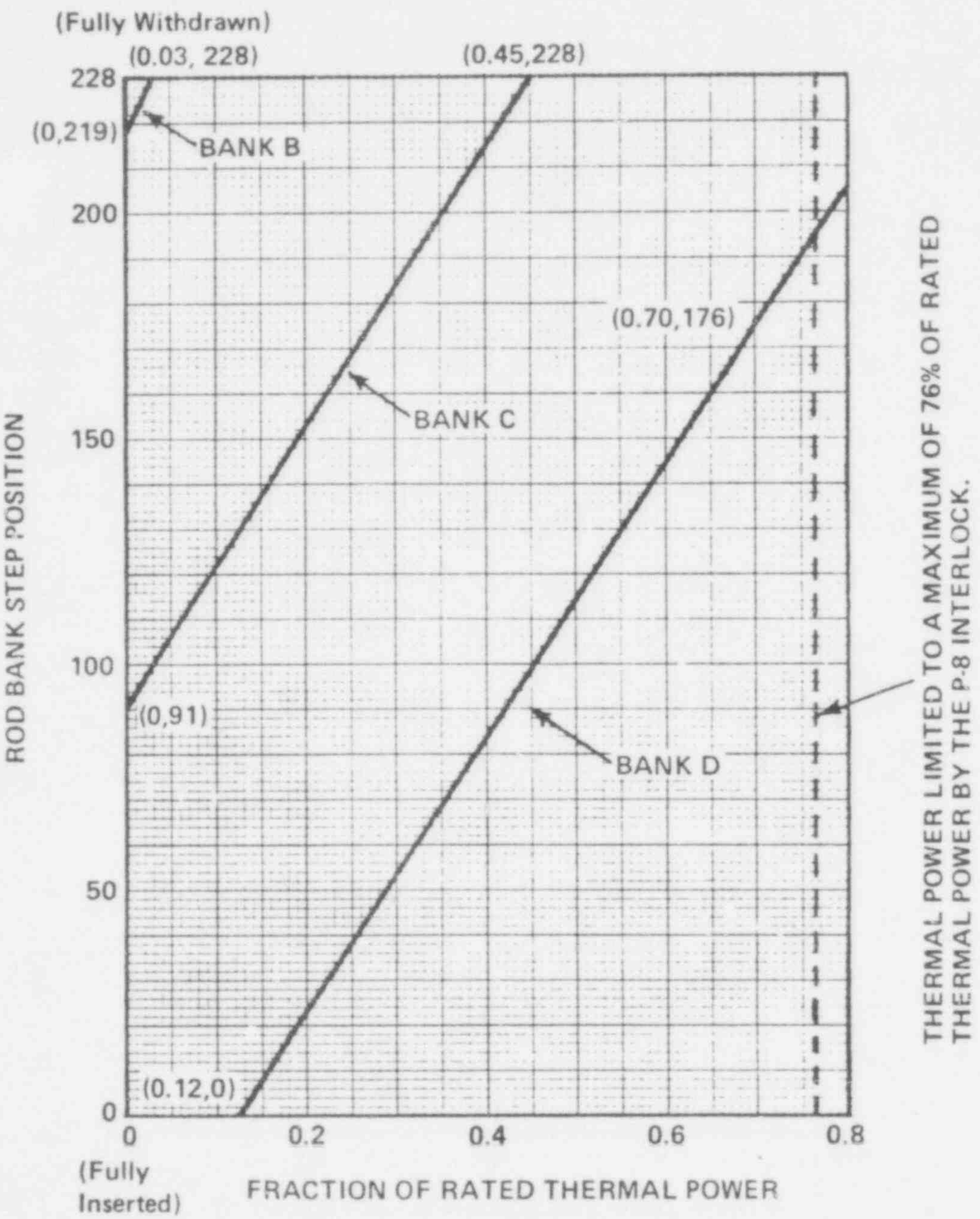

FIGURE 3.1-2 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER THREE LOOP OPERATION

