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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. NPF-14

PENNSYLVANIA POWER AND LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-387

1.0 INTRODUCTION

In a letter from N. W. Curtis to A. Schwencer, NRC, dated July 22, 1983, as supplemented on July 26, August 2, November 22, 1983, June 21, and November 13, 1984, Pennsylvania Power and Light Company requested a change in the Technical Specification high setpoint for the main steam line radiation monitors from three times background to seven times background. The change was requested because of a number of spurious reactor trips.

The amendment, as proposed by the licensee, changes the Unit 1 Technical Specifications to revise the Main Steam Line Radiation-High Setpoint from three times normal background (three times setting) to seven times normal background (seven times setting). The purpose of the proposed Technical Specification change increasing the setpoint for Main Steam Line Radiation-High was to prevent unwarranted reactor scrams caused by N-16 spikes in the main steam line. The change was initiated by PP&L to correct operating problems experienced at Susquehanna Unit 1 with the condensate polishing system, which sometimes resulted in the releases of resin fines (impurities) to the reactor vessel. These impurities produced a transient increase in N-16 activity in the steam, which caused radiation levels to exceed the setpoint. As a result the Pennsylvania Power and Light Company requested a setpoint change to provide additional margin against unnecessary main steam line isolations. Subsequent to this request, the problem with the condensate polishing system was corrected by a design modification.

Since the licensee's original request the problem which resulted in these spurious trips has been corrected. The licensee, however, has since requested that the setpoint be changed permanently in the Technical Specifications. The licensee has provided a safety analysis to support this change. The staff's evaluation of this analysis follows.

2.0 EVALUATION

The licensee has stated that the main steam line radiation monitors are not relied upon for any design basis accident analyses. The staff agrees with this statement. Furthermore, the staff finds that the purpose of these monitors is to provide a quick means of detecting gross degradation of the fuel. The current setpoint for the radiation monitors (three times background) corresponds to the cladding failure of approximately 15 fuel rods. The requested setpoint of seven times background would increase this value to an equivalent cladding failure of approximately 44 fuel rods. When compared with the total number of fuel rods in the core (47,368), the failures associated with either the three times setting or the seven times setting corresponds to a very small percentage of failed rods. The consequences of the failure of 44 fuel rods can be conservatively determined by comparison with the control rod drop design basis accident which is assumed to produce cladding failure of 770 fuel rods. The projected thyroid and whole body doses for this design basis control rod drop accident at the Exculsion Area and Low Population Zone Boundaries are 1.47 Rem and less than 0.1 Rem, and 0.32 Rem and less than 0.1 Rem, respectively. Because the projected failed fuel value for the seven times setting is better than a factor of ten less than that projected for the rod drop accident, the projected doses would be more than a factor of ten less. Consequently, the doses for the new setpoint would be much less than a Rem for both the thyroid and whole body at both the Exclusion Area and Low Population Zone boundaries for design basis accidents.

An additional consideration is that fuel design and fuel management employed at current boiling water reactors (BWR's) has resulted in significantly less failed fuel during operation than existed at the time the monitoring requirement was promulgated.

The staff concludes that on the basis that 1) the monitors perform no functions for mitigating the consequences of design basis accidents; 2) the setpoint change would result in only a small incremental increase in the projected doses; 3) the current vintage BWR fuel is performing in a manner superior to the fuel in place prior to the requirement for monitor installation, the requested change is acceptable. The staff believes that such accident monitors should be set high enough to avoid spurious reactor trips, but low enough to detect significant fuel degradation. The staff believes that the licensee meets this criterion with a setpoint of seven times background and therefore, finds the licensee's request acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or 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 findings. 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 (48 FR 39540) on August 31, 1983, 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 Contributors: Mari-Josette Campagnone, Project Directorate No. 3, DBL
Theodore Quay, Plant Systems Branch, LPWR-A

Dated: April 15, 1986

AMENDMENT NO. 58TO FACILITY OPERATING LICENSE NO. NPF-14
SUSQUEHANNA STEAM ELECTRIC, UNIT 1

DISTRIBUTION:

Docket No. 50-387
NRC PDR
Local PDR
PRC System
NSIC
BWD-3 r/f
HDenton/DEisenhut
RBernero
ORAS
EHylton (2)
MCampagnone (4)
Attorney, OELD
EJordan
BGrimes
JPartlow
TBarnhart (4)
OPA
LFMB
RDiggs, LFMB
LHarmon, I&E

LICENSE AUTHORITY FILE COPY

DO NOT REMOVE

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

Posted
Amdt. 57
to NPF-14

APR 1 : 1986

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:

Subject: Amendment No. 57 to Facility Operating License No. NPF-14, Susquehanna Steam Electric Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 57 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. The amendment is in response to your letter dated January 16, 1986, as supplemented on March 18, 1986.

This amendment revises the Unit 1 Technical Specifications to include operational control for Cycle 3 operation.

A copy of the related safety evaluation supporting Amendment No. 57 to Facility Operating License NPF-14 is enclosed.

