Docket No. 50-388

April 11, 1994

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

Dear Mr. Byram:

SUBJECT: POWER UPRATE WITH INCREASED CORE FLOW, SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 (PLA-4055) (TAC NO. M88311)

The Commission has issued the enclosed Amendment No. 103 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your letter dated November 24, 1993, as supplemented by letters dated January 7 and February 14, 1994.

This amendment raises the authorized power level from 3293 MWt to a new limit of 3441 MWt. The amendment also approves changes to the Technical Specifications to implement uprated power operation.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely, Original signed by Richard J. Clark

Richard J. Clark, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 103 to License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

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OFFICE	PDI-2/LA	PDI-2/PM	TECH ED*	SPLB/C	SRXB/C	EMEB/C	SCSB/C	PRPB/C	RSGB
NAME	MO'Brien	RClark:rb	RSanders	CMcCracken	~~RJones~~ JCollins	JNorberg	RBarrett	LCunningham	JWermiel
DATE	2/25/94	02/25/94	02/10/94	2/25/94	3/7/94	3/1/94	3/1/94	3/14/94	2/25/94

OFFICE	OGC	PDI-2/D	DRPE/AD	DRPE/D	ADT	ADP	NRR/DD	NRR/D
NAME	[illegible]	CMiller	JCalvo	SVarga	AThadani	LReyes (Acting)	FMiraglia	WRussell
DATE	4/15/94	4/16/94	4/16/94	4/18/94	4/7/94	4/11/94	4/11/94	4/18/94

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Richard J. Clark, Senior Project Manager
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See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 103
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 November 24, 1993, as supplemented by letters dated January 7 and February 14, 1994, 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, Facility Operating License No. NPF-22 paragraph 2.C.(1) is hereby amended to read as follows:

   (1) Maximum Power Level

   Pennsylvania Power and Light Company (PP&L) is authorized to operate the facility at reactor core power levels not in excess of 3441 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

3. Further, 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. 103 , 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.

4. This license amendment is effective as of its date of issuance and is to be implemented prior to startup in Cycle 7, currently scheduled for May 21, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Page 3 of License*
2. Changes to the Technical Specifications

Date of Issuance: April 11, 1994

*Page 3 is attached, for convenience, for the composite license to reflect this change.

(3) PP&L, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) PP&L, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) PP&L, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Pennsylvania Power & Light Company (PP&L) is authorized to operate the facility at reactor core power levels not in excess of 3441 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 103, 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.

ATTACHMENT TO LICENSE AMENDMENT NO. 103

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Appendix A Technical Specifications with the 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
1-5	1-5*
1-6	1-6
2-1	2-1
2-2	2-2*
2-3	2-3*
2-4	2-4
B 2-1	B 2-1
B 2-2	B 2-2*
B 2-7	B 2-7
-	-
3/4 1-19	3/4 1-19*
3/4 1-20	3/4 1-20
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3*
3/4 3-3	3/4 3-3*
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20

ATTACHMENT TO LICENSE AMENDMENT NO. 103

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
3/4 3-25	3/4 3-25*
3/4 3-26	3/4 3-26
3/4 3-41	3/4 3-41*
3/4 3-42	3/4 3-42
3/4 3-53	3/4 3-53*
3/4 3-54	3/4 3-54
3/4 4-1	3/4 4-1*
3/4 4-1a	3/4 4-1a
3/4 4-1b	3/4 4-1b
3/4 4-1c	3/4 4-1c
3/4 4-1d	3/4 4-1d*
3/4 4-1e	3/4 4-1e
3/4 4-1f	3/4 4-1f
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4*
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6*
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8*
3/4 4-19	3/4 4-19*
3/4 4-20	3/4 4-20
3/4 5-3	3/4 5-3*
3/4 5-4	3/4 5-4

ATTACHMENT TO LICENSE AMENDMENT NO. 103

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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
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4*
B 3/4 4-5	B 3/4 4-5*
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2*
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-4*
5-5	5-5*
5-6	5-6
5-7	5-7*
5-8	5-8
6-20a	6-20a*
6-20b	6-20b

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.29 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.30 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.31 The PROCESS CONTROL PROGRAM (PCP) shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE - PURGING

1.32 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.33 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3441 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.35 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.36 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.3.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows, resilient material seals, or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1a.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With MCPR less than 1.06* and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

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

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

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

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

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	≤ 1087 psig	≤ 1093 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	≤ 7.0 x full power background	≤ 8.4 x full power background
7. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 88 gallons	≤ 88 gallons
b. Float Switch	≤ 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.

\# See Specification 3.4.1.1.2.a for single loop operation requirements.

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2 for SNP fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. XN-NF-524(A) Revision 1).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the XN-3 correlation is valid for critical power calculations at pressure greater than 580 psig and bundle mass fluxes greater than 0.25×10^6 lbs./hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity Safety Limit is established by a limiting condition on core THERMAL POWER with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to assure a minimum bundle flow for all fuel assemblies which have a relatively high power and potentially can approach a critical heat flux condition. For the SNP 9 x 9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the SNP 9 x 9 design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At 25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of approximately 3.0 which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressures below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR).

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. XN-NF-524 (A), Revision 1 and PL-NF-90-001 describe the methodologies used in determining the Safety Limit MCPR.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation (refer to Section B 2.1.1), the assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that during sustained operation at the Safety Limit MCPR there would be no transition boiling in the core. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicates that LWR fuel can survive for an extended period of time in an environment of boiling transition.

SNP fuel is monitored using the XN-3 Critical Power Correlation. SNP has determined that this correlation provides sufficient conservatism to preclude the need for any penalty due to channel bow. The conservatism has been evaluated by SNP to be greater than the maximum expected ΔCPR (0.02) due to channel bow in C-lattice plants using channels for only one fuel bundle lifetime. Since Susquehanna SES Unit 2 is a C-lattice plant and uses channels for only one fuel bundle lifetime, monitoring of the MCPR limit with the XN-3 Critical Power Correlation is conservative with respect to channel bow and addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 5.5% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient assuming the turbine bypass valves operate.

This function is not required when THERMAL POWER is below 30% of RATED THERMAL POWER. The Turbine Bypass System is sufficient at this low power to accommodate a turbine stop valve closure without the necessity of shutting down the reactor. This function is automatically bypassed at turbine first stage pressures less than the analytical limit of 147.7 psig, equivalent to THERMAL POWER of about 30% RATED THERMAL POWER. Turbine first stage pressure of 147.7 psig is equivalent to 22% of rated turbine load.

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a faster closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section 15.2 of the Final Safety Analysis Report.

This function is not required when THERMAL POWER is below 30% of RATED THERMAL POWER. The Turbine Bypass System is sufficient at this low power to accommodate a turbine control valve closure without the necessity of shutting down the reactor. This function is automatically bypassed at turbine first stage pressures less than the analytical limit of 147.7 psig, equivalent to THERMAL POWER of about 30% RATED THERMAL POWER. Turbine first stage pressure of 147.7 psig is equivalent to 22% of rated turbine load.

11. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the standby liquid control system otherwise inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

2. With the standby liquid control system otherwise inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-2.

3. The heat tracing circuit is OPERABLE by actuating the test feature and determining that the power available light on the local heat tracing panel energizes.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Verifying the continuity of the explosive charge.

2. Determining that the available weight of sodium pentaborate is greater than or equal to 5500 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-2 by chemical analysis.*

3. Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1224 psig is met.

d. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.

2. **Demonstrating that all heat traced piping is unblocked by pumping from the storage tank to the test tank and then draining and flushing the discharge piping and test tank with demineralized water.

3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized.

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* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

** This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

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

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 core flow of 100 million lbs/hr,

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. The FRACTION OF LIMITING POWER DENSITY (FLPD) for SNP fuel is the actual LHGR divided by the applicable LINEAR HEAT GENERATION RATE for APRM Setpoints limit specified in the CORE OPERATING LIMITS REPORT.

