January 6, 1989

Docket No: 50-374

DISTRIBUTION		
Docket file	NRC PDRs	GPA/PA
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EJordan	BGrimes	
TMeek (4)	WJones	
EButcher	ACRS (10)	

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

Subject: ISSUANCE OF AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NPF-18- LASALLE COUNTY STATION, UNIT 2 (TAC NO. 69368)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 41 to Facility Operating License No. NPF-18 for the LaSalle County Station Unit 2. This amendment is in response to your letter dated September 14, 1988.

This amendment revises the LaSalle County Station, Unit 2 Technical Specifications in support of the second reload (Cycle 3) for Unit 2.

A copy of the related Safety Evaluation supporting Amendment No. 41 to Facility Operating License No. NPF-18 is enclosed. The Notice of Issuance will be included in the Commssion's biweekly Federal Register notice.

Sincerely,

Paul C. Shemanski

Paul C. Shemanski, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Enclosures:
1. Amendment No. 41 to License No. NPF-18
2. Safety Evaluation

cc w/enclosure:
See next page

	P.S.		
PDIII-2:LA	PDIII-2:PM	PDIII-2:PD	OGC Barth
LLuther:dmj	PShemanski	DMuller	
1/4/89	1/4/89	1/6/89	1/5/89

DF01 1/1

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

January 6, 1989

Docket No: 50-374

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

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cc w/enclosure:
See next page

Mr. Henry E. Bliss
Commonwealth Edison Company

LaSalle County Nuclear Power Station
Units 1 & 2

cc:

Phillip P. Steptoe, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Assistant Attorney General
100 West Randolph Street
Suite 12
Chicago, Illinois 60601

Resident Inspector/LaSalle, NPS
U.S. Nuclear Regulatory Commission
Rural Route No. 1
P. O. Box 224
Marseilles, Illinois 61341

Chairman
LaSalle County Board of Supervisors
LaSalle County Courthouse
Ottawa, Illinois 61350

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

Chairman
Illinois Commerce Commission
Leland Building
527 East Capitol Avenue
Springfield, Illinois 62706

Mr. Michael C. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

John W. McCaffrey
Chief, Public Utilities Division
SOIC
100 West Randolph Street
Chicago, Illinois 60601

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated September 14, 1988 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 the Facility Operating License No. NPF-18 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. 41, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: January 6, 1989

ENCLOSURE TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
XIX	XIX
2-4	2-4
--	2-4a
B2-2	B2-2
B2-3	B2-3
B2-4	B2-4
B2-5	B2-5
B2-6	B2-6
B2-7	B2-7
3/4.1-16	3/4.1-16
3/4.2-1	3/4.2-1
3/4.2-2b	3/4.2-2b
3/4.2-3	3/4.2-3
3/4.2-4	3/4.2-4
--	3/4.2-4a
3/4.2-5	3/4.2-5
3/4.2-5a	3/4.2-5a
3/4.2-7	3/4.2-7
3/4.3-39	3/4.3-39
3/4.3-53	3/4.3-53
3/4.3-54	3/4.3-54
3/4.4-1	3/4.4-1
3/4.4-6	3/4.4-6
3/4.6-1	3/4.6-1
3/4.7-34	3/4.7-34
3/4.10-4	3/4.10-4
3/4.10-7	3/4.10-7
B3/4.2-1	B3/4.2-1
B3/4.2-2	B3/4.2-2
B3/4.2-3	B3/4.2-3
B3/4.2-4	B3/4.2-4
B3/4.2-5	B3/4.2-5
B3/4.2-6	B3/4.2-6
--	B3/4.2-7
B3/4.3-3	B3/4.3-3
B3/4.3-4	B3/4.3-4
B3/4.4-1a	B3/4.4-1a
B3/4.5-2	B3/4.5-2
B3/4.7-5	B3/4.7-5
5-4	5-4

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% with a maximum of	≤ 0.58W + 62% 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% with a maximum of	≤ 0.58W + 57.3% 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

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

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
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	≥ 1134 psig
b. Delay Timer	≤ 10 seconds	≤ 10 seconds

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.

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

THERMAL POWER, High Pressure and High Flow (Continued)

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.

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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:

a. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 prior to reaching 20% of RATED THERMAL POWER 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.

#See Special Test Exception 3.10.8

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, and 3.2.1-3.