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SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

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

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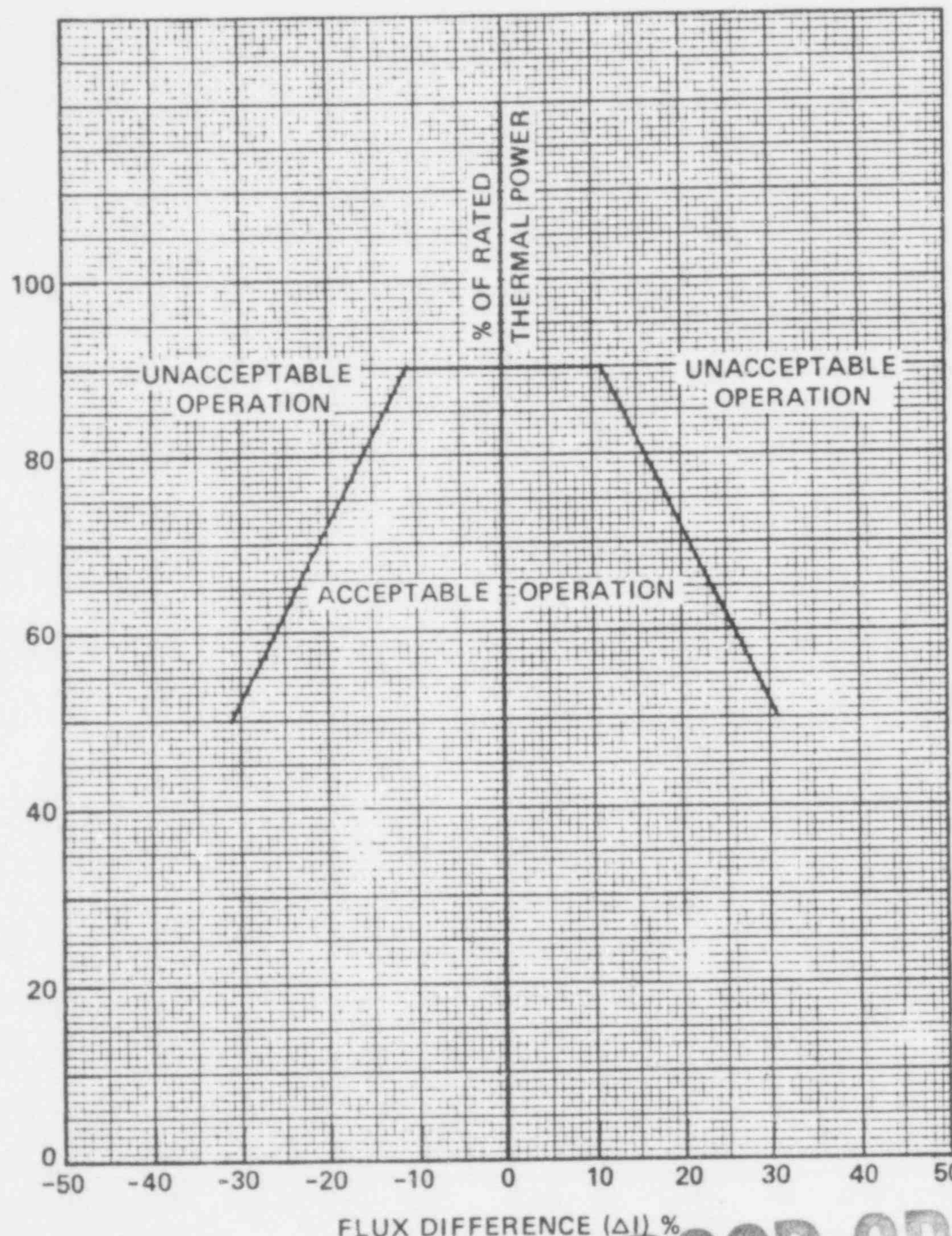

