

May 7, 1987

Docket No. 50-387

Mr. Harold W. Keiser
Vice President
Nuclear Operations
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

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SUBJECT: ISSUANCE OF AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE
NO. NPF-14 - SUSQUEHANNA STEAM ELECTRIC STATION, UNIT NO. 1

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 64 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, (SSES) Unit 1. This amendment is in response to your letter dated December 12, 1986.

This amendment revises the Susquehanna Unit 1 Technical Specifications to change the SSES MAPLHGR and MCPR limits, preclude single loop operation, and change the affected Technical Specification bases.

A copy of the related safety evaluation supporting Amendment No. 64 to Facility Operating License No. NPF-14 is enclosed.

Sincerely,

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:

1. Amendment No. 64 to NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

8705150386 870507
PDR ADDCK 05000387
P PDR

Previously concurred*:

LA:BWD-3:DBL*
EHylton/hmc
03/26/87

BWD-3:DBL*
MThadani
03/26/87

D:BWD-3:DBL*
EAdensam
03/26/87

PDI-2/D
WButler
05/7/87

WB



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

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Sincerely,

A handwritten signature in cursive script, reading "Walter R. Butler".

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:

1. Amendment No. 64 to NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:
 - A. The application for the amendment filed by the Pennsylvania Power & Light Company (PP&L), dated December 12, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-14 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. 64, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

8705150398 870507
PDR ADOCK 05000387
P PDR

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

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 7, 1987

Previously concurred*:

LA:BWD-3:DBL*
EHylton/hmc
03/26/87

BWD-3:DBL
MThadani
03/26/87

OGC*
MYoung
03/31/87

D:PDI-2:DRP
WButler
05/7/87

WB

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FOR THE NUCLEAR REGULATORY COMMISSION

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Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 7, 1987

ENCLOSURE TO LICENSE AMENDMENT NO: 64

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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

