December 5, 19

Docket No. 50-388

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

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Dear Mr. Keiser:

SUBJECT: A ONE-TIME RELIEF FROM THE REQUIREMENTS OF SECTION 3.6.3 OF THE TECHNICAL SPECIFICATIONS FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 (TAC NO. 77900)

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment has been prepared and issued on an emergency basis in response to your letter dated October 30, 1990.

On October 24, 1990, the staff granted a Temporary Waiver of Compliance which was immediately effective. This action was verbally authorized and confirmed by a followup letter on October 30, 1990.

On November 9, 1990, during a telephone conversation with Pennsylvania Power and Light Company, the staff proposed a revision to the Technical Specifications change which would clarify that change. The Pennsylvania Power and Light Company agreed with the staff proposal. Accordingly, the revised Technical Specification change is incorporated.

This amendment changes the Technical Specifications to permit relief from the requirements of action "a" of Section 3.6.3 related to a reactor water cleanup isolation valve. The relief is granted on a one time only basis.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/S/

Edward G. Greenman, Acting
Assistant Director for Region I Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 70 to License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

[SU AMDT. TAC NO. 77900]
*Previously concurred

PDI-2/LA	*EMEB/C	*SPLB/C	PDI-2/PM	PDI-2/D	OGC
MO'Brien	LMarsh	CMcCracken	MThadani:tlc	WButler	
/1/90	11/23/90	11/09/90	12/3/90	12/3/90	11/26/90
		w/comments			
REGION I	Acting ADRI		*SRXB		
Telecon* DIV. DIV	EGreenman		RJones		
10/30/90	12/5/90		11/23/90		

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

December 5, 1990

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Senior Vice President-Nuclear
Pennsylvania Power and Light Company
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Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 70 to
License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

cc:

Jay Silberg, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street N.W.
Washington, D.C. 20037

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

Mr. J. M. Kenny
Licensing Group Supervisor
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. Scott Barber
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 35
Berwick, Pennsylvania 18603-0035

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

Mr. Jesse C. Tilton, III
Allegheny Elec. Cooperative, Inc.
212 Locust Street
P.O. Box 1266
Harrisburg, Pennsylvania 17108-1266

Mr. S. B. Ungerer
Joint Generation
Projects Department
Atlantic Electric
P.O. Box 1500
1199 Black Horse Pike
Pleasantville, New Jersey 08232

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Harold G. Stanley
Superintendent of Plant
Susquehanna Steam Electric Station
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. Herbert D. Woodeshick
Special Office of the President
Pennsylvania Power and Light Company
1009 Fowles Avenue
Berwick, Pennsylvania 18603

Mr. Robert G. Byram
Vice President-Nuclear Operations
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

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. 70
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 October 30, 1990 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. 70 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 license amendment became effective on October 30, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Edward G. Greenman, Acting
Assistant Director for Region I Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1990

Concurred on 10/30/90 Div. Dir.

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D	Region I	Acting ADRI
MO'Brien	for MThadani:tlc		WButler		EGreenman
11/8/90	11/8/90	11/28/90	12/3/90 WB	/ /90	12/5/90

3. This license amendment became effective on October 30, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Edward G. Greenman

Edward G. Greenman, Acting
Assistant Director for Region I Reactors
Division of Reactors Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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 page is identified by Amendment number and contains vertical lines indicating the area of change. The overleaf page is provided to maintain document completeness.*

REMOVE	INSERT
3/4 6-19	3/4 6-19
3/4 6-20	3/4 6-20*

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

*For the RWCU HV-244 F001 valve, the ACTION statement 3.6.3.a need not be followed for the period beginning October 30, 1990 until an outage of sufficient duration to revise the current torque switch setting occurs.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

1.0 INTRODUCTION

By letter dated October 30, 1990, Pennsylvania Power & Light Company requested an amendment, on an emergency basis, to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. The proposed amendment would grant relief from the requirements of action "a" of Technical Specification Section 3.6.3 related to a reactor water cleanup system isolation valve.