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 determined above, 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:

---

* With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

\# See Specification 3.4.1.1.2.a for single loop operation requirements.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

4.2.2 (Continued)

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 rector is operating with MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
7. Drywell Pressure - High	1, 2[h]	2	1
8. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	5[i]	2	3
b. Float Switch	1, 2	2	1
	5[i]	2	3
9. Turbine Stop Valve - Closure	1[j]	4[k]	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1[j]	2[k]	6
11. Reactor Mode Switch Shutdown Position	1, 2	1	1
	3, 4	1	7
	5	1	3
12. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1	Be in at least HOT SHUTDOWN within 12 hours.
ACTION 2	Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.
ACTION 3	Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.
ACTION 4	Be in at least STARTUP within 6 hours.
ACTION 5	Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.
ACTION 6	Initiate a reduction in THERMAL POWER within 15 minutes, and reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within 2 hours.
ACTION 7	Verify all insertable control rods to be inserted within 1 hour.
ACTION 8	Lock the reactor mode switch in the Shutdown position within 1 hour.
ACTION 9	Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

TABLE 3.3.1-1 (Continued) REACTOR PROTECTION SYSTEM INSTRUMENTATION TABLE NOTATIONS	
(a)	A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon determination that a trip setpoint cannot be restored to within its specified value during performance of the CHANNEL CALIBRATION, the appropriate ACTION, 3.3.1a or 3.3.1b, shall be followed.
(b)	This function is automatically bypassed when the reactor mode switch is in the Run position.
(c)	The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.
(d)	The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.
(e)	An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.
(f)	This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.
(g)	This function is automatically bypassed when the reactor mode switch is not in the Run position.
(h)	This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.
(i)	With any control rod withdrawn.*
(j)	This function shall not be automatically bypassed when turbine first stage pressure is greater than an allowable value of 136 psig.
(k)	Also actuates the EOC-RPT system.

* Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ 0.09**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switch	NA
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant.

#Measured from actuation of fast-acting solenoid.

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	≤ 7.0 x full power background	≤ 8.4 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	≤ 7.0 x full power background	≤ 8.4 x full power background
c. Main Steam Line Pressure - Low	≥ 861 psig**	≥ 841 psig**
d. Main Steam Line Flow - High	≤ 113 psid**	≤ 121 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	≤ 197°F	≤ 200°F
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. RWCU Δ Flow - High	≤ 60 gpm	≤ 80 gpm
b. RWCU Area Temperature - High	≤ 147° F or 131°F#	≤ 154°F or 137°F#
c. RWCU/Area Ventilation Δ Temperature - High	≤ 69°F or 40.5°F#	≤ 72°F or 43.5°F#*
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
f1. RWCU Flow - High	≤ 462 gpm	≤ 472 gpm
f2. Non-Regenerative Heat Exchanger Discharge Temperature - High	≤ 144°F	≤ 150°F
g. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Δ Pressure - High	≤ 138" H_2O	≤ 143" 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

* These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)		
d. RCIC Equipment Room Temperature - High	≤ 167°F	≤ 174°F
e. RCIC Equipment Room Δ Temperature - High	≤ 89°F	≤ 98°F*
f. RCIC Pipe Routing Area Temperature - High	≤ [illegible]	≤ 174°F##
g. RCIC Pipe Routing Area Δ Temperature - High	≤ [illegible]	≤ 98°F##*
h. RCIC Emergency Area Cooler Temp. - High	≤ 1[illegible]	≤ 174°F
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	≤ 387 inches H_2O	≤ 399 inches H_2O
b. HPCI Steam Supply Pressure - Low	≥ 104 psig	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. HPCI Equipment Room Temperature - High	≤ 167°F	≤ 174°F
e. HPCI Equipment Room Δ Temperature - High	≤ 89°F	≤ 98°F
f. HPCI Emergency Area Cooler Temp. - High	≤ 167°F	≤ 174°F
g. HPCI Pipe Routing Area Temperature - High	≤ 167°F##	≤ 174°F##
h. HPCI Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##*
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

* These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

TABLE 3.3.2-2 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches *	≥ 11.5 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 98 psig	≤ 108 psig
c. RHR Flow - High	≤ 25,000 gpm	≤ 26,000 gpm
d. Manual Initiation	NA	NA
e. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

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

** Initial value. Final value to be determined based on Power Uprate startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.

\# Lower setpoints for TSH-G33-2N600 E, F and TDSH-G33-2N602 E, F.

\#\# 15 minutes time delay.

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
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	NA	M	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3*
f. RCIC Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
g. RCIC Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3*
h. RCIC Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	NA	M	Q	1, 2, 3
b. HPCI Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3

*These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

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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
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
d. HPCI Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3
f. HPCI Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
h. HPCI Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3****
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	Q	1, 2, 3
c. RHR Flow - High	S	M	R	1, 2, 3
d. Manual Initiation	NA	R	NA	1, 2, 3
e. Drywell Pressure - High	NA	M	R	1, 2, 3

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When any turbine stop valve is open.

*** When VENTING or PURGING the drywell per Specification 3.11.2.8.

**** This trip function need not be OPERABLE from October 19, 1989 to January 19, 1990.

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2. LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The instrument response time portion of the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be measured at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE-RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limit.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured at least once per 60 months.

TABLE 3.3.4.2-1
END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [a]
1. Turbine Stop Valve - Closure	2 [b]
2. Turbine Control Valve - Fast Closure	2 [b]

(a) A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.
(b) This function shall not be automatically bypassed when turbine first stage pressure is greater than an allowable value of 136 psig.

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 1 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 1 hour.

NOTES

* With THERMAL POWER ≥ 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.

*** Not required when eight or fewer fuel assemblies (adjacent to the SRMs) are in the core.

(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 ≥ 100 cps or the IRM channels are on range 3 or higher.

(c) This function is automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function is automatically bypassed when the IRM channels are on range 3 or higher.

(e) This function is 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##	≤ 0.63 W + 41%	≤ 0.63 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##		
1) Flow Biased	≤ 0.58 W + 50%	≤ 0.58 W + 53%
2) High Flow Clamped	≤ 108% of RATED THERMAL POWER	≤ 111% of RATED THERMAL POWER
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	≥ 0.7 cps **	≥ 0.5 cps **
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 division of full scale	≤ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 division 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	≤ 114/125 divisions of full scale	≤ 117/125 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.

** Provided signal-to-noise ratio is ≥ 2. Otherwise, 3 cps as trip setpoint and 2.8 cps for allowable value.

See Specification 3.4.1.1.2.a for single loop operation requirements.

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/cm^2 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.

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 a core flow of 109.5 million lbm/hr and 110.5 million lbm/hr respectively, 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.

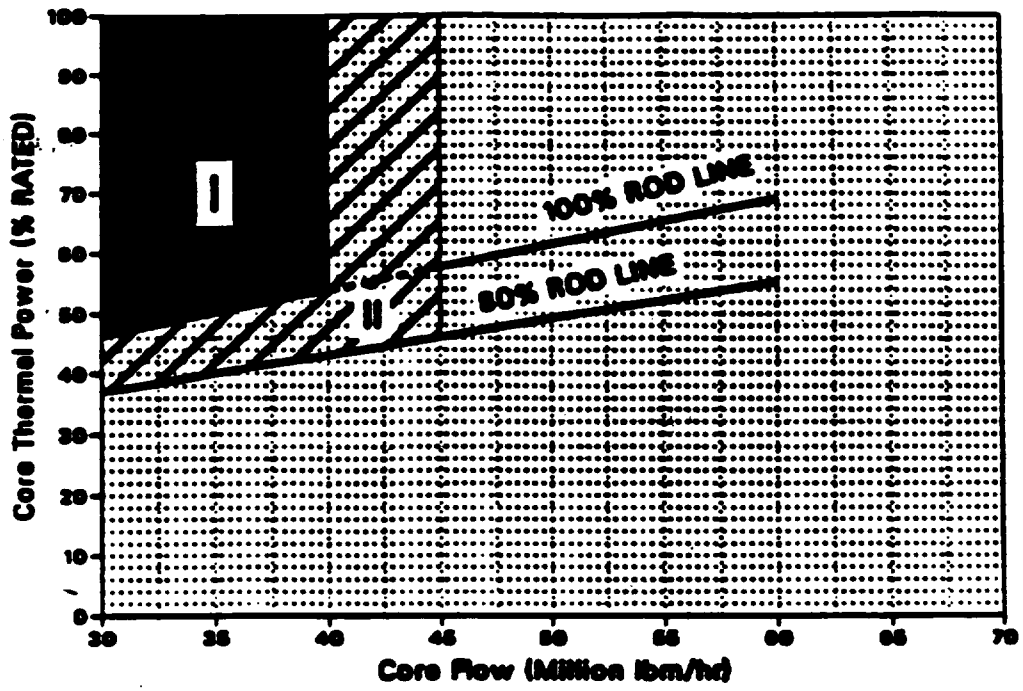

Figure 3.4.1.1.1-1
THERMAL POWER STABILITY RESTRICTIONS

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 ≤ 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:

Trip Setpoint	Allowable Value
≤ 0.58W + 54%	≤ 0.58W + 57%

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

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

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.
5. Specification 3.2.4: The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the applicable Single Loop Operation LHGR limit as specified in the CORE OPERATING LIMITS REPORT.
6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.63W + 35%	≤ 0.63W + 37%
	Trip Setpoint	**Allowable Value**
b. APRM - Flow Biased	≤ 0.58W + 45%	≤ 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:

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/cm^2 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 OPERALBE,

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.

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 ≤ 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 ≤ 50%**** of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,
b.## ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and
c.## ≤ 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 stops shall be demonstrated OPERABLE with overspeed setpoints less than or equal to a core flow of 109.5 million lbm/hr and 110.5 million lbm/hr respectively, at least once per 18 months.

4.4.1.1.2.6 During single recirculation loop operation, all jet pumps, including those in the inoperable 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.

SURVEILLANCE REQUIREMENTS (Continued)

b. The indicated total core flow differs by more than 10% from the established total core flow value from single recirculation loop flow measurements.

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 Power Uprate startup testing. Any required change to this value shall be submitted to the Commission within 90 days of Power Uprate startup test program 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.

At least once per 18 months (555 days), 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 required surveillances.

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

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 million lbm/hr.

b. 10% of each other with core flow less than 75 million lbm/hr.

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.

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* See Special Test Exception 3.10.4.

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.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:* **

2 safety-relief valves @ 1175 psig ±1%
6 safety-relief valves @ 1195 psig ±1%
8 safety-relief valves @ 1205 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, AND 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool water temperature is 105°F or greater, place the reactor mode switch in the Shutdown position.

c.### With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2### The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.25 of the full open noise level# by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a

b. Calibration in accordance with procedures prepared in conjunction with its manufacturer's recommendations at least once per 18 months.##

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* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

** Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

\# Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

\## The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

\### Compliance with these requirements for the "S" SRV acoustic monitor is not required for the period beginning January 21, 1994, until the next unit shutdown of sufficient duration to allow for containment entry, not to exceed the sixth refueling and inspection outage.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 At least the following reactor coolant system leakage detection systems shall be OPERABLE:

a. Two drywell floor drain sump level channels, and

b. One primary containment atmosphere gaseous radioactivity monitoring system channel and one containment atmosphere particulate radioactivity monitoring system channel aligned to the drywell.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or both channels of the drywell floor drain sump level monitoring system inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both channels of the gaseous radioactivity monitoring system inoperable or with both channels of the particulate radioactivity monitoring system inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours. If at least one channel of the affected monitoring system cannot be returned to OPERABLE status and aligned to the drywell within 30 days, or the grab samples are not obtained and analyzed as required, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Drywell floor drain sump level monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.
b. 5 gpm UNIDENTIFIED LEAKAGE.
c. 25 gpm total leakage averaged over any 24-hour period.
d. 1 gpm leakage at a reactor coolant system pressure of 1035 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.
e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b. and/or c., above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm pressure at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric particulate and gaseous radioactivity at least once per 4 hours, and

b. Monitoring the drywell floor drain sump level at least once per 4 hours.

c. Determining the total IDENTIFIED LEAKAGE at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with the alarm setpoints per Table 3.4.3.2-1 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

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REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1050 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1050 psig, reduce the pressure to less than 1050 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1050 psig at least once per 12 hours.

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* Not applicable during anticipated transients.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the CSS and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$ 100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to $\leq$ 100 psig within the next 24 hours.

e. With a CSS header ΔP instrumentation channel inoperable, restore the inoperable channel to OPERABLE status with 72 hours or determine the ECCS header ΔP locally at least once per 12 hours; otherwise, declare the CSS inoperable.

f. In the event an ECCS system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

g. With the condensate transfer pump discharge low pressure alarm instrumentation inoperable, monitor the CSS, LPCI, and HPCI pressure locally at least once per 24 hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling system shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the CSS, the LPCI system, and the HPCI system:

a) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water by:

1. Venting at the high point vents
2. Performing a CHANNEL FUNCTIONAL TEST of the condensate transfer pump discharge low pressure alarm instrumentation.

b) Verifying that each valve, manual, power-operated, or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct** position.

2. For the CSS, performance of a CHANNEL FUNCTIONAL TEST of the core spray header ΔP instrumentation.

3. For the LPCI system, verifying that at least one LPCI system subsystem cross-tie valve is closed with power removed from the valve operator.

4. For the HPCI system, verifying that the pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a total flow of at least 6350 gpm against a test line pressure of ≥ 282 psig, corresponding to a reactor vessel steam dome pressure of ≥ 105 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 12,200 gpm against a test line pressure of ≥ 222 psig, corresponding to a reactor vessel to primary containment differential pressure ≥ 20 psid.

3. The HPCI pump develops a flow of at least 5000 gpm against a test line pressure of ≥ 1140 psig when steam is being supplied to the turbine at 920, +140, -20 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

** Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

This function is not required when THERMAL POWER is below 30% of RATED THERMAL POWER. The Turbine Bypass System is sufficient at this low power to accommodate a turbine stop valve or control valve closure without the necessity of tripping the reactor recirculation pumps. This function is automatically bypassed at turbine first stage pressures less than the analytical limit of 147.7 psig, equivalent to THERMAL POWER of about 30% RATED THERMAL POWER. Turbine first stage pressure of 147.7 psig is equivalent to 22% of rated turbine load.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 175 ms.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Rod Block Monitor (RBM) portion of the control rod block instrumentation contains multiplexing circuitry which interfaces with the reactor manual control system. The RBM is a redundant system which includes two channels of information which must agree before rod motion is permitted. Each of these redundant channels has a self-test feature which is implicitly tested during the performance of surveillance pursuant to this specification as well as the control rod operability specification (3/4.1.3.1).

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figure 3.4.6.1-1, curves C and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through 1972.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

