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

April 7, 1987

Docket No. 50-354

Mr. Corbin A. McNeill, Jr., Vice President - Nuclear
Public Service Electric & Gas Company
Nuclear Administration Building
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

Subject: Issuance of Amendment No. 3 to Facility Operating License No. NPF-57 - Hope Creek Generating Station

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 3 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment is in response to your letter dated May 30, 1986, as supplemented by letters dated December 24, 1986, and February 6, 1987.

This amendment modifies the Hope Creek Technical Specifications to permit long-term operation with one recirculation loop out of service.

A copy of the related safety evaluation supporting Amendment No. 3 to Facility Operating License No. NPF-57 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosures:
1. Amendment No. 3 to Facility Operating License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

8704140536 870407
PDR ADOCK 05000354
P PDR

Mr. C. A. McNeill
Public Service Electric & Gas Co.

Hope Creek Generating Station

cc:
Gregory Minor
Richard Hubbard
Dale Bridenbaugh
MHB Technical Associates
1723 Hamilton Avenue, Suite K
San Jose, California 95125

Troy B. Conner, Jr., Esquire
Conner & Wetterhahn
1747 Pennsylvania Avenue N.W.
Washington, D.C. 20006

Richard Fryling, Jr., Esquire
Associate General Solicitor
Public Service Electric & Gas Company
P.O. Box 570 T5E
Newark, New Jersey 07101

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 241
Hancocks Bridge, New Jersey 08038

Richard F. Engel
Deputy Attorney General
Division of Law
Environmental Protection Section
Richard J. Hughes Justice Complex
CN-112P
Trenton, New Jersey 08625

Mr. R. S. Salvesen
General Manager-Hope Creek Operation
Public Service Electric & Gas Co.
P.O. Box A
Hancocks Bridge, New Jersey 08038

Mr. B. A. Preston
Public Service Electric & Gas Co.
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Susan C. Remis
Division of Public Interest Advocacy
New Jersey State Department of
the Public Advocate
Richard J. Hughes Justice Complex
CN-850
Trenton, New Jersey 08625

Office of Legal Counsel
Department of Natural Resources
and Environmental Control
89 Kings Highway
P.O. Box 1401
Dover, Delaware 19903

Ms. Rebecca Green
New Jersey Bureau of Radiation
Protection
380 Scotch Road
Trenton, New Jersey 08628

Mr. Anthony J. Pietrofitta
General Manager
Power Production Engineering
Atlantic Electric
1199 Black Horse Pike
Pleasantville, New Jersey 08232

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated May 20, 1986, as supplemented by letters dated December 24, 1986, and February 6, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 3, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1987

ENCLOSURE TO LICENSE AMENDMENT NO. 3

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

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

REMOVE	INSERT
ix	ix (overleaf)
x	x
xvii	xvii (overleaf)
xviii	xviii
2-1	2-1
2-2	2-2 (overleaf)
2-3	2-3 (overleaf)
2-4	2-4
B 2-1	B 2-1
B 2-2	B 2-2 (overleaf)
B 2-3	B 2-3
B 2-4	B 2-4 (overleaf)
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2 (overleaf)
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8 (overleaf)
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60 (overleaf)
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
------	3/4 4-2a
------	3/4 4-2b
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6 (overleaf)

REMOVE	INSERT
B 3/4 1-1	B 3/4 1-1 (overleaf)
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1 (overleaf)
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2
------	B 3/4 4-2a

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

BASES

BASES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 with two recirculation loop operation and shall not be less than 1.07 with single recirculation loop operation, in both cases with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06 with two recirculation loop operation or less than 1.07 with single recirculation loop operation and in both cases with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	$\leq$ 120/125 divisions of full scale	$\leq$ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. ~~Flow Biased Simulated Thermal Power-Upscale~~		
1) Flow Biased	$\leq$ 0.66(w-Δw)+51%** with a maximum of	$\leq$ 0.66(w-Δw)+54%** with a maximum of
2) High Flow Clamped	$\leq$ 113.5% of RATED THERMAL POWER	$\leq$ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
e. Downscale	$\geq$ 4% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1037 psig	$\leq$ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches above instrument zero*	$\geq$ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 12% closed
6. Main Steam Line Radiation - High, High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background

*See Bases Figure B 3/4 3-1.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. Δw = 0 for two recirculation loop operation. Δw = "To be determined at a later date" for single recirculation loop operation.