Sincerely,

Elinor G. Adensam

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

Enclosures:
1. Amendment No. 57 to NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

01002003З

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

cc:

Jay Silberg, Esq.
Shaw, Pittman, Potts, & Trowbridge
1800 M Street, N. W.
Washington, D.C. 20036

Bryan A. Snapp, Esq.
Assistant Corporate Counsel
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. William E. Barberich
Manager-Nuclear Licensing
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. R. Jacobs
Resident Inspector
P.O. Box 52
Shickshinny, Pennsylvania 18655

Mr. R. J. Benich
Services Project Manager
General Electric Company
1000 First Avenue
King of Prussia, Pennsylvania 19406

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Resources
Commonwealth of Pennsylvania
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Robert W. Alder, Esquire
Office of Attorney General
P.O. Box 2357
Harrisburg, Pennsylvania 17120

Mr. William Matson
Allegheny Elec. Cooperative, Inc.
212 Locust Street
P. O. Box 1266
Harrisburg, Pennsylvania 17108-1266

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

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

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

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. 57
License No. NPF-14

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company (the licensee), dated January 16, 1986, as supplemented on March 18, 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 attachment to this license amendment and paragraph 2.C.(2) of the 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. 57, 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.

3. This amendment is effective upon startup following the Unit 1 second refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: APR 1 1 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 57

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
iii	iii (overleaf)
iv	iv
xxi	xxi
xxii	xxii
B 2-5	B 2-5
B 2-6	B 2-6 (overleaf)
3/4 1-1	3/4 1-1 (overleaf)
3/4 1-2	3/4 1-2
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6 (overleaf)
3/4 2-7	3/4 2-7 (overleaf)
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9 (overleaf)
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10a
3/4 2-10b	3/4 2-10b
B 3/4 1-1	B 3/4 1-1 (overleaf)
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3 (overleaf)
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

INDEX

LIST OF FIGURES

LIST OF TABLES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale 118% setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the monitored core K_{eff} and the predicted core K_{eff} shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity difference greater than 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity difference between the monitored core K_{eff} and the predicted core K_{eff} shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 700 MWD/MT of core exposure during POWER OPERATION.

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 for GE fuel and AVERAGE BUNDLE EXPOSURE for Exxon fuel shall not exceed the limits shown in Figures 3.2.1-1, and 3.2.1-2.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

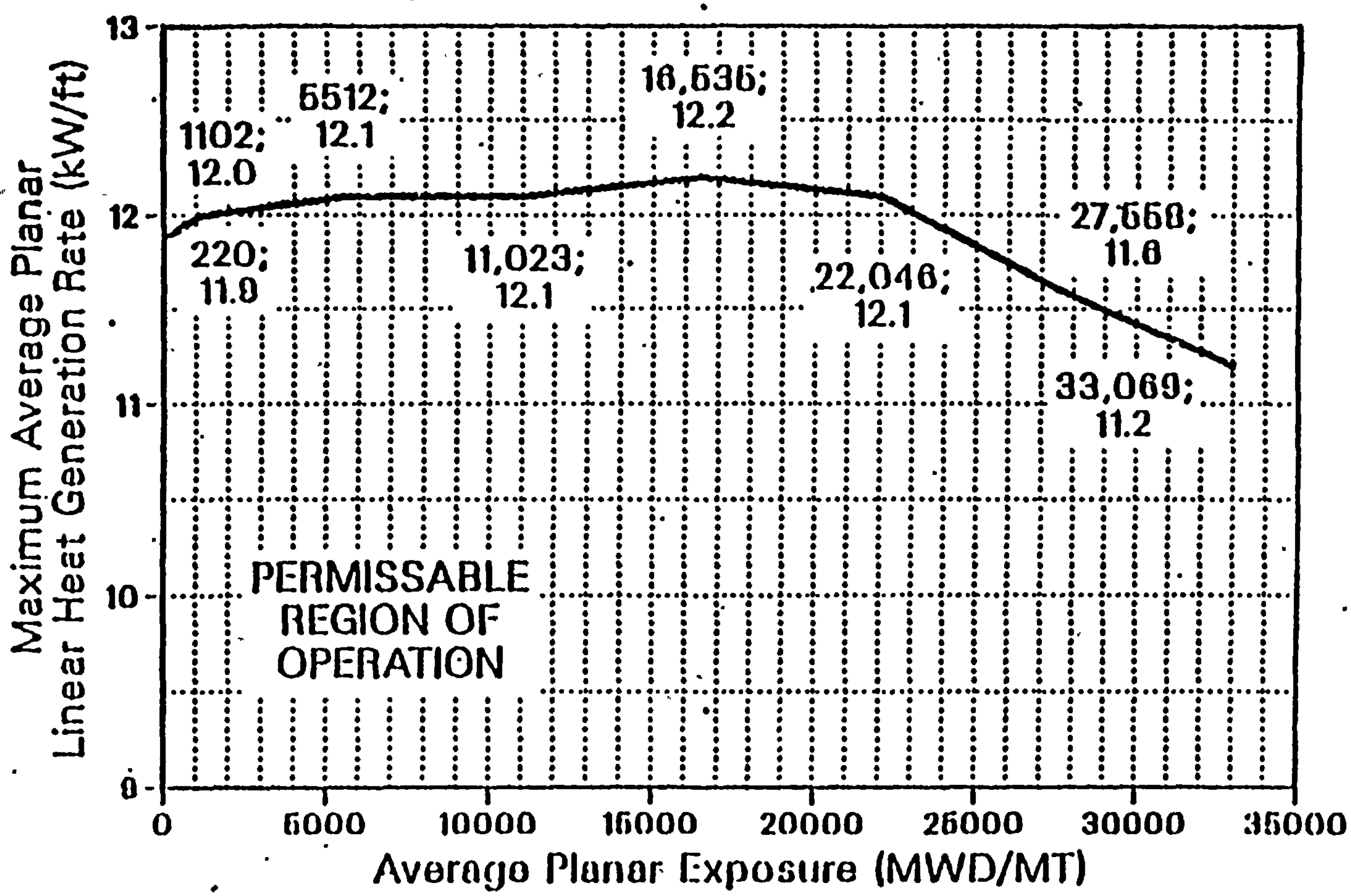
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