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xp}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods, and

2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, abovt, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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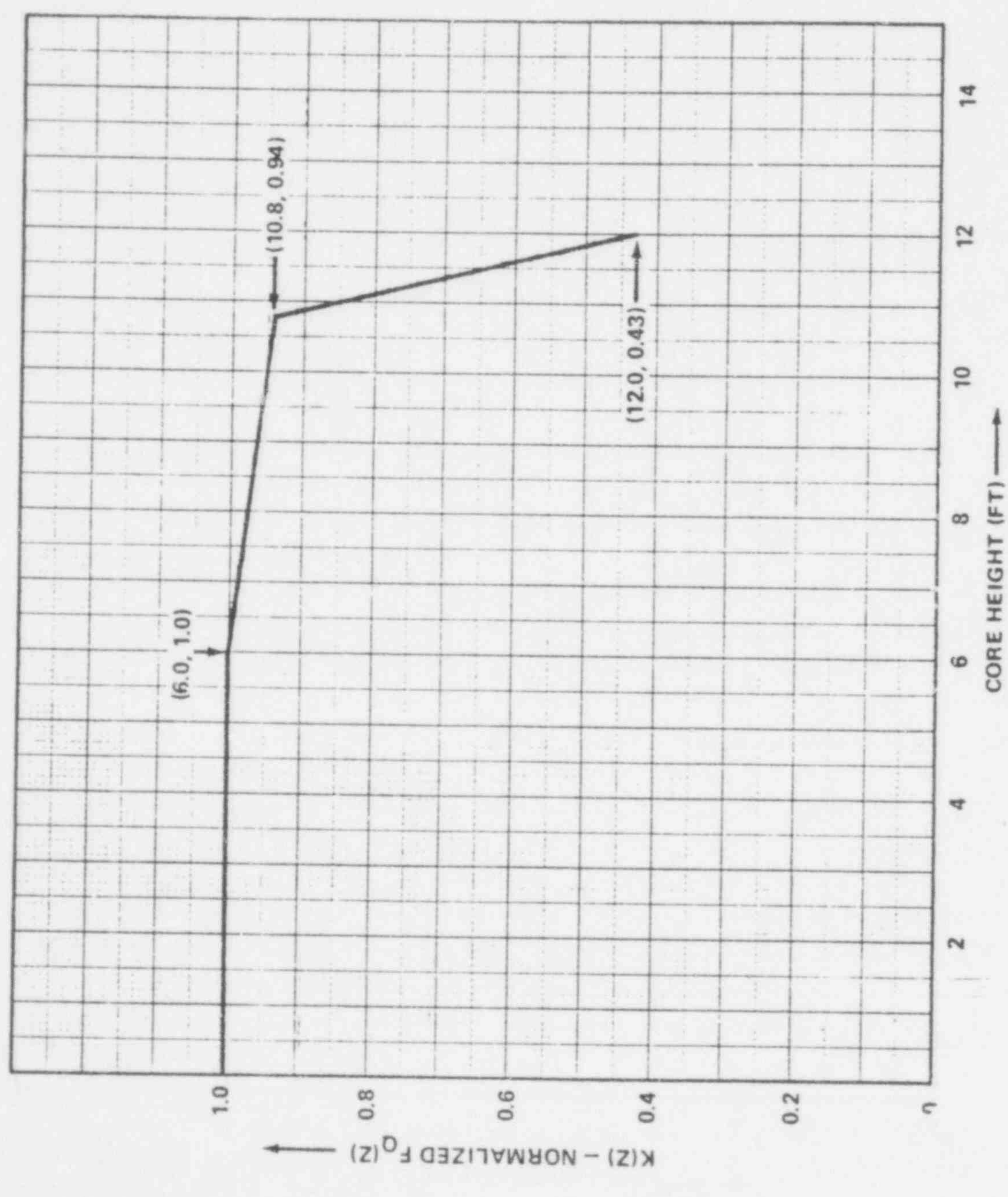

FIGURE 3.2-2

K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

   1. Within 2 hours:

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

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

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

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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

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

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

*See Special Test Exception 3.10.2.