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation. MCPR greater than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), (GEXL), correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

---

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Amendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

Bases Table B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	
Two Recirculation Loop Operation	2.5
Single Recirculation Loop Operation	6.0
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	
Two Recirculation Loop Operation	6.3
Single Recirculation Loop Operation	6.8
R Factor	1.5
Critical Power	3.6

* The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core. The values herein apply to both two recirculation loop operation and single recirculation loop operation, except as noted.

Bases Table B2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

THERMAL POWER	3323 MW
Core Flow	108.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1089 ft^2
R-Factor	High enrichment - 1.043 Medium enrichment - 1.039 Low enrichment - 1.030

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5. The limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5 shall be reduced to a value of 0.86 times the two recirculation loop operation limit when in single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, or 3.2.1-5, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

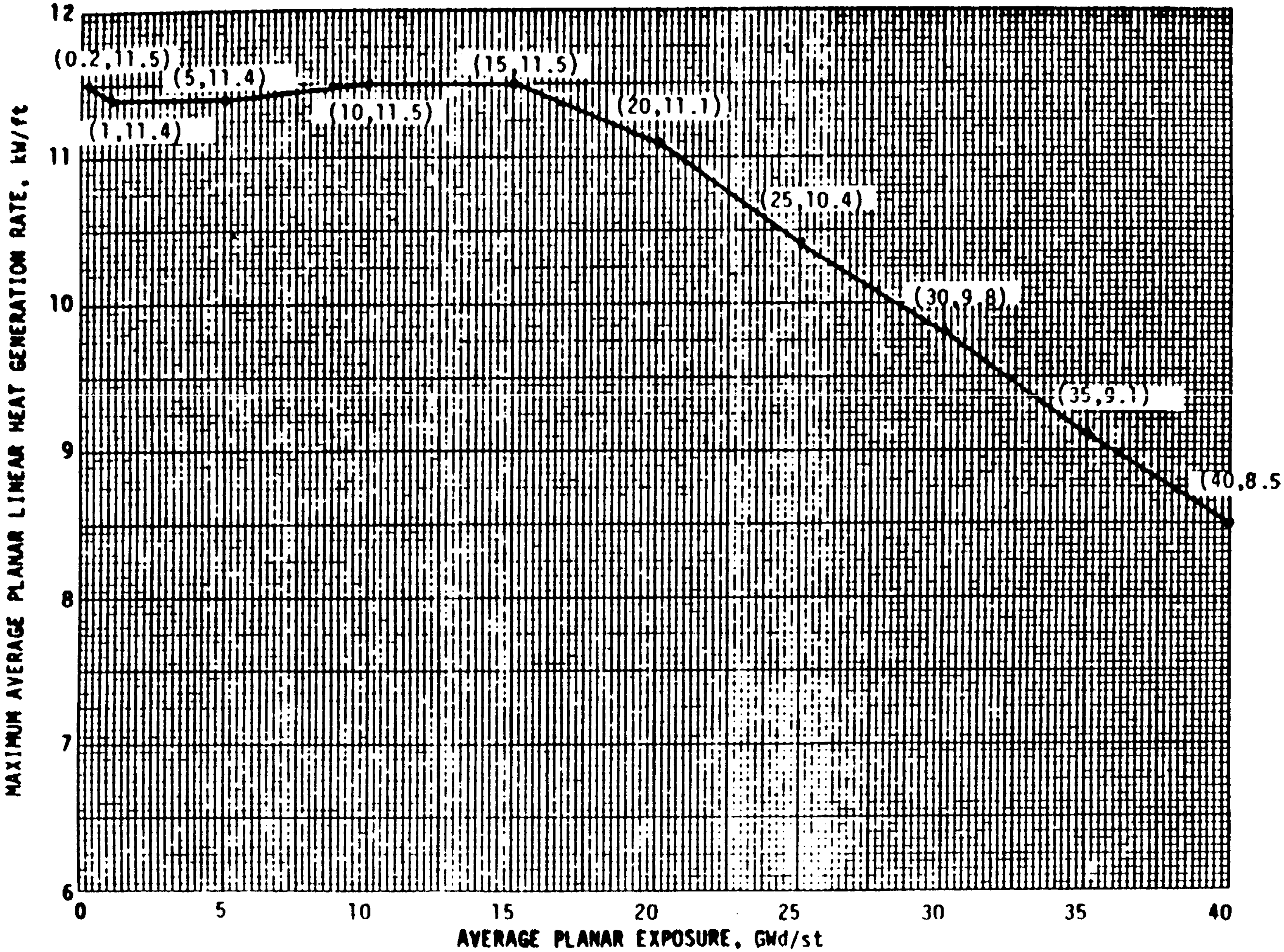
4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE P8CIB071
Figure 3.2.1-1