The licensee has identified that the "closing" torque switch on the Unit 2 RWCU HV-244F001 Inboard Containment Isolation Valve was incorrectly set at 1½ turns rather than the specified 1 3/4 turns. As a result of this lower torque setting, the valve may not fully close when it experiences the maximum differential pressure of 1000 psid. Under these conditions, the valve would be limited to a 97% closure position. Contrary to the requirements of the Technical Specifications, further manual action would be required to fully close the valve. As a result, the valve was declared inoperable and appropriate actions taken to allow continued operation for an interim period of time.

For pressure differentials of 890 psid or less, the valve will function normally and is expected to fully isolate the penetration. Therefore, there are a limited number of potential break locations that would prevent the valve from complete closure. The only design basis accident or operating scenario that the licensee has determined to be capable of generating these high pressure differentials is a large break in the RWCU suction line outboard of the valve in question. In addition, there is a limiting break size below which the differential pressure would be below the 890 psid value. Therefore, the valve would function normally for these smaller break sizes. However, the licensee has not taken credit for these limited break sizes in assessing the impact of continued operation.

The licensee has requested the above stated temporary relief to allow continued operation until March, 1991 to permit it to correct the torque setting and fully restore the valve to an operable status during an outage of suitable duration.

On October 24, 1990, the staff granted a Temporary Waiver of Compliance which was immediately effective. This action was verbally authorized and confirmed by a followup letter on October 30, 1990.

2.0 EVALUATION

To support the request for a one-time relief from the Technical Specification Section 3.6.3, the licensee evaluated the consequences of an accident with only 97% closure of the valve, from two perspectives: the probability of a line break which would result in this limited closure and the consequences of such an event if it were to occur.

For the period of time between now and March 1991, the licensee has computed the probability of a line rupture in the suction area (6-inch carbon steel line in the penetration room) of the RWCU system coincident with a random failure of the outboard isolation valve to be 4.3×10^{-5}. Although the staff does not agree with the assumptions used in this calculation, the staff agrees that the probability of this occurrence is expected to be low.

A leak detection system is also installed to detect leakage from the RWCU System. The leak detection system provides for automatic isolation on high room temperature or high delta-T between ventilation inlet and exhaust. The set points are low enough to allow timely detection of a 5 gpm leak. In addition, high flow rate, high change in flow rate, or a low reactor water level will initiate an automatic isolation of the RWCU System. Area radiation monitors provide alarms on increasing radiation in the RWCU rooms which will alert operators to the possibility of leaks developing in the RWCU System.

The licensee has also reviewed existing procedures to determine if the consequences of the limiting break can be minimized. As a result of this review, the licensee has taken the the following actions:

1. Made provisions for bypassing the isolation signal per revised emergency operating procedures and removed the open seal-in feature in order to reset the torque switch without moving the valve to the full open position. This will allow remote-manual full closure of the valve after the differential pressure has been reduced below 890 psid.

2. Developed procedures for remote-manual valve closure as described in 1. above.

3. Provided operator training to make operators aware of degraded conditions and the special procedures in place to assure manual valve closure.

Based on these added procedures, the licensee has indicated that complete closure of the valve under the most limiting conditions can be accomplished within 19 minutes of the initiation of the event. Using this time for closure, the licensee has performed a dose-consequence analysis which demonstrated that the main steam line rupture remains as the most limiting pipe rupture.

The licensee has indicated that the motor and actuator can accomodate the loads during the closure stroke when the torque switch is bypassed, but has not discussed the availability of thermal overload protection for the motor. The capability of the motor actuator as well as the availability of the thermal overload will be addressed by the licensee in its response to Supplement 3 of Generic Letter (GL) 89-10.

To assure protection of the motor, actuator, and valve, the licensee has decided not to make temporary modifications to the electrical circuits and bypass the torque switch. The staff agrees with the licensee's decision not to bypass the torque switch and thus protect the valve against possible damage.

The staff is concerned about the licensee's procedures for torque switch setting, which have resulted in two incorrect torque switch settings during the past three years. The licensee should review and, if needed, improve its methods for torque switch settings, and avoid any future possibilities that torque switches may be incorrectly set.

Based on the above, the staff finds that the licensee can continue power operation with the degraded torque switch setting of the reactor water cleanup system inboard isolation valve. The staff's conclusion is based on the consideration that the probability of a pipe rupture which would cause the valve to close slightly off the seat is very low, and the consequences of such an accident would be bounded by the radiological doses calculated for the design basis rupture of the main steam line, and the dose guidelines of 10 CFR Part 100 will be met, should a pipe rupture downstream of the outboard isolation valve of the reactor water cleanup system occur.