REMOVE

INSERT

xxi	xxi
xxii	xxii (overleaf)
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5 (overleaf)
3/4 2-6	3/4 2-6
3/4 2-8 Deleted	-----
3/4 2-9	3/4 2-9
-----	3/4 2-9a
3/4 2-10	3/4 2-10 (overleaf)
3/4 3-53	3/4 3-53 (overleaf)
3/4 3-54	3/4 3-54
3/4 4-1b	3/4 4-1b (overleaf)
3/4 4-1c	3/4 4-1c
B 3/4 2-1	B 3/4 2-1 (overleaf)
B 3/4 2-2	B 3/4 2-2
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2 (overleaf)
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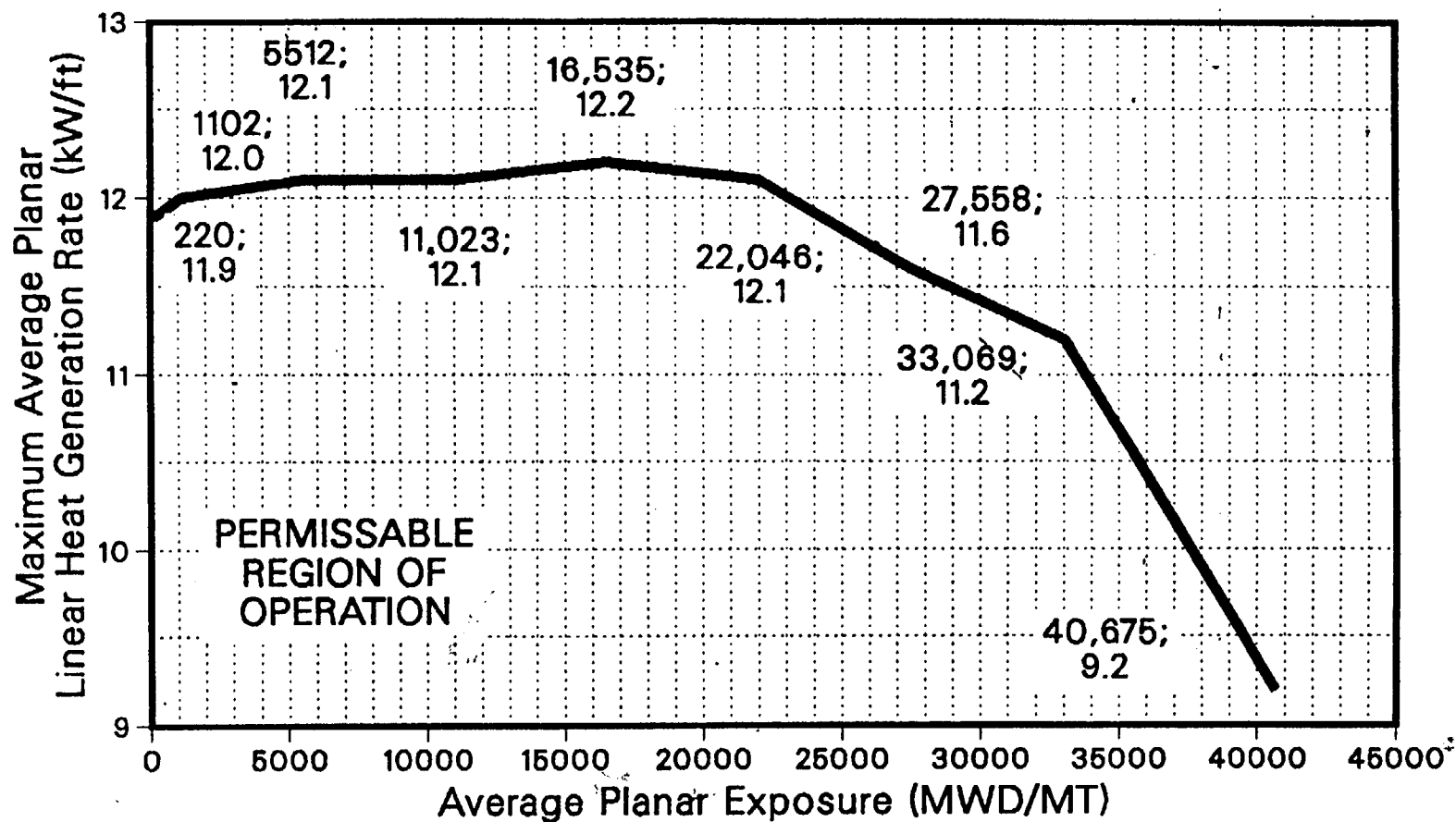
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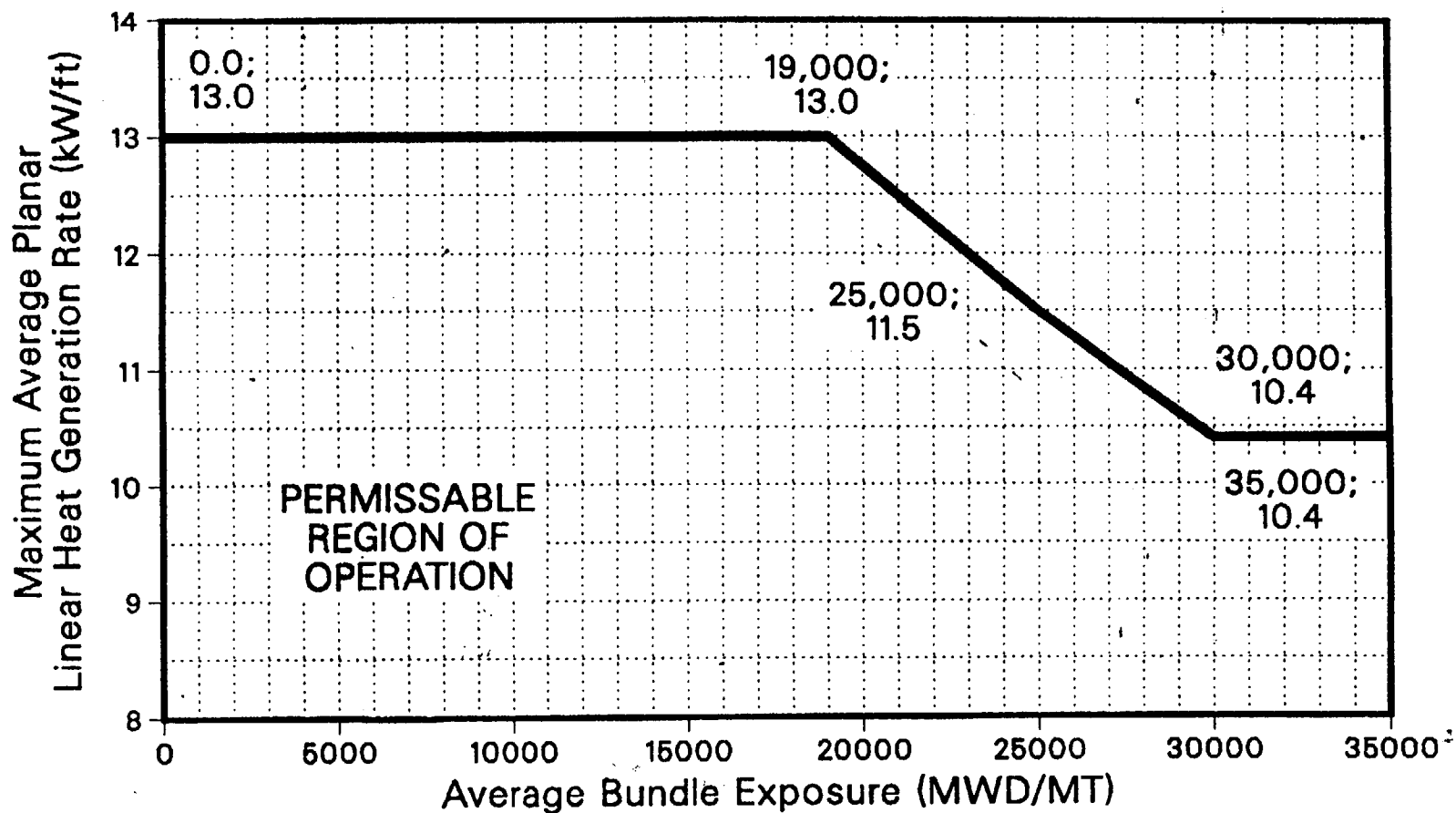
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MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
GE FUEL TYPES 8CR233 (2.33% ENRICHED)
FIGURE 3.2.1-1