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LIMITING CONDITION FOR OPERATION (Continued)

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

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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

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

   2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

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

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

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

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any 5 of the 10 following locations and shall be determined at least once per 24 hours:

Location (Containment Perimeter)

a. Elev. 84' - North
b. Elev. 106' - North
c. Elev. 78' - Northeast
d. Elev. 84' - East
e. Elev. 106' - East
f. Elev. 136' - South
g. Elev. 84' - South
h. Elev. 106' - Southwest
i. Elev. 121' - West
j. Elev. 136' - West

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective action taken.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a feedwater isolation test signal, each feedwater isolation valve isolates to its isolation position.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each Purge and Pressure-Vacuum Relief valve actuates to its isolation position.

4.6.3.1.3 At least once per 18 months, verify that on a main steam isolation test signal, each main steam isolation valve specified in Table 3.6-1 actuates to its isolation position.

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TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME
A.	PHASE "A" ISOLATION		
1.	1 PR 17*	Pressurizer Relief Tk.-Gas Analyzer Conn.	$\leq$ 10 Sec.
2.	1 PR 18*	Pressurizer Relief Tk.-Gas Analyzer Conn.	$\leq$ 10 Sec.
3.	1 NT 25*	Pressurizer Relief Tk.-N_2 Connection	$\leq$ 10 Sec.
4.	1 WR 80*	Pressurizer Relief Tk.-Primary Water Conn.	$\leq$ 10 Sec.
5.	1 CV 3	CVCS - Letdown Line	$\leq$ 10 Sec.
6.	1 CV 4	CVCS - Letdown Line	$\leq$ 10 Sec.
7.	1 CV 5	CVCS - Letdown Line	$\leq$ 10 Sec.
8.	1 CV 7	CVCS - Letdown Line	$\leq$ 10 Sec.
9.	1 CV 68##	CVCS - Charging Line	$\leq$ 10 Sec.
10.	1 CV 69##	CVCS - Charging Line	$\leq$ 10 Sec.
11.	1 CV 284	CVCS - RCP Seals	$\leq$ 10 Sec.
12.	1 CV 116	CVCS - RCP Seals	$\leq$ 10 Sec.
13.	1 CC 215	Comp. Cooling to Excess Letdown Hx	$\leq$ 10 Sec.
14.	1 CC 113	Comp. Cooling to Excess Letdown Hx	$\leq$ 10 Sec.
15.	1 WL 96*	RC Drain Tk - Gas Analyzer Conn.	$\leq$ 10 Sec.
16.	1 WL 97*	RC Drain Tk - Gas Analyzer Conn.	$\leq$ 10 Sec.
17.	1 WL 98*	RC Drain Tk - Vent Header Conn.	$\leq$ 10 Sec.
18.	1 WL 99*	RC Drain Tk - Vent Header Conn.	$\leq$ 10 Sec.
19.	1 WL 108*	RC Drain Tk - N_2 Connection	$\leq$ 10 Sec.
20.	1 WL 12*	RC Drain Tank Pumps	$\leq$ 10 Sec.
21.	1 WL 13*	RC Drain Tank Pumps	$\leq$ 10 Sec.
22.	1 NT 32*	Accumulator N_2 Supply	$\leq$ 10 Sec.
23.	1 SJ 123*	SI Test Line	$\leq$ 10 Sec.
24.	1 SJ 60*	SI Test Line	$\leq$ 10 Sec.
25.	1 SJ 53*	SI Test Line	$\leq$ 10 Sec.
26.	1 SS 103*	Accumulator Sampling	$\leq$ 10 Sec.