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66(w-\Delta w)^{**} + 51\%)T$	$S \leq (0.66(w-\Delta w)^{**} + 54\%)T$
$S_{RB} \leq (0.66(w-\Delta w)^{**} + 42\%)T$	$S_{RB} \leq (0.66(w-\Delta w)^{**} + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint values* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

*With CMFLPD greater than the FRTP, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. Δw = 0 for two recirculation loop operation. Δw = "To be determined at a later date" for single recirculation loop operation.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the sum of the MCPR limit shown in Figure 3.2.3-1 plus the feedwater heating capacity adjustment given in Table 3.2.3-1 times the K_f shown in Figure 3.2.3-2, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.688 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{\frac{1}{2}}(0.052),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66(w-Δw) + 40%	≤ 0.66(w-Δw) + 43%
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66(w-Δw) + 42%*	≤ 0.66(w-Δw) + 45%*
b. Inoperative	NA	NA
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High (Float Switch)	109'1" (North Volume) 108'11.5" (South Volume)	109'3" (North Volume) 109'1.5" (South Volume)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108% of rated flow	≤ 111% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

*The rod block function is varied as a function of recirculation loop flow (w) and Δw which is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. The trip setting of the Average Power Range Monitor Rod Block function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b)(c), M(c)	SA	1*
b. Inoperative	NA	S/U(b)(c), M(c)	NA	1*
c. Downscale	NA	S/U(b)(c), M(c)	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA	S/U(b), M	SA	1
b. Inoperative	NA	S/U(b), M	NA	1, 2, 5
c. Downscale	NA	S/U(b), M	SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b), M	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W	NA	2, 5
b. Upscale	NA	S/U(b), W	SA	2, 5
c. Inoperative	NA	S/U(b), W	NA	2, 5
d. Downscale	NA	S/U(b), W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W	NA	2, 5
b. Upscale	NA	S/U(b), W	SA	2, 5
c. Inoperative	NA	S/U(b), W	NA	2, 5
d. Downscale	NA	S/U(b), W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High (Float Switch)	NA	M	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b), M	SA	1
b. Inoperative	NA	S/U(b), M	NA	1
c. Comparator	NA	S/U(b), M	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R	NA	3, 4

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or

b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a) Place the recirculation flow control system in the Local Manual mode, and

b) Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.07 per Specification 2.1.2, and

d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value of 0.86 times the two recirculation loop limit per Specification 3.2.1, and

e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1, 3.2.2 and 3.3.6, and

f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 30% ** of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% ** of rated loop flow.

2. The provisions of Specification 3.0.4 are not applicable.

3. Otherwise be in at least HOT SHUTDOWN within the next 12 hours.

---

*See Special Test Exception 3.10.4.