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, 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, 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

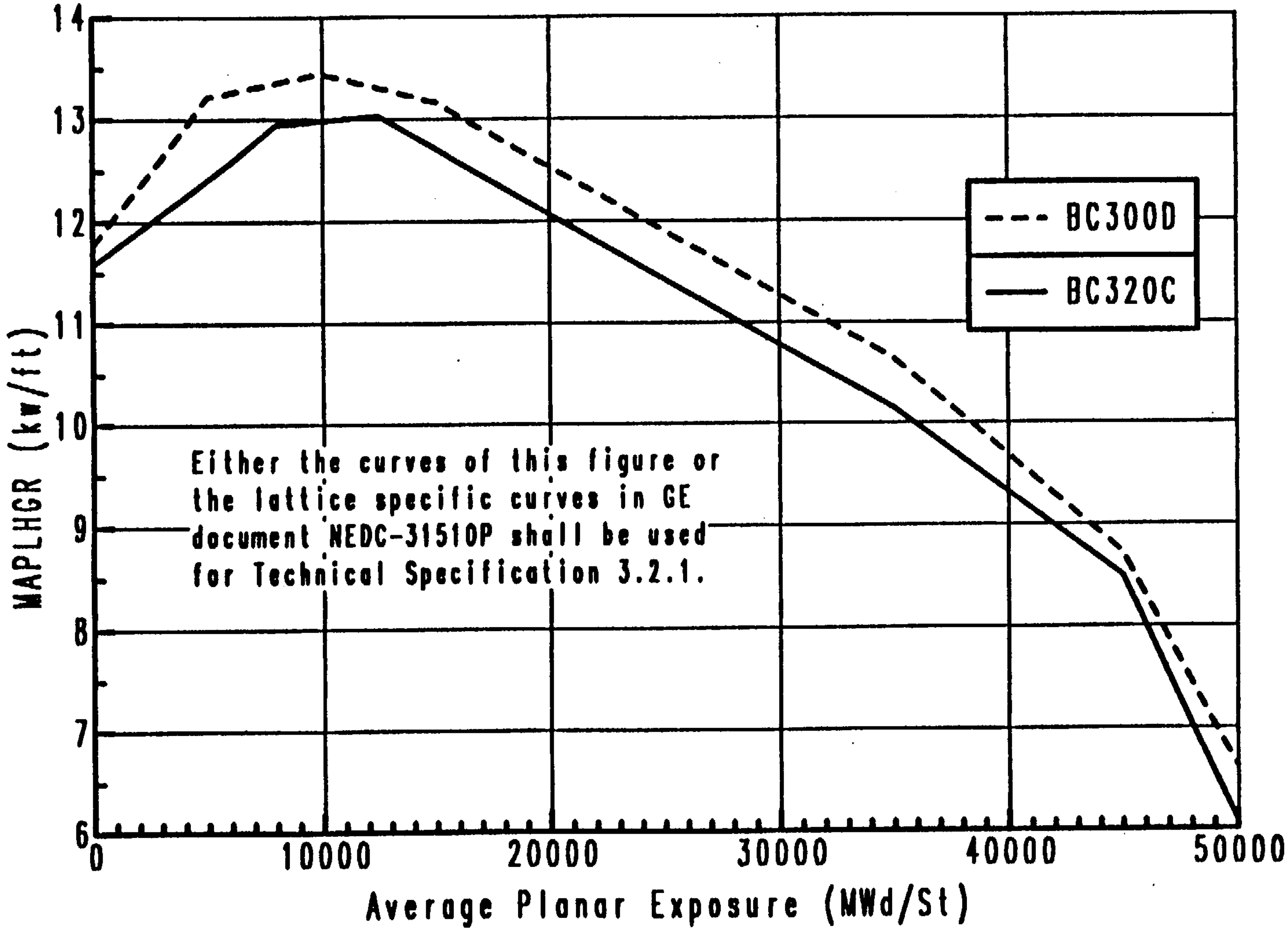


Figure 3.2.1-3

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 (0.58W + 59%)T

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

b. Single Recirculation Loop Operation

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

S_{RB} less than or equal to (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.

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:

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 inoperation as directed by Specification 3.3.4.2 (any RBM setpoint determined per Specification Table 3.3.6-2 may be used).

APPLICABILITY:

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

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.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