27.	1 SS 27*	Accumulator Sampling	$\leq$ 10 Sec.
28.	1 SS 104*	RC Sampling	$\leq$ 10 Sec.
29.	1 SS 33*	RC Sampling	$\leq$ 10 Sec.
30.	11 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
A. PHASE "A" ISOLATION (Continued)		
31. 12 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.
32. 13 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.
33. 14 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.
34. 1 SS 107*	Pressurizer Liquid Sampling	$\leq$ 10 Sec.
35. 1 SS 49*	Pressurizer Liquid Sampling	$\leq$ 10 Sec.
36. 1 SS 110*	Pressurizer Steam Sampling	$\leq$ 10 Sec.
37. 1 SS 64*	Pressurizer Steam Sampling	$\leq$ 10 Sec.
38. 1 VC 7	Containment Radiation Sampling	$\leq$ 10 Sec.
39. 1 VC 8	Containment Radiation Sampling	$\leq$ 10 Sec.
40. 1 VC 11	Containment Radiation Sampling	$\leq$ 10 Sec.
41. 1 VC 12	Containment Radiation Sampling	$\leq$ 10 Sec.
42. 11 CA 330	Instrument Air Supply	$\leq$ 10 Sec.
43. 12 CA 330	Instrument Air Supply	$\leq$ 10 Sec.
44. 1 DR 29	Demineralized Water Supply	$\leq$ 10 Sec.
45. 11 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
46. 12 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
47. 13 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
48. 14 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
49. 1 WL 16	Containment Sump Discharge	$\leq$ 10 sec.
50. 1 WL 17	Containment Sump Discharge	$\leq$ 10 sec.
51. 1 FP 147*	Fire Protection System	$\leq$ 10 sec.
B. PHASE "B" ISOLATION		
1. 1 CC 118	Component Cooling to RCP	$\leq$ 10 sec.
2. 1 CC 117	Component Cooling to RCP	$\leq$ 10 sec.
3. 1 CC 187	Component Cooling to RCP	$\leq$ 10 sec.
4. 1 CC 136	Component Cooling to RCP	$\leq$ 10 sec.
5. 1 CC 190	Component Cooling to RCP	$\leq$ 10 sec.
6. 1 CC 131	Component Cooling to RCP	$\leq$ 10 sec.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
C. MAIN STEAM ISOLATION		
1. 11 MS 7#	Main Steam Drain	$\leq$ 10 Sec.
2. 12 MS 7#	Main Steam Drain	$\leq$ 10 Sec.
3. 13 MS 7#	Main Steam Drain	$\leq$ 10 Sec.
4. 14 MS 7#	Main Steam Drain	$\leq$ 10 Sec.
5. 11 MS 18#	Main Steam Bypass	$\leq$ 10 Sec.
6. 12 MS 18#	Main Steam Bypass	$\leq$ 10 Sec.
7. 13 MS 18#	Main Steam Bypass	$\leq$ 10 Sec.
8. 14 MS 18#	Main Steam Bypass	$\leq$ 10 Sec.
D. FEEDWATER ISOLATION		
1. 11 BF 19#	Main Feedwater Isolation	$\leq$ 8 Sec.
2. 12 BF 19#	Main Feedwater Isolation	$\leq$ 8 Sec.
3. 13 BF 19#	Main Feedwater Isolation	$\leq$ 8 Sec.
4. 14 BF 19#	Main Feedwater Isolation	$\leq$ 8 Sec.
5. 11 BF 40#	Main Feedwater Isolation	$\leq$ 8 sec.
6. 12 BF 40#	Main Feedwater Isolation	$\leq$ 8 Sec.
7. 13 BF 40#	Main Feedwater Isolation	$\leq$ 8 Sec.
8. 14 BF 40#	Main Feedwater Isolation	$\leq$ 8 Sec.
E. CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF		
1. 1 VC 1*	Purge Supply	$\leq$ 2 Sec.
2. 1 VC 2*#	Purge Supply	$\leq$ 2 Sec.
3. 1 VC 3*#	Purge Exhaust	$\leq$ 2 Sec.
4. 1 VC 4*	Purge Exhaust	$\leq$ 2 Sec.
5. 1 VC 5*	Pressure-Vacuum Relief	$\leq$ 2 Sec.
6. 1 VC 6*#	Pressure-Vacuum Relief	$\leq$ 2 Sec.