**Initial values. Final values to be determined during Startup Testing based upon the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With one or two reactor coolant system recirculation loops in operation and total core flow less than 45% but greater than 39%# of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

1. Determine the APRM and LPRM* noise levels (Surveillance 4.4.1.1.4):

   a) At least once per 8 hours, and

   b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

2. With the APRM or LPRM* neutron flux noise levels greater than three times their established baseline noise levels, within 15 minutes initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

d. With one or two reactor coolant system recirculation loops in operation and total core flow less than or equal to 39%# and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1, within 15 minutes initiate corrective action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 or increase core flow to greater than 39%# within 4 hours.

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

a. Reactor THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is less than or equal to 90% of rated pump speed, and

d. Core flow is greater than 39%# when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1-1.

---

*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

#Initial values. Final values to be determined during Startup Testing (core flow with both recirculation pumps at a minimum pump speed).

SURVEILLANCE REQUIREMENTS

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30%# of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50%# of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements or Specifications 4.4.1.1.2b and 4.4.1.1.2c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

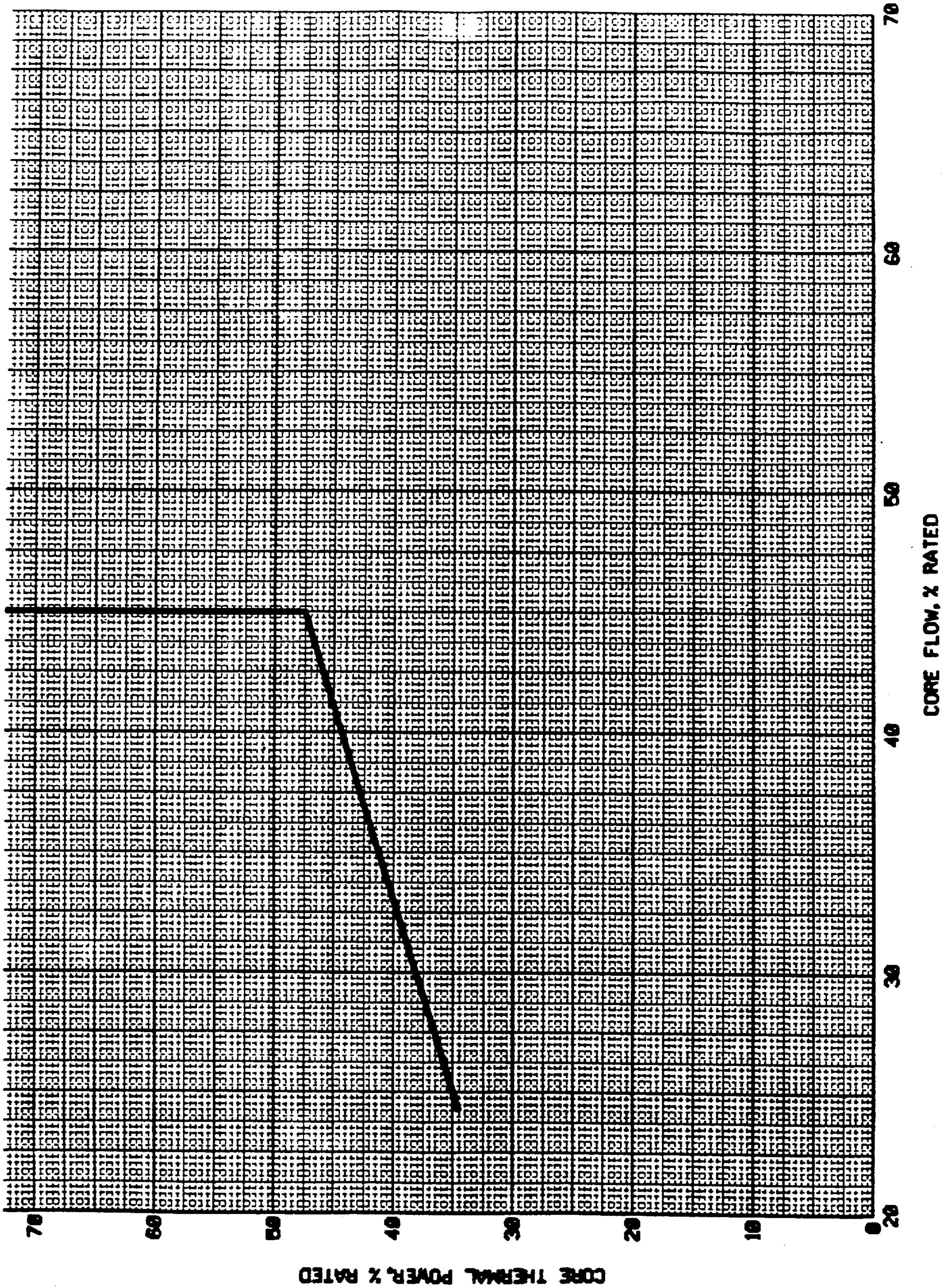
4.4.1.1.3 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 105% and 102.5%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.4 Establish a baseline APRM and LPRM* neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