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,

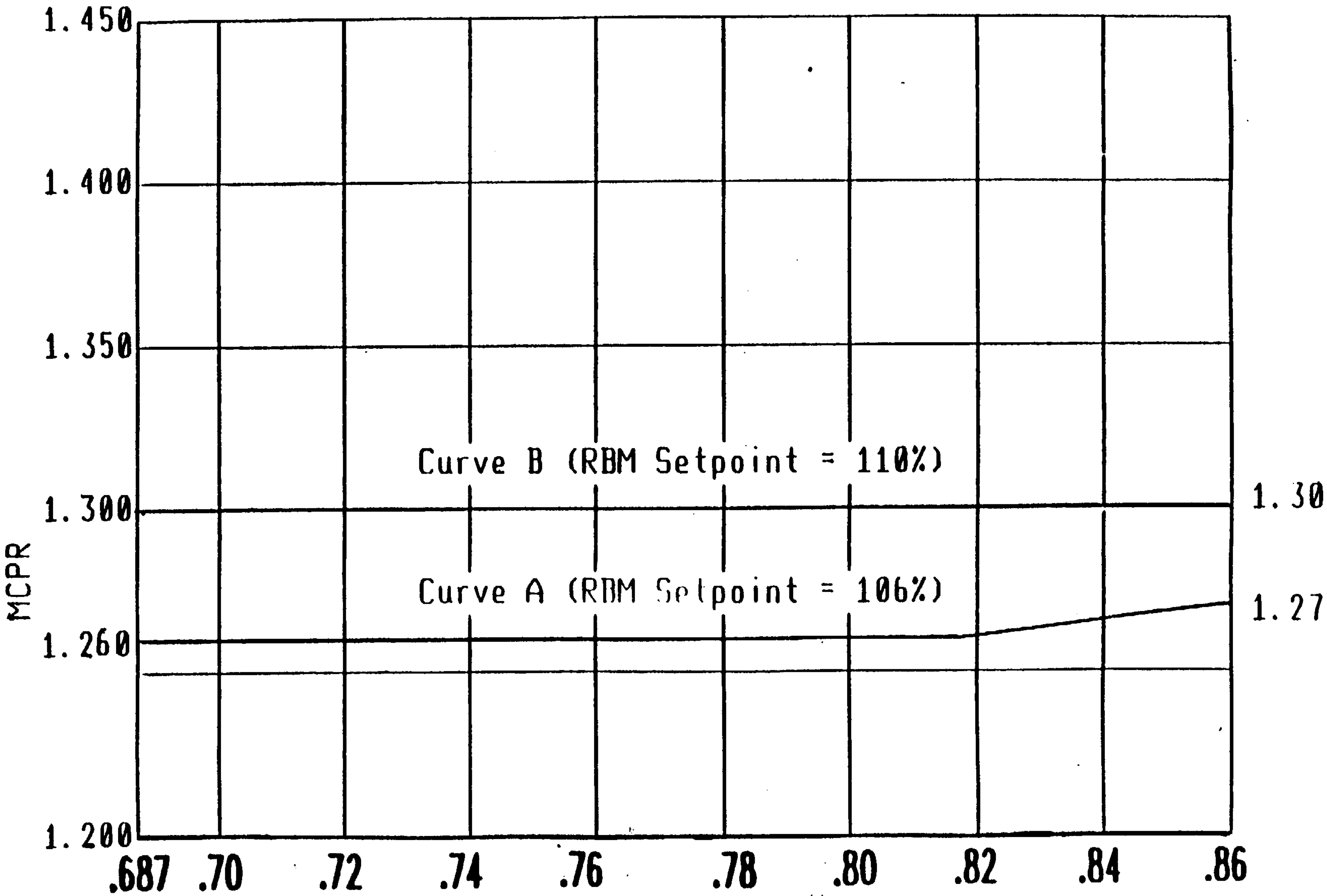
shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

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.

