ATTACHMENT C

LIST OF AFFECTED TECHNICAL SPECIFICATION PAGES

Changed Page #	Comment
XIX	
2-4	Table 2.2.1-1
B2-2	
B2-3	
B2-4	Deleted
B2-5	Deleted
B2-6	Deleted
B2-7	Deleted
3/4.1-16	
3/4.2-1	
3/4.2-2b	(New) Figure 3.2.3-1b
3/4.2-3	
3/4.2-4	Insert "A"
3/4.2-5	Redrawn Figure 3.2.3-1a
3/4.2-5a	(New) Figure 3.2.3-1b
3/4.2-7	Insert "B"
3/4.3-39	
3/4.3-53	Insert C1 & C2 to Table 3.3.6-2
3/4.3-54	Table 3.3.6-2 (Cont'd)
3/4.4-1	
3/4.4-2	Insert "D"
3/4.4-6	
3/4.6-1	
3/4.7-34	Insert "E"
3/4.10-4	Deleted
3/4.10-7	Deleted
B3/4.2-1	Insert "F"
B3/4.2-2	Deleted
B3/4.2-3	Insert "G" (2 Pages)
B3/4.2-4	
B3/4.2-6	
B3/4.3-3	Insert "H"
B3/4.4-1	Insert "I"
B3/4.5-2	
B3/4.7-5	Insert "J"
5-4	Design Features, Insert "K"

0017

ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATION CHANGES

5102K

LIST OF FIGURES

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120 divisions of full scale	≤ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	0.58W + 59% ≤ ~~0.66W + 51%~~ with a maximum of	0.58W + 62% ≤ ~~0.66W + 54%~~ with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	0.58W + 54.3% ≤ ~~0.66W + 45.7%~~ with a maximum of	0.58W + 57.3% ≤ ~~0.66W + 48.7%~~ with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1043 psig	≤ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High	≤ 3 x full power background	≤ 3.6 x full power background
7. Primary Containment Pressure - High	≤ 1.69 psig	≤ 1.89 psig
8. Scram Discharge Volume Water Level - High	≤ 767' 5¼"	≤ 767' 5¼"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 414 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A
13. Control Rod Drive		
a. Charging Water Header Pressure-Low	≥ 1157 psig	≥ 113[illegible] psig
b. Delay Timer	≤ 10 seconds	≤ 10 seconds

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

00 19

SAFETY LIMITS

BASES

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. These conditions are:~~

~~Pressure:~~	~~800 to 1400 psia~~
~~Mass Flow:~~	~~0.1×10^6 to 1.25×10^6 lb/hr-ft^2~~
~~Inlet Subcooling:~~	~~0 to 100 Btu/lb~~
~~Local Peaking:~~	~~1.61 at a corner rod to 1.47 at an interior rod~~

Le

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

SAFETY LIMITS

BASES

THERMAL POWER, High Pressure and High Flow (Continued)

Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	64 Rods in an 8 x 8 array	

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1, the nominal values of the core parameters listed in Bases Table B2.1.2-2, and the relative assembly power distribution shown in Bases Table B2.1.2-3. Bases Table B2.1.2-4 shows the R-factor distributions that are input to the statistical model which is used to establish the Safety Limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

e

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 Admendment 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	8.7
Single Recirculation Loop Operation	6.8
R Factor	1.6
Critical Power	3.6

DELETE PAGE

*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	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
R-Factor	1.038 - 0 GWD/t 1.031 - 7 GWD/t 1.030 - 15 GWD/t 1.033 - 20 GWD/t

DELETE
PAGE

Bases Table B2.1.2-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.375 to 1.425	5.1
1.325 to 1.375	7.3
1.275 to 1.325	7.8
1.225 to 1.275	9.8
1.175 to 1.225	7.3
1.125 to 1.175	11.8
1.075 to 1.125	4.7
1.025 to 1.075	4.7
≤1.025	41.5
	100.0

DELETE PAGE

C

Bases Table B2.1.2-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

8x8 Rod Array

R-Factor	Rod Sequence No.
1.038	1
1.038	2
1.037	3
1.037	4
1.035	5
1.035	6
1.030	7
≤ 1.030	8 through 64

DELETE PAGE

00 26

REACTIVITY CONTROL SYSTEM

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1# and 2*#, when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER, the minimum allowable low power setpoint.

ACTION:

a. With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

reaching 20% of RATED THERMAL POWER

a. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpo of making the reactor critical, and in OPERATIONAL CONDITIC prior to ~~RWM automatic initiation~~ when reducing THERMAL POWER, by verifying proper annunciation of the selection error of at least one out-of-sequence control rod.

b. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within 1 hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying the control rod patterns and sequence input to the RWM computer is correctly loaded following any loading of the program into the computer.

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

#

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, ~~and~~ 3.2.1-2, and 3.2.1-3. ~~The limits of Figures 3.2.1-1 and 3.2.1-2 shall be reduced to a value of 0.85 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 Figures 3.2.1-1, ~~and~~ 3.2.1-2, and 3.2.1-3 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, ~~and~~ 3.2.1-2, and 3.2.1-3:

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.

