

November 22, 1989

Docket No. 50-388

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

Dear Mr. Keiser:

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SUBJECT: TECHNICAL SPECIFICATION CHANGES TO ADDRESS THERMAL HYDRAULIC INSTABILITY (TAC NO. 73586)

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

The Commission has issued the enclosed Amendment No. 60 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your letter dated June 23, 1989.

This amendment revises the Technical Specifications to reflect resolutions arrived at in response to NRC Bulletin 88-07 and its Supplement No. 1, related to, "Power Oscillations in Boiling Water Reactors".

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/s/

Mohan C. Thadani, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 60 to License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

[KEISER LET]

MO'Brien
PDI-2/PA
MO'Brien
11/22/89

MThadani
PDI-2/PM
MThadani:tr
10/18/89

WButler
PDI-2/D
WButler
11/22/89

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OGC
CBarth concurred on amendment
10/15/89

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PDR ADOCK 05000388
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 22, 1989

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Senior Vice President-Nuclear
Pennsylvania Power and Light Company
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Sincerely,

A handwritten signature in cursive script, reading "Mohan C. Thadani".

Mohan C. Thadani, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 60 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

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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. 60
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 June 23, 1989 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. 60 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 is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 22, 1989

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PDI-2/D
WButler
11/22/89

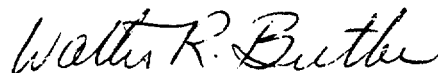
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MThadani:tr
10/16/89

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O&C
10/15/89

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PDI-2/D
WButler
11/22/89

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **November 22, 1989**

ATTACHMENT TO LICENSE AMENDMENT NO. 60

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 overleaf pages are provided to maintain document completeness.*

REMOVE

INSERT

xxi
xxii

xxi*
xxii

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3/4 4-1a

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3/4 4-1b
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B 3/4 4-1
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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS - TWO LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.1 Two reactor coolant system recirculation loops shall be in operation with the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1.

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

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With:

- a) No reactor coolant system recirculation loops in operation, or
- b) Region I of Figure 3.4.1.1.1-1 entered, or
- c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:
 - 1) Two or more APRM readings oscillating with at least one oscillating greater than or equal to 10% of RATED THERMAL POWER peak-to-peak, or
 - 2) Two or more LPRM upscale alarms activating and deactivating with a 1 to 5 second period, or
 - 3) Observation of a sustained LPRM oscillation of greater than 10 w/cm² peak-to-peak with a 1 to 5 second period, or
- d) Region II of Figure 3.4.1.1.1-1 entered and less than 50% of the required LPRM upscale alarms OPERABLE,

immediately place the reactor mode switch in the shutdown position.

*See Special Test Exception 3.10.4.

#See Specification 3.4.1.1.2 for single loop operation requirements.

+The LPRM upscale alarms are not required to be OPERABLE to meet this specification in OPERATIONAL CONDITION 2.

