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

APR 1 1986

Docket No. 50-388

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. 25 to Facility Operating License No. NPF-22, Susquehanna Steam Electric Station, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. The amendment is in response to your letter dated December 19, 1985.

This amendment revises the Technical Specifications to change the setpoint for main steam isolation valve (MSIV) closure on reactor vessel water low level from Level 2 to Level 1.

A copy of the related safety evaluation supporting Amendment No. 25 to Facility Operating License No. NPF-22 is enclosed.

Sincerely,

Elinor G. Adensam

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

Enclosures:
1. Amendment No. 25 to NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

DESIGNATED ORIGINAL

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-22

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, dated December 19, 1985, 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-22 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. 25 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 2 first refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: APR 01 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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 page is also provided to maintain document completeness.

REMOVE	INSERT
3/4 3-11	3/4 3-11 (overleaf)
3/4 3-12	3/4 3-12
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18 (overleaf)
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22 (overleaf)
3/4 3-23	3/4 3-23 (overleaf)
3/4 3-24	3/4 3-24
3/4 6-19	3/4 6-19
3/4 6-20	3/4 6-20 (overleaf)
B3/4 3-7	B3/4 3-7 (overleaf)
B3/4 3-8	B3/4 3-8

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	A	2	1, 2, 3	20
2) Low Low, Level 2	B	2	1, 2, 3	20
3) Low Low Low, Level 1	X	2	1, 2, 3	20
b. Drywell Pressure - High	Y,Z	2	1, 2, 3	20
c. Manual Initiation	NA	1	1, 2, 3	24
d. SGTS Exhaust Radiation - High	R	1	1, 2, 3, 4***, 5***	20
e. Main Steam Line Radiation - High	C	2	1, 2, 3	20
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	**	2	1, 2, 3 and *	25
b. Drywell Pressure - High	**	2	1, 2, 3	25
c. Refuel Floor High Exhaust Duct Radiation - High	**	2	*	25
d. Railroad Access Shaft Exhaust Duct Radiation - High	**	2	*	25
e. Refuel Floor Wall Exhaust Duct Radiation - High	**	2	*	25
f. Manual Initiation	NA	1	1, 2, 3 and *	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	X	2	1, 2, 3	21
b. Main Steam Line Radiation - High	C	2	1, 2, 3	21
c. Main Steam Line Pressure - Low	P	2	1	22
d. Main Steam Line Flow - High	D	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	UA	2	1, 2, 3	21
f. Reactor Building Main Steam Line Tunnel Temperature - High	E	2	1, 2, 3	21
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	E	2	1, 2, 3	21
h. Manual Initiation	NA	1	1, 2, 3	24
i. Turbine Building Main Steam Line Tunnel Temperature - High	E	2	1, 2, 3	21
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	J	1	1, 2, 3	23
b. RWCU Area Temperature - High	W	3	1, 2, 3	23
c. RWCU Area Ventilation Δ Temperature - High	W	3	1, 2, 3	23
d. SLCS Initiation	I	2	1, 2, 3	23
e. Reactor Vessel Water Level - Low Low, Level 2	B	2	1, 2, 3	23
f. RWCU Flow - High	J	1	1, 2, 3	23
g. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥13.0 inches*	≥ 11.5 inches
2) Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Manual Initiation	NA	NA
d. SGTS Exhaust Radiation - High	≤ 23.0 mR/hr	≤ 31.0 mR/hr
e. Main Steam Line Radiation - High	≤ 3 X full power background	≤ 3.6 X full power background
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Refuel Floor High Exhaust Duct Radiation - High	≤ 2.5 mR/hr**	≤ 4.0 mR/hr**
d. Railroad Access Shaft Exhaust Duct Radiation - High	≤ 2.5 mR/hr**	≤ 4.0 mR/hr**
e. Refuel Floor Wall Exhaust Duct Radiation - High	≤ 2.5 mR/hr**	≤ 4.0 mR/hr**
f. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Main Steam Line Radiation - High	≤ 3 X full power background	≤ 3.6 X full power background
c. Main Steam Line Pressure - Low	≥ 861 psig	≥ 841 psig
d. Main Steam Line Flow - High	≤ 107 psid	≤ 110 psid