POWER DISTRIBUTION LIMITS
MCPR



MCPR VERSUS τ AT RATED FLOW

Figure 3.2.3-1a

POWER DISTRIBUTION LIMITS
MCPR

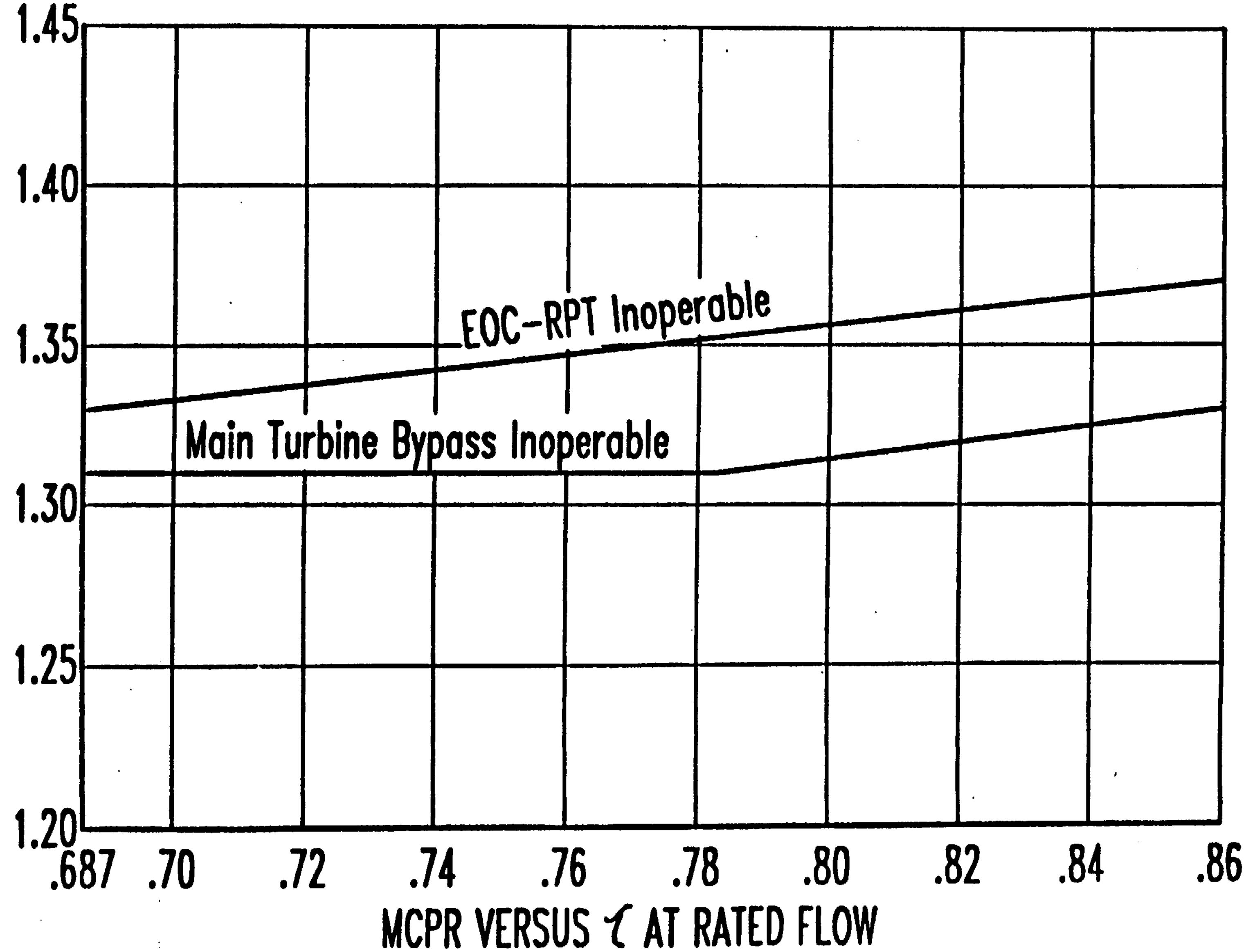


Figure 3.2.3-1b

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed:

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

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.

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.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 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. 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,

   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. Reduce THERMAL POWER to less than 30% RATED THERMAL POWER within the next 6 hours.

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 Loop Operation		
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%**
2) Single Recirculation Loop Operation		
a. When using the 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%**
b. Inoperative	N.A.	N.A.
c. Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER

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

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation Loop Operation	≤0.58 W + 47%*	≤0.58 W + 50%*
2) Single Recirculation Loop Operation	<0.58W + 42.3%*	<0.58 W + 45.3%*
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
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

*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.

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 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.

   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.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

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.

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.

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.

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:

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 RATIO (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.

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 automatic 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.

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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 calculational procedure used to establish the APLHGR values for the initial cycle and first reload fuel shown on Figure 3.2.1-1 and 3.2.1-2 are 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.

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 compromised and, consequently, fuel rod mechanical integrity is maintained.

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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 $\geq$ 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.

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 established 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,

MINIMUM CRITICAL POWER RATIO (Continued)

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 recirculation 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 analysis 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 analysis for the feedwater controller failure event wil 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.

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 events are described

MINIMUM CRITICAL POWER RATIO (Continued)

in NEDE-24011-P-A-US (Reference 4). The outputs of these programs along with the initial MCPR form the input for further analyses of the thermally limiting bundle (Reference 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 described further in Reference 2, 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 = .672

σ = standard deviation of above distribution = .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

MINIMUM CRITICAL POWER RATIO (Continued)

The value for τ_B used in Specification 3.2.3 is 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.

That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

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. The K_f factors were derived using THERMAL POWER and core flow corresponding to 105% of rated steam flow.

The K_f factors were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to 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.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

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. "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. "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.

3/4.2.4 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.

3/4.3.4 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.

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) value 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 RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

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.

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.4.2 SAFETY/RELIEF VALVES

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. Analysis has shown that with the safety function of one of the eighteen safety/relief valves inoperable the reactor pressure is limited to within ASME III allowable values for the worst case upset transient. 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.

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.

ECCS-OPERATING and SHUTDOWN (Continued)

the suppression pool into the reactor, but no credit is taken in the hazards analyses for the condensate storage tank water.

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 components 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 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 (See Figure B 3/4.6.2-1). 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

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 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.

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

5.3.2 The reactor core shall contain 185 control rod assemblies. 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 a cruciform shaped stainless steel sheath, and the second type contains 143 inches of absorber material of which the first 6 inches are hafnium and the remainder is B_4C.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNIT 2

DOCKET NO. 50-374

1.0 INTRODUCTION

By letter from C. M. Allen, Commonwealth Edison Company (CECo), to USNRC, dated September 14, 1988 (Ref. 1), Technical Specification changes were proposed for the operation of LaSalle County Station Unit 2 for Cycle 3 (LS2C3) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested Technical Specification (TS) changes and reports (including Reference 2 through 4) discussing the reload and analyses to support and justify Cycle 3 operation including an increased flow operating region and equipment-out-of-service.