MAPLHGR Vs Average Planar Exposure
Fuel Types BC300D and BC320B

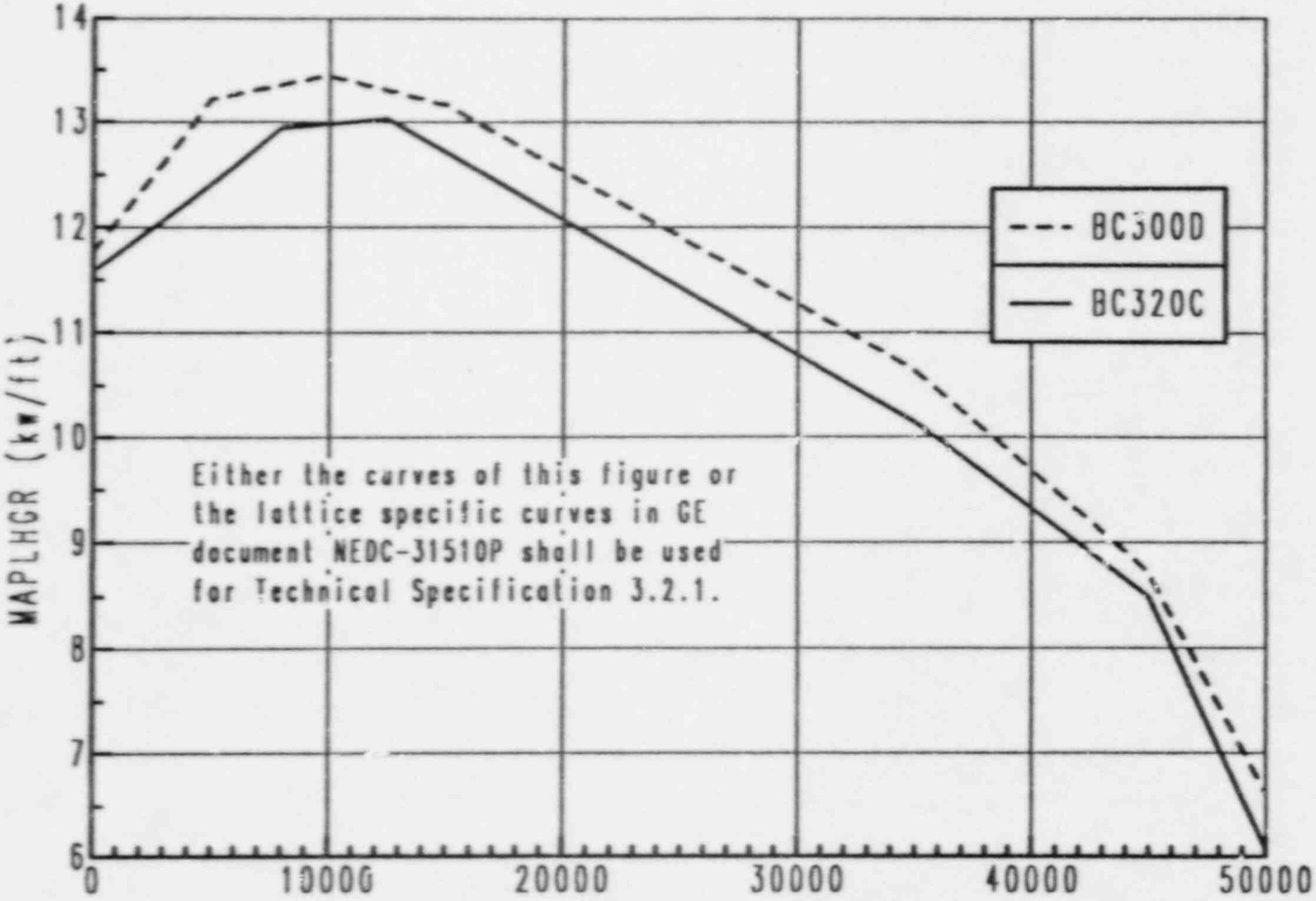


Figure 3.2.1-3

POWER DISTRIBUTION LIMITS

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 simulated thermal power-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

a. Two Recirculation Loop Operation

S less than or equal to ~~(0.66W + 51%)T~~ (0.58W + 59%)T

S_{RB} less than or equal to ~~(0.66W + 42%)T~~ (0.58W + 47%)T

b. Single Recirculation Loop Operation

S less than or equal to ~~(0.66W + 42.7%)T~~ (0.58W + 54.3%)T

S_{RB} less than or equal to ~~(0.66W + 36.7%)T~~ (0.58W + 42.3%)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 108.5 million lbs/hr,

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY or the value 1.0. T is always less than or equal to 1.

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 simulated thermal power-upscale control rod block trip setpoint set less conservatively than S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} 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

4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and control rod block trip setpoint 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 MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

POWER DISTRIBUTION LIMITS

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 MCPR limit determined from Figure 3.2.3-1 times the K_f determined from Figure 3.2.3-2 for two recirculation loop operation and shall be equal to or greater than the MCPR limit determined from Figure 3.2.3-1 + 0.01 times the K_f determined from Figure 3.3.3-2 for single recirculation loop operation.~~ e

Insert A

APPLICABILITY:

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

ACTION

~~With MCPR less than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.~~ e

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ_{ave} = 0.86 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

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 MCPR.

Insert A

3.2.3 - The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit determined from:

a. Single Recirculation Loop Operation
Figure 3.2.3-1a (Curve A for a RBM setpoint of 106% or Curve B for a RBM setpoint of 110%) plus 0.01, times the K_f determined from Figure 3.2.3-2.

b. Two Recirculation Loop Operation
Figure 3.2.3-1a (Curve A for a RBM setpoint of 106% or Curve B for a RBM setpoint of 110%) times the K_f determined from Figure 3.2.3-2.

c. Two Recirculation Loop Operation with Main Turbine Bypass Inoperable
Figure 3.2.3-1b times the K_f determined from Figure 3.2.3-2, for two recirculation loop operation, with the main turbine bypass system inoperable per Specification 3.7.10 (any RBM setpoint determined per Specification Table 3.3.6-2 may be used).

d. Two Recirculation Loop Operation with End-of-Cycle Recirculation Pump Trip System Inoperable
Figure 3.2.3-1b times the K_f determined from Figure 3.2.3-2, for two recirculation loop operation, with the end-of-cycle recirculation pump trip system inoperable as directed by Specification 3.3.4.2 (any RBM setpoint determined per Specification Table 3.3.6-2 may be used).