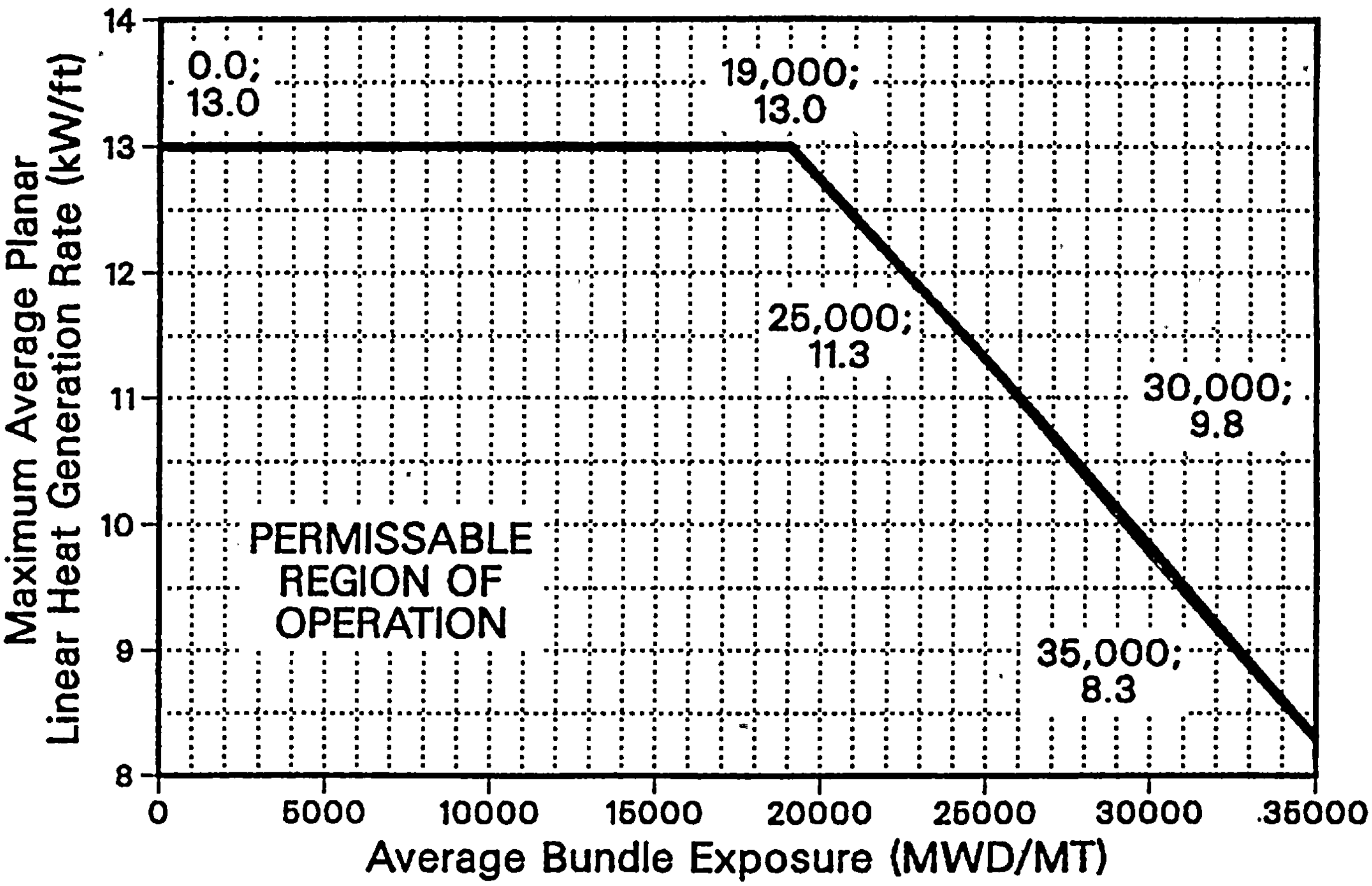
d. The provisions of Specification 4.0.4 are not applicable.

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

THIS PAGE INTENTIONALLY LEFT BLANK



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

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

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

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 = 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.