The reload for Cycle 3 is generally a normal reload with no unusual core features or characteristics. Proposed TS changes relate to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits for the new fuel, MAPLHGR and Minimum Critical Power Ratio (MCPR) limits for all of the fuel using Cycle 3 core and transient parameters, extended operating regions and conditions, and new approved analytical methods. The new fuel is the extended burnup type which has been approved for use in several recent GE reloads.

The submittal proposed an extension of the current allowable operating region on the reactor power-flow map via an increased core flow (ICF) extension. Extended Load Line Limit Analysis (ELLLA) and associated TS have also been proposed for LaSalle Unit 2.

Also proposed for the cycle and supported with GE analyses is operation with "equipment-out-of-service" and extended operating modes including feedwater heaters out of service (FWHOOS), final feedwater temperature reduction (FFWTR), relief valve out of service (RVOOS), main turbine bypass system out of service (TBOOS), and recirculation pump trip system out of service (RPTOOS). TS MCPR limits bounding analyzed combinations of these conditions have been proposed.

2.0 EVALUATION

2.1 RELOAD DESCRIPTION

The LS2C3 will retain 40 8CRB176 and 260 CRB219 GE fuel assemblies from Cycle 1, 224 BP8CRB299L GE fuel assemblies from Cycle 2, and add 240 new GE8x8EB fuel assemblies (96 BC320C and 144 BC300D). The reload is based on a Cycle 2 end of cycle core nominal average exposure of 17,990 MWd/MT and Cycle 3 assumed end of cycle exposure of 19,377 MWd/MT. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery.

2.2 FUEL DESIGN

The new fuel for Cycle 3 is the GE extended burnup fuel GE8x8EB. The fuel designations are BC320C and BC300D. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Refs. 5 & 6). The specific description of this fuel has been submitted in Amendment 18 to GESTAR II, which has been accepted by the staff in Reference 7.

LOCA analyses have been performed for the retained and reload fuel using the improved SAFER/GESTR-LOCA methods approved by the staff (Ref. 8). The initial condition MAPLHGR values used in these analyses are less restrictive than those used in the fuel mechanical integrity design analyses. Thus the multi-axial region MAPLHGR TS used in some other recent reload applications of extended burnup fuel are unnecessary, and only a single set of burnup dependent values, for each fuel type, as determined by the mechanical design are required. The MAPLHGR values for both the reload and retained fuel have been calculated with approved methodology (GESTAR II, Reference 6, Section 2 of Vol. 1) and are acceptable.

The proposed LHGR limit for the GE8x8EB fuel is 14.4 KW/ft (rather than the 13.4 for other GE fuel). The LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Ref. 5). This LHGR is acceptable for the GE fuel in LS2C3.

2.3 NUCLEAR DESIGN

The nuclear design for LS2C3 has been performed by GE with the approved methodology described in GESTAR II (Ref. 6). The results of these analyses are given in the GE reload report (Ref. 2) in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 4.0% delta-k at the beginning of cycle and 1.0% delta-k at the minimum conditions, thus fully meeting the required 0.38% delta-k shutdown margin. The standby liquid control system also meets shutdown requirement with a reasonable shutdown margin of 3.7% delta-k. Since the LS2C3 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design for LS2C3 has been performed by GE with the approved methodology described in GESTAR II (Ref. 6) and the results are given in the GE reload report (Ref. 2). The GEMINI/ODYN transient analysis methodology (Ref. 6) was used.

The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, which, for standard conditions, are usually Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF) and Load Rejection Without Bypass (LRNBP). The analyses of these events for LS2C3, using the standard approved GEMINI/ODYN Options A and B approach for pressurization transients in standard and extended operating regions and with analyzed equipment out of service combinations, provide new Cycle 3 TS values of OLMCPR as a function of average scram time. For all standard operating conditions the non-pressurization local transient RWE is controlling, giving an OLMCPR value of 1.26 at and below a scram time of 0.818 seconds. Between scram times of 0.818 and 0.860 seconds, the LRNBP becomes the limiting core-wide transient and the OLMCPR limit varies linearly from 1.26 to 1.27 based on an Option A analysis. The OLMCPR limits are illustrated in the proposed TS Figure 3.2.3-1a for a Rod Block Monitor (RBM) setpoint of 106 percent. Also illustrated in Figure 3.2.3-1a is a limit curve for an RBM setpoint of 110 percent with a constant OLMCPR limit of 1.30. The use of two curves allows for more efficient use of the extended operating domain (EOD) requested by this amendment. To accommodate the extended operating region and equipment out-of-service conditions the OLMCPR limit has also been analyzed for those conditions (Ref. 4). This has resulted in an OLMCPR limit of 1.26 for ODYN Option A and 1.24 for Option B associated with the feedwater heater out-of-service (FWHOOS) analyses and an increase to 1.37 for Option A and 1.33 for Option B associated with recirculation pump trip out of service analyses. The FWHOOS limits are bounded by the RWE limits and are not needed in the TS. The RPT limits appear on Figure 3.2.3-1b of the proposed TS. Approved methods (Ref. 6) were used to analyze these events and the results are acceptable because they fall within expected ranges.