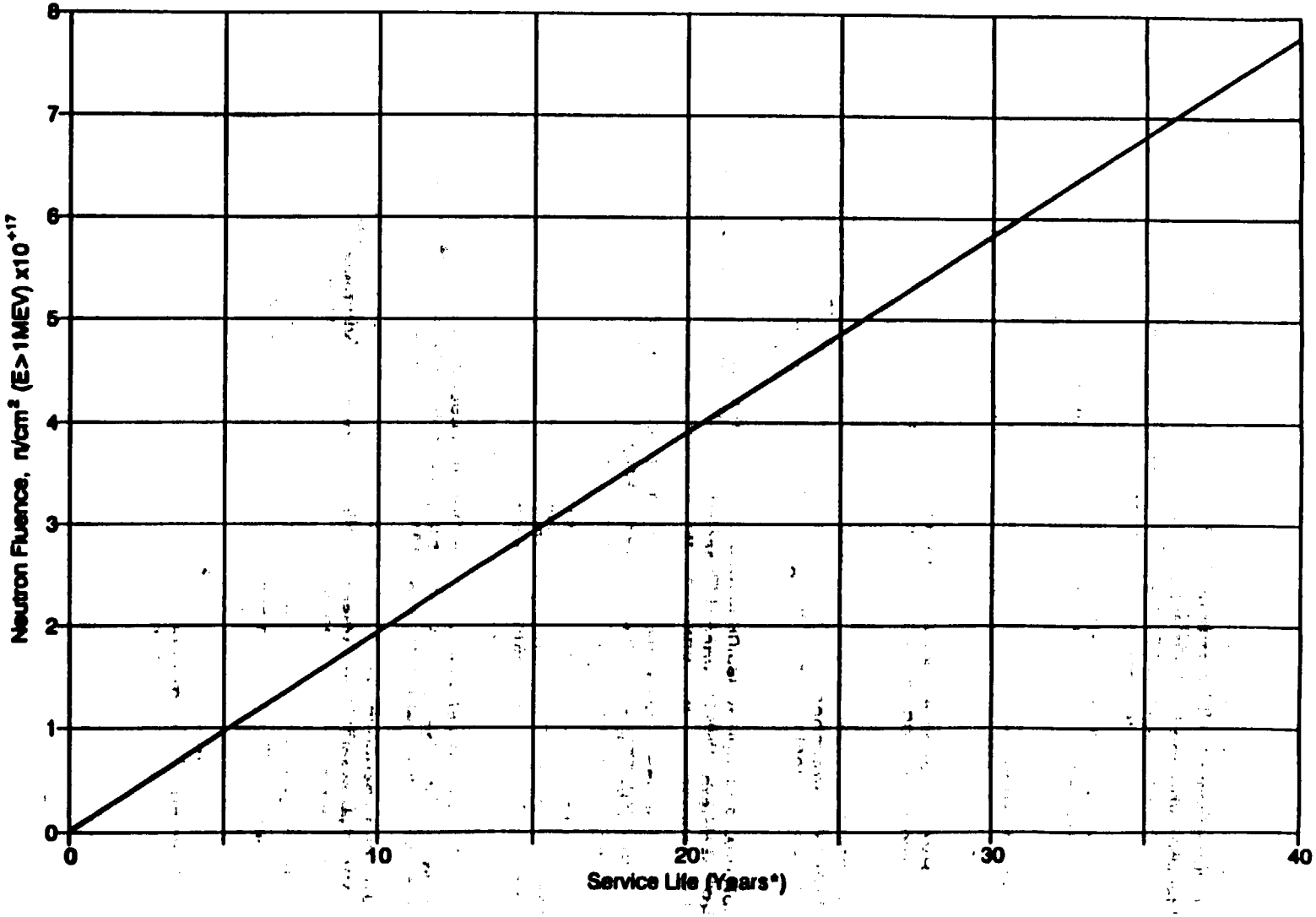
BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

Limiting Beltline Component	Weld Seam I.D. or Mat'l. Type	Heat/Slab or Heat/Lot	CU(%)	Ni(%)	Starting RT_{NDT} (°F)	ΔRT_{NDT} (°F)*	Min. Upper Shelf (Lft-Lbs)	Max. RT_{NDT} (°F)
Plate	SA-533 GR B CL.1	C2421-3	0.13	0.68	-10	56.7	N/A	46.7
Weld	N/A	624263/ E204A27A	0.06	0.89	-20	50	N/A	+30

NOTE: * These values are given only for the benefit of calculating the 32 EFPY RT_{NDT} per R.G. 1.99 Rev. 2.

NON-BELTLINE COMPONENT	MATERIAL TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring #5	SA-533 GR B Cl.1	All	+10
Bottom Head Dome	"	C0462	+20
Bottom Head Torus	"	C0472	+10
Top Head Side Plates	"	C0473-1	+10
Top Head Flange	SA-508, Cl.2	125H446	+10
Vessel Flange	"	2L2393	+10
Feedwater Nozzle	"	Q2Q62W	-10
Steam Outlet Nozzle	"	Q2Q64W	+30
Weld	Bottom Head Flanges to Shell Top Head Other Non-Beltline	All All All	-20 -20 0
Closure Studs	SA-540 GR B24	All	Meet requirements of 45 ft-lbs and 25 mils lateral expansion at +10°F



Fast Neutron Fluence (E> 1 Mev) at I.D. Surface as a Function of Service Life*
Bases Figure B 3/4.4.6-1

* At 90% of RATED THERMAL POWER and 90% Availability

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING AND SHUTDOWN

The core spray system (CSS) is provided to assure that the core is adequately cooled following a loss-of-coolant accident, and together with the LPCI mode of the RHR system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the automatic depressurization system (ADS).

The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CSS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two subsystems, each with two pumps, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which CS system operation or LPCI mode of the RHR system operation maintains core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5000 gpm at reactor pressures between 1187 and 150 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls six selected safety-relief valves although the safety analysis only takes credit for five valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI, CSS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is also required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume, vortex prevention plus a safety margin for conservatism.

3/4 6.2 DEPRESSURIZATON SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 53 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1053 psia. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 53 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 45.0 psig which is below the design pressure of 53 psig. Maximum water volume of 133,540 ft^3 results in a downcomer submergence of 12 feet and the minimum volume of 122,410 ft^3 results in a submergence approximately 24 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 128°F immediately following blowdown which is below the 170°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

During a LOCA, potential leak paths between the drywell and suppression chamber airspace could result in excessive containment pressures, since the steam flow into the airspace would bypass the heat sink capabilities of the pool. Potential sources of bypass leakage are the suppression chamber-to-drywell vacuum breakers (VBs), penetrations in the diaphragm floor, and cracks in the diaphragm floor/liner plate and downcomers located in the suppression chamber airspace. The containment pressure response to the postulated bypass leakage can be mitigated by manually actuating the suppression chamber sprays. An analysis was performed for a design bypass leakage area of $A/(k)^{1/2}$ equal to 0.0535 ft^2 to verify that the operator has sufficient time to initiate the sprays prior to exceeding the containment design pressure of 53 psig. The limit of 10% of the design value of 0.0535 ft^2 ensures that the design basis for the steam bypass analysis is met.

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.3 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test is not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 70% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 70% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10CFR 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically

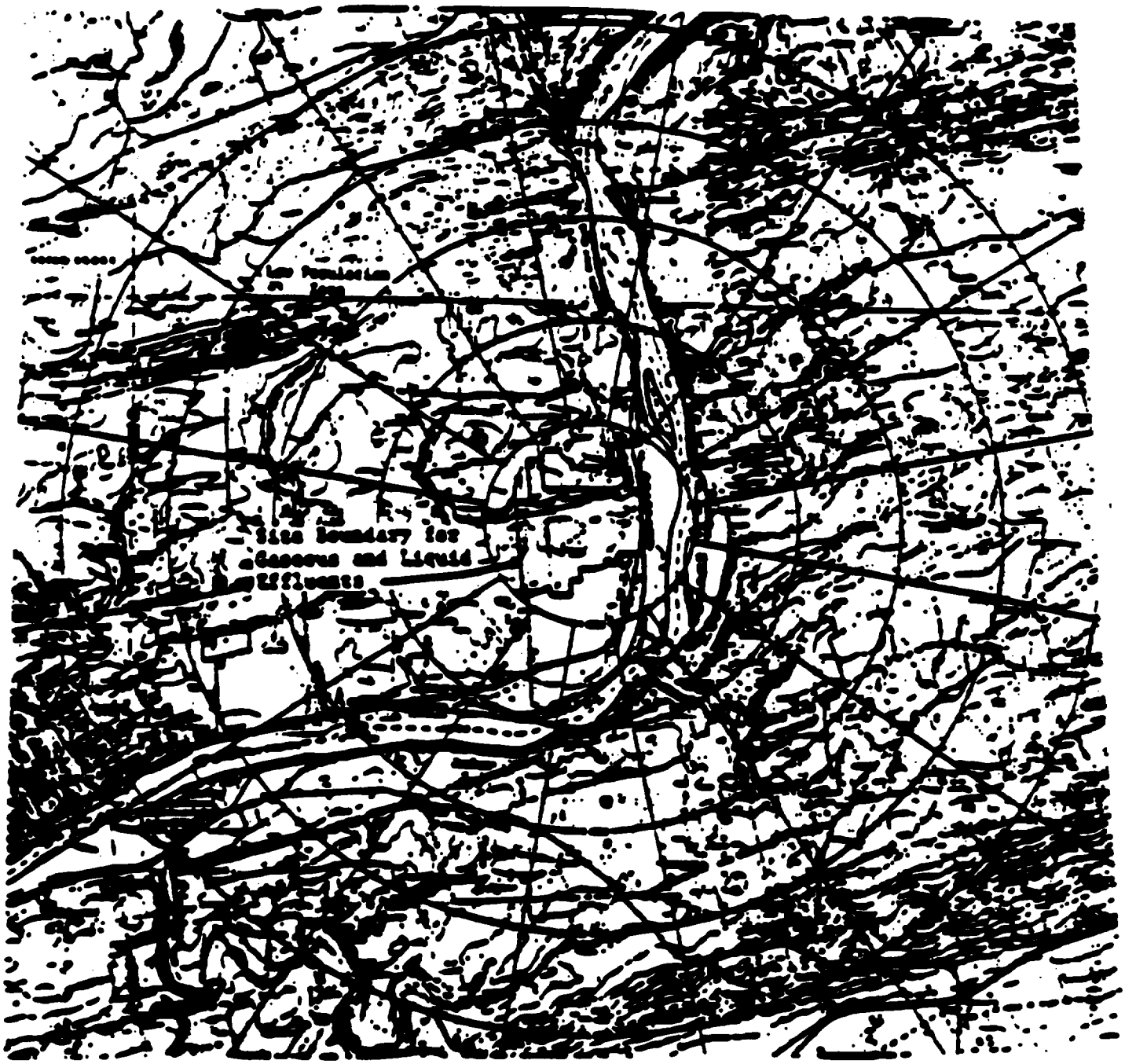


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of non-enriched or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of Zirconium alloy filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in non-limiting core regions. Each fuel rod shall have a nominal active fuel length of 150 inches. Reload fuel shall have a maximum average enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C), and/or Hafnium metal. The control rod shall have a nominal axial absorber length of 143 inches. Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 532°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes all calculational biases and uncertainties as described in Section 9.1.2 of the FSAR.

b. A nominal 6.625 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 816'9".

CAPACITY

5.6.3.1 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2840 fuel assemblies.

5.6.3.2 A multi-purpose storage rack may be used to store up to 10 sound and/or defective fuel assemblies and/or other reactor internals.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 551°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 hydrostatic pressure and leak tests	Pressurized to ≥ 930 psig and ≤ 1250 psig.

CORE OPERATING LIMITS REPORT (Continued)

6.9.3.2 The analytical methods used to determine the core operating limits shall be those topical reports and those revisions and/or supplements of the topical report previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For Susquehanna SES the topical reports are:

1. PL-NF-87-001-A, "Qualification of Steady State Core Physics Methods for BWR Design and Analysis," July, 1988.

2. PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," July, 1992.

3. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

4. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc., June 1986.

5. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc., September 1986.

6. PLA-3407, "Proposed Amendment 132 to License No. NPF-14: Unit 1 Cycle 6 Reload," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), July 2, 1990.

7. Letter from Elinor G. Adensam (NRC) to H. W. Keiser (PP&L), "Issuance of Amendment No. 31 to Facility Operating License No. NPF-22 - Susquehanna Steam Electric Station, Unit 2," October 3, 1986.

8. PLA-3533, Revised Proposed Amendment 67 to License No. NPF-22: Unit 2 Cycle 5 Reload," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), March 7, 1991.

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question: Seismic/LOCA Analysis of U2C2 Reload," Letter from H. W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

12. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc., March 1983.

13. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1983.

CORE OPERATING LIMITS REPORT (Continued)

14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.

15. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

16. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

17. NRC SER on PP&L Power Uprate LTR (November 30, 1993).

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

By letter dated November 24, 1993, as supplemented by letters of January 7 and February 14, 1994, the Pennsylvania Power and Light Company (PP&L or the licensee) submitted a request for a revision of Facility Operating License No. NPF-22 for Susquehanna Steam Electric Station (SSES), Unit 2, to uprate the current licensed power level from 3293 MWt to a new limit of 3441 MWt. The amendment application also submitted a number of changes to the Technical Specifications (TSs) to implement uprated power operation.

The supplemental letter of January 7, 1994, corrected a single typographical error. The February 14, 1994, letter transmitted an affidavit. The supplemental letters did not affect the application or the staff's initial proposed no significant hazards consideration determination.

2.0 EVALUATION

PP&L's letter of June 15, 1992, submitted "Licensing Topical Report NE-092-001, Revision 0, for Power Uprate With Increased Core Flow," for Susquehanna Steam Electric Station (SSES), Units 1 and 2. The report was submitted to support future proposed amendments to the Units 1 and 2 licenses to permit a 4.5-percent increase in reactor thermal power and an 8-percent increase in core flow for each unit. The initial submittal was revised and supplemented by letters of July 24, September 17, and December 18, 1992, and January 8, January 25, April 2, August 5, August 12, and September 29, 1993.

On November 30, 1993, the Director, Office of Nuclear Reactor Regulation, issued a letter, supported by an enclosed safety evaluation, which informed PP&L that the revised licensing topical report adequately supported the proposed power uprate for SSES. Therein the staff concluded that SSES could operate safely with the proposed 8-percent increase in core flow, the proposed 4.5-percent increase in reactor thermal power, the corresponding 5-percent

increase in main turbine inlet steam flow, and the corresponding increases in flows, temperatures, pressures, and capacities required in supporting systems and components at these uprated conditions, but that authorization for any increase in reactor thermal power would be based on a review of the TS changes submitted with the amendment application. The safety evaluation and letter are attached.

As stated in the conclusion section of the November 30, 1993, safety evaluation, there were four open items that PP&L was to address in the proposed license amendment application. These four items were (1) the startup test plan, (2) the anticipated transient without scram (ATWS) analysis, (3) the pipe whip and jet impingement evaluation, and (4) the program to upgrade the emergency operating procedures. PP&L addressed each of these items in the subject amendment application.

2.1 Post-Power Uprate Startup Test Program

PP&L plans to perform a post-power uprate startup test program similar in nature to the original Susquehanna startup test program described in Chapter 14 of the Final Safety Analysis Report (FSAR), but with the scope of testing limited to those tests or portions of tests affected by power uprate or increased core flow. The test program will be conducted in four separate segments or test plateaus. Each test plateau will contain one or more test conditions which defines uprate power levels and core flows at which the tests are to be performed. The test plateaus and test conditions were described in the application. The current 100-percent power level (3293 MWt) represents about 95.7-percent power of the proposed maximum uprated power level (i.e., 100% power equals 3441 MWt). One of the test plateaus will bracket this point (i.e., 95-96% of the uprate power level) with varying core flow. The last two plateaus are at 97-98-percent and 99-100-percent of the proposed uprated power level. Generally, all tests scheduled to be performed in one test condition are to be completed before proceeding to the next higher test condition. After all testing in each plateau is completed, the test results for all tests will be reviewed by the Plant Operations Review Committee (PORC) before operations authorization is given to proceed to the next test plateau.

The requirements for power-uprate startup testing come from a review of Chapter 14 of the FSAR, the General Electric (GE) Power Uprate Startup Test Specification, the proposed TSs for power uprate and the Susquehanna Licensing Topical Report NE-092-001 described previously. The tests which will be performed for the power-uprate startup test program were described in five tables in Attachment 1 to PP&L's application. The staff has reviewed the proposed test program and finds it acceptable. It is recognized that changes to the test program may occur as it is executed.

2.2 High Energy Line Breaks

In Section 3.9.1 of the November 30, 1993, safety evaluation, the staff reported that the licensee was still evaluating the calculations supporting the disposition of potential targets of pipe whip and jet impingement from

postulated high energy line breaks (HELBs) to determine the effects of the power uprate. The staff also stated that the licensee expected the evaluation to confirm the adequacy of the existing design under power-uprate conditions.