#Initial values. Final values to be determined during Startup Testing based upon the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

THIS PAGE INTENTIONALLY LEFT BLANK



THERMAL POWER VERSUS CORE FLOW
FIGURE 3.4.1.1-1

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS*

4.4.1.2 All jet pumps shall be demonstrated OPERABLE as follows:

a. Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation pumps are operating in accordance with Specification 3.4.1.3.

   1. The indicated recirculation loop flow differs by more than 10% from the established pump speed-loop flow characteristics.

   2. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

   3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established patterns by more than 10%.

b. During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:

   1. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* pump speed-loop flow characteristics.

   2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from single recirculation loop flow measurements.

   3. The indicated difference-to-lower plenum differential pressure of any individual jet pump differs from established* single recirculation loop patterns by more than 10%.

c. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

*During startup following any refueling outage and in order to obtain single loop or two loop operation baseline data, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon conclusion of the baseline data analysis.

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated core flow with effective core flow** greater than or equal to 70% of rated core flow.

b. 10% of rated core flow with effective core flow** less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare the recirculation loop of the pump with the slower flow not in operation and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

**Effective core flow shall be the core flow that would result if both recirculation loop flows were assumed to be at the smaller value of the two loop flows.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

3/4.1.1 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.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% delta k/k change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% delta k/k would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

3/4.1.3 CONTROL RODS

The specifications of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a withdrawn control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding Safety Limit during the limiting power transient analyzed in Section 15.4 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding Safety Limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER .... 3430 Mwt* which corresponds to 105% of rated steam flow

Vessel Steam Output .... 14.87×10^6 lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure............. 1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 4.1 ft^2

b. Small Breaks 0.09 ft^2,

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.20**

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

**For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 seconds after LOCA regardless of initial MCPR.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-2 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

The K_f factors values shown in Figure 3.2.3-2 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Corrected heat capacity of reactor internals heat nodes.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

For plant operation with single recirculation loop, the MAPLHGR limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5 are multiplied by 0.86. The constant factor 0.86 is derived from LOCA analysis initiated from single loop operation to account for earlier transition at the limiting fuel node compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and the flow biased neutron flux-upscale control rod block trip setpoints must be adjusted to ensure that the MCPR does not become less than the fuel cladding Safety Limit or that > 1% plastic strain does not occur in the degraded situation. The scram setpoints and rod block setpoints are adjusted in accordance with the formula in Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single loop operation is permitted if the MCPR fuel cladding Safety Limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively. MAPLHGR limits are decreased by the factor given in Specification 3.2.1, and MCPR operating limits are adjusted per Specification 3/4.2.3.