REACTOR COOLANT SYSTEM

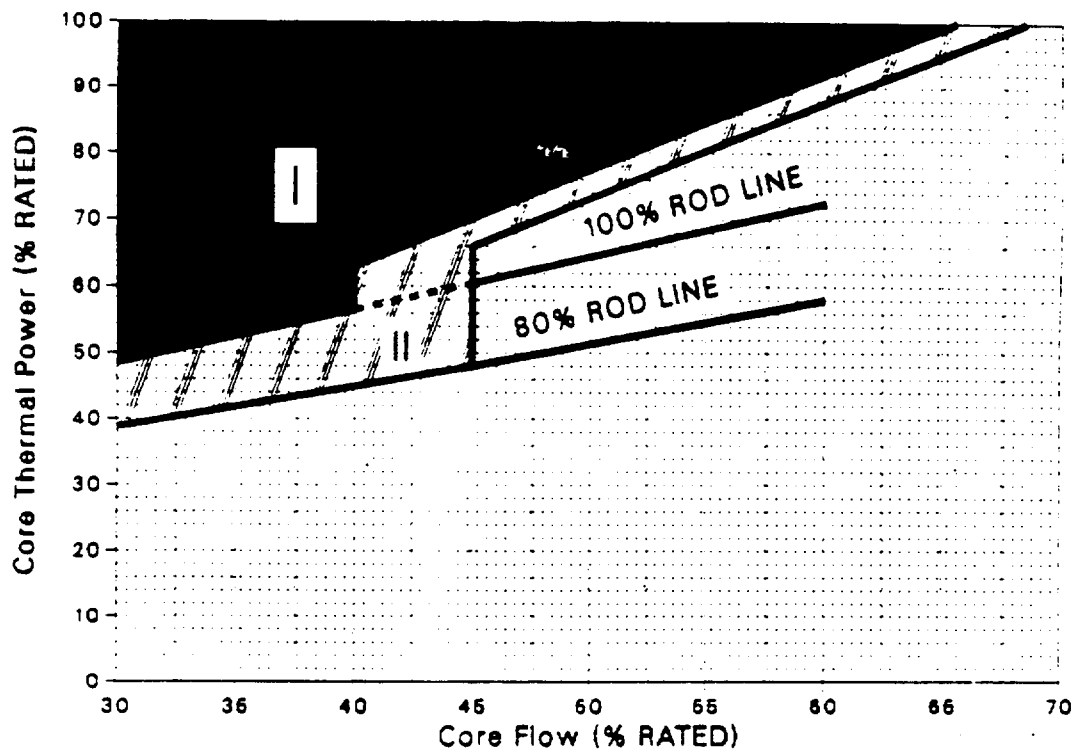
SURVEILLANCE REQUIREMENTS

ACTION: (Continued)

2. If Region II of Figure 3.4.1.1.1-1 is entered and greater than or equal to 50% of the required LPRM upscale alarms OPERABLE, immediately exit the region by:
 - a) inserting a predetermined set of high worth control rods, or
 - b) increasing core flow.
 3. With less than 50% of the required LPRM upscale alarms OPERABLE, follow ACTION a.1.d upon entry into Region II of Figure 3.4.1.1.1-1.
 - b. In OPERATIONAL CONDITION 2 with no reactor coolant system recirculation loops in operation, return at least one reactor coolant system recirculation loop to operation, or be in HOT SHUTDOWN within the next 6 hours.
 - c. With any pump discharge valve not OPERABLE remove the associated loop from operation, close the valve and comply with the requirements of Specification 3.4.1.1.2.
 - d. With any pump discharge bypass valve not OPERABLE close the valve and verify closed at least once per 31 days.
- 4.4.1.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.
- 4.4.1.1.1.2 Each pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5 and 105%, respectively, of rated core flow, at least once per 18 months.
- 4.4.1.1.1.3 At least 50% of the required LPRM upscale alarms shall be determined OPERABLE by performance of the following on each LPRM upscale alarm:
- 1) CHANNEL FUNCTIONAL TEST at least once per 92 days, and
 - 2) CHANNEL CALIBRATION at least once per 184 days.

**If not performed within the previous 31 days.

Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS



REACTOR COOLANT SYSTEM

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed \leq 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

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

<u>Trip Setpoint</u>	<u>Allowable Value</u>
$\leq 0.58W + 54\%$	$\leq 0.58W + 57\%$

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

<u>Trip Setpoint</u>	<u>Allowable Value</u>
$S \leq (0.58W + 54\%)T$	$S \leq (0.58W + 57\%)T$
$S_{RB} \leq (0.58W + 45\%)T$	$S_{RB} \leq (0.58W + 48\%)T$

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the largest of the following values:
 - a. the MCPR determined from Figure 3.2.3-1 plus 0.01, and
 - b. the MCPR determined from Figure 3.2.3-2 plus 0.01.
5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

a. RBM - Upscale	<u>Trip Setpoint</u> $\leq 0.66W + 36\%$	<u>Allowable Value</u> $\leq 0.66W + 39\%$
b. APRM-Flow Biased	<u>Trip Setpoint</u> $\leq 0.58W + 45\%$	<u>Allowable Value</u> $\leq 0.58W + 48\%$

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

ACTION:

- a. In OPERATIONAL CONDITION 1:
 1. With
 - a) no reactor coolant system recirculation loops in operation, or
 - b) Region I of Figure 3.4.1.1.1-1 entered, or
 - c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- 1) Two or more APRM readings oscillating with at least one oscillating greater than or equal to 10% of RATED THERMAL POWER peak-to-peak, or
 - 2) Two or more LPRM upscale alarms activating and deactivating with a 1 to 5 second period, or
 - 3) Observation of a sustained LPRM oscillation of greater than 10 w/cm² peak-to-peak with a 1 to 5 second period, or
 - d) Region II of Figure 3.4.1.1.1-1 entered and less than 50% of the required LPRM upscale alarms OPERABLE,
immediately place the reactor mode switch in the shutdown position.
2. If Region II of Figure 3.4.1.1.1-1 is entered and greater than or equal to 50% of the required LPRM upscale alarms are OPERABLE, immediately exit the region by:
 - a) inserting a predetermined set of high worth control rods, or
 - b) increasing core flow by increasing the speed of the operating recirculation pump.
 3. With less than 50% of the required LPRM upscale alarms OPERABLE, follow ACTION a.1.d upon entry into Region II of Figure 3.4.1.1.1-1.
- b. In OPERABLE CONDITION 2 with no reactor coolant system recirculation loops in operation, return at least one reactor coolant system recirculation loop to operation, or be in HOT SHUTDOWN within the next 6 hours.
 - c. With any of the limits specified in 3/4.1.1.2a not satisfied:
 1. Upon entering single loop operation, comply with the new limits within 6 hours or be in at least HOT SHUTDOWN within the following 6 hours.
 2. If the provisions of ACTION c.1 do not apply, take the ACTION(s) required by the referenced Specification(s).
 - d. With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.
 - e. With any pump discharge valve not OPERABLE remove the associated loop from operation, close the valve and verify closed at least once per 31 days.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