Because the licensee had not completed these calculations, the staff could not reach any conclusion regarding the impact of the uprated power level operation on HELBs.

In the November 24, 1993, letter, the licensee submitted information to indicate that these calculations were complete. The results of the licensee's analysis showed that the effects of power uprate on HELBs were proportional to the increase in reactor vessel pressure which resulted in higher loads, stresses, and displacements on the piping, supports, and whip restraints. However, the increases were relatively small and, as expected, the original design-basis HELB commitments in the FSAR were still satisfied. The staff has reviewed the results of the licensee's analysis and concurs with the licensee's conclusions that, for the power uprate, no further action is required regarding protection against the effects of pipe whip and jet impingement due to HELBs. The results of the analysis are consistent with the results of analyses performed at other plants during similar power uprates. The staff, therefore, concludes that protection against the effects of postulated breaks in HELBs will remain acceptable after the power uprate.

2.3 Anticipated Transient Without Scram Analysis (ATWS)

PP&L had not addressed the Susquehanna ATWS analysis for power-uprate conditions in the Susquehanna Licensing Topical Report NE-092-001 because the licensee had not completed the calculations and analyses when the topical report was submitted. Although GE has performed generic bounding ATWS analyses, these analyses cannot be used for Susquehanna because the licensee: (1) uses non-GE fuel and (2) has taken exceptions to Revision 4 of the emergency procedure guidelines (EPGs) for responding to ATWS, which are assumed in the GE generic analyses.

The results of the ATWS analysis for SSES, Unit 2, for power-uprate conditions were sent with the November 24, 1993, submittal. Seven limiting events were analyzed. All events were initiated at the extended load line limit, 100 percent of uprated power (3441 MWt) and 87 percent of rated core flow (87 MLb/hr).

The licensee's ATWS analysis predicts that the most limiting transient is rapid closure of the main steam isolation valves (MSIVs). In this pressurization transient, the computer analyses predict that the peak reactor pressure vessel pressure could reach 1317 psig, the peak suppression pool temperature could rise to 178.9 °F and the peak fuel cladding temperature could be 1463 °F. The staff has reviewed the licensee's ATWS analysis for Unit 2 for power-uprate conditions and has determined that the results are acceptable.

2.4 Emergency Operating Procedures

Emergency operating procedures (EOPs) to support uprated power operation are under development with implementation, to include operator training, scheduled to take place before startup in Fuel Cycle No. 7. Presently, the plant-specific technical guidance has been revised and verified. All EOPs to support power uprate have been revised and reviewed by shift management and training personnel. Comments are being resolved, and five of the six affected EOPs have been completed and are being verified. The sixth EOP is in the comment resolution stage. The final revised EOPs for power uprate will be reviewed in the same manner as other changes to the EOPs are being reviewed during the normal inspection programs.

2.5 Proposed TS Changes

Operation with a 4.5-percent increase in reactor thermal power and an 8-percent increase in core flow results in a 5-percent increase in main turbine inlet steam flow, approximately a 30 psig increase in design reactor pressure and other changes in system pressures, temperatures and flows. To implement the power uprate, the licensee submitted a number of changes to the TSs to revise such parameters as the authorized power level, core flow, reactor pressure, steam pressures and flows, turbine first-stage pressure setpoints, average power range monitor (APRM) setpoints for two-loop and single-loop operation, changes in some reactor protection system (RPS) setpoints (such as the turbine pressure that initiates the recirculation pump trip system), high pressure coolant injection (HPCI) steamline flow and pump discharge pressure, thermal power stability restrictions, and resetting the safety/relief valve setpoints. The specific TS changes are as follows:

1. Change Definition 1.33 to redefine rated thermal power as 3441 megawatts thermal. The staff's safety evaluation of November 30, 1993, evaluated all aspects of operation of the Susquehanna units at an increased thermal power of 3441 megawatts including: the reactor thermohydraulic and neutronic performance, thermal-hydraulic stability, the ability of the control rod drive system to control core reactivity at the increased reactor pressure, the structural integrity of the reactor coolant and connected systems, overpressure protection with the new safety-relief valve settings, the effect of revised LOCA loads on the reactor system, containment systems and emergency core cooling system performance, the effect of increased core flow on reactor internals and pumps, the performance of the steam, feedwater and auxiliary systems, the capability of the High Pressure Coolant Injection, Reactor Core Isolation Cooling, Residual Heat Removal and Core Spray Systems, the impact of the increased thermal power on containment system and standby gas treatment system performance, the changes to the plants' instrumentation and control systems, the functioning of all safety-related service water systems, the capability of the non-safety-related cooling systems, the impact of the increased thermal power on the heating, ventilating and air conditioning systems, the impact on the radwaste systems, the impact of the increased thermal power on postulated design basis accidents, the environmental

qualification of mechanical and electrical equipment under the increased pressures, temperatures and humidity and the effect of the increased power on generic issues. The staff also issued an environmental assessment, dated March 11, 1994, that evaluated the potential impact of operation at the increased thermal power with respect to potential radiological and non-radiological effects on the environment. As part of the power uprate program, the licensee conducted an extensive design-basis reconstitution and design basis upgrade program. The NRC staff in effect performed a licensing review of all systems that would be effected by operation at increased thermal power and the associated increased core, feedwater and steam system flows and pressures. As a result of the extensive evaluation, the staff concluded that the Susquehanna units can operate safely with a 4.5 percent increase in reactor thermal power, an 8 percent increase in core flow, the corresponding 5 percent increase in steam flow and the corresponding increases in flows, temperatures, pressures and capacities required in supporting systems and components. The proposed increase in thermal power from 3293 MWt to 3441 MWt is acceptable.

2. In Section 2.1.1 and 2.1.2, replace the reference to 10 percent of rated core flow with a reference to the actual core flow of 10 million lbs/hr under power uprate conditions. The references to "rated core flow" in TS 2.1.1 and 2.1.2 have been deleted to avoid confusion since allowable core flow is being increased by 8 percent. As discussed in the Bases for TS 2.1.1, boiling transition will not occur in fuel bundles if core power is less than 25 percent of rated thermal power, regardless of pressure or core flow. Specifying a specific minimum core flow before exceeding 25 percent power is more precise than specifying a percentage of maximum core flow and is acceptable.

3. In Table 2.2.1-1, Reactor Protection System Instrumentation Setpoints, Item 3, change the trip setpoint and allowable value for Reactor Vessel Steam Dome Pressure-High to ≤ 1087 psig and ≤ 1093 psig, respectively, to reflect the higher reactor pressure with power uprate. This scram function is designed to terminate a pressure increase transient not terminated by direct scram or high flux scram. The nominal trip setpoint is maintained above the reactor vessel maximum operating pressure. The allowable value is set below the analytical limit used in the transient analyses. For the uprated transient analyses, the licensee used 1105 psig. The results of the overpressure protection analyses using this revised analytical limit showed that the peak pressure remained below the 1375 psig American Society of Mechanical Engineers (ASME) limit and met all licensing requirements. The 36 psig increase in the allowable value to ≤ 1093 psig is acceptable as well as the new increased trip setpoint.

4. In the Bases for Section 2.1.1 on Thermal Power, change the value on fuel bundle radical peaking factor at 25 percent thermal power from "greater than 3.0" to "approximately 3.0" because of the higher thermal power with power uprate. This is still significantly higher than the expected peaking factor and is acceptable.

5. In the Bases for the Reactor Protection System Instrumentation Setpoints, add a paragraph to 2.2.1.9 on Turbine Stop Valve - Closure and 2.2.1.10 on Turbine Control Valve Fast Closure to clarify that the anticipating scram function is not required when Thermal power is below 30 percent, since the turbine bypass valves can bypass up to 30 percent of the steam flow directly to the condenser to alleviate a potential pressurization transient. The added Bases also notes that the new analytical limit, used in the transient analyses, is 147.7 psig, which is equivalent to 30 percent rated thermal power under uprated power conditions. The added paragraphs are clarifications rather than changes to the present Bases and are acceptable.