ACTION:

a. With MCPR less than the applicable MCPR limit as determined for one of the above conditions:

1. Initiate corrective action within 15 minutes, and

2. Restore MCPR to within the required limit within 2 hours.

3. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

b. When operating in a condition not identified above, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within 4 hours.

replace

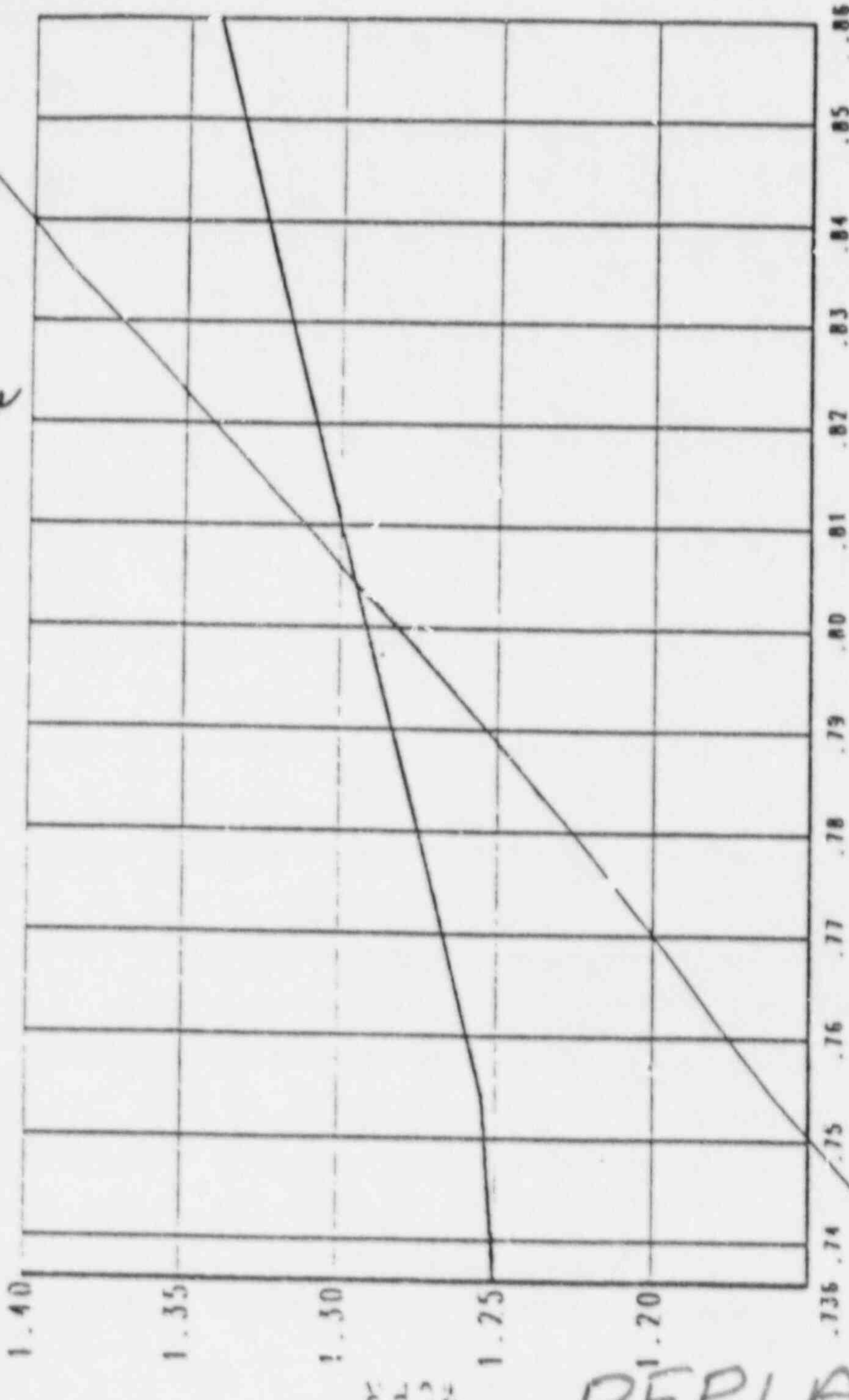
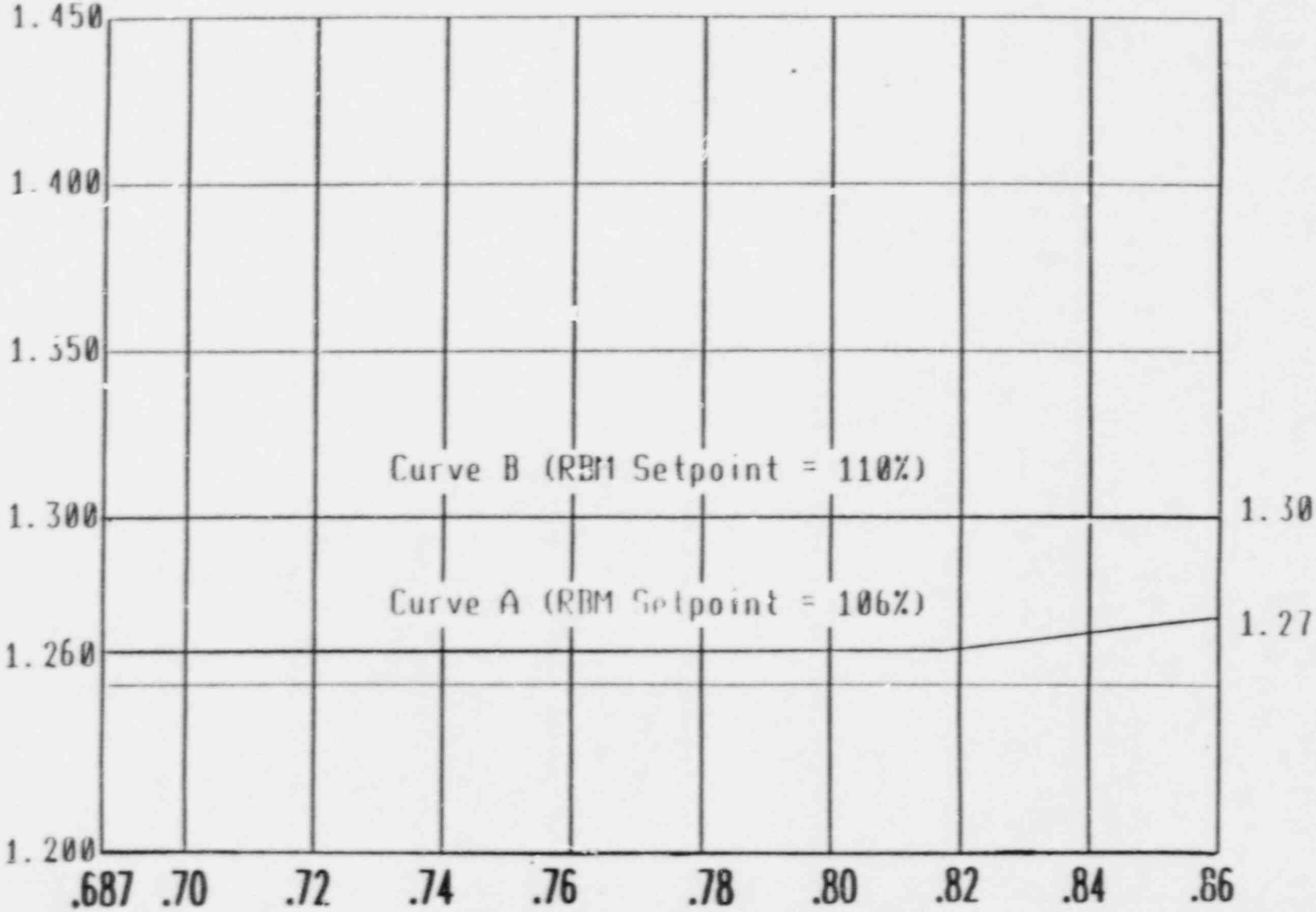