- f. With any pump discharge bypass valve not OPERABLE close the valve and verify closed at least once per 31 days.

SURVEILLANCE REQUIREMENTS

- 4.4.1.1.2.1 Upon entering single loop operation and at least once per 24 hours thereafter, verify that the pump speed in the operating loop is \leq 80% of the rated pump speed.
- 4.4.1.1.2.2 At least 50% of the required LPRM upscale alarms shall be determined OPERABLE by performance of the following on each LPRM upscale alarm.
- 1) CHANNEL FUNCTIONAL TEST at least once per 92 days, and
 - 2) CHANNEL CALIBRATION at least once per 184 days.
- 4.4.1.1.2.3 Within 15 minutes prior to either THERMAL POWER increase resulting from a control rod withdrawal or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $<$ 30%**** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is \leq 50%**** of rated loop flow:
- a. \leq 145°F between reactor vessel steam space coolant and bottom head drain line coolant,
 - b.## \leq 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and
 - c.## \leq 50°F between the reactor coolant within the loop not in operation and operating loop.
- 4.4.1.1.2.4 The pump discharge valve and bypass valve in both loops shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.
- 4.4.1.1.2.5 The pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5% and 105%, respectively, of rated core flow, at least once per 18 months.
- 4.4.1.1.2.6 During single recirculation loop operation, all jet pumps, including those in the operable loop, shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:###
- a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established single recirculation pump speed-loop flow characteristics.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patterns by more than 10%.

4.4.1.1.2.7 The SURVEILLANCE REQUIREMENTS associated with the specifications referenced in 3.4.1.1.2a shall be followed.

- * See Special Test Exception 3.10.4.
- ** If not performed within the previous 31 days.
- **** Initial value. Final value to be determined based on startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.
- # See Specification 3.4.1.1.1 for two loop operation requirements.
- ## This requirement does not apply when the loop not in operation is isolated from the reactor pressure vessel.
- ### During startup testing following each refueling outage, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the performance of subsequent required surveillances.
- + The LPRM upscale alarms are not required to be OPERABLE to meet this specification in OPERATIONAL CONDITION 2.

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed mismatch shall be maintained within:

- a. 5% of each other with core flow greater than or equal to 75% of rated core flow.
- b. 10% of each other with core flow less than 75% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* when both recirculation loops are in operation.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

- a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or
- b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION required by Specification 3.4.1.1.1. |

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation pump speed mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F, and:

- a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or
- b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F, the operating loop flow rate is less than or equal to 50% of rated loop flow, and the reactor is operating at a THERMAL POWER/core flow condition below the 80% Rod Line shown in Figure 3.4.1.1.1-1.

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

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

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

LOCA analyses for two loop operating conditions, which result in Peak Cladding Temperatures (PCTs) below 2200°F, bound single loop operating conditions. Single loop operation LOCA analyses using two-loop MAPLHGR limits result in lower PCTs. Therefore, the use of two-loop MAPLHGR limits during single loop operation assures that the PCT during a LOCA event remains below 2200°F.

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO).

For single loop operation, the RBM and APRM setpoints are adjusted by a 8.5% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

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

Specifications have been provided to prevent, detect, and mitigate core thermal hydraulic instability events. These specifications are prescribed in accordance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," dated December 30, 1988. The boundaries of the regions in Figure 3.4.1.1.1-1 are determined using ANF decay ratio calculations and supported by Susquehanna SES stability testing.