6. Revise specification 4.1.5.C to require the Standby Liquid Control pumps to develop a discharge pressure greater than or equal to 1224 psig versus the current requirement of 1190 psi. The increased discharge pressure acceptance criteria is based on the increased reactor pressure with power uprate and takes into account that operating with increased core flow will result in additional friction losses through the core and a slightly larger core differential pressure (approximately 4 psi). The 34 psig increase in Standby Liquid Control pump test discharge pressure acceptance criteria ensures that the pumps will inject sufficient sodium pentaborate into the core at the approximately 30 psig increased reactor pressure to bring the reactor subcritical. The increased acceptance criteria is acceptable.

7. TS 3.2.2 on Average Power Range Monitor (APRM) Setpoints contains the definition of "W" for the flow biased APRM scram equation. The word "rated" is being deleted from the definition of "W" since rated core flow is being increased. The definition of "W" is not altered. The change is being made for editorial purposes and is acceptable.

8. Action 6 in Table 3.3.1-1 on Reactor Protection System Instrumentation is being revised to clarify the current requirements. The revision does not change the intent. Action 6 currently reads: "Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first-stage pressure until the function is automatically bypassed within 2 hours." As noted in Item 5 above, the turbine bypass valves can bypass up to 30 percent of the steam flow directly to the condenser. The licensing basis analysis for the Minimum Critical Power Ratio (MCPR) operating limit (for the Generator Load Rejection Without Bypass) transient takes credit for operation of the anticipating scram on control valve fast closure at greater than 30 percent of rated thermal power. The revision to Action 6 clarifies that the action only applies when the Reactor Protection System (RPS) scram functions and End-of-Cycle Recirculation Pump Trip (EOC-RPT) on turbine main stop valve closure or control valve fast closure are not automatically bypassed. The revised Action 6 reads: "Initiate a reduction in THERMAL POWER within 15 minutes and reduce THERMAL POWER to less than 30 percent of rated THERMAL POWER within 2 hours." The revisions to the action statement clarify the current requirements; they do not change their intent and are acceptable.

9. Note (j) in Table 3.3.1-1 on Reactor Protection System Instrumentation is being revised to increase the scram bypass limit to 136 psig from 108 psig to reflect the higher steam pressure with power uprate. The setpoint change is related to Item 8 above. Setting the value of first-stage turbine pressure at 136 psig ensures that the analytical limit of 147.7 psig, which represents 30 percent rated thermal power, is not exceeded.

   The proposed revision to Table 3.3.1-1, Note (j), and Table 3.3.4.2-1, Note (b), does not change the operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30 percent power. The turbine first stage pressure switches will still be calibrated in the same manner, and, by procedure, the reactor operator will not exceed 30 percent power if the trip bypass annunciator does not clear.

   The setpoints for the RPS and EOC-RPT bypass functions were selected to allow sufficient operating margin to avoid scrams during low power turbine generator trips. This small absolute setpoint increase maintains the safety basis for the setpoint and is acceptable.

10. In Table 3.3.2-2, the main steam line flow high differential pressure setpoint is being changed from ≤ 107 psid to ≤ 113 psid and the allowable value is being changed from ≤ 110 psid to ≤ 121 psid to reflect the higher steam line pressure with power uprate. Footnote "**" is being added to Table 3.3.2-2 to indicate that these values will be confirmed during the power uprate startup testing. If revisions to the setpoint and allowable value are required, they will be forwarded to the NRC for approval within 90 days of the completion of the test program.

    The main steam line flow high differential pressure setpoint changes reflect the redefinition of rated main steam line flow that occurs with power uprate. The allowable value is maintained at the same percentage of rated steam flow as the differential pressure changes due to the increased uprate steam flow. The analytical limit of 140 percent of uprated steam flow is maintained for the uprated analyses. The relationship between the allowable value and the analytical limit was retained to ensure that a trip avoidance margin is maintained for the normal plant testing of MSIV's and turbine stop valves. The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for a main steam line break accident which satisfies the original intent of the design. The proposed main steam line flow high differential pressure setpoint changes are acceptable.

11. In Table 3.3.2-2, the Reactor Water Cleanup (RWCU) system flow-high isolation trip setpoint is being changed from 426 gpm to 462 gpm and the allowable value is being changed from 436 gpm to 472 gpm. RWCU flow is being increased by 10 percent to maintain reactor coolant water chemistry at the higher power level and increased core flow.

    The basis for the RWCU flow-high isolation is to ensure a RWCU System isolation in case of a pipe break. The high flow setpoint is set high

enough to avoid spurious trips from normal operating transients but low enough to ensure an isolation during a pipe break. The proposed TS limits will result in a negligible reduction in the margin between the RWCU isolation setpoint and the 4350 gpm flow postulated during an RWCU line break and will avoid spurious isolations. The proposed change in the trip setpoint maintains the original design intent with the 10 percent increase in the purification rate and is acceptable.

12. In Table 3.3.2-2, on Isolation Actuation Instrumentation Setpoints, the High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation (RCIC) steam line flow-high are being changed to account for changes in steam conditions and flows that result from operation at uprated conditions. For the RCIC system, the trip setpoint and allowable value for the high delta pressure in the steam line are being increased to less than or equal to 138" H_2O and less than or equal to 143" H_2O, respectively. The trip setpoint and allowable value for the HPCI steam line flow-high are being increased to less than or equal to 387" H_2O and less than or equal to 399" H_2O, respectively. The setpoint and allowable value are set so that isolation occurs at greater than 272% normal steam flow and less than 300% steam flow. Setting the isolation at less than or equal to 300% of normal flow ensures that the isolation will occur if a steam line were to rupture.d

    The original setpoints were calculated using information obtained during the Susquehanna startup program. The revised setpoints and allowable values were calculated using the same startup data and adjusted for uprate conditions. The revised setpoints maintain the current design intent and are acceptable.

13. In Table 4.3.2.1-1, on Isolation Actuation Instrumentation Surveillance Requirements, footnote "**" is being revised to delete reference to reactor pressure. The original purpose of this footnote was to describe the functioning of the permissive circuitry that allowed the main steam isolation valves (MSIV) low condenser pressure isolation to be bypassed. In the startup phase of the Susquehanna units, GE deleted the reactor pressure setpoint input to the bypass circuitry. This change is being made to have the footnote conform to the installed configuration. This change is editorial in nature and is acceptable.

14. In Table 3.3.4.2-1, on End-of-Cycle Recirculation Pump Trip System Instrumentation, note "(b)" is being revised to specify that the EOC-RPT shall not be automatically bypassed when turbine first-stage pressure is greater than an allowable value of 136 psig for the reason stated in item 9, above. This setpoint provides adequate margin between the analytical limit of 147.7 psig, which represents 30 percent rated thermal power (under power-uprate conditions) to ensure that the trip is not bypassed above 30 percent power. This maintains the current design requirement under uprate conditions and is acceptable.

15. In Table 3.3.6-2 (Page 3/4 3-54) on Control Rod Block Instrumentation

Setpoints, and Specification 3.4.1.1.2.a.6.a (Page 3/4 4-1c), on Single Loop Operation, the rod block monitor (RBM) flow biased rod blocks are being changed. In the table, item 1.a is being revised to change the trip setpoint and allowable value to less than or equal to 0.63 W + 41% and less than or equal to 0.63 W + 43%, respectively. In the new specification 3.4.1.1.2.a.6.a, the trip setpoint and allowable values will be less than or equal to 0.63 W + 35% and less than or equal to 0.63 W + 37%, respectively. The downward rescaling is made necessary by the re-definition of rated thermal power. These TS changes do not represent a change from current limits.

The RBM flow biased rod blocks are used in the Rod Withdrawal Error (RWE) analysis. In order to maintain Critical Power Ratio (CPR) margins similar to previous Susquehanna cycles, the flow biased rod blocks were changed in terms of megawatts thermal but the change was not appreciable. The rescaling of the RBM flow biased rod block to reflect the re-definition of rated thermal power maintains the same level of protection as previously provided. The proposed change to the RBM trip setpoints and allowable value maintain the current level of protection and are acceptable.