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 30%* THERMAL POWER or 50%* rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculating pump and vessel bottom head during the extended operation of the single recirculation loop mode.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference > 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

---

*Initial values. Final values will be determined during Startup Testing based upon the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head, preventing saturation.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservation decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1-12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline date taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 13 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown. The safety/relief valves will be removed and either set pressure tested or replaced with spares which have been previously set pressure tested and stored in accordance with manufacturer's recommendations at the specified frequency.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated May 30, 1986, as supplemented by letters dated December 24, 1986, and February 6, 1987, Public Service Electric and Gas Company (PSE&G), requested changes to the Hope Creek Generating Station (HCGS) Technical Specifications (TS) to permit reactor operation with one of the two recirculation loops out of service (single-loop operation, (SLO)). Presently, the TS require that the reactor be in at least hot shutdown if an idle recirculation loop cannot be returned to service within 12 hours. Resolution of Generic Issue B-19 regarding thermal-hydraulic stability has provided a basis to permit operation in the single loop mode with appropriate restrictions relating to stability concerns. General Electric Company (GE), in SIL No. 380, Revision 1, addressed these concerns by providing the boiling water reactor licensees generic guidance for actions which suppress thermal-hydraulic instability induced neutron flux oscillations.

The proposed changes requested by the licensee consist of: (1) deletion of the Technical Specification requirement restricting SLO which involves limiting the allowable pump speed during SLO, increasing the Minimum Critical Power Ratio (MCPR) Safety Limit by 0.01, establishing appropriate Average Power Range Monitor (APRM) Flow Biased Scram Trip setpoints, revising the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits and revising the Rod Block Monitor (RBM)/APRM Control Rod Block setpoints; (2) for single-loop operation, incorporating requirements in the Technical Specifications which should result in the detection and suppression of thermal-hydraulic instability induced neutron flux oscillations if they should occur; and (3) including an applicability section and appropriately revising SURVEILLANCE requirements and the BASES.

The proposed amendment was noticed in the FEDERAL REGISTER on January 28, 1987 (52 FR 2889). The May 30, 1986, letter had requested that the words "for initial core loading only" be added to the double-asterisk footnote on Table 3.3.6-2 on page 3/4 3-59 of the Technical Specifications. Initial core loading for HCGS was completed on April 27, 1986.

In a letter dated February 6, 1987, the licensee requested the note be removed in its entirety, because the note, if revised as requested in the May 30, 1986, letter, would no longer be applicable in the post-initial core load condition. The request to delete the note clarifies the note's applicability; therefore, the February 6, 1987, revision to the amendment request does not affect the proposed no significant hazards consideration discussed in the Federal Register notice.

2.0 EVALUATION

In its letter dated May 30, 1986, the licensee provided a General Electric (GE) report entitled, "Hope Creek Single-Loop Operation Analysis." This report evaluated the SLO safety issues in order to justify extended operation with one recirculation loop out of service. The staff evaluation of the SLO safety issues and the proposed Technical Specification changes follow.

2.1 Minimum Critical Power Ratio (MCPR) Fuel Cladding Integrity Safety Limit

For SLO, the MCPR fuel cladding integrity safety limit is increased by 0.01 to account for increased uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings. The limiting transients were analyzed to verify that there is more than enough margin during SLO to compensate for this increase in safety limit. The proposed change is acceptable.

2.2 MCPR Operating Limit - Accidents (Other Than Loss of Coolant Accident (LOCA)) and Transients Affected by One Recirculation Loop Out of Service

- One Pump Seizure Accident

  A plant specific analysis was not performed for this event. The licensee stated that GE completed SLO analyses for 23 domestic BWRs and 4 overseas BWRs thus establishing a data base of SLO information and analysis techniques which could be applied to HCGS. The results of the SLO analyses for the other plants have always demonstrated that SLO was a non-limiting event and boiling transition was not experienced during recirculation pump seizure. Also, there was significant margin to the safety limit MCPR, even though a small fraction of the fuel is permitted to exceed the safety limit MCPR for this event. Because the safety limit MCPR is not exceeded, the 10 CFR 100 "small fraction" limits are satisfied for pump seizure during SLO. Since significant margin to the safety limit MCPR has been previously demonstrated for other BWRs, a plant specific analysis for HCGS is not necessary.