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 FRTP 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:

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

*With MFLPD greater than the FRTP 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.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be:

a. greater than or equal to the applicable MCPR limit determined from Table 3.2.3-1 during steady state operation at rated core flow, or

b. greater than or equal to the greater of the two values determined from Table 3.2.3-1 and Figure 3.2.3-1 during steady state operation at other than rated core flow.

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

ACTION:

With MCPR less than the applicable MCPR limit determined from Table 3.2.3-1 and Figure 3.2.3-1, 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 Table 3.2.3-1 and Figure 3.2.3-1:

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

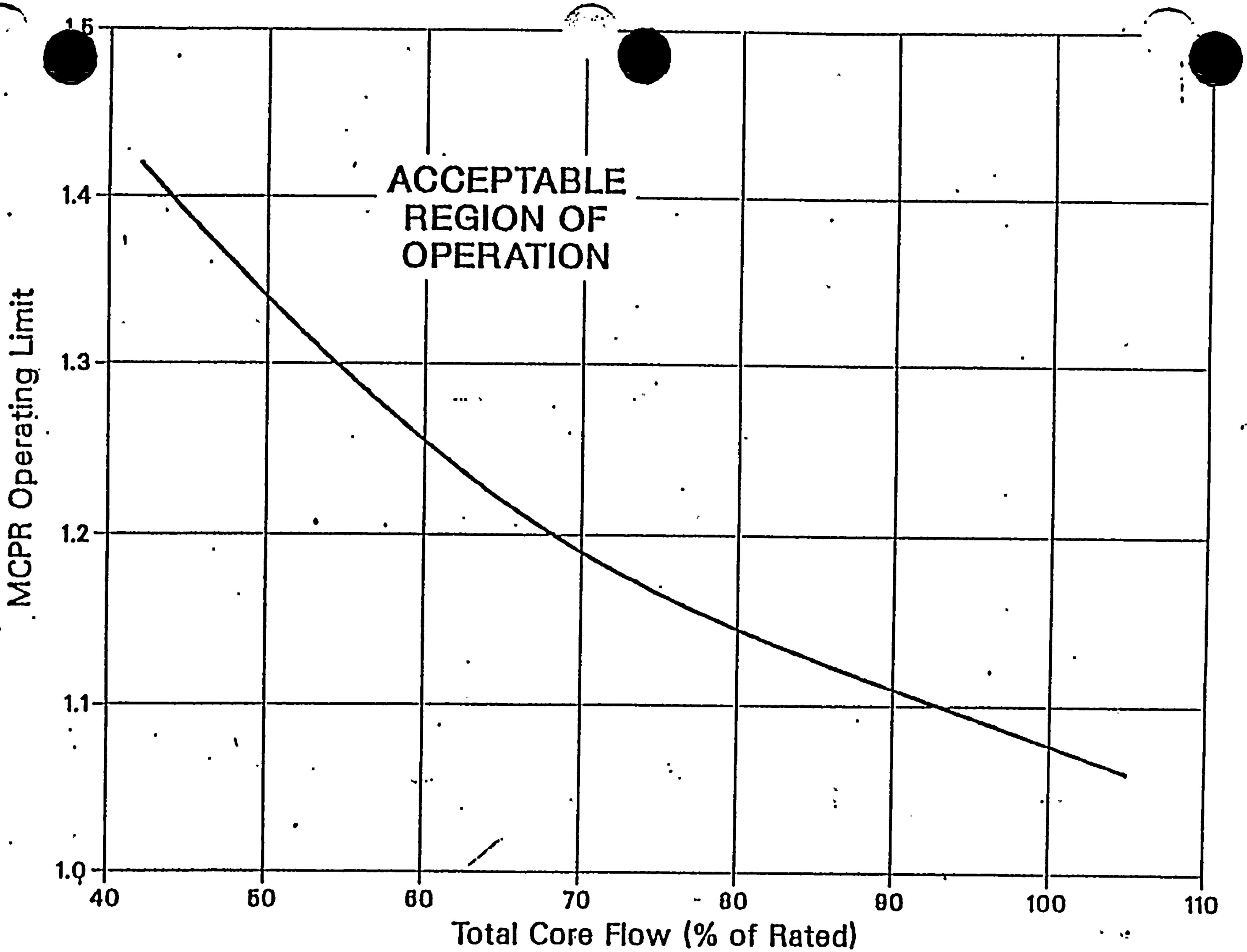
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

THIS.PAGE INTENTIONALLY LEFT BLANK.

TABLE 3.2.3-1

MCPR OPERATING LIMITS FOR RATED CORE FLOW

EQUIPMENT STATUS	MCPR OPERATING LIMIT
1. EOC-RPT and Main Turbine Bypass OPERABLE, RBM setpoint ≤ 108%	1.29
2. EOC-RPT Inoperable, Main Turbine Bypass OPERABLE, RBM setpoint ≤ 108%	1.33
3. Main Turbine Bypass Inoperable, EOC-RPT OPERABLE, RBM Setpoint ≤ 108%	1.29
4. EOC-RPT and Main Turbine Bypass OPERABLE, RBM Setpoint ≤ 106%	1.25
5. EOC-RPT Inoperable, Main Turbine Bypass OPERABLE, RBM Setpoint ≤ 106%	1.33
6. Main Turbine Bypass Inoperable, EOC-RPT OPERABLE, RBM Setpoint ≤ 106%	1.26



REDUCED FLOW MCPR OPERATING LIMIT
FIGURE 3.2.3-1

3/4.2.4 LINEAR HEAT GENERATION RATE

GE FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.1 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

4.2.4.1 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.