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
EXXON 8X8 FUEL
FIGURE 3.2.1-2

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

<u>Trip Setpoint[#]</u>	<u>Allowable Value[#]</u>
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,

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

T (Exxon fuel) = 1.0

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

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB} , as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* 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 F RTP and the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

- At least once per 24 hours,
- Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to F RTP.
- The provisions of Specification 4.0.4 are not applicable.

*With MFLPD greater than the F RTP during power ascension 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.

[#]See Specification 3.4.1.1.2.a for single loop operation requirements.

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 greater than or equal to the greater of the two values determined from Figure 3.2.3-1 and Figure 3.2.3-2

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

ACTION:

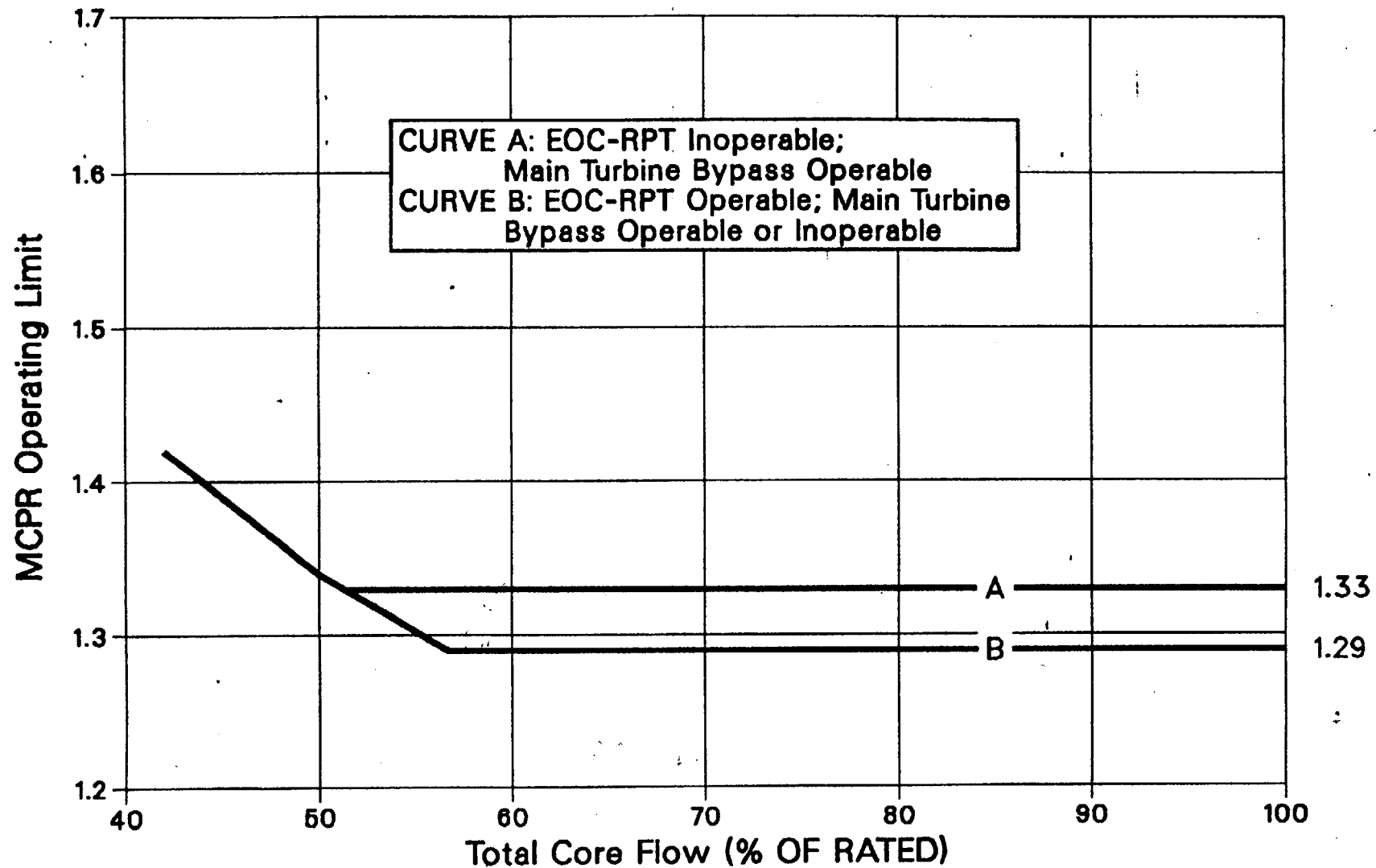
With MCPR less than the applicable MCPR limit determined above, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