GE has calculated the core stability decay ratio at the point of minimum stability (the intersection of the natural circulation line and the extended APRM block line) for LS2C3. The calculated value of reactor core stability decay ratio is 0.72 (LaSalle 2 Cycle 2 was 0.60). In the past, this has indicated a stable core since it is less than the accepted value of 0.8 (for approved GE methods). However, due to the LaSalle 2 instability event and the continuing investigation regarding decay ratio calculations, the licensee was informed that previous calculation results are unacceptable. By letter dated April 26, 1988 (Ref. 9), TS changes were proposed to provide additional requirements for stability monitoring and actions to be taken by an operator if oscillations are observed in the high power/low flow region of the power to flow map. The revised Technical Specifications require immediate insertion of high worth rods and observation of APRM/LPRM noise when no pumps are operating and power is above the 80% Rod Control Line. The reactor is to be tripped immediately whenever instability is suspected. It is expected that the time available (greater than 5 minutes) to instability following a two pump transient is sufficient to permit manual power reduction, avoiding the need for reactor trip unless the core is unstable by a large margin. Procedures consistent with these Technical Specifications have been approved by the staff for both LaSalle Units 1 and 2 (Ref. 10).

2.5 TRANSIENT AND ACCIDENT ANALYSIS

The transient and accident analysis methodologies used for LS2C3 are described and NRC approval indicated in GESTAR II (Ref. 6). The GEMINI/ODYN method was used for the core wide transient analysis which includes load rejection without bypass (LRNBP), loss of feedwater heating and feedwater controller failure. The local rod withdrawal error (RWE) was analyzed on a plant and cycle specific basis and rod block setpoints of 106% and 110% were selected to provide a flexible OLMCPR range of 1.26 to 1.30 for all fuel types. This bounds the core wide events. The limiting MCPR events for LS2C3 are indicated in Section 2.4. The core wide and the local transient analysis methodologies and results are acceptable because they fall within expected ranges.

The limiting pressurization event, the main steam isolation valve closure with flux scram, analyzed with standard GESTAR II methods, gave results for peak steam dome and vessel pressures for standard and extended operating regions and equipment out-of-service conditions well under required limits. These are acceptable methodologies and results.

Banked position withdrawal sequence and rod patterns are used for LaSalle 2. For plants using this system the Rod Drop Accident (RDA) event has been statistically analyzed generically and it was found that with a high degree of confidence the peak fuel enthalpy would not approach the NRC limit of 280 cal/gm for this event. This approach and analysis has been approved by NRC (Ref. 6). This approach is acceptable for LS2C3.

The LOCA analyses for LS2C3 were performed using the SAFER/GESTR-LOCA methodology. This methodology (Refs. 6 & 8) has been approved by the staff and used and approved in several recent reload applications. The licensee has reported the results of these analyses (Ref. 3) which are required to meet the necessary conditions (Ref. 8). Specifically, the analyses include break sizes from 0.05 ft^2 to the maximum DBA recirculation suction line break (3.10 ft^2). Seven different break sizes were analyzed (for either nominal input or Appendix K values) in conjunction with ECCS failure combinations. A total of 24 cases were evaluated to establish the trend of peak clad temperature (PCT) curves (nominal and Appendix K) versus break size.

The input parameters for both the nominal and Appendix K cases are within those used in the approved generic analyses. The ECCS configuration of LaSalle 2 (3 Low Pressure Coolant Injection, Low Pressure Core Spray, High Pressure Coolant Injection, Automatic Depressurization System) is consistent with the ECCS configuration of a generic BWR-5/6. The results show that the DBA recirculation suction line break with diesel generator failure is the limiting case, which is consistent with BWR-5/6 generic conditions. The plant-specific Appendix K calculations demonstrate that the LPCS diesel generator failure is limiting for the P8x8R fuel, which is the limiting fuel type. The calculated PCT is 650°F when nominal input values are used and 1138°F when Appendix K input values (plus adder) are used. The latter value lies between the generic BWR-5/6 upper bound (best-estimate nominal plus uncertainties) PCT of 1100°F and the 10 CFR 50.46 limit of 2200°F and therefore demonstrates acceptable conservatism. Because the accident analyses have been performed using approved methods, and the results meet the staff's acceptance criteria, we conclude that these analyses are acceptable.

LOCA sensitivity studies or specific calculations were examined to consider the effect of extended or equipment out-of-service operation (Refs. 3 & 4). This included the full range discussed in Section 2.6 below. The changes to peak cladding temperature were generally small (or the condition was included in the base calculations, e.g., RVOOS) compared to the large margins available, so that no modifications to MAPLHGR limits are required for these conditions. These results are reasonable and acceptable. The results indicate that the TS MAPLHGR limits are not set by the LOCA calculations but by the thermal-mechanical design calculations.