- Abnormal Operating Transients

  The SLO abnormal operating transients were analyzed assuming an initial power of 75% of rated and 60% of rated core flow. The most limiting events were pressurization transients resulting from Feedwater Controller Failure-Maximum Demand (FWCF) and Generator Load Rejection with Bypass Failure (LRBPF). The corresponding MCPRs were 1.17 and 1.16. Although the increased uncertainties in core total flow and TIP readings resulted in a 0.01 increase in the MCPR fuel cladding integrity safety limit to 1.07 during SLO, the limiting transients analyzed in the GE report indicate that there is more than enough MCPR margin during SLO to compensate for this increase in safety limit. For SLO at off-rated conditions, the steady-state operating MCPR limit is established by flow dependent K_f curves.

Since the maximum core flow runout during SLO is only about 60% of rated, the current flow dependent MCPR limits which are generated based on the flow runout up to rated core flow are also adequate to protect the flow runout events during SLO. Since the SLO transient analysis is bounded by the two-loop transient analysis, power dependent MCPR curves used for two-loop operation are also applicable for SLO.

- Rod Withdrawal Error

The rod withdrawal error at rated power is analyzed in the FSAR for the initial core load. This analysis was performed to demonstrate that, even if the operator had ignored all instrument indications and alarms during the course of the transient, the rod block system would stop rod withdrawal at a MCPR which is higher than the fuel cladding integrity safety limit. Correction of the rod block equation for single-loop operation assures that the MCPR safety limit is not violated.

Additionally, SLO results in backflow through 10 of the 20 jet pumps while flow is being supplied to the lower plenum from the 10 active jet pumps. Because of this backflow through the inactive jet pumps, the present rod block equation and Average Power Range Monitor (APRM) settings were modified. The licensee has also modified the two-pump rod block equation and APRM settings that exist in the TS for SLO.

The staff finds that one-loop transients and accidents other than LOCA, which is discussed in Section 2.4 of this evaluation, are bounded by the two-loop operation analysis, which has been found to be acceptable.

2.3 Loss of Coolant Accident Analysis

The licensee evaluated the LOCA for SLO. That evaluation utilized the GE methodology outlined in NEDO-20566-2, Rev. 1. Results show the maximum average planar linear heat generation rate

(MAPLHGR) needed to be multiplied by a factor of 0.86. The methodology for the SLO MAPLHGR assumes a boiling transition time of 0.1 seconds. The evaluation methodology was approved by the staff in a letter dated March 5, 1986 (H. Berkow to J. Quirk, GE). The licensee's use of this methodology and its evaluation are acceptable.

2.4 Stability Analysis

With one recirculation loop not in service, the primary contributing factors to stability performance are the power/flow ratio and the recirculation loop characteristics. At forced circulation with one recirculation loop not in operation, the reactor core stability is influenced by the inactive recirculation loop. Staff evaluations have considered whether increased noise in SLO was being caused by reduced stability margin as SLO core flow was increased. Results of analyses and tests indicate that the SLO stability characteristics are not significantly different from two-loop operation. At low core flows, SLO may be slightly less stable than two-loop operation but as core flow is increased and reverse flow is established, the stability performance is similar. At higher core flows with substantial reverse flow in the inactive recirculation loop, the effect of cross flow on the flow noise results in an increase in system noise (jet pump, core flow and neutron flux noise); however, core thermal-hydraulic stability margin remains high, similar to two-loop operation. GE has developed Service Information Letter-380, Revision 1 (February 10, 1984) informing plant operators how to recognize and suppress unanticipated oscillations when encountered during plant operation. The NRC staff has approved the GE generic stability analysis for application to SLO in a letter dated April 24, 1985 (C. Thomas, NRC to H. Pfefferlen, GE), provided that the recommendations of SIL-380 have been incorporated into the plant TS.

The staff compared these GE recommendations with the proposed HCGS TS for SLO. The proposed changes are in conformance with the SIL-380, Revision 1 recommendations and are, therefore, acceptable.