Figure 3.2.3-1 MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW

REPLACE WITH NEW PAGE

POWER DISTRIBUTION LIMITS
MCPR



MCPR VERSUS τ AT RATED FLOW

Figure 3.2.3-1a

00 33

POWER DISTRIBUTION LIMITS
MCPR

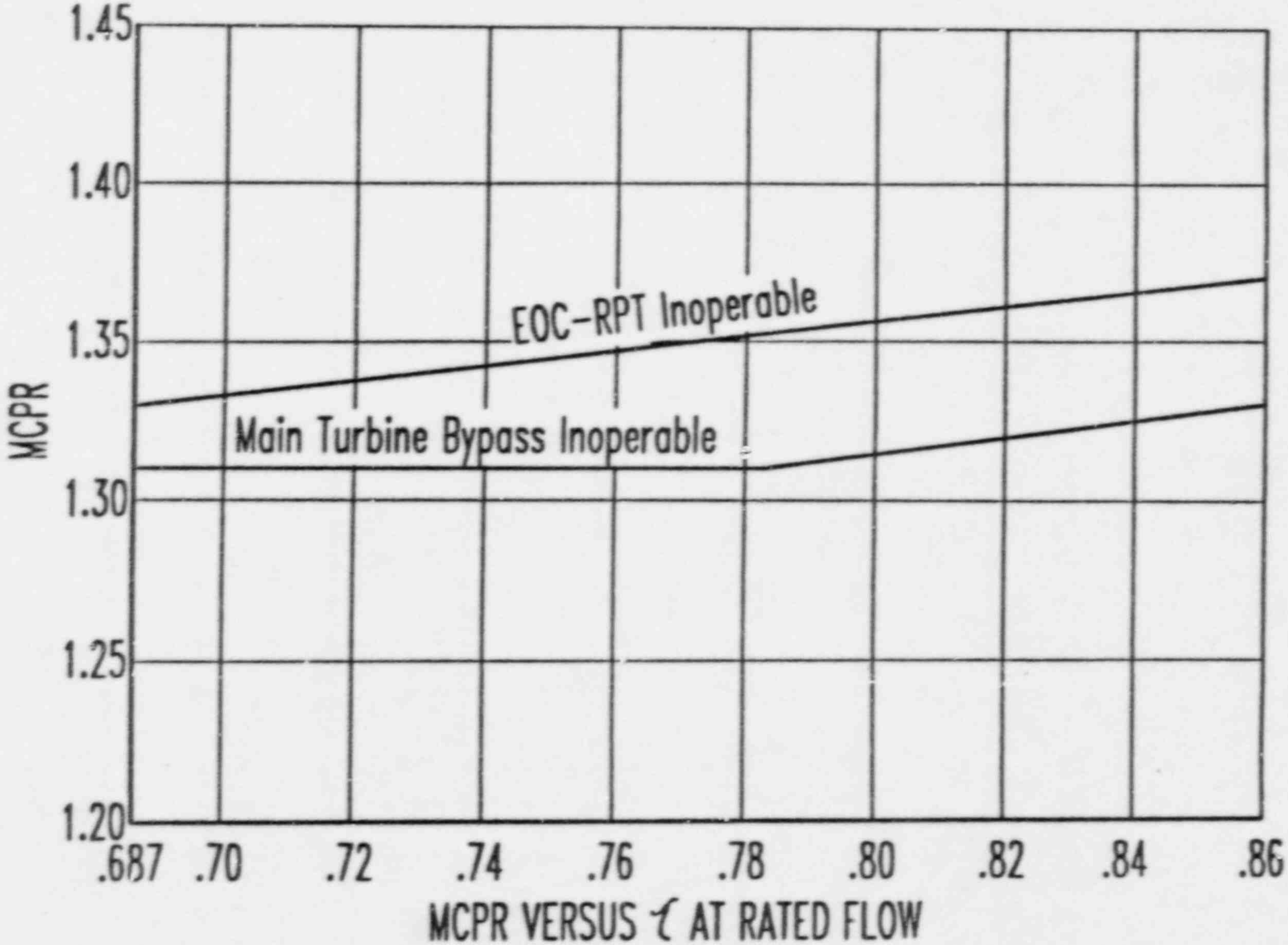


Figure 3.2.3-1b

00 34

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

Insert B

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed, ~~13.4 kw/ft.~~

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

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. Intially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

Insert B

a. 13.4 kw/ft for fuel types:

   1. 8CRB176

   2. 8CRB219

   3. BP8CRB299L

b. 14.4 kw/ft for fuel types:

   1. BC300D

   2. BC320C

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement(s) for one trip system and:

    1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 1 hour.

