November 2, 1983

Docket No.: 50-387

Mr. Norman W. Curtis
Vice President
Engineering and Construction - Nuclear
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Curtis:

Subject: Amendment No. 17 to Facility Operating License No. NPF-14 - Susquehanna Steam Electric Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 17 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. The amendment is in response to your letters dated February 3, 1983 and March 11, 1983. This amendment changes Technical Specification 3.2.2 and Tables 2.2.1-1 and 3.3.6-2 to allow the expansion of the operating region of the power/flow map for the initial fuel cycle. This amendment also changes Technical Specification Table 3.3.7.9-1 to correct administrative errors, decrease the number of detectors in two fire zones, and add fire zones that warrent inclusion in the Technical Specifications, reflecting as-built conditions.

A copy of the related safety evaluation supporting Amendment No. 17 to Facility Operating License NPF-14 is enclosed.

Sincerely,

Original signed

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 17 to NPF-14
2. Safety Evaluation

cc w/ enclosures:
See next page

DOBE 11/2/83 NONE

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	DL:AD/L
RLPerch:pt	EGHylton	ASchwencer		TMNovak
10/4/83	10/[illegible]/83	10/11/83	10/[illegible]/83	11/2/83

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Susquehanna

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U. S. Environmental Protection Agency
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PENNSYLVANIA POWER AND LIGHT COMPANY
ALLEGHENY ELECTRIC COOPERATIVE, INC.
DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. NPF-14

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

   A. The applications for amendments filed by the Pennsylvania Power and Light Company, dated February 3, 1983 and March 11, 1983, comply 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;

   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.17 , 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1983

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	DL:AD/L
RLPerch:pt	EGWynton	ASchwencer		TMNovak
10/4/83	10/4/83	10/11/83	10/24/83	10/2/83

ATTACHMENT TO LICENSE AMENDMENT NO. 17
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.

REMOVE	INSERT
2-3	2-3
2-4	2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54
3/4 3-77	3/4 3-77
3/4 3-78	3/4 3-78
3/4 3-79	3/4 3-79
3/4 3-80	3/4 3-80

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

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

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.58 W+59%, with a maximum of	≤ 0.58 W+62%, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero*	≥ 11.5 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10% closed	≤ 11% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High	≤ 88 gallons	≤ 88 gallons
9. Turbine Stop Valve - Closure	≤ 5.5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 460 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

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

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

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

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.

---

* 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.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit determined from Figure 3.2.3-1 times the K_f shown in Figure 3.2.3-2, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2, with:

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

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

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

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

where:

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

N_i = number of active control rods measured in the ith surveillance tests,

τ_i = average scram time to notch 39 of all rods measured in the ith surveillance test, and

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

APPLICABILITY:

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

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 one 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 one 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.

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 shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + 40%	≤ 0.66 W + 43%
b. Inoperative	NA	NA
c. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 of divisions full scale
2. APRM		
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. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 2×10^5 cps	≤ 4×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 2 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	≤ 44 gallons	≤ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
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.
**For initial loading and startup the count rate may be > 0.5 cps.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.9 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.9-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.9-1:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside an inaccessible zone, then inspect the area surrounding the inaccessible zone at least once per hour.