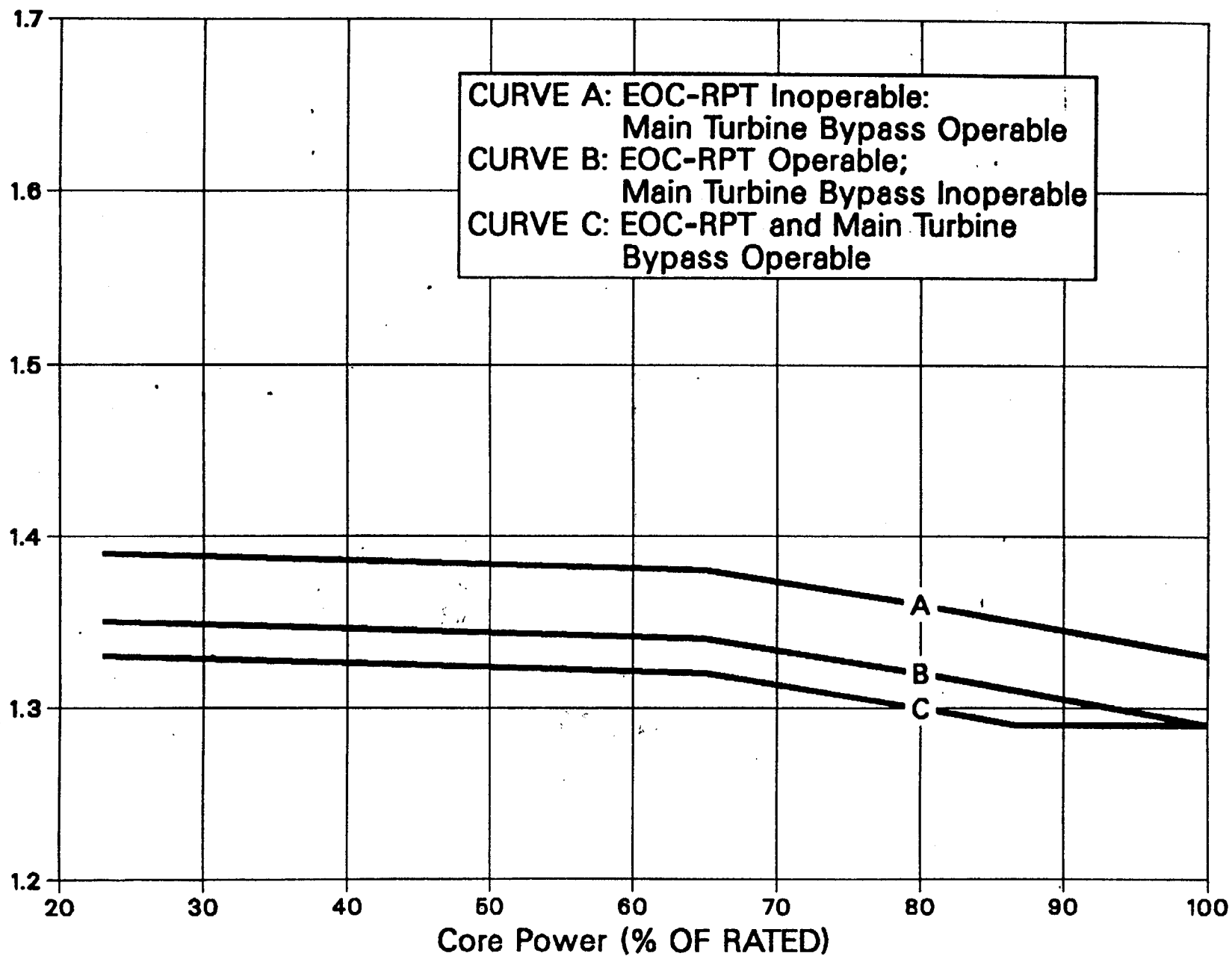
4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figure 3.2.3-2:

- 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.
- d. The provisions of Specification 4.0.4 are not applicable.