    2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

(insert into both d+e)

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours, ~~or~~ reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour, ~~or~~ reduce THERMAL POWER to less than 30% RATED THERMAL POWER within the next 6 hours.

Otherwise, either:

1. Increase the MINIMUM CRITICAL POWER (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 hour, or

2.

f. The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
1) Two Recirculation		
Insert C1 → Loop Operation	~~≤0.66 W + 38%~~	~~≤0.66 W + 41%~~
2) Single Recirculation		
Insert C2 → Loop Operation	~~≤0.66W + 32.7%~~	~~≤0.66W + 35.7%~~
b. Inoperative	N.A.	N.A.
c. Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation	≤0.58 W + 47%*	≤0.58 W + 50% *
Loop Operation	~~≤0.66 W + 42%*~~	~~≤0.66 W + 45%*~~
2) Single Recirculation	≤0.58W + 42.3%*	≤0.58W + 45.3% *
Loop Operation	~~≤0.63W + 36.7%*~~	~~≤0.66W + 39.7%*~~
b. Inoperative	N.A.	N.A.
c. Downscale	>5% of RATED THERMAL POWER	>3% of RATED THERMAL POWER
d. Neutron Flux-High	≤12% of RATED THERMAL POWER	≤14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$<2 \times 10^5$ cps	$<5 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥0.7 cps	≥0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	<108/125 of full scale	<110/125 of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥5/125 of full scale	≥3/125 of full scale

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

00 38

Insert C1

a. When using the MCPR LCO from Curve A of Figure 3.2.3-1a or the curves from Figure 3.2.3-1b.	≤ 0.66W + 37%**	≤ 0.66W + 40%**
b. When using the MCPR LCO from Curve B of Figure 3.2.3-1a or the curves from Figure 3.2.3-1b.	≤ 0.66W + 41%**	≤ 0.66W + 44%**

Insert C2

a. When using the MCPR LCO from Curve A of Figure 3.2.3-1a	≤ 0.66W + 31.7%**	≤ 0.66W + 34.7**
b. When using the MCPR LCO from Curve B of Figure 3.2.3-1a	≤ 0.66W + 35.7%**	≤ 0.66W + 38.7%**

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 765' 5¼"	≤ 765' 5¼"
b. Scram Discharge Volume Switch in Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	< 108/125 of full scale	< 111/125 of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

** Clamped, with an allowable value not to exceed the allowable value for a recirculation loop flow (W) of 100%.

00 40

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.

APPLICABILITY : OPERATIONAL CONDITIONS 1 AND 2

ACTION

a. With only one (1) reactor coolant system recirculation loop ~~not~~ in operation , comply with Specification 3.4.1.5 and:

1. Within four (4) hours:

   a) Place the recirculation flow control system in the Master Manual mode or lower, and

   b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and

   c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,

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

   ~~e) Reduce the MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) limit to a value of 0.85 times the two recirculation loop operation limit per Specification 3.2.1.~~

2. The provisions of Specification 3.0.4 are not applicable.

3. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

b. With no reactor coolant recirculation loops in operation:

1. Take the ACTION required by Specification 3.4.1.5, and

2. Be in at least HOT SHUTDOWN within the next six (6) hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic power units, and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

NEW PAGE

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

~~3.4.2 The safety valve function of 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings.* #~~ Insert D

a. 4 safety/relief valves @ 1205 psig + 1%, -3%
b. 4 safety/relief valves @ 1195 psig + 1%, -3%
c. 4 safety/relief valves @ 1185 psig + 1%, -3%
d. 4 safety/relief valves @ 1175 psig + 1%, -3%
e. 2 safety/relief valves @ 1150 psig + 1%, -3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With ~~one or more~~ of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

INSERT D

3.4.2 - The safety valve function of 17 of the below listed 18 reactor coolant system safety/relieve valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2,* and 3. 1 e

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at Pa, 39.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

~~#See Special Test Exception 3.10.7.~~ e

PLANT SYSTEMS

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10 The main turbine bypass system shall be OPERABLE.

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

ACTION: ~~With the main turbine bypass system inoperable, within 2 hours restore the system to OPERABLE status or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.~~

Insert E

SURVEILLANCE REQUIREMENTS

4.7.10 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel.

b. 18 months by:

1. Performing a system functional test which includes simulated atuomatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to 200 milliseconds.

INSERT E

A. With the main turbine bypass system inoperable:

1. If at least four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:

   a. Within 2 hours, either:

      1) Restore the system to OPERABLE status, or

      2) Increase the MINIMUM CRITICAL POWER RATION (MCPR) Limiting Condition for Operation (LCO) to the main turbine bypass inoperable value per Specification 3.2.3.

   b. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

2. If less than four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:

   a. Within 2 hours, increase the MCPR LCO to the main turbine bypass inoperable value per Specification 3.2.3, and

   b. Within the next 12 hours, restore the system to OPERABLE status.

   c. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

B. The provisions of Specification 3.0.4 are not applicable.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

Delete

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specification 3.4.1.1 that recirculation loops be in operation may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during first fuel cycle PHYSICS TESTS and the Initial Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

DELETE
PAGE

DELETE

SPECIAL TEST EXCEPTIONS

3/4.10.7 CONFIRMATORY FLOW INDUCED VIBRATION TEST

LIMITING CONDITION FOR OPERATION

3.10.7 The provisions of Specifications 3.6.1.1 and 3.7.3 may be suspended to permit the drywell head to be removed and the RCIC system to be inoperable with a nitrogen supply line connected to the reactor vessel at the RCIC injection connection in order to perform the confirmatory flow induced vibration test prior to first reactor criticality. In addition, the provisions of the following specifications which are applicable during HOT SHUTDOWN may be suspended so that the unit may be brought to HOT SHUTDOWN and maintained in HOT SHUTDOWN for the duration of the test by non-nuclear heatup provided that initial reactor criticality has not occurred. Upon successful completion of the test or initial reactor criticality, whichever occurs first, this specification is cancelled.