3/4.2.4 LINEAR HEAT GENERATION RATE

ENC FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.2 The LINEAR HEAT GENERATION RATE (LHGR) for ENC fuel shall not exceed the LHGR limit determined from Figure 3.2.4.2-1.

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

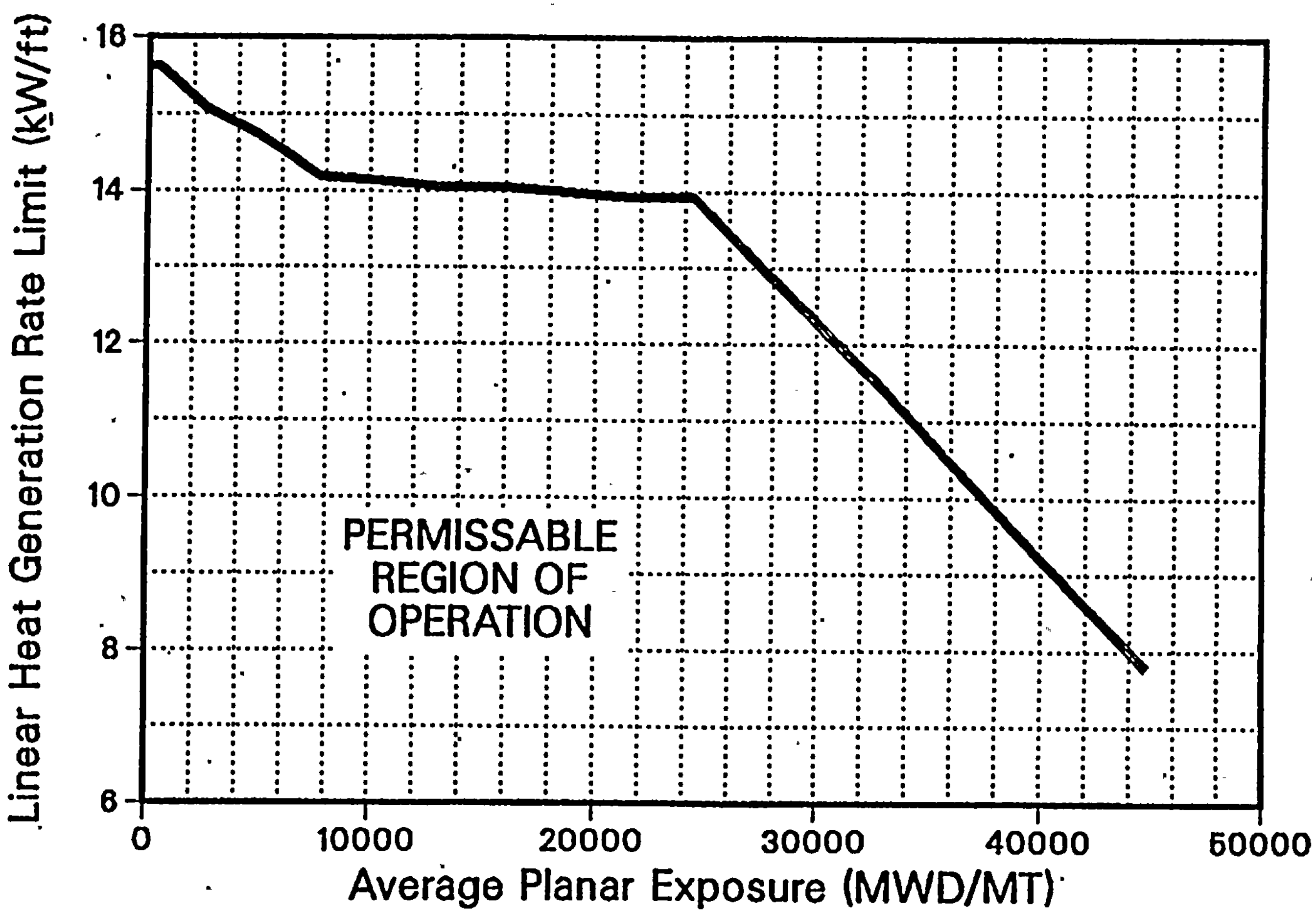
4.2.4.2 LHGRs for ENC fuel shall be determined to be equal to or less than the limit:

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.



EXP	LHGR
0	15.62
500	15.62
2,500	15.10
5,070	14.71
7,730	14.19
10,290	14.13
13,090	14.06
15,910	14.06
18,750	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,960	10.47
38,900	9.55
41,830	8.65
44,760	7.77

LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
EXXON 8X8 FUEL
FIGURE 3.2.4.2-1

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

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

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

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

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

3/4.1.2 Reactivity Anomalies

Since the SHUTDOWN MARGIN requirement is small, a careful check on actual reactor conditions compared to the predicted conditions is necessary. Any changes in reactivity from that of the predicted (predicted core k_{eff}) can be determined from the core monitoring system (monitored core k_{eff}). In the absence of any deviation in plant operating conditions or reactivity anomaly, these values should be essentially equal since the calculational methodologies are consistent. The predicted core k_{eff} is calculated by a 3D core simulation code as a function of cycle exposure. This is performed for projected or anticipated reactor operating states/conditions throughout the cycle and is usually done prior to cycle operation. The monitored core k_{eff} is the k_{eff} as calculated by the core monitoring system for actual plant conditions.