2.6 OPERATING EXTENSIONS AND EQUIPMENT OUT-OF-SERVICE

The LS2C3 reload submittal proposes extensions to standard operating regions and equipment out-of-service in the GESTAR II standard category of "Operating Flexibility or Margin Improvement Options." The selected options are ICF, FFWTR, FWHOOS, RVOOS, TBOOS and RPTOOS. These have become commonly selected and approved options for a number of reactors in recent years. These options and associated analyses, including relevant transients and accidents, are described and discussed in Reference 4. Included in the analysis and discussion is the application for operation beyond nominal end of cycle with ICF (or decreased flow) and FFWTR, and coastdown to lower power levels (as low as 40 percent is assumed).

For ICF the analyses are performed at the bounding condition of 108% of rated core flow (the ICF capacity of LS2 is only feasible up to 105% core flow due to jet pump vibration considerations) (Ref. 4). The proposed operating region is bounded by the 108% APRM rod block limit line (0.58 W + 50%), the rated power line and the rated rod line. The region of operation above the rated rod line is known as the Extended Load Line Limit Analysis (ELLLA) region. The safety evaluation for this operating region includes operation beyond normal end-of-cycle, up to 100°F FFWTR (with ICF or reduced flow) and power coastdown (40 percent assumed in the analysis). Conservative power profiles were assumed. The transient analyses were used to determine OLMCPR values for these operating conditions. As discussed in Section 2.4, the OLMCPR for LS2C3 is determined by the analysis of FWHOOS. The LOCA examination concluded that the effects on MAPLHGR were insignificant compared to the large margin available. The core stability is addressed in Section 2.4 in response to the LaSalle 2 instability event. The effects of ICF and FFWTR related loads, vibration and fatigue on various reactor internals and the impact on containment LOCA response was examined and were found to be within allowable design limits except for (as is usually the case) a possible need for a slightly reduced feedwater nozzle refurbishment interval (based on seal leakage). Throughout these analyses the transients and accident examined, the methodologies and the results were similar to those reviewed on previous approved ICF-FFWTR applications for other reactors. The analyses and results and operation in this extended region are acceptable for LaSalle 2.

The FWHOOS was analyzed in a manner similar to FFWTR except for potential duration and time of occurrence in cycle which can affect core parameters to a greater extent. As indicated in Section 2.4, the extreme conditions used for analysis result in setting the OLMCPR for LS2C3. The increased limit is caused primarily by changes in axial power distribution and resulting effectiveness of scram action. The review concludes that operation with FWHOOS is acceptable for LS2C3.

The RVOOS the limiting pressurization event was evaluated with the most limiting relief valve out. The impact on MCPR is negligible. Standard sensitivity studies also show the effect on overpressure is small and results in adequate margin. The effect of a relief valve out of service was included in the LOCA analyses. It is concluded that operation with one RVOOS is acceptable.

The single loop operation (SLO) analysis was previously reviewed and approved by the NRC. The previous SLO analysis demonstrated that, within the normal operating domain and without equipment out-of-service, the consequences of abnormal operation transients from one-loop operation will be considerably less severe than those analyzed for a two-loop operation mode. The MAPLHGR changes for LS2 are not necessary since the LOCA analysis for SLO (using the new methodology) provides peak cladding temperature well below limits. The stability issue for the LaSalle Unit 2 core (GE8x8EB fuel) should follow the proposed TS changes stated in Section 2.4 of this SER.

2.7 TECHNICAL SPECIFICATION

The following TS changes have been proposed for LaSalle 2 to implement the reload analyses and operation changes which have been discussed. The reason or bases for the changes have been for the most part already discussed and approved and the changes will only be briefly described as follows:

1. TECHNICAL SPECIFICATION TABLES 2.2.1-1 AND 3.3.6-2 AND TECHNICAL SPECIFICATION 3/4.2.2
   The reactor protection system APRM flow biased scram trip setpoint and allowable values, for two loop and single loop operation, are revised to incorporate the extended load line limit analyzed region (a reduction of 4.7% during SLO based on a slope of 0.58 at 8% flow difference). The control rod withdrawal block instrumentation setpoints (RBM and APRM upscale) have been revised to reflect the change in the selected setpoint based on the rod withdrawal error including the operating region analyzed in the extended load line limit analyses. A footnote has been added to indicate that the RBM setpoint is clamped at 100% drive flow to prevent the RBM setpoint from exceeding the analyzed value. This is acceptable since an approved method was used. The variable "T" has been revised for clarification which is acceptable.

2. BASES 2.1.2
   The input for the GEXL correlation and the GETAB statistical model including Tables B2.1.2-1 through B2.1.2-4 was removed because they are overly detailed for inclusion in the TS. The detailed description is provided in GESTAR II and the deletion is acceptable.