b. Restore the minimum number of instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.9.1 Each of the above required fire detection instruments which are accessible during unit operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during unit operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.7.9.2 The supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION			INSTRUMENTS OPERABLE					
FIRE ZONE	ROOM OR AREA	ELEV.	HEAT TOTAL	HEAT MIN.	IONIZATION TOTAL	IONIZATION MIN.	PHOTO-ELECTRIC TOTAL	PHOTO-ELECTRIC MIN.
a. Control Building								
0-24D	Lower Relay Room	698'-1"	4	2	4	2	NA	NA
0-24D	PGCC	698'-1"	54	27	30	15	NA	NA
0-25A	Lower Cable Spreading Rm.	714'-0"	26	13	6	3	NA	NA
0-25E	Lower Cable Spreading Rm.	714'-0"	20	10	6	3	NA	NA
0-26H	Control Rm. (Under Flr. Unit 1)*	729'-1"	NA	NA	18	9	NA	NA
0-26H	Control Room (Under Flr. Unit 2)*	729'-1"	NA	NA	15	7	NA	NA
0-26H	Control Room	729'-1"	NA	NA	10	5	NA	NA
0-26H	Control Rm. (Above Clg)*	729'-1"	NA	NA	9	4	NA	NA
0-27C	Upper Cable Spreading Rm.	753'-0"	29	14	8	4	NA	NA
0-27B	Upper Cable Spreading Rm.	753'-0"	28	14	5	2	NA	NA
0-27E	Upper Relay Room	754'-1"	4	2	2	1	NA	NA
0-27E	PGCC	754'-1"	55	27	30	15	NA	NA
0-28K	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28L	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28M	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28N	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28I	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28J	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28G	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28F	Battery Room	771'0"	NA	NA	1	1	NA	NA
0-28E	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28C	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28D	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28T	Battery Room	771'-0"	NA	NA	1	1	NA	NA

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION			INSTRUMENTS OPERABLE					
			HEAT		IONIZATION		PHOTO-ELECTRIC	
FIRE ZONE	ROOM OR AREA	ELEV.	TOTAL	MIN.	TOTAL	MIN.	TOTAL	MIN.
b. Reactor Building								
1-1B	Core Spray Pump Room	645'-0"	NA	NA	5	2	NA	NA
1-1A	Core Spray Pump Room	645'-0"	NA	NA	7	3	NA	NA
1-1E	RHR Pump Room	645'-0"	NA	NA	NA	NA	13	7
1-1F	RHR Pump Room	645'-0"	NA	NA	NA	NA	15	8
1-1D	RCIC Pump Room	645'-0"	2	1	NA	NA	5	2
1-1C	HPCI Pump Room	645'-0"	2	1	NA	NA	7	3
1-1G	Sump Room	645'-0"	NA	NA	2	1	NA	NA
1-2D	Remote Shutdown Panel Rm.	670'-0"	NA	NA	2	1	NA	NA
1-4C	Switchgear Room	719'-0"	NA	NA	2	1	NA	NA
1-4D	Switchgear Room	719'-0"	NA	NA	2	1	NA	NA
1-4A	Containment Access Area	719'-0"	NA	NA	27	13	4	2
1-5F	Load Center Room	749'-1"	NA	NA	2	1	NA	NA
1-5G	Load Center Room	749'-1"	NA	NA	2	1	NA	NA
1-2A	Access Area	670'-0"	NA	NA	7	3	NA	NA
1-3A	Access Area	683'-0"	NA	NA	7	3	NA	NA
1-3B	Access Area	683'-0"	NA	NA	12	6	NA	NA
1-3C	Access Area	683'-0"	NA	NA	NA	NA	13	6
1-4B	Pipe Penetration Room	719'-1"	NA	NA	1	1	NA	NA
1-4G	Main Steam Piping	719'-1"	NA	NA	NA	NA	6	3
1-5B	Valve Access Area	761'-10"	NA	NA	NA	NA	2	1
1-5D	RWCU Pumps & Heat Exchangers	749'-1"	NA	NA	NA	NA	12	6
1-5E	Penetration Room	749'-1"	NA	NA	NA	NA	2	1
1-6A	Access Area	779'-1"	NA	NA	9	4	NA	NA
1-6I	Fuel Pool Holding Pump Room	779'-1"	NA	NA	NA	NA	2	1