In order to continue power operation, the licensee requested the following footnote to be added to the Table 3.6.3-1.

> "The RWCU HV-244 F001 Valve may be considered OPERABLE with its current minimum torque switch setting for the period beginning October 30, 1990 until an outage of sufficient duration to revise the setting occurs".

Although the staff has found that the licensee can continue power operation with the degraded torque switch setting of the valve HV-244 F001, the staff finds that the valve HV-244 F001 must not be considered OPERABLE. On November 9, 1990, the staff informed the licensee that the staff would find the Technical Specification change acceptable if it modified the footnote to read as follows:

> "For the RWCU HV-244 F001 valve, the ACTION statement 3.6.3.a need not be followed for the period beginning October 30, 1990 until an outage of sufficient duration to revise the current torque switch setting occurs".

The licensee verbally agreed to the staff proposal to change the footnote. Therefore, the revised footnote may be incorporated in the Technical Specifications.

3.0 EMERGENCY BASIS

The licensee has provided the following basis for the existence of an emergency required by 10 CFR 50.91.

> [Guidance is provided under] 10 CFR 50.91 on what information the NRC requires in support of an application for an emergency change.
>
> First, it requires the applicant to justify that an emergency exists, i.e. "... failure to act in a timely way would result in derating or shutdown of a nuclear power plant ...". Unit 2 is currently operating at full power. Since the affected valve is inside containment, the torque switch cannot be reset without a unit shutdown and a containment entry. Isolation of the affected RWCU penetration as required by Technical Specification 3.6.3 Action "a" would result in a unit shutdown due to exceeding Chemistry limits on reactor water conductivity (reference Technical Specifications 3.4.4) in approximately 72 hours.
>
> Secondly, 10 CFR 50.91 requires the licensee to "... explain why this emergency situation occurred and why it could not avoid this situation ...". The improper setting was discovered during a records search pursuant to a request from the BWR Owners' Group. After some evaluation, discussion and clarification of guidance, the valve was declared inoperable and a temporary waiver of compliance was applied for. Application in advance of this situation was impossible since the problem was discovered during full power operation. Based on the time necessary to evaluate the problem and to prepare and review this proposal internally, we believe that this application has been submitted in a timely fashion.

The staff believes that the licensee made a timely application after determining on October 24, 1990 that the torque setting was incorrect. The staff agrees with the above basis for an emergency provided by the licensee. Therefore, the staff finds that the proposed Technical Specification changes should be processed on an emergency basis.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The staff has reviewed the licensee's request and concurs with the following basis and conclusions provided by the licensee in its October 30, 1990 submittal.

The proposed change does not:

I. Involve a significant increase in the probability or consequences of an accident previously evaluated.

As delineated above, the only event of concern given the current condition of the F001 valve is a RWCU line break outside containment. The probability analysis indicates that this is an unlikely event over the slightly more than four months that the proposed change could be in effect. The radiological consequences were conservatively determined to be the same as the bounding FSAR analysis of a Main Steamline break outside containment, but the increase is insignificant given the implicit error in such calculations and since both numbers are such a small fraction of 10 CFR Part 100 limits.

II. Create the possibility of a new or different kind of accident from any accident previously evaluated.

No hardware or procedural changes are proposed that would create a new event requiring evaluation.

III. Involve a significant reduction in a margin of safety.

As delineated above, the event of concern has been shown to be unlikely, and its consequences, both in terms of affect on safety-related equipment and radiologically, have been shown to be acceptable from a regulatory standpoint. Therefore, the margin of safety is not significantly reduced due to operation with the improper torque switch setting until the Unit 2 Fourth Refueling and Inspection Outage.

5.0 STATE CONSULTATION

The Commonwealth of Pennsylvania was contacted on November 5, 1990 and had no Comments on the Amendment.

6.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change to a requirement with respect to 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 made a final no significant hazards finding with respect to this amendment. 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.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) create the possibility of a new or different kind of accident from any previously evaluated or (c) significantly reduce a safety margin and, therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: J. Kudrick, T. Scarborough, and M. Thadani

Dated: December 5, 1990