2.5 Jet Pump Surveillance

Significant increase in APRM noise and core plate Δp fluctuations have been observed in some plants during operation at high flow rates while in SLO. Even though the fluctuations are no longer associated with thermal-hydraulic stability (see Section 2.4), there is a remaining concern that excessive vibration leading to mechanical failure of a jet pump could result. Since this could lead to unacceptable consequences if the condition existed during a LOCA, we require that an acceptable jet pump surveillance requirement be in place during single-loop operation. A similar concern was addressed in IE Bulletin 80-07 for the resolution of a generic problem relating to the integrity of the hold down beams for BWR jet pumps. In response to this bulletin, many licensees incorporated jet pump surveillance requirements into the Technical Specifications. Hope Creek has incorporated such requirements into its Technical Specifications and we find them acceptable for assurance of jet pump integrity while in SLO.

2.6 Technical Specification Changes

The proposed TS changes for SLO are as follows:

1. Incorporation of the safety limit MCPR for SLO in TS 2.1.2, Bases for Safety Limits, Section 2.1, Bases for APRM Set Points, Section 3/4.2.2, Bases for Control Rods Section 3/4.1.3, and Bases for Minimum Critical Power Ratio Section 3/4.2.3.

2. Revision of the APRM scram, APRM Rod Block and Rod Block Monitor setpoints and allowable values to include SLO in Table 2.2.1-1, TS 3.2.2 and Table 3.3.6-2.

3. Incorporation of uncertainties for core total flow and TIP readings for SLO in the Bases Table B2.1.2-1.

4. Incorporation of a MAPLHGR multiplier for SLO in TS 3.2.1 and Bases for Average Planar Liner Heat Generation Rate, Section 3/4.2.1.

5. Incorporation of 0.1 sec time for loss of nucleate boiling during SLO in Bases Table B 3.2.1-1 of significant parameters to LOCA.

6. Revision of the jet pump surveillance requirements, TS 4.4.1.2 to include SLO and baseline data.

7. Revision of the section describing the recirculation system recirculation loops, T.S 3/4.4.1 to include SLO.

8. Revision of Figure 3.4.1.1-1, which illustrates thermal power vs. core flow to be compatible with the TS.

9. Revision of section 3/4.4.1 Bases for the recirculation system to include factors necessary for SLO. Also revised in this section was the basis of temperature differences greater than 145°F between the reactor vessel bottom head coolant and coolant in the upper region of the reactor vessel as related to vessel stress.

We have reviewed the changes and find them consistent with GE-SIL-380 and the results of the GE analysis and also with the SLO TS approved for other BWR 4/5s (e.g., Susquehanna 1 and 2). Accordingly, the proposed changes are acceptable.

The licensee also proposed other TS changes as follows:

10. TS 3.4.1.3 and 4.4.1.3 to be restricted in applicability to two loop operation and also to reflect loop flows instead of recirculation pump speeds.

11. TS 3.2.2 footnote to be revised to extend above 90% thermal power, the region in which APRM gain adjustment rather than APRM setpoint adjustment is allowed. This revision has also been accepted on other BWRs.

12. Table 3.3.6-2 to be revised to eliminate the footnote for the source range monitor downscale trip setpoint option of 0.7 cps. The footnote was developed for and used during the initial core loading only.

We have reviewed the changes and find them consistent with GE-SIL-380 and the results of the GE analysis and also with SLO TS changes approved for other BWR/4s (e.g., Susquehanna 1 and 2).

We have reviewed these changes and find them acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative 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. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 2889) on January 28, 1987, and consulted with the state of New Jersey. No public comments were received, and the state of New Jersey did not have any comments.

We have concluded, based on the considerations 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.

Principal Contributors: Don Katze, Reactor Systems Branch, DBL
David Wagner, BWR Project Directorate No. 3, DBL

Dated: April 7, 1987

AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. NPF-57
HOPE CREEK GENERATING STATION

DISTRIBUTION:

Docket No. 50-354
NRC PDR
Local PDR
PRC System
NSIC
BWD-3 r/f
DWagner (2)
EHylton
EAdensam
Attorney, OELD
CMiles
RDiggs
JPartlow
EJordan
BGrimes
LHarmon
TBarnhart (4)
EButcher