a. Specification 3.3.2, Table 3.3.2-1 for Trip Function A.1.c.1, Main Steam Line Radiation - High Monitor.
b. Specification 3.3.7.10, Table 3.3.7.10-1 for Instrument 1.a., Liquid Radwaste Effluent Line Monitor.
c. Specification 3.3.7.11, Table 3.3.7.11-1 for Instrument 1.a, Noble Gas Activity Monitor.
d. Specification 3.4.3.1 for the primary containment atmosphere particulate and gaseous radioactivity monitoring systems.
e. Specification 3.5.1 for the ADS valves and "B" LPCI loop.
f. Specification 3.6.1.1, 3.6.1.2, 3.6.1.3, and 3.6.1.4.
g. Specification 3.6.2.1.
h. Specification 3.6.3, Table 3.6.3-1 for valves in a.1, Main Steam Isolation Valves; a.3, Reactor Coolant System Sample Line Valves; a.10, LPCS, HPCS, RCIC, and RHR Injection Testable Check Bypass Valves; a.12, Drywell Pneumatic Valves; and a.14, TIP Guide Tube Valve Ball Valve.
i. Specification 3.4.3.2.d, isolation valve leakage for "B" LPCI check valve 1E12 F041B.

APPLICABILITY: OPERATIONAL CONDITION 3, during performance of the confirmatory flow induced vibration test.

ACTION: With the provisions of the above specification not satisfied, be in COLD SHUTDOWN within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.7 The reactor shall be verified not to have been critical with any fuel assembly presently in the core within 24 hours prior to performance of the test.

DELETE PAGE

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

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. This specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

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 Figure 3.2.1-1 for two recirculation loop operation. These values shall be multiplied by a factor of 0.85 for single recirculation loop operation. This multiplier is determined from comparison of the limiting analysis between two recirculation loop and single recirculation loop operation.~~ e

The calculational procedure used to establish the APLHGR values for the initial cycle and first reload fuel shown on Figures 3.2.1-1 and 3.2.1-2 are ~~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 Part 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) the analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.2.1-1, (2) fission product decay is computed assuming an energy release rate of 200 MeV/fission; (3) pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; and (4) the effects of core spray entrainment and counter-current flow limitation as described in Reference 2, are included in the reflooding calculations.

Insert F ~~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.~~ e

Insert F

The APLHGR values for the reload fuel shown in Figure 3.2.1-3 are based on the fuel thermal-mechanical design analysis. The improved SAFER/GESTR-LOCA analysis (Reference 3) performed for Cycle 3 used bounding MAPLHGR values of 13.0 and 14.0 kw/ft, independent of nodal exposure. These MAPLHGR values are higher than the expected "thermal-mechanical MAPLHGR" for both BP8x8R and GE8x8EB fuel. Therefore, SAFER/GESTR established that for all BP8x8R and GE8x8EB fuel designs the MAPLHGR values are not expected to be limited by LOCA/ECCS considerations. However, MAPLHGR values are still required to assure that the LHGR limits are not compomised and, consequently, fuel rod mechanical integrity is maintained.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

DELETE

Plant Parameters:

Core THERMAL POWER .... 3489 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 3.1 ft^2.

b. Small Breaks 0.10 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	1.18

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 15.0-1 of the FSAR.

*This power level meets the Appendix 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.

DELETE PAGE

POWER DISTRIBUTION SYSTEMS

BASES

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 control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation 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 settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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 and presented in Figure 3.2.3-1a.

Insert G →

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-1 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate ~~pressurization~~ events ~~is~~ are described in ~~NEDO-24154(1)~~ NEDE-24011-P-A-US (Reference 4). ~~and the program used in nonpressurization events is described in NEDO-10802(2)~~. The outputs of ~~this~~ these programs along with the initial MCPR (Reference 4) 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 need to adjust the MCPR operating limit as a function of scram time arises from the statistical approach used in the implementation of the ODYN computer code for analyzing rapid pressurization events. Generic statistical analyses were performed for plant groupings of similar design which considered the statistical variation in several parameters, i.e., initial power level, CRD scram insertion time, and model uncertainty. These analyses, which are

Insert G

When the Rod Withdrawal Error is the limiting transient event, two MCPR limits may be provided. These limits are a function of the Rod Block Monitor (RBM) setpoint. The appropriate limit will be chosen based on the current RBM setpoint. The flexibility of the variable RBM setpoint/MCPR limit allows efficient use of the extended operating domain (ELLLA region), while maintaining transient protection with the more restrictive MCPR limit.

Analyses have been performed to determine the effects on CRITICAL POWER RATIO (CPR) during a transient assuming that certain equipment is out of service. A detailed description of the analyses is provided in Reference 5. The analyses performed assumed a single failure only and establised the licensing bases to allow continuous plant operation with the analyzed equipment out of service. The following single equipment failures are included are part of the transient analyses input assumptions:

1. main turbine bypass system out of service,
2. recirculation pump trip system out of service,
3. safety/relief valve (S/RV) out of service, and
4. feedwater heater out of service (corresponding to a 100 degree F reduction in feedwater temperature).