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
MAIN STEAM LINE ISOLATION (Continued)		
e. Condenser Vacuum - Low	≥ 9.0 inches Hg vacuum	≥ 8.8 inches Hg vacuum
f. Reactor Building Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	≤ 99°F	≤ 108°F
h. Manual Initiation	NA	NA
i. Turbine Building Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. RWCU Δ Flow - High	≤ 60 gpm	≤ 80 gpm
b. RWCU Area Temperature - High	≤ 147°F or 118.3°F#	≤ 154°F or 125.3°F#
c. RWCU/Area Ventilation Δ Temperature - High	≤ 69°F or 35.3°F#	≤ 78°F or 44.3°F#
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
f. RWCU Flow - High	≤ 426 gpm	≤ 436 gpm
g. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Δ Pressure - High	≤ 153" H_2O**	≤ 165" H_2O**
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	$\leq 10^{(a)}$
2) Low Low, Level 2	$\leq 1.0^{*}/\leq 10^{(a)**}$
3) Low Low Low, Level 1	$\leq 10^{(a)}$
b. Drywell Pressure - High	$\leq 10^{(a)}$
c. Manual Initiation	NA
d. SGTS Exhaust Radiation - High$^{(b)}$	$\leq 10^{(a)}$
e. Main Steam Line Radiation - High$^{(b)}$	$\leq 10^{(a)}$
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level-Low Low, Level 2	$\leq 10^{(a)}$
b. Drywell Pressure - High	$\leq 10^{(a)}$
c. Refuel Floor High Exhaust Duct Radiation - High$^{(b)}$	$\leq 10^{(a)}$
d. Railroad Access Shaft Exhaust Duct Radiation - High$^{(b)}$	$\leq 10^{(a)}$
e. Refuel Floor Wall Exhaust Duct Radiation -High$^{(b)}$	$\leq 10^{(a)}$
f. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level- Low Low Low, Level 1	$\leq 10^{(a)}$
b. Main Steam Line Radiation - High$^{(b)}$	$\leq 1.0^{*}/\leq 10^{(a)**}$
c. Main Steam Line Pressure - Low	$\leq 1.0^{*}/\leq 10^{(a)**}$
d. Main Steam Line Flow-High	$\leq 0.5^{*}/\leq 10^{(a)**}$
e. Condenser Vacuum - Low	NA
f. Reactor Building Main Steam Line Tunnel Temperature - High	NA
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	NA
h. Manual Initiation	NA
i. Turbine Building Main Steam Line Tunnel Temperature - High	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	$\leq 10^{(a)\#\#}$
b. RWCU Area Temperature - High	NA
c. RWCU Area Ventilation Temperature ΔT - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low Low, Level 2	$\leq 10^{(a)}$
f. RWCU Flow - High	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure - High	$\leq 10^{(a)\#\#\#}$
b. RCIC Steam Supply Pressure - Low	$\leq 10^{(a)}$
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Temperature - High	NA
e. RCIC Equipment Room Δ Temperature - High	NA
f. RCIC Pipe Routing Area Temperature - High	NA
g. RCIC Pipe Routing Area Δ Temperature - High	NA
h. RCIC Emergency Area Cooler Temperature - High	NA
i. Manual Initiation	NA
j. Drywell Pressure - High	$\leq 10^{(a)}$

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Flow - High	$\leq 10^{(a)}$####
b. HPCI Steam Supply Pressure - Low	$\leq 10^{(a)}$
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Equipment Room Temperature - High	NA
e. HPCI Equipment Room Δ Temperature - High	NA
f. HPCI Emergency Area Cooler Temperature - High	NA
g. HPCI Pipe Routing Area Temperature - High	NA
h. HPCI Pipe Routing Area Δ Temperature - High	NA
i. Manual Initiation	NA
j. Drywell Pressure - High	$\leq 10^{(a)}$
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION	
a. Reactor Vessel Water Level - Low, Level 3	$\leq 10^{(a)}$
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
c. RHR Equipment Area Δ Temperature - High	NA
d. RHR Equipment Area Temperature - High	NA
e. RHR Flow - High	NA
f. Manual Initiation	NA
g. Drywell Pressure - High	$\leq 10^{(a)}$

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed for MSIV Valves.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

##With time delay of 45 seconds.

###With time delay of 3 seconds.