16. In Table 3.3.6-2, Control Rod Block Instrumentation Setpoints, item 2.a., the Average Power Range Monitor (APRM) rod block upscale value has been changed to add a high flow clamp setpoint at 108% of rated thermal power with a high flow clamped allowable value at 111%. The addition of the high flow clamp to the flow biased APRM rod block function maintains the normal margins between the rod block and the scram power levels in the increased core flow (ICF) regions. When the reactor core flow is greater than 100 million lbm/hr, the APRM clamp provides an alarm to help the operator avoid scrams while operating in the ICF region. The additional APRM trip provides an additional margin of safety in the ICF regions and is acceptable.

17. In Table 3.3.6-2, Control Rod Block Instrumentation Setpoints, item 6.a., the reactor coolant system recirculation flow upscale rod block trip setpoint and allowable value are being increased to 114/125 divisions of full scale and 117/125 divisions of full scale, respectively. The upscale rod block setpoint and allowable value are being increased to allow operation in the ICF region. The purpose of the Reactor Coolant System recirculation flow upscale rod block is to prevent rod movement when an abnormally high increase in reactor recirculation flow causes an increase in neutron flux that results in an increase in reactor power. However, this increase in neutron flux is monitored by the Neutron Monitoring System that can provide a rod block. No design basis accident or transient analysis takes credit for rod block signals initiated by the Reactor Coolant Recirculation System. The increase in the upscale trip setpoint from 108/125 divisions to 114/125 divisions of full scale and the increase in the allowable value from 111/125 divisions to 117/125 divisions is necessary to operate with increased core flow and is acceptable.

18. Surveillance Requirements 4.4.1.1.1.2 and 4.4.1.1.2.5 on the Reactor Coolant System are being revised to allow core flows in the ICF region of up to 108 million lbm/hr. The reactor recirculation pump motor generator set scoop tube electrical and mechanical overspeed stop setpoints are being increased to a core flow of 109.5 million lbm/hr. and 110.5 million lbm/hr., respectively. The electrical stop is maintained above the maximum operating core flow and below the mechanical stop. The 109.5 million lbm/hr. electrical stop setpoint, specified by General Electric, is based on BWR operating history. The electrical stop is a system design feature and is not used in any safety analysis. The 110.5 million lbm/hr. mechanical stop setpoint is used in transient analysis to limit core flow during a recirculation pump controller failure. The 110.5 million lbm/hr. mechanical stop setpoint, specified by General Electric, is also based on BWR operating history. The cycle specific analyses, performed for power uprate, used the 110.5 million lbm/hr. mechanical stop setpoint. The 110.5 million lbm/hr setpoint was used by the licensee in the Unit 2, Cycle 7, reload analysis and is acceptable.

19. Figure 3.4.1.1.1-1 on Thermal Power Stability Restrictions has been redrawn to reflect the new definition of Rated Thermal Power to retain the same stability operating restrictions in terms of megawatts thermal as currently prescribed by this graph. The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are redefined to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, power uprate has no detrimental effect on the level of protection provided by the TSs. The revised figure precludes operation in the region of potential thermal-hydraulic instability and is acceptable.

20. A new specification, 3.4.1.1.2.5 is being added to the Limiting Condition for Operation (LCO) on the Reactor Coolant System, Recirculation Loops - Single Loop Operation, to specify that a 0.70 Linear Heat Generation Rate (LHGR) multiplier has been added to Specification 3.2.4 when in single recirculation loop operation. Operation with one recirculation loop out of service is allowed, but is not considered a normal mode of operation. Single loop operation (SLO) is a special operational condition when only one of the two recirculation loops is operable. In this operating condition, the reactor power will be limited to less than 80 percent of rated by the maximum achievable core flow, which is typically less than 60 percent of rated core flow. A postulated LOCA (Loss of Coolant Accident) occurring in the active recirculation loop during SLO would cause a more rapid coastdown of the recirculation flow than would occur in two loop operation, where one active loop would remain intact. This rapid coastdown causes an earlier boiling transition and deeper penetration of boiling transition into the bundle, which tends to increase the calculated PCT (Peak Clad Temperature). However, the PCT effects of early boiling transition are substantially offset by the mitigating effect of the lower power level achievable at the start of such an event. An LHGR reduction (multiplier) of 0.70 will be imposed when the plant is in SLO. The SLO results are less limiting (i.e., lower PCT's) than the results for the two

loop DBA LOCA. Thus, the licensing PCT is based appropriately on two loop operation rather than SLO. As discussed in Section 3.3.3 of the staff's safety evaluation of November 30, 1993, the licensee used the staff-approved SAFER/GESTR (S/G) methodology to assess the Emergency Core Cooling System (ECCS) capability for meeting the 10 CFR 50.46 criteria. The addition of an LHGR reduction of 0.70 when the plant is in SLO provides an additional margin of safety and is acceptable.

21. Footnote **** to Specification 4.4.1.1.2.7 on the Reactor Coolant System is being changed to reference the power uprate startup test program as distinguished from the initial startup test program when the unit was first licensed. This footnote provided a mechanism for changing the power limits specified if the results of the initial startup test program determined that it was necessary. The footnote is being modified to allow operation at uprated power with the present power limits. Should the power uprate startup test program determine a need to change the power limits, they will be submitted to the Commission within 90 days as required by the revised footnote. This is consistent with the original BWR startup test program requirement and is acceptable.

22. Specifications 4.4.1.1.1.2, 4.1.1.2.5, 3.4.1.3, and Figure 3.4.1.1.1-1 specify performance requirements and limits for the Reactor Recirculation System. These specifications are referenced to 102.5 percent and 105 percent of the current rated core flow. The references to "rated core flow" are being replaced with actual equivalent core flows. As discussed in item 18 above, the electrical and mechanical stops will be set at 109.5 million lbm/hr. and 110.5 million lbm/hr., respectively. The specifications are equivalent and unchanged. This change is being made for editorial purposes to avoid confusion since rated core flow is being increased. These changes are also consistent with the changes made in Section 2.1. As discussed in the staff's safety evaluation of November 30, 1993, the staff evaluated operation of the Susquehanna units at increased core flows of up to 110.5 million lbm/hr and determined that the new mechanical and electrical setpoints were acceptable.

23. Specification 3.4.2, Reactor Coolant System, Safety Relief Valves (SRV) is being changed to reduce the number of setpoint groups from 5 to 3. Two valves will be set at 1175 psig plus or minus 1 percent, 6 will be set at 1195 psig plus or minus 1 percent and 8 will be set at 1205 psig plus or minus 1 percent. Also, the number of Operable safety valves are being increased from 10 to 12. The staff's assessment of the licensee's reactor overpressure protection analysis was discussed in Section 3.2.2 of the November 30, 1993, safety evaluation. The licensee's analysis showed that for the most limiting pressurization transient, Main Steam Isolation Valve (MSIV) closure with failure of the valve position scram, the peak pressure remained below the 1375 psig ASME limit and met all licensing requirements.

    The margin between peak allowable pressure and the maximum safety setpoints (1205 psig $\pm$ 1 percent) is unchanged. The difference is that in

the present TSs, 3 of the 16 SRVs are set at 1205 psig, whereas with the power uprate TSs, 8 of the 16 SRVs will be set at 1205 psig. The licensee performed analysis on the effects of the setpoint changes for the design conditions and the emergency and faulted conditions. The increased RPV dome pressure does not affect the design condition and, therefore, stresses remain unchanged. With the changed setpoints, there will be reduction in the simmer margin which will be compensated for by more stringent leak test requirements during valve refurbishment. The Crosby SRVs used at Susquehanna have not had the problems of "weeping" associated with the Target Rock SRVs used at some other BWRs. Since the licensee's analysis demonstrates that reactor pressure will be limited to within ASME Section III allowable values for the worst case upset transient, the revised SRV lift settings are acceptable.

24. Specification 3.4.3.2.d, Reactor Coolant System, Operational Leakage, is being revised to indicate that the 1 gpm leakage rate limit currently applicable applies at the uprated maximum allowable pressure of 1035 psig. The steam dome pressure for leakage is being increased by 35 