For the main turbine bypass and recirculation pump trip systems, specific cycle-independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values are established to allow continuous plant operation with these systems out of service. A bounding end-of-cycle exposure condition was used to develop nuclear input to the transient analysis model. The bounding exposure condition assumes a more top-peaked axial power distribution than the nominal power shape, thus yielding a bounding scram response with reasonable conservatisms for the MCPR LCO values in future cycles. The cycle independent MCPR LCO values shown in Figure 3.2.3-1b for the main turbine bypass and recirculation pump trip systems out of service are valid provided:

1. The cycle specific analysis for the Load Reject Without Bypass and Turbine Trip Without Bypass events yield MCPR LCO values less than or equal to 1.33 and 1.29 for Options A and B, respectively.
2. The cycle specific analysis for the Feedwater Controller Failure event yields MCPR LCO values less than 1.25 and 1.21 for Options A and B, respectively, when analyzed with normal feedwater temperature.

The analysis for main turbine bypass and reciculation pump trip systems inoperable allows operation with either system inoperable, but not both at the same time.

For operation with the feedwater heater out of service, a cycle specific analysis will be performed. With reduced feedwater temperature, the Load Reject Without Bypass event will be less severe because of the reduced core steaming rate and lower initial void fraction. Consequently, no further analyis is needed for that event. However, the feedwater controller failure event becomes more severe with a feedwater heater out of service and could become the limiting transient for a specific cycle. Consequently, the cycle specific analycis for the feedwater controller failure event will be performed with a 100 degree F feedwater temperature reduction. The calculated change in CPR for that event will then be used in determining the cycle specific MCPR LCO value.

In the case of a single S/RV out of service, transient analysis results showed that there is no impact on the calculated MCPR LCO value. The change in CPR for this operating condition will be bounded by reload licensing calculations and no further analyses are required. The analysis for a single S/RV out of service is valid in conjunction with dual and single recirculation loop operation

POWER DISTRIBUTION SYSTEMS

BASES

MINIMUM CRITICAL POWER RATIO (Continued)

described further in Reference ~~8~~ a, produced generic Statistical Adjustment Factors which have been applied to plant and cycle specific ODYN results to yield operating limits which provide a 95% probability with 95% confidence that the limiting pressurization event will not cause MCPR to fall below the fuel cladding integrity Safety Limit.

As a result of this 95/95 approach, the average 20% insertion scram time must be monitored to assure compliance with the assumed statistical distribution. If the mean value on a cycle cumulative, running average, basis were to exceed a 5% significance level compared to the distribution assumed in the ODYN statistical analyses, the MCPR limit must be increased linearly, as a function of the mean 20% scram time, to a more conservative value which reflects an NRC determined uncertainty penalty of 4.4%. This penalty is applied to the plant specific ODYN results, i.e. without statistical adjustment, for the limiting single failure pressurization event occurring at the limiting point in the cycle. It is not applied in full until the mean of all current cycle 20% scram times reaches the 0.86 seconds value of Specification 3.1.3.3. In practice, however, the requirements of 3.1.3.3 would most likely be reached, i.e., individual data set average > 0.86 secs, and the required actions taken well before the running average exceeds 0.86 secs.

The 5% significance level is defined in Reference 4 as:

$$\tau_B = \mu + 1.65 \left(N_1 / \sum_{i=1}^{n} N_i\right)^{1/2} \sigma$$

where μ = mean value for statistical scram time distribution to 20% inserted = ~~.608~~ .672

σ = standard deviation of above distribution = ~~.0[illegible]~~ .016

N_1 = number of rods tested at BOC, i.e., all operable rods

$\sum_{i=1}^{n} N_i$ = total number of operable rods tested in the current cycle

The value for τ_B used in Specification 3.2.3 is ~~0.700~~ 0.687 seconds which is conservative for the following reason:

For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^{n} N_i$ of 598 was used. This represents one full core data set at BOC plus one full core data set following a 120 day outage plus twelve 10% of core, 19 rods, data sets. The 12 data sets are equivalent to 24 operating months of surveillance at the increased surveillance frequency of one set per 60 days required by the action statements of Specifications 3.1.3.2 and 3.1.3.4.

MINIMUM CRITICAL POWER RATIO (Continued)

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

~~2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).~~

~~3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.~~

~~4. TASC-01-A Computer Program For the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.~~

2 ~~5~~. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" General Electric Co. Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Vol. III as supplemented by letter dated September 5, 1980, from R. H. Buchholz (GE) to P. S. Check (NRC).

3/4.2.4 LINEAR HEAT GENERATION RATE

The specification assures that the LINEAR HEAT GENERATION RATE (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.

3. "LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analyses", General Electric Co. Report NEDC-31510P, December 1987.

4. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A, (latest approved revision).

5. "Extended Operating Domain and Equipment Out-of-Service for LaSalle County Nuclear Station Units 1 and 2", NEDC-31455, November 1987.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

Insert H

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Insert H

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the ˉ T function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

2060

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and been found to be acceptable provided the unit is operated in accordance with the single recirculation loop operation Technical Specifications herein.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design-basis-accident by increasing the blowdown area and reducing 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 scheduled for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criterion. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Where the recirculation loop flow mismatch limits can not be maintained during the recirculation loop operation, continued operation is permitted in the single recirculation loop operation 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. Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

The possibility of thermal hydraulic instability in a BWR has been investigated since the startup of early BWRs. Based on tests and analytical models, it has been identified that the high power-low flow corner of the power-to-flow map is the region of least stability margin. This region may be encountered during startups, shutdowns, sequence exchanges, and as a result of a recirculation pump(s) trip event.

To ensure stability, single loop operation is limited in a designated restricted region (Figure 3.4.1.1-1) of the power-to-flow map. Single loop operation with a designated surveillance region (Figure 3.4.1.1-1) of the power-to-flow map requires monitoring of APRM and LPRM noise levels.

3/4.4.2 SAFETY/RELIEF VALVES

Analysis has shown that with the safety function of one of the eighteen

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. ~~A total of 18 OPERABLE~~ safety/relief valves, ~~is required to limit~~ reactor pressure, to within ASME III allowable values for the worst case upset transient. is limited

inoperable the

(Insert I)

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

Insert I

Therefore, operation with any 17 SRV's capable of opening is allowable, although all installed SRV's must be closed and have position indication to ensure that integrity of the primary coolant boundary is known to exist at all times.