5653C1

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
F. MANUAL		
1. (2 valves)#	Pressurizer Dead-Weight Calibrator	Not Applicable
2. 11 CV 98#	CVCS - RCP Seals	Not Applicable
3. 12 CV 98#	CVCS - RCP Seals	Not Applicable
4. 13 CV 98#	CVCS - RCP Seals	Not Applicable
5. 14 CV 98#	CVCS - RCP Seals	Not Applicable
6. 1 SJ 71#	CVCS Flushing Connection	Not Applicable
7. 11 SS 93*#	Steam Generator Sampling	Not Applicable
8. 12 SS 93*#	Steam Generator Sampling	Not Applicable
9. 13 SS 93*#	Steam Generator Sampling	Not Applicable
10. 14 SS 93*#	Steam Generator Sampling	Not Applicable
11. 1 SA 118#	Compressed Air Supply	Not Applicable
12. 1 WL 190#	Refueling Canal Supply	Not Applicable
13. 1 SF 36#	Refueling Canal Supply	Not Applicable
14. 1 WL 191#	Refueling Canal Discharge	Not Applicable
15. 1 SF 22#	Refueling Canal Discharge	Not Applicable
16. 1 VC 9*#	Containment Radiation Sampling	Not Applicable
17. 1 VC 10*#	Containment Radiation Sampling	Not Applicable
18. 1 VC 13*#	Containment Radiation Sampling	Not Applicable
19. 1 VC 14*#	Containment Radiation Sampling	Not Applicable
20. - #	Fuel Transfer Tube	Not Applicable

565302

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
G. CHECK		
1. 1 CV 74	CVCS - Charging Line	Not Applicable
2. 1 CV 296	CVCS - RCP Seals	Not Applicable
3. 1 CC 186	Component Cooling to RCP	Not Applicable
4. 1 CC 208	Component Cooling to RCP	Not Applicable
5. 1 NT 26	Pressurizer Relief Tk. - Nitrogen Conn.	Not Applicable
6. 1 WR 81	Pressurizer Relief Tk. - Primary Water Conn.	Not Applicable

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* May be opened on an intermittent basis under administrative control.

\# Not subject to Type C leakage tests.

\#\#Either valve 1 CV 68 or 1 CV 69 must be OPERABLE.

565303

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. Zero volume percent hydrogen, balance purging air without any free hydrogen, and

b. Two volume percent hydrogen, balance air.

565304

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 10 gpm of 20,100 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 10 gpm of 20,100 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

565305

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained $\leq$ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be $\leq$ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

565306

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4, and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Rangt Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

565307

SPECIAL TEST EXCEPTION

NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set $\leq$ 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

565308

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200°F$, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

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

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the accident and transient analyses.

5653C9

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

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental changes in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value - 3.8×10^{-4} $\Delta k/k/°F$. The MTC value of -2.9×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value -3.8×10^{-4} $k/k/°F$.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains with its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

565 310

3/4.1.2 BORATION SYSTEMS

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

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

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.6% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 5106 gallons of 20,100 ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000 ppm borated water from the refueling water storage tank. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,100 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

5653100

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 541°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

565312

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

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

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

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

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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

565313

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm 5\%$ target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD are derived from the plant nuclear instrumentation system through the AFD Monitor Alarm. A control room recorder continuously displays the auctioneered high flux difference and the target band limits as a function of power level. A first alarm is received any time the auctioneered high flux difference exceeds the target band limits. A second alarm is received if the AFD exceeds its allowable limits for a cumulative time of one hour during any 24 hour time period starting with the occurrence of the first alarm. Time outside the target band is graphically presented on the strip chart.