####With time delay of 3 seconds.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1) Low, Level 3	S	M	R	1, 2, 3
2) Low Low, Level 2	S	M	R	1, 2, 3
3) Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure - High	NA	M	R	1, 2, 3
c. Manual Initiation	NA	R	NA	1, 2, 3
d. SGTS Exhaust Radiation - High	S	M	R	1, 2, 3, 4, 5
e. Main Steam Line Radiation - High	S	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and *
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Refuel Floor High Exhaust Duct Radiation - High	S	M	R	*
d. Railroad Access Shaft Exhaust Duct Radiation - High	S	M	R	*
e. Refuel Floor Wall Exhaust Duct Radiation - High	S	M	R	*
f. Manual Initiation	NA	R	NA	1, 2, 3 and *

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	NA	M	Q	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2**, 3**
f. Reactor Building Main Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	NA	M	Q	1, 2, 3
h. Manual Initiation	NA	R	NA	1, 2, 3
i. Turbine Building Main Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	S	M	R	1, 2, 3
b. RWCU Area Temperature - High	NA	M	Q	1, 2, 3
c. RWCU Area Ventilation Δ Temperature - High	NA	M	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
f. RWCU Flow - High	S	M	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)	ISOLATION SIGNAL(S)[a]
a. Automatic Isolation Valves		
MSIV		
HV-241F022 A,B,C,D	5	X,C,D,E,P,UA
HV-241F028 A,B,C,D	5	X,C,D,E,P,UA
MSL Drain		
HV-241F016	10	X,C,D,E,P,UA
HV-241F019	10	X,C,D,E,P,UA
RCIC Steam Supply		
HV-249F007	20	K,KB
HV-249F008	20	K,KB
HV-249F088	3	K,KB
HPCI Steam Supply		
HV-255F002	50	L,LB
HV-255F003	50	L,LB
HV-255F100	3	L,LB
RHR - Shutdown Cooling Suction		
HV-251F008	52	A,M,UB
HV-251F009	52	A,M,UB
RWCU Suction[b]		
HV-244F001	30	B,J,W
HV-244F004	30	I,B,J,W
RHR - Reactor Vessel Head Spray		
HV-251F022	30	A,M,UB,Z
HV-251F023	20	A,M,UB,Z

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

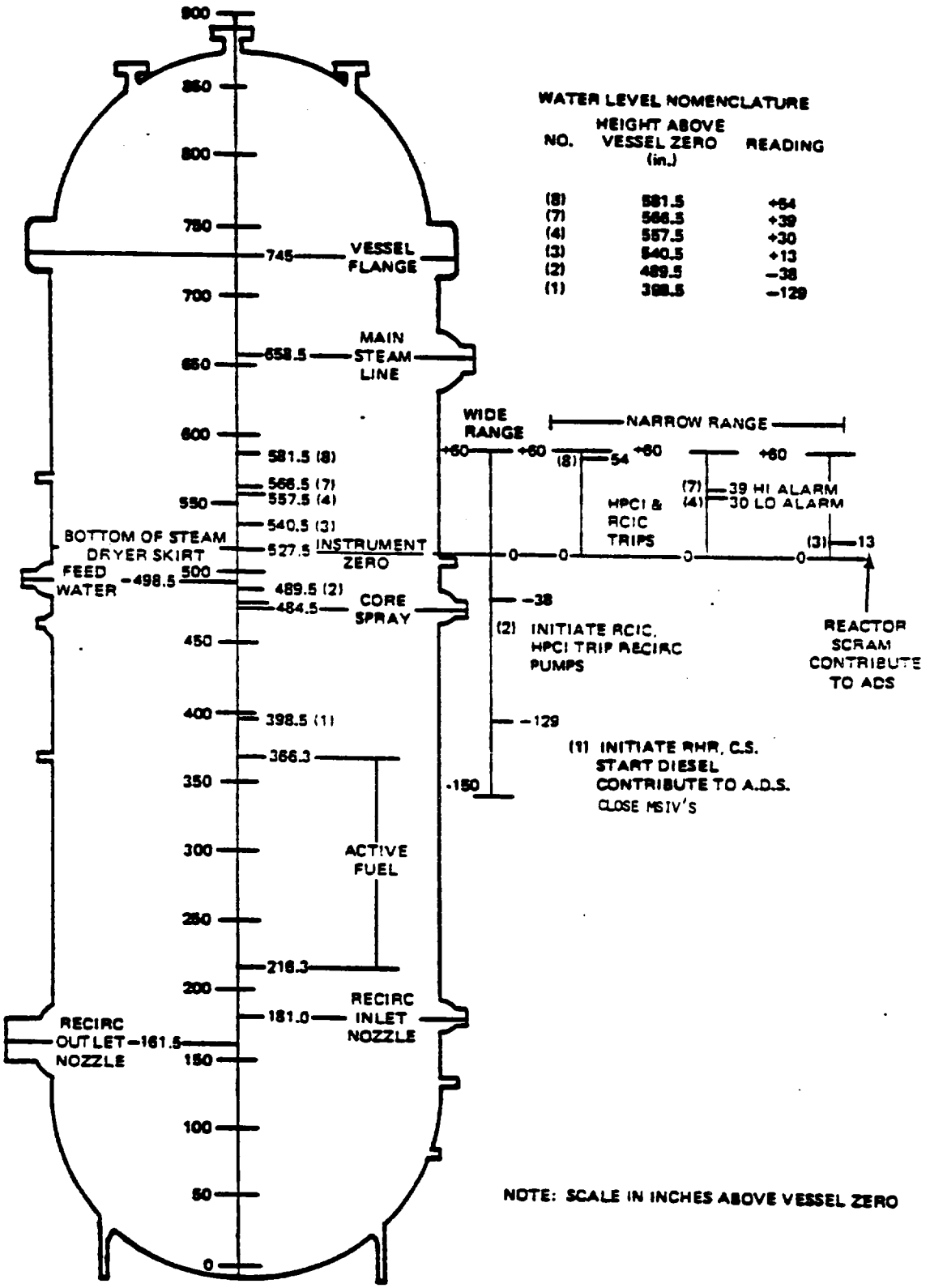
VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)	ISOLATION SIGNAL(S)[a]
Automatic Isolation Valves (Continued)		
Containment Instrument Gas		
HV-22603	20	X,Z
SV-22605	N/A	X,Z
SV-22651	N/A	X,Z
SV-22661	N/A	Y,B
SV-22671	N/A	Y,B
RBCCW		
HV-21313	30	X,Z
HV-21314	30	X,Z
HV-21345	30	X,Z
HV-21346	30	X,Z
Containment Purge		
HV-25703	15	B,Y,R
HV-25704	15	B,Y,R
HV-25705	15	B,Y,R
HV-25711	15	B,Y,R
HV-25713	15	B,Y,R
HV-25714	15	B,Y,R
HV-25721	15	B,Y,R
HV-25722	15	B,Y,R
HV-25723	15	B,Y,R
HV-25724	15	B,Y,R
HV-25725	15	B,Y,R
RHR - Drywell Spray[c]		
HV-251F016 A,B	90	X,Z
RB Chilled Water		
HV-28781 A1,A2,B1,B2	40	X,Z
HV-28782 A1,A2,B1,B2	6	X,Z
HV-28791 A1,A2,B1,B2	15	Y,B
HV-28792 A1,A2,B1,B2	4	Y,B