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION			INSTRUMENTS OPERABLE					
			HEAT		IONIZATION		PHOTO-ELECTRIC	
FIRE ZONE	ROOM OR AREA	ELEV.	TOTAL	MIN.	TOTAL	MIN.	TOTAL	MIN.
Reactor Building (Continued								
1-6D	H&V Equipment Room	779'-1"	NA	NA	12	6	NA	NA
1-6E	Recirculation Fans Area	779'-1"	NA	NA	2	1	NA	NA
0-6G	Surge Tank Vault	779'-4"	NA	NA	2	1	NA	NA
1-7A	H&V Fan and Filter Rooms	799'-1"	24	12	15	7	NA	NA
0-8A	Refueling Floor	818'-1"	NA	NA	NA	NA	30	15
c. ESSW Pumphouse								
0-51	Pump Room	685'-6"	NA	NA	6	3	NA	NA
0-52	Pump Room	685'-6"	NA	NA	6	3	NA	NA
					INFRA-RED (FLAME)			
					TOTAL	MIN.		
d. Diesel Generator Building								
0-41A	Diesel Generator Rooms	660'-0" 677'-0"	22	11	2	1	15	7
0-41C	Diesel Generator Rooms	660'-0" 677'-0"	22	11	2	1	15	7
0-41B	Diesel Generator Rooms	660'-0" 677'-0"	23	11	2	1	15	7
0-41D	Diesel Generator Rooms	660'-0" 677'-0"	22	11	2	1	15	7

*Not accessible.

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

SAFETY EVALUATION
AMENDMENT NO. 17 TO NPF-14
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1
DOCKET NO. 50-387

Introduction

The licensee in letters dated February 3, 1983 and March 11, 1983, proposed changes to the Technical Specifications of the operating license for Susquehanna Steam Electric Station, Unit 1 which are as follows:

a) Changes to Technical Specification 3.2.2 and Tables 2.2.1-1 and 3.3.6-2 to allow rated power operation at any core flow rate between 87% and 100% of the rated flow as a result of extension of the load line limits. The purpose of this change is to improve operating flexibility during power ascension.

b) Changes to Technical Specification Table 3.3.7.9-1 to correct administrative errors, decrease the number of detectors in two fire zones, and add fire zones that warrant inclusion in the Technical Specifications, reflecting as-built conditions.

Evaluation

a) Power/Flow Map Expansion

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) safety limit minimum critical power ratio, (2) operating limit MCPR, (3) thermal hydraulic stability, and (4) changes to Table 2.2.1-1, Section 3.2.2 and Table 3.3.6-2 of the Technical Specifications.

The licensee submitted the analysis report (Ref. 1) to support the proposed Technical Specification change. Discussion of the review concerning the thermal-hydraulic design for extended load line operation for Cycle 1 follows:

Safety Limit MCPR

The safety limit MCPR has been imposed to assure that 99.9% of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in reference 2, the safety limit MCPR is 1.06 for the fresh fuel. The safety limit of 1.06 is used for the Susquehanna-1 Cycle 1 operation.

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Operating Limit MCPR (OLMCPR)

The maximum value of operating limit MCPR, as indicated in the Technical Specifications, is 1.30. This is calculated by using the ODYN methods for the most limiting transient, generator load rejection without bypass event, at 105% power and 100% flow condition (licensing basis) for Cycle 1 operation.

The licensee submitted a revised power/flow map as shown in Figure 1-1 of reference 1. The proposed extension of the power/flow map is to allow ascension along the 108% APRM rod block line to 100% power at 87% flow and allow rated power operation at any core flow rate between 87% and 100%.

The ODYN results in reference 1 indicate the calculated Δ CPRs for the most limiting transient, generator load rejection without bypass event, are 0.19 and 0.18 for the power/flow map at 105/100 and 100/87 points respectively. The calculated results also show that the transients originated from the reduced flow condition result in lower values of Δ CPR due to an improved scram effect. Therefore, it concludes in reference 1 that (1) the ODYN results for the power/flow at 100/87 point are bounded by the licensing basis results which are based on the power/flow map at 105/100 point, and (2) the OLMCPR's specified in the Technical Specifications are applicable to the rated power operation at any core flow rate between 87% and 100%. We have reviewed the operating limit MCPR results (Ref. 3) and extended load line limit analysis (Ref. 1) discussed above. The NRC staff finds that the ODYN methods were used and results have shown an acceptable margin of safety from conditions which could lead to fuel damage during any anticipated operational transient.

Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Ref. 1) show that the core has the smallest stability margin for the power/flow map at the point where the extrapolated rod block line intercepts the natural circulation line and the corresponding maximum decay ratio is 0.96. Since (1) the calculated maximum stability ratio is less than that of some of the operating plants (for example, Peach Bottom Units 2 and 3 have decay ratio of 0.98) and (2) there will be added margin to the core stability because the Technical Specifications prohibit the natural circulation operation, the NRC staff concludes that the thermal-hydraulic stability results are acceptable for extension of the load line limits for Cycle 1 operation.

Change to the Technical Specifications

Tables 2.2.1-1, 3.3.6-2 and Section 3.2.2 of the proposed Technical Specifications have been modified to include the extended operating power/flow map given in Figure 1.1 of reference 1. The modifications include the change of the APRM flow biased trip set points from 0.66 W + 51% to o.58 W + 59%, where W is the loop recirculation flow as a percentage of the loop recirculation flow which produced a rated core flow of 100 million lbs/hr.

Staff Conclusion

The NRC staff finds that approved methods have been used, and that the results of the extended load line limit analysis support the current operating limit MCPRs, which avoid violation of the safety limit MCPR for design transients.

The NRC staff concludes that this core design will not adversely affect the capability to operate the Susquehanna Steam Electric Station, Unit 1 safely during Cycle 1 and that proposed changes to Tables 2.2.1-1, 3.3.6-2 and Section 3.2.2 of the Technical Specifications discussed above are acceptable.

References

1. NEDO-22128, General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Susquehanna Steam Electric Station Unit 1, May 1982.

2. NEDO-24011-A-4, General Electric Boiling Water Reactor Generic Reload Fuel Applications dated January 1982.

3. Susquehanna Steam Electric Station, Units 1 and 2, FSAR, Docket No. 50-387 and 50-388.

b) Fire Zones/Detectors

The licensee has performed a field inspection of detector locations in conjunction with designated fire zones and room numbers. Due to administrative errors, the licensee was unable to correlate detector locations with the appropriate fire zones and room numbers. The proposed change allows designation of fire zones to permit correct identification, and deletes room numbers from the table because they are not needed.

The proposed change also decreases the number of detectors in two fire zones:

Zone 1-48 (piping penetration room) - This zone is 600 square feet and has one ionization detector. Based on NFPA standard 72 E which recommends 1 detector per 900 square feet, one detector is sufficient.

Zone 1-7B (recirculation fan room) - This zone is being deleted from the table because it was evaluated and found not to present a fire exposure hazard to safety-related systems necessary to accomplish or maintain a safe shutdown condition.

The NRC staff agrees with the licensee that Zone 1-48 (pipe penetration room) meets our guidelines with one detector, and that Zone 1-7B (recirculation fan room) can be deleted from the table as the area does not present fire exposure to safe shutdown equipment.

Three additional changes have been proposed:

Zones 0-24D & 0-27E - The Power Generation Control Complex (PGCC) areas associated with the Unit 1 Upper and Lower Relay Rooms are proposed to be added since they protect essential equipment and the Halon systems associated with them are presently in the Technical Specifications.

Zones 0-25A, 0-25E - Due to a typographical error, the instrument totals for these two zones were reversed.

Zone 1-6I - The Fuel Pool Holding Pump Room was formed due to the subdivision of Zone 1-6A. Although there is no redundant safety-related equipment in Zone 1-6I, this zone contains cables necessary to accomplish and maintain a safe shutdown condition, and therefore this zone needs to be included in the Technical Specifications.

The NRC staff agrees with the licensee that these additions and changes should be made.

Conclusion

We find that the proposed changes to Table 3.3.7.9-1 are either administrative in nature, or where technical changes have been proposed to reduce the number of detectors in an area, the level of protection provided meets our guidelines. Based on the above considerations, the NRC staff concludes the requested Technical Specification revision is acceptable.

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and enviromental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 2, 1983

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DATED: November 2, 1983

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