Figure B 3/4 2-1 shows a typical monthly target band.

565314

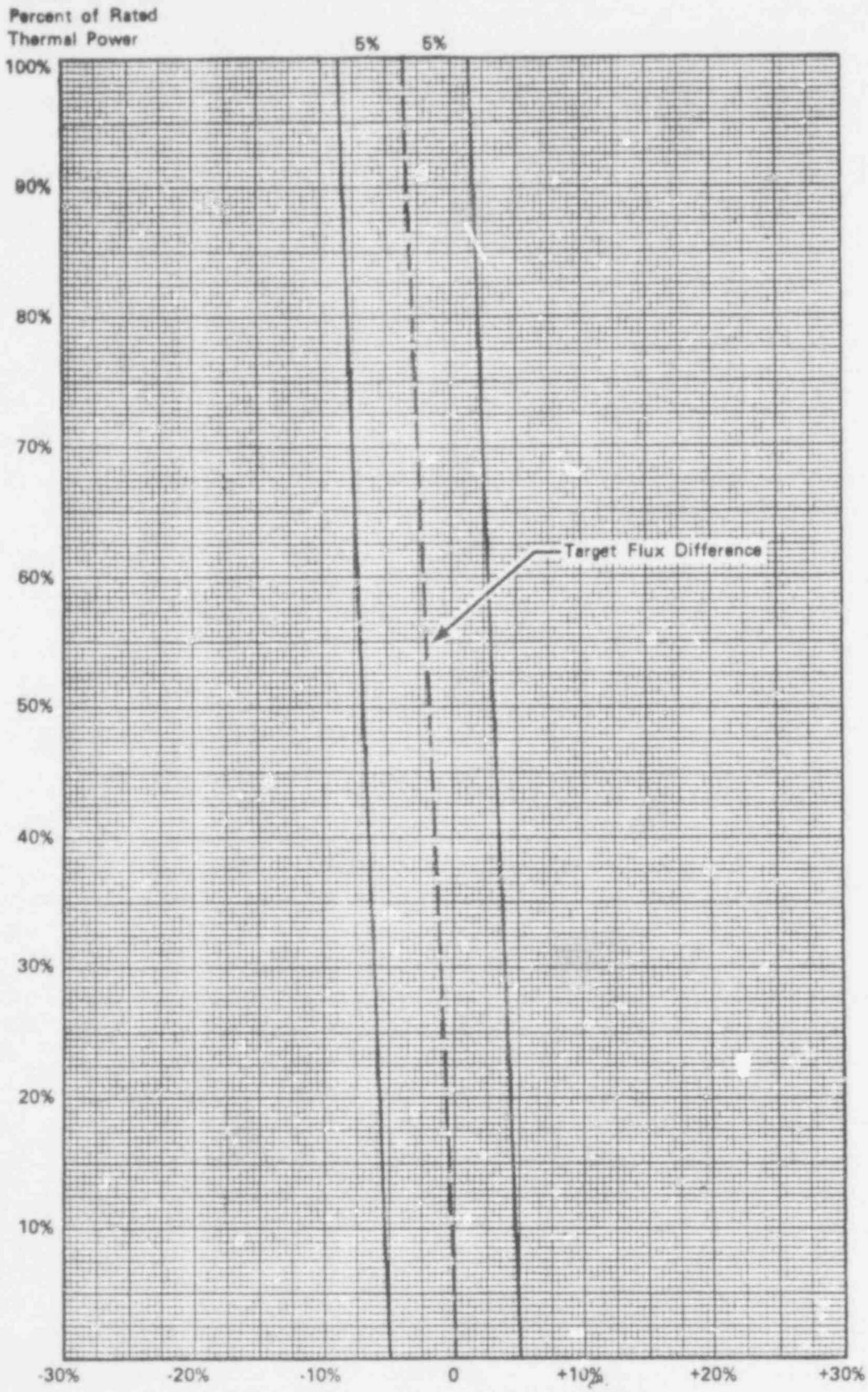

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

565315

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS-

$F_Q(Z)$ and $F^N_{\Delta H}$

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

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

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

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

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

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

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

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

565316

LOW POPULATION ZONE
FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 47 psig and an air temperature of 271°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 143.7 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