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of failure of feedwater controller under maximum demand.



BASES FIGURE B 3/4 3-1
REACTOR VESSEL WATER LEVEL

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

The licensee proposed changes to the Technical Specifications of the operating license for Susquehanna Steam Electric Station, Unit 2 to lower the setpoint for main steam isolation valve (MSIV) closure from low water level 2 (Level 2) to low water level 1 (Level 1).

2.0 EVALUATION

The purpose of lowering the setpoint for MSIV closure is to reduce the number of challenges to the safety-relief valves (SRVs). This modification is consistent with staff recommendations for resolving Action Plan Item II.K.3.16 "Reduction of Challenges and Failures of Relief Valves-Feasibility Study and System Modifications." The licensee reviewed the FSAR transient and accident analyses to assess the impact of the setpoint change on the events. Thermal safety limits for those transients involving MSIV closure on low water level are not affected because the limiting thermal conditions are reached and reduced well before water level falls to Level 2. For loss-of-coolant accidents involving loss of offsite power, MSIV closure will occur on the loss of power signal regardless of water level. For a loss-of-coolant accident where offsite power is available, lowering the setpoint will delay closing of the MSIVs. This will result in a more rapid depressurization which leads to earlier injection into the vessel by the low pressure Emergency Core Cooling (ECC) systems.

The proposed modification will eliminate MSIV closures for many transient events where water level falls below Level 2 but is recovered prior to reaching Level 1. This will reduce challenges to the safety-relief valves. As discussed above, previously analyzed transients and accidents are not adversely impacted by the change.

For the above reasons, the proposed technical specification change is 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. 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 or 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 6829) on February 26, 1986, and consulted with the State of Pennsylvania. No public comments were received, and the State of Pennsylvania did not have any comments.

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.

Principal Contributors: George Thomas, Reactor Systems Branch, DBL
Mari-Josette Campagnone, Project Directorate No. 3, DBL

Dated: APR [illegible]