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**FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1**



REDUCED POWER MCPR OPERATING LIMIT

Figure 3.2.3-2

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel 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 LHGRs for GE fuel 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. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.3.6-1 (Continued)
CONTROL ROD BLOCK INSTRUMENTATION

ACTION

- ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.
- ACTION 61 - With the number of OPERABLE Channels:
- a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.
 - b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 1 hour.
- ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 1 hour.

NOTES

- * With THERMAL POWER \geq 30% of RATED THERMAL POWER.
- ** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
- *** Not required when eight or fewer fuel assemblies (adjacent to the SRMs) are in the core.
- a. The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.
- b. This function shall be automatically bypassed if detector count rate is \geq 100 cps or the IRM channels are on range 3 or higher.
- c. This function is automatically bypassed when the associated IRM channels are on range 8 or higher.
- d. This function is automatically bypassed when the IRM channels are on range 3 or higher.
- e. This function is automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. <u>ROD BLOCK MONITOR</u>		
a. Upscale	< 0.66 W + 42%	< 0.66 W + 45%
b. Inoperative	NA	NA
c. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 of divisions full scale
2. <u>APRM</u>		
a. Flow Biased Neutron Flux - Upscale	< 0.58 W + 50%*	< 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. <u>SOURCE RANGE MONITORS</u>		
a. Detector not full in	NA	NA
b. Upscale	< 2×10^5 cps	< 4×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 0.7 cps**	≥ 0.5 cps**
4. <u>INTERMEDIATE RANGE MONITORS</u>		
a. Detector not full in	NA	NA
b. Upscale	< 108/125 divisions of full scale	< 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. <u>SCRAM DISCHARGE VOLUME</u>		
a. Water Level - High	≤ 44 gallons	≤ 44 gallons
6. <u>REACTOR COOLANT SYSTEM RECIRCULATION FLOW</u>		
a. Upscale	< 108/125 divisions of full scale	< 111/125 divisions of full scale
b. Inoperative	NA	NA
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.

**Provided signal-to-noise ratio is >2. Otherwise, 3cps as trip setpoint and 2.8cps for allowable value.

##See Specification 3.4.1.1.2.a for single loop operation requirements.