3. TECHNICAL SPECIFICATION 3/4.1.4
   Surveillance Requirement 4.1.4.1a has been revised to require the rod worth minimize (RWM) to be demonstrated operable prior to reaching 20% of rated thermal power when reducing thermal power, rather than prior to RWM automatic initiation. This administrative change is acceptable since the RWM does not function prior to automatic initiation.

4. TECHNICAL SPECIFICATION 3/4.2.1 AND 3/4.2.4, AND BASES 3/4.2.1
   The required APLHGR and MAPLHGR limit reduction of 0.85 during SLO has been deleted based on the results of the SAFER/GESTR-LOCA analysis. This is acceptable since an approved method was used. Also MAPLHGR plots for the two new reload fuel types have been added as Figure 3.2.1-3. The LHGR limit of 14.4 KW/ft for the GE8x8EB reload fuel has been added to Section 3.2.4. These are administrative changes due to the new fuel type for the Cycle 3 reload and are acceptable.

5. TECHNICAL SPECIFICATIONS 3/4.2.3 AND 3/4.3.4.2, AND BASES 3/4.2.3 AND 3/4.3.4
The references have been revised to include licensing analyses used for LS2C3 and the discussion has been revised to incorporate the changes due to the new ODYN methods and the RBM setpoint dependent MCPR and MCPR penalties for operation with particular equipment out-of-service. The proposed revision also includes continued operation with the EOC-RPT system-out-of-service provided the MCPR limit is increased within 2 hours to the limit specified in TS 3/4.2.3. These changes are acceptable on the basis of the approved document and methods used.

6. TECHNICAL SPECIFICATIONS 3/4.6.1.1, 3/4.10.4 AND 3/4.10.7
Deletion of the footnote "see Special Test Exception 3.10.7" and Special Text exception is acceptable since they are no longer applicable.

7. TECHNICAL SPECIFICATION 3/4.4.2 AND BASES 3/4.4.2
As a result of the one safety/relief valve out-of-service analysis, the LCO statement was reworded to reflect that only 17 SRVs are required for safety valve actuation. Clarification is added to indicate that each of the required valves must be closed with position indication operable. This is acceptable.

8. TECHNICAL SPECIFICATION 3/4.7.10 AND BASES 3/4.7.10
Revision has been proposed to allow continued operation with the main turbine bypass system inoperable per the surveillance requirements, provided at least four turbine bypass valves are capable of accepting steam flow and the MCPR limit for this condition of operation is met within 2 hours per Specification 3.2.3. This is acceptable since the analyses were performed using an approved method.

9. DESIGN FEATURES 5.3.2
This section includes a new control rod assembly (ASEA-ATOM control rod). This is acceptable since the ASEA-ATOM control rod has been approved for plant specific use.

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 these amendments involve 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 CONCLUSIONS

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the FEDERAL REGISTER (53 FR 46140) on November 16, 1988, and consulted with the state of Illinois. No public comments were received, and the state of Illinois did not have any comments.

We have reviewed the reports submitted for LaSalle Unit 2 Cycle 3 operation with extended operating regions and equipment out-of-service. Based on this review we conclude that appropriate material was submitted and that the fuel nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

The staff has 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.

5.0 REFERENCES

1. Letter from C. M. Allen (CECO) to USNRC dated September 14, 1988, "LaSalle County Station Unit 2 Proposed Amendment to Technical Specification for Facility Operating License NPF-18 - Reload Licensing Package for Cycle 3, NRC Docket No. 50-374."

2. 23A5841, Revision 0, July, 1988, "Supplemental Reload Licensing Submittal for LaSalle County Station Unit 2 Reload 2 (Cycle 3)."

3. NEDC-31510P, December 1987, "LaSalle County Station Units 1 and 2 SAFER-GESTR-LOCA Loss-of-Coolant Accident Analysis."

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

5. Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-A-A-6, Amendment 10."

6. GESTAR II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel."

7. Letter, A. C. Thadani (ONRR) to J. S. Charnley (GE) dated May 12, 1988, Acceptance for Referencing of Amendment 18 to Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

8. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident" Volume I, II, and III, General Electric Company, June 1984.

9. Letter from C. M. Allen (CECO) to USNRC dated April 26, 1988, "LaSalle County Station Units 1 and 2 Proposed Amendment to Technical Specifications for Facility Operating Licenses NPF-11 and NPF-8 - Core Performance Monitoring, NRC Docket Nos. 50-373 and 50-374."

10. Letter from P. Shemanski (NRC) to H. E. Bliss (CECO) dated September 7, 1988, "Issuance of Amendment Nos. 60 and 40 to Facility Operating Licenses NPF-11 and NPF-18 - LaSalle County Station, Units 1 and 2."

Principal Contributor: M. McCoy, NRR/SRXB
P. Shemanski, NRR/PDIII-2

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