Since the comparisons are easily done, frequent checks are not an imposition on normal operation. A 1% deviation in reactivity from that of the predicted is larger than expected for normal operation, and therefore should be throughly evaluated. A deviation as large as 1% would not exceed the design conditions of the reactor.

3/4.1.3 CONTROL RODS

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

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

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

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

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the limit specified in Specification 2.1.2 during the core wide transient analyzed in the cycle specific transient analysis report. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the limit specified in Specification 2.1.2. The occurrence of scram times longer then those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

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

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

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

Parametric Control Rod Drop Accident analyses have shown that for a wide range of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient, effective delayed neutron fraction, and maximum four-bundle local peaking factor are compared with the inputs to the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

280 cal/gm design limit to demonstrate compliance for each operating cycle. If cycle-specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle-specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in XN-NF-80-19 Volume 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs. of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement for the sodium penetrate solution is necessary to ensure that the sodium penetaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

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 and 3.2.1-2.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1 and 3.2.1-2 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, the APRM setpoints must be adjusted to ensure that $\geq$1% plastic strain and fuel centerline melting do not occur during the worst anticipated operational occurrence (AOO), including transients initiated from partial power operation.

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 Table 3.2.3-1.

When the less operationally limiting Rod Block Monitoring trip setpoint (.66W + 42% from Table 3.3.6-2) is used, a more limiting MCPR valve Table 3.2.3-1 is applicable due to a larger delta MCPR from the limiting Rod Withdrawal Error (RWE) transient.

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. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of Figure 3.2.3-1 is to define MCPR operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the maximum of the rated flow MCPR determined from Table 3.2.3-1 and the reduced flow MCPR determined from Figure 3.2.3-1. The reduced flow MCPR assures that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The reduced flow MCPR is only calculated for the manual flow control mode. Therefore, automatic flow control operation 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

MINIMUM CRITICAL POWER RATIO (Continued)

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.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any fuel rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 57 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 January 16, as supplemented on March 18, 1986, from H. Keiser, Pennsylvania Power and Light Company, to E. Adensam, NRC, (Reference 1), Technical Specification changes were proposed for the operation of Susquehanna Unit 1 for Cycle 3 (SIC3) using Exxon manufactured fuel assemblies and Exxon analyses and methodologies for Cycle 3 operation. In order to support the requested Technical Specification changes needed for operation of Susquehanna Unit 1 during Cycle 3, the licensee provided as attachments the Susquehanna Steam Electric Station (SES) Unit 1 Cycle 3 Proposed Startup Physics Test Summary Descriptions and those reports and materials contained in References 2, 3, and 4.

2.0 EVALUATION

Cycle 3 is the second reload for Susquehanna using Exxon fuel and analyses. Since the fuel and methodologies for this reload are nearly the same as those for the previous reload (and for previous Dresden reloads) the relevant methodology reviews have previously been completed as needed for this review. Furthermore, the Susquehanna Unit 1, Cycle 2 (SIC2) reload submittal and review included numerous Technical Specification changes to match the Exxon methodologies and analyses. Therefore, the changes in Technical Specifications needed for SIC3 are minimal compared to the previous reload (SIC2). Beyond the switch to additional Exxon fuel (similar but not identical to the previous Exxon reload fuel) there is nothing unusual about SIC3, and the proposed Technical Specification changes. The requested Technical Specification changes are entirely related to the use of the new fuel and slightly different core parameters.

A. Reload Description

The SIC3 reload will remove the GE medium enrichment fuel assemblies, retain the 276 twice burned GE high enrichment fuel assemblies and 192 Exxon 2.72 percent U235 XN-1 assemblies of the previous cycle, and add 296 new Exxon 2.89 percent U235 XN-2 assemblies. The XN-2 assemblies are similar to the XN-1 fuel. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. The loading and cycle analyses are based on a Cycle 2 end of cycle exposure of 13.9 to 15.3 Gigawatt Days/Metric Ton of Uranium (GWD/MTU). Cycle 3 will be an 18 month cycle. This is generally a normal reload with no unusual core features or characteristics. Technical Specification changes are few and primarily related to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), Linear Heat Generation Rate (LHGR) and Minimum Critical Power Ratio (MCPR) limits, and Average Power Range Monitor (APRM) setpoint changes associated with Cycle 3 operation.

B. Fuel Design

The design for the XN-2 fuel is very similar to the XN-1 fuel used for SIC2. The primary difference is a small increase in average U235 enrichment and a different, although similar, enrichment and gadolinium pin distribution pattern. The design and analysis of the XN-2 fuel support a batch average burnup of 30 GWD/MTU, the same as that for the XN-1.