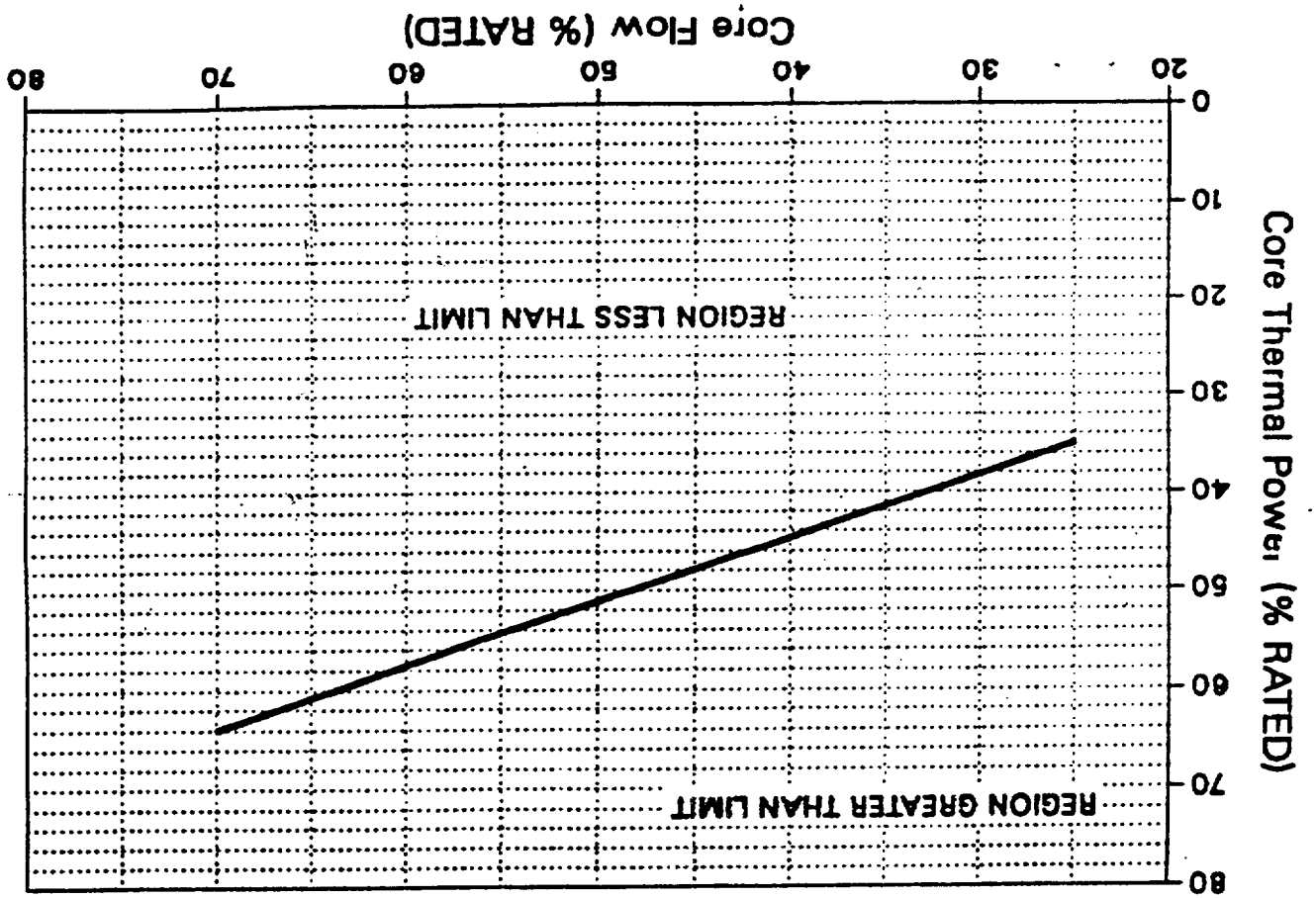


Figure 3.4.1.1.1-1
THERMAL POWER LIMITATIONS

REACTOR COOLANT SYSTEM

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed $\leq 80\%$ of the rated pump speed, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.
2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

<u>Trip Setpoint</u>	<u>Allowable Value</u>
$\leq 0.58W + 55\%$	$\leq 0.58W + 58\%$

3. Specification 3.2.1: The MAPLHGR limits shall be as follows:

- a. GE fuel: the limits specified in Figure 3.2.1-1 multiplied by 0.81.
- b. Exxon fuel: the limits specified in Figure 3.2.1-2 multiplied by 0.0.

4. Specification 3.2.2: the APRM Setpoints shall be as follows:

<u>Trip Setpoint</u>	<u>Allowable Value</u>
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

a. RBM - Upscale	<u>Trip Setpoint</u> $\leq 0.66W + 37\%$	<u>Allowable Value</u> $\leq 0.66W + 40\%$
b. APRM-Flow Biased	<u>Trip Setpoint</u> $\leq 0.58 + 46\%$	<u>Allowable Value</u> $\leq 0.58W + 49\%$

- b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1.
- c. Total core flow shall be greater than or equal to 42 million lbs/hr when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during two loop operation.#

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.

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. For GE fuel, 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) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for Exxon fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, for GE fuel, the APRM setpoints must be adjusted to ensure 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. For the Exxon fuel, no adjustment is required since operation within the MCPR and MAPLHGR operating limits assures that fuel mechanical design criteria are not violated.

POWER DISTRIBUTION LIMITS

BASES

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 Figures 3.2.3-1 and 3.2.3-2.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to a Exxon-core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71 and XN-NF-84-105. The principal result of this evaluation is the reduction in MCPR caused by the transient.

Figure 3.2.3-1 defines core flow dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded during a flow increase transient resulting from a motor-generator speed control failure. The flow dependent MCPR is only calculated for the manual flow control mode. Therefore, automatic flow control operation is not permitted. Figure 3.2.3-2 defines the power dependent MCPR operating limit which assures that the Safety Limit MCPR will not be exceeded in the event of a feedwater controller failure initiated from a reduced power condition.

Cycle specific analyses are performed for the most limiting local and core wide transients to determine thermal margin. Additional analyses are performed to determine the MCPR operating limit with either the Main Turbine Bypass inoperable or the EOC-RPT inoperable. Analyses to determine thermal margin with both the EOC-RPT inoperable and Main Turbine Bypass inoperable have not been performed. Therefore, operation in this condition is not permitted.

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

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

For single loop operation, the MAPLHGR limits for Exxon fuel are multiplied by a factor of 0.0. This multiplication factor precludes extended operation with one loop out of service.

For single loop operation, the RBM and APRM setpoints are adjusted by a 7% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits on THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

THERMAL POWER, core flow, and neutron flux noise level limitations are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability," dated February 10, 1984.

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

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

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4 REACTOR COOLANT SYSTEM

BASES (Continued)

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. A total of 10 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

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.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

PLANT SYSTEMS

BASES

3/4 7.4 SNUBBERS (continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, 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. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4 7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources required leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4 7.6 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO₂ systems, Halon systems and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

PLANT SYSTEMS

BASES

3/4 7.6 FIRE SUPPRESSION SYSTEMS (continued)

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight and pressure of the tanks.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4 7.7 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4 7.8 MAIN TURBINE BYPASS SYSTEM

The required OPERABILITY of the main turbine bypass system is consistent with the assumptions of the feedwater controller failure analysis in the cycle specific transient analysis.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPF-14

PENNSYLVANIA POWER & LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-387

1.0 INTRODUCTION

By letter dated December 12, 1986, Pennsylvania Power and Light Company (PP&L or the licensee) proposed to amend Appendix A of Susquehanna Steam Electric Station (SSES) Unit 1 Facility Operating License No. NPF-14. The requested amendment furnished information to support extended operation with the resident GE 8x8 fuel up to a fuel exposure of 40,675 MWD/MT and provided a revision to single loop operation (SLO) provisions in the Technical Specifications. Furthermore, Technical Specification changes were made to the previous Exxon fuel Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits and operating limit Minimum Critical Power Ratios (MCPRs) to reflect present ENC methodology and analyses.

2.0 EVALUATION

The staff evaluation of the licensee's proposed Technical Specification changes follows:

- (1) MAPLHGR limits for the resident GE fuel bundle types 8CR233 are extended from an average planar exposure limit of 33,069 MWD/MT to 40,675 MWD/MT. The resulting peak cladding temperature (PCT) limit and local oxidation fraction were calculated by GE based on the same plant conditions and systems analysis used to derive the current MAPLHGR limits defined in the SSES FSAR. The calculated values are well within the 10 CFR 50.46 Appendix K limits.

During review of a proposed revision to GESTAR II (NEDE-24011-P-A-1, "Generic Reload Fuel Application," dated August 1979), the staff approved an increase in the peak pellet exposure limit used as a fuel design analysis input parameter to 50,000 MWD/STU (Letter, R. Tedesco, NRC, to R. Engel, GE, dated November 7, 1980), which typically corresponds to a peak average planar exposure of 50,000 MWD/MTU. Although this limit bounds the proposed extension of average planar exposure in the SSES MAPLHGR limit Technical Specification figure to 40,675 MWD/MTU for the GE fuel, in review of future reload amendments, the staff will consider extended burnup experience, methods and surveillance data for specific methodology for extended burnup (NEDE-22148(P), "Extended Burnup Evaluation-Methodology, General Electric Company, June 1982) in approval of exposure levels above batch average burnups of 40,000 MWD/MTU. In addition, the staff will review and evaluate the radiological consequences of the Fuel Handling Accident involving both ENC and GE fuel

assemblies for future proposed burnup levels above those proposed in this amendment. Based on the application of approved methodology and design criteria for LOCA analyses and our previous approval of the proposed extended burnup level, the staff finds the proposed changes to the MAPLHGR limit curves for GE fuel to be acceptable.

- (2) The proposed MAPLHGR limits for the EXXON nuclear fuel are based on LOCA analysis results which were reviewed and approved in the most recent reload amendment for SSES Unit 1 Cycle 3 (S1C3) (Amendment 57 to License No. NPF-14, dated April 11, 1986). Since the calculated values for PCT and local oxidation fraction are within the 10 CFR 50.46 Appendix K limits, the staff finds the proposed change acceptable.

It is noted that the SSES Unit 1 proposed change results in separate Linear Heat Generation Rate and MAPLHGR versus average planar exposure Figures to distinguish between the fuel mechanical design analyses results and the Loss of Coolant Accident (LOCA) analysis results as related to the Limiting Conditions for Operation. This is consistent with the staff Safety Evaluation for the S1C3 reload amendment (Amendment 57) which approved the LHGR limit as a function of burnup for the ENC fuel types XN-1 and XN-2. The LHGR operating limit is based on a power profile used in the fuel design analysis as prescribed in XN-NF-81-21(A), Revision 1 "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1982. In addition, the staff notes that the design and analysis methodologies for the Exxon fuel design are the same as those used and approved for both Cycle 2 and Cycle 3 reload amendments. These methodologies include a modified RODEX 2 calculation as required by the staff safety evaluation of XN-NF-81-21. This modified analysis is necessary to confirm that the calculated end of life rod internal pressure does not exceed the system pressure. Thus, the previous approval of the present LHGR curve remains in effect. As stated in the staff safety evaluation for the previous reload amendments, the LHGR operating limit assures compliance with fuel design assumptions.

- (3) The Minimum Critical Power Ratio (MCPR) operating limits have been reevaluated by the licensee to reflect the results of transient thermal-hydraulic core analyses with the XCOBRA-T computer code. The staff has found the use of this code acceptable for BWR licensing calculations (Letter, G. Lainas, NRC to G. N. Ward, ENC dated October 27, 1986 "Acceptance for Referencing of Licensing Topical Report XN-NF-84-105, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis").

Exxon has reexamined certain transients discussed in the previous S1C3 submittal. These included Generator Load Rejection without Bypass (LRWB) and Feedwater Controller Failure (FWCF). These transients were

analyzed with End-of-Cycle Recirculation Pump Trip (EOC-RPT) operable and inoperable. The previous S1C3 analyses identified the Rod Withdrawal Error (RWE) as the limiting event for the determination of the operating limit MCPR. The previous analyses (reported in XN-NF-85-132, Rev 1, "Susquehanna Unit 1 Cycle 3 Reload Analysis" December 1985) were for Rod Block Monitor (RBM) setpoints of 106 and 108%. The proposed amendment considers an RBM setpoint of 108% only. Under this condition, the RWE remains the limiting event with a calculated delta-CPR of 0.23; this establishes the operating limit MCPR of 1.29 which is used in the revised Technical Specification Figures in the proposed Amendment. The XCOBRA-T analysis for the Generator Load Rejection transient without bypass and with inoperable EOC-RPT resulted in a calculated delta-CPR of 0.27 which is incorporated in the revised Technical Specification Figures as an operating limit MCPR of 1.33 for the additional conditions.

The licensee has taken into account the impact of reduced flow and reduced power on transient response. This is reflected in the new flow dependent and power dependent MCPR operating limits incorporated in the SSES Unit 1 Technical Specifications. The automatic flow control mode of operation is still not permitted for SSES Unit 1.

Our review of the transient and accident analyses done for the proposed amendment indicates that appropriate methodology and input have been used and the results provide a suitable basis for the SSES Unit 1 Technical Specification changes. The proposed MCPR operating limit changes are, therefore, acceptable.

- (4) The licensee has proposed a modification to the present Technical Specification Limiting Condition for Operation (LCO) for the Single Loop Operation (SLO) mode. The proposed change consists of setting the MAPLHGR limit multiplier to 0.0 for extended SLO. The effect of this change is to preclude SLO for an extended period of time. This is an interim measure until ENC can provide revised analyses to justify applicability of the General Electric operating limits as specified in General Electric Service Information Letter 380 Revision 1 to Exxon fuel loadings. Revised analyses with current approved methodology are to be provided in a future submittal and should include a specific analysis of the one-pump seizure accident. The approach and changes to the Limiting Conditions for Operation are the same as those previously reviewed and approved by the staff in connection with our review of SSES Unit 2 reload submittal (Amendment No. 31 to Facility Operating License No. NPF-22, dated October 3, 1986) and are acceptable.

The following specification changes have been requested to accommodate the previously discussed extended MAPLHGR limits for GE 8x8 fuel, modified MAPLHGR limits for the Exxon 8x8 fuel, revised operating limit MCPR limits with consideration of approved ENC methodology, modifications to SLO operational limits and additional discussion in the Bases pages related to the above specifications.

- (1) Figures 3.2.1-1 and 3.2.1-2: These revised Figures replace the previous Figures and reflect the results of additional GE and ENC LOCA analyses.
- (2) Figures 3.2.3-1 and 3.2.3-2; Table 3.2.3-1: These Figures define core flow dependent MCPR operating limits and power dependent MCPR operating limits which consider the results of revised ENC analyses using recently approved methodology and the prior SIC3 analysis of the Rod Withdrawal Error transient. The licensee has proposed the deletion of Table 3.2.3-1 which summarized the MCPR results applicable to the SIC3 reload analysis. Since some of the conditions defined in this Table are no longer applicable (e.g., a Rod Block Monitor Trip Point setting of 106%), the staff finds the deletion acceptable. It is noted that this approach is consistent with that used and approved for Susquehanna, Unit 2 in Amendment 31 to License No. NPF-22 (October 1986).
- (3) LCO 3.4.1.1.2 and Table 3.3.6-2: Changes were made to restrict operation in the single loop mode until additional analyses can be provided to justify applicability of the GE operating limits to Exxon fuel loadings.
- (4) LCO page 3/4 2-6: Editorial changes were made to reflect references to the revised MCPR Figures and to provide consistency with the Susquehanna SES Unit 2 Technical Specifications.
- (5) Bases pages B 3/4 2-2, B 3/4 4-1 and B 3/4 7-4: Revised and expanded text was provided to reflect the bases for the proposed changes.

The staff has reviewed the material submitted by the licensee for the proposed changes discussed above. Based on the results of our review, we find that, as discussed earlier, sufficient basis has been provided to allow extension of the MAPLHGR limits for the resident GE fuel, revision of the MAPLHGR limits for the resident ENC fuel, updated operating limit MCPR Figures and interim restrictions on operation in the single loop operation (SLO) mode. The proposed TS changes are therefore acceptable for SSES Unit 1.

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 the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 4414) on February 11, 1987, and consulted with the State of Pennsylvania. No public comments were received, and the state of Pennsylvania did not have any comments.

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 nor to the health and safety of the public.

Principal Contributor: M. McCoy, RSB, DBL

Dated: May 7, 1987