LPRM upscale alarms are required to detect reactor core thermal hydraulic instability events. The criteria for determining which LPRM upscale alarms are required is based on assignment of these alarms to designated core zones. These core zones consist of the level A, B and C alarms in 4 or 5 adjacent LPRM strings. The number and location of LPRM strings in each zone assure that with 50% or more of the associated LPRM upscale alarms OPERABLE sufficient monitoring capability is available to detect core wide and regional oscillations. Operating plant instability data is used to determine the specific LPRM strings assigned to each zone. The core zones and required LPRM upscale alarms in each zone are specified in appropriate procedures.

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

REACTOR COOLANT SYSTEM

BASES

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

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

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 10 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

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

3/4.4.3.2 OPERATIONAL LEAKAGE

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

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

REACTOR COOLANT SYSTEM

BASES

3/4.4.4 CHEMISTRY

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

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR Part 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

REACTOR COOLANT SYSTEM

BASES

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT} . The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1 includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 1.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 60 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

In a letter dated June 23, 1989, Pennsylvania Power and Light Company (PPLC) proposed Technical Specification (TS) changes for Susquehanna Steam Electric Station, Unit 2 (SSES2). The proposed changes alter current TS associated with thermal hydraulic stability (THS) concerns by defining revised regions on the operating power-flow map pertinent to THS and new or revised operating requirements and restrictions on activities relating to these regions.

The changes are intended to avoid problems with thermal hydraulic instability, which have been a focus of NRC attention following the LaSalle instability event of March 1988. This attention resulted in the issuance of NRC Bulletin 88-07 and Supplement 1 to that bulletin, which request the utilities to provide for operator training, instrumentation verification and operating procedures intended to minimize instability potential or consequences. The NRC requests did not specifically include TS changes since it is expected that long term solution implementation, to replace the interim recommendations of the bulletin supplement, will begin within a year. However, PPLC elected to change the stability related TS for both Susquehanna Units 1 and 2, to provide operational compatibility. The current TS differ in many characteristics and details from the restrictive regions, requirements and procedures requested in the bulletin supplement, and changes are needed to prevent conflict between the TS and procedural compliance with the requested changes.

2.0 EVALUATION

The proposed changes are primarily to TS 3/4.4.1.1.1, Recirculation Loops - Two Loop Operation (TLO) and TS 3/4.4.1.1.2, Recirculation Loops - Single Loop Operation (SLO). In addition there are changes to the associated Bases, a minor related change to TS 3.4.1.4, changes to the Recirculation Loops TS which are administrative changes transferring items from Surveillance to Action, and a change to the Index page.

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The TS changes proposed for SSES2 are exactly the same as those for Susquehanna Steam Electric Station, Unit 1 (SSES1) which were previously submitted and were reviewed and approved by the staff. This similarity covers the complete range of changes outlined in the preceding paragraph, including the changed region boundaries suitably modified for the SSES2 core loading pattern. The staff Safety Evaluation Report for SSES1 describes the review and evaluation for these changes and is fully applicable for SSES2 as well. Therefore, the evaluation will not be repeated in detail here. The SSES1 review found the proposed TS changes and the material submitted to support the changes to be fully acceptable. That conclusion is also applicable to the proposed SSES2 changes.

It is noted that SSES2, for the upcoming Cycle 4, has a full core loading of Advanced Nuclear Fuel (ANF) 9x9 fuel assemblies, compared to a 61 percent 9x9 and 39 percent 8x8 assembly loading for SSES1 in its current cycle. The ANF calculation results for SSES2 Cycle 4 decay ratios are about 0.1 greater than those calculated for SSES1 at similar power-flow points. Thus for SSES2 the region boundaries given in Figure 3.4.1.1.1-1 are more conservative than those for SSES1. This difference is not unexpected (for the increased 9x9 loading) and is acceptable. It is also noted that the analysis of SSES2 end-of-cycle 3 reactor noise data (leading to measured decay ratios) by the NRC staff consultants from Oak Ridge National Laboratory have been completed and the results fall within expected ranges, indicating acceptable decay ratios for the power-flow operating points examined.

The overall conclusion of the review is that the proposed TS changes and the material submitted to support the changes are acceptable. It should be noted, however, that the NRC staff, its consultants, the BWR Owners' Group (BWROG), GE and others are continuing the review of THS concerns. The BWROG is developing several long term solutions for the problem. It is expected that a solution will be announced by the end of 1989. Any new requirements resulting from the continuing generic review of THS concerns and BWROG long term solutions will be applicable to SSES2 and may impact some of the operations, systems surveillance or TS found to be acceptable in this review.

3.0 ENVIRONMENTAL CONSIDERATION

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

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 31114) on July 26, 1989 and consulted with the State of Pennsylvania. No public comments were received, and the State of Pennsylvania did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: H. Richings

Dated: November 22, 1989