The design and analysis methodologies used by the licensee are contained in References 5, 6, and 7. These are the same methodologies which were used and approved for SIC2, including the RODEX 2 calculations using a modified RODEX2 approved in the Cycle 2 review. The design and analysis of the XN-2 fuel for SIC3 is thus acceptable.

Two areas related to the LHGR operating limit have been changed in the Technical Specifications for Cycle 3 to assure compliance with fuel design assumptions. For GE fuel, the Technical Specifications provide for (1) a limit on the peak operating LHGR of 13.4 kw/ft and (2) a setdown of APRM scram and rod block settings when there are excessively high power peaking factors at less than 100 percent power operation. Previously for Exxon fuel, operation without Exxon specific specifications was approved. Recent discussions with Exxon on the need for an Exxon fuel LHGR limit in the Technical Specification has resulted in an addition of an LHGR operating limit as a function of burnup for the SIC3 specifications. This is based on the power profile used in the fuel design analysis (Figure 5.10 of Reference 5) and satisfies staff concerns in this area. Also, the analysis for SIC3 indicated the need for an APRM setpoint adjustment for the Recirculation Flow Controller Failure with Increasing Flow event. Thus, the APRM scram and rod block setpoints will be reduced for Exxon, as well as GE fuel, by the Fraction of Rated Power/Maximum Fraction Limiting Power Density (FRP/MFLPD) ratio used for GE fuel. These are acceptable changes to the Technical Specifications.

The mechanical response of Exxon fuel assemblies to design Seismic-LOCA events is essentially the same as that for the GE assemblies. As for the previous cycle, the channel boxes were manufactured for the assemblies to GE design criteria and dimensions. The licensee's analyses which indicates that design limits are not exceeded are acceptable.

C. Thermal-Hydraulic Design

The Exxon thermal-hydraulic methodology and criteria used for the SIC3 design and analysis is the same as that used and approved for the previous cycle. As for SIC2, statistical aspects of the methodology were not needed since bounding transient analyses were used. The previous SIC2 review concluded that hydraulic compatibility between GE and Exxon fuel, the calculation of core bypass flow, and the safety limit MCPR are all acceptable. The MCPR safety limit for SIC3 continues to be 1.06. Susquehanna has Technical Specifications implementing surveillance for detection and suppression of power oscillation. These Technical Specifications are still acceptable for SIC3.

D. Nuclear Design

Exxon nuclear design methodologies for SIC3 are the same as those previously used and approved for SIC2. The SIC3 reload replaces about one third of Cycle 2 fuel with new Exxon fuel. The loading pattern is a normal type of scattered configuration. The axial maximum planar average enrichment of the new assembly is 2.98 percent U235.

The beginning of cycle shutdown margin is calculated to be 2.26 percent Δk, the R factor is 0.27 percent Δk, and thus the cycle minimum shutdown margin is 1.99 percent Δk, well in excess of the required 0.38 percent Δk. The Standby Liquid Control System also fully meets shutdown requirements.

The existing new fuel storage calculations are based on k_∞ of the assembly. If the maximum enrichment zone is such that k_∞ is less than 1.30 at limiting state conditions then the required criticality limits are met. For the new Exxon fuel k_∞ under these conditions is 1.11 and the criterion is met. The existing spent fuel pool criticality calculations have met criteria using a U235 assembly average enrichment of 3.25 percent and no burnable poison. Since the maximum corresponding enrichment of the new fuel is 2.98 percent the previous calculations are still acceptable.

Susquehanna will continue to use the Exxon POWERPLEX core monitoring system to monitor reactor parameters. We have not specifically reviewed details of this system (nor have we in the past reviewed details of the GE process computer monitoring system), but we have reviewed the principal methodologies involved in the system and consider them to be appropriate and acceptable. The system has been in use during Cycle 1 and 2 and has provided suitable monitoring and predictive results.

E. Transient and Accident Analyses

The Exxon transient methodology is contained in Reference 8. This methodology as applied in SIC2 and SIC3 is acceptable. (Aspects of the methodology review not yet completed involve statistical analyses. This is not used in the SIC2 or SIC3 analyses. Instead bounding parameters were used in the calculations.)

Exxon examined the design events discussed in the SIC2 submittal, and in Reference 8. The SIC3 submittals provided by the Pennsylvania Power and Light Company presented results for more limiting events than those analyzed by Exxon. These included Generator Load Rejection without Bypass (LRWOB), Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LOFWH) and Rod Withdrawal Error (RWE). These transients were analyzed with End of Cycle Recirculation Pump Trip (EOC-RPT) or with Main Turbine Bypass (MTB) inoperable. The RWE was analyzed for a range of Rod Block Monitor settings, including values of 1.06 and 1.08 used in the Technical Specifications.

These various analyses were used to determine the Technical Specification MCPR limits. The RWE is the limiting event for normal operation with RPT

and MTB operable. For SIC2, the Standard Technical Specification scram times were used for all analyses so no scram speed adjustments to the MCPR limits in the present specifications are necessary for SIC3.

Reduced flow operation for manual flow control was presented for SIC2 and this analysis is still applicable for SIC3, including the operating limit plot of MCPR vs core flow. The automatic flow control mode of operation is still not permitted for SIC3.