565318

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 THERMAL

2.1.1a MAXIMUM ΔT ACROSS CONDENSER DURING NORMAL OPERATION

Objective

To limit thermal stress to the aquatic ecosystem by limiting the maximum ΔT across the condenser during normal operation.

Specification

The maximum ΔT across the condenser shall not exceed 16.5°F during normal operation with all six circulating water pumps operating. In the event that the specification is exceeded corrective action shall be taken to reduce the ΔT to within specification. Such corrective action could include cleaning condenser water boxes or reduction of unit power level, unless an emergency need for power exists.

Monitoring Requirement

The temperature differential across the condenser shall be monitored every hour utilizing the computer printout of the intake and discharge temperature measurements. The intake temperature is measured at each of the two inlets to each condenser shell. The discharge temperature is measured at a point downstream of the condenser in each of the two 84-inch ID discharge lines from each condenser shell. Instrumentation accuracy is ±0.5°F between the range of 32°F and 150°F.

If the plant computer is out of service, the intake and discharge temperatures shall be monitored every two hours utilizing local reading instrumentation until the plant computer is returned to service.

565319

2. The maximum ΔT across the condenser shall not exceed 16.5°F for more than 72 consecutive hours for reasons of pump failure.

3. At no time will the ΔT across the condenser exceed 27.5°F. In the event that either specification is exceeded, corrective action shall be taken to reduce the ΔT to within specification. Such corrective action could include cleaning condenser water boxes or reduction of limit power level, unless an emergency need for power exists.

Monitoring Requirement

The temperature differential across the condenser shall be monitored every hour utilizing the computer printout of the intake and discharge temperature measurements. The intake temperature is measured at each of the two inlets to each condenser shell. The discharge temperature is measured at a point downstream of the condenser in each of the two 84-inch ID discharge lines from each condenser shell. Instrumentation accuracy is ±0.5°F between the range of 32°F and 150°F.

If the plant computer is out of service, the intake and discharge temperatures shall be monitored every two hours utilizing local reading instrumentation until the plant computer is returned to service.

565320

Bases

The condenser cooling water system was designed to operate with a ΔT that would minimize thermal stress to organisms. The U.S. Environmental Protection Agency has set a limit of 27.5°F as a maximum ΔT permitted under the NPDES

2. In the event that fewer than six circulating water pumps are in operation, the maximum condenser discharge water temperature shall not exceed 115°F for more than eight consecutive hours within any 24 hour period.

3. In the event specifications 2.1.2.1 or 2.1.2.2 are exceeded corrective action shall be taken to reduce the condenser discharge water temperature to within specification. Such corrective action could include cleaning condenser water boxes or reduction of unit power level, unless an emergency need for power exists.

Monitoring Requirement

Discharge temperature shall be monitored every hour utilizing the average of the computer printout of the discharge temperature measurements. The discharge temperature is measured at a point downstream of the condenser in each of the two 84-inch ID discharge lines from each condenser shell. Instrumentation accuracy is ±0.5°F between the range of 32°F and 150°F.

If the plant computer is out of service, the discharge temperature shall be monitored every two hours utilizing local reading instrumentation until the plant computer is returned to service.

565324

Bases

Ichthyological Associates (IA) studies performed from June 1968 through December 1973 show 25 records of river temperatures ≥ 84°F. Twenty-one of