EMERGENCY CORE COOLING SYSTEMS

BASES

ECCS-OPERATING and SHUTDOWN (Continued)

the suppression pool into the reactor, b[illegible] no credit is taken in the hazards analyses for the condensate storage tank [illegible]ter.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the hazards analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active com[illegible]ents are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly, if required, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 122 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls seven selected safety-relief valves. Six valves are required to be OPERABLE ~~although the hazards analysis only takes credit for five valves~~ since the LOCA analysis assumes 6 ADS valves in addition to a single failure. It is therefore appropriate to permit one of the required valves to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is also required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume, vortex prevention plus a 2'-4" safety margin for conservatism.

SNUBBERS (Continued)

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

~~The main turbine bypass system is required OPERABLE as assumed in the feedwater controller failure analysis.~~ ℓ

(Insert J)

0064

INSERT J

A generic analysis, which provides for continued operation with the main turbine bypass system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the main turbine bypass system is inoperable. The MCPR LCO values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the main turbine bypass system inoperable. Although analysis supports operation with all five turbine bypass valves inoperable, the specification provides for continued operation only if at least 4 bypass valves are capable of accepting steam flow. The analysis results are further discussed in the bases for Specification 3.2.3.

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.89 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

INSERT K

5.3.2 The reactor core shall contain 185 control rod assemblies~~, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath~~.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

0066

INSERT K

There are two possible types of control rods, one consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C powder, surrounded by cruciform shaped stainless steel sheath, and the second typo contains 143 inches of absorber material of which the first 6 inches are hafnium and the remainder is B_4C.

ATTACHMENT E

SIGNIFICANT HAZARDS EVALUATION

Commonwealth Edison proposes to amend Facility Operating License NPF-18 for LaSalle Unit 2 to support the Cycle 3 core reload. The proposed reload fuel and analyses including the previously approved SAFER/GESTR-LOCA Loss-of-Coolant Accident (LOCA) Analysis, changes resulting from analyses performed to expand the operating region and allow equipment out-of-service and changes that are administrative or provide clarification. The proposed changes for LaSalle Unit 2 are identical to those previously submitted and approved for use at LaSalle Unit 1, except for minor calculation differences in the results for transient analyses.

DESCRIPTION OF AMENDMENT REQUEST

The Technical Specification changes for the LaSalle Unit 2 Cycle 3 (L2C3) reload include:

1. Provision for operation in the expanded operating domain including revised APRM and RBM setpoint changes incorporated using standard and previously approved methodology.

2. Use of extended burnup fuel (GE 8x8EB) with increased LHGR limit of 14.4 Kw/ft.

3. Use of improved transient and LOCA analysis methods which allow use of a lower tau-B value in determining the MCPR operating limit as a function of scram time, and deletion of the single loop MAPLHGR limit multiplier of 0.85.

4. Provision for operation with certain equipment inoperable or out of service. Specifically, one of the following systems or components may be out of service when the appropriate Technical Specification ACTIONs are satisfied:

    a. Turbine Bypass System
    b. End-of-Cycle Recirculation Pump Trip (EOC-RPT)
    c. One Safety Relief Valve (SRV)
    d. Feedwater Heaters

5. Several changes for clarification or administrative purposes are proposed including:

    a. Deletion of GEXL correlation and GETAB statistical model in the bases of the safety limit section.

    b. Revision to the Control Rod Program Controls Technical Specification to require the RWM to be demonstrated operable in Operational Condition 1, prior to reaching 20% power, when reducing thermal power.

0068

BASIS FOR PROPROSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Commonwealth Edison has evaluated the proposed Technical Specifications and determined that they do not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92(c), operation of LaSalle Unit 2 Cycle 3 in accordance with the proposed changes will not:

a. Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   (1) The use of the proposed operating limits are specifically analyzed to ensure the input assumptions of all existing transient and accident analyses remain valid. These analyses are performed using a methodology which has received review and approval for other similar plants including LaSalle Unit 1.

   (2) The Technical Specification ACTIONs included in the proposed revisions do not significantly affect the probability of an accident previously analyzed because the required time intervals for corrective action are consistent with the existing specifications.

b. Create the possibility of a new or different kind of accident from any accident previously evaluated because:

   (1) The proposed MCPR, MAPLHGR, and LHGR limits represent limitations on reactor operating state which do not directly affect the operation, or function of any system or component. As a result, there is no impact on or addition of any systems or equipment whose failure could initiate an accident.

   (2) The proposed operating domain is evaluated to retain the originally required design margins to system integrity during normal operation, transients and accidents and therefore do not cause significant new loads or stresses on mechanical systems or boundaries.

   (3) The proposed allowances for operation with prescribed equipment inoperable or out-of-service do not cause physical changes to any systems and therefore do not induce new failure modes.

c. Involve a significant reduction in the margin of safety because:

   (1) No change to Safety Limits are involved.

(2) The analyses used to evaluate reactor and system performance are performed using standard methods and the calculated operating limits maintain conservative margin to safety limits to accommodate the anticipated performance during transients and accidents.

(3) No changes to protective system logic or design are involved.

(4) Changes which are administrative in nature do not affect the operating limits of the plant or the consequences of analyzed transients.

Guidance has been provided in 51 Fr 7744 for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not likely considered to involve significant hazards considerations. This amendment request is similar to example (iii) of the examples that are not likely to involve significant hazards consideration.

Example (iii) "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

This change clearly falls within this example as the reload fuel for Unit 2 Cycle 3 is of the same design as reviewed and approved for LaSalle 1 Cycle 3.

This proposed amendment does <u>not</u> involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the the Federal Register and the criteria established in 10 CFR 50.92(e), the proposed change does not constitute a significant hazards consideration.

0070

ATTACHMENT F

SUPPLEMENTAL INFORMATION

GE Document 23A5841, "Supplemental Reload Licensing Submittal for LaSalle County Station Unit 2 Reload 2 (Cycle 3)", dated July, 1988.

5102K