Compliance with the ASME code overpressurization criterion was demonstrated by analyses of the MSIV closure event with MSIV position switch failure. Maximum pressure was 102 percent of vessel design pressure, well under the 110 percent criterion.

The LOCA analyses for SIC2 using Exxon methodology was approved in the Cycle 2 review. The same methodology was used for Cycle 3 analysis to provide MAPLHGR limits for the XN-2 fuel. These calculations showed that the previous limits for XN-1 fuel provide acceptable limits for XN-2. Thus, a single MAPLHGR limit as a function of burnup for Exxon fuel is contained in the Technical Specifications. The LOCA analysis for SIC2 at full power operation and 89 percent core flow resulted in a slightly lower Peak Clad Temperature (PCT) than that at full flow. This analysis is applicable to SIC3. Thus, operation in the expanded power/flow region is acceptable.

The rod drop accident was analyzed for SIC2 using approved methodology. The resulting maximum fuel enthalpy of 83 cal/gm is well below the limit of 280 cal/gm. The analysis assumed a control rod reactivity worth which requires the use of GE's Banked Position Withdrawal Sequence.

Our review of the transient and accident analyses done for SIC3 reload indicates that appropriate methodologies and assumptions have been used. Therefore, the Technical Specifications are acceptable.

F. Technical Specification Changes

The following specification changes have been requested and approved in order to support re-loading with XN-2 fuel. The changes approved for Cycle 2 operation have accomplished the major aspects of the transfer to Exxon fuel using methodologies. Except for the LHGR limit addition only minor cycle specific parameter changes are reflected in the Cycle 3 Technical Specification changes.

(1) 3.1.2, Action b.: This is an administrative change to correct an error. It is an acceptable change.

(2) 3/4.2.1 and related figures: This change removes the figure for MAPLHGR vs exposure for 1.8 percent enriched GE fuel (not used in Cycle 3) and provides a (similar) figure for Exxon 8x8 fuel for both XN-1 and

XN-2 type fuel. The Exxon figure for XN-2 fuel is the same as that used for XN-1 fuel up to 25 GWD/MTU, because the LOCA analyses for XN-2 fuel is calculated to fall within the same limits. Beyond 25 GWD/MTU the curve is more restrictive than required by the LOCA analyses in order to maintain the power history assumed in the approved mechanical design analysis. These changes are acceptable.

(3) 3.3.2: This change requires the reduction of the APRM scram and rod block trip setpoints by T (where T is related to FRP/MFLPD) for both GE and Exxon fuel. This is an acceptable change.

(4) Table 3.2.3-1: The changes to this table of MCPR operating limits are the results of the core wide transient and local event analyses resulting in MCPR changes for the limiting events for Cycle 3. The limits are for normal operation, operation with EOC-RPT inoperable or Main Turbine Bypass inoperable at RBM setpoints of 106 percent and 108 percent. The limits are applicable to both GE and Exxon fuel. The analyses were based on approved methodologies. Therefore, the changes are acceptable for Cycle 3.

(5) 3/4.2.4.1 and .2 and Figure 3.2.4.2-1: These changes provide a LHGR limit for Exxon fuel in addition to the existing GE fuel limit. The Exxon limit is a function of exposure and provides information to find the value of T in specification 3.2.2. It is to be noted that the correct figure is provided in Reference 4. Reference 4 contains only a small correction over a limited burnup range.

(6) There are also minor changes to the index and to Bases related to the above changed specifications. The Bases changes are primarily administrative, reflecting the above technical changes and are acceptable.

G. Summary

We have reviewed the reports submitted for the Cycle 3 reload of Susquehanna Unit 1 with Exxon fuel and with Exxon methodologies and analyses. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the changes in Cycle 3 fuel loading and the operating limits associated with these changes and reload parameters.

H. References

1. Letter from H. Keiser, Pennsylvania Power & Light Co., to E. Adensam, NRC, "Susquehanna Steam Electric Station, Proposed Amendment 78 to License No. NPF-14," January 16, 1986

2. XN-NF-85-130, Rev. 1, "Susquehanna Unit 1 Cycle 3 Plant Transient Analyses," December 1985

3. XN-NF-85-132, Rev. 1, "Susquehanna Unit 1 Cycle 3 Reload Analysis," December 1985

4. Letter from H. Keiser, PP&L, "Susquehanna Steam Electric Station, Revision to Proposed Amendment 78," March 18, 1986

5. XN-NF-81-21(A), Rev. 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1982

6. XN-NF-81-21(P), Rev. 1,, Supplement 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR 8x8 Reload Fuel," March 1985

7. XN-NF-81-58(A), Supplement 1 & 2, Rev. 2, "RODEX 2 Fuel Rod Thermal-Mechanical Response Evaluation Model," March 1984

8. XN-NF-79-71(P), Rev. 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactor : EXEM BWR ECCS Evaluation Model," September 1982

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (51 FR 8599) on March 12, 1986, 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 Contributors: Howard Richings, Reactor Systems Branch, DBL
Larry Phillips, Reactor Systems Branch, DBL
Mari-Josette Campagnone, Project Directorate No. 3, DBL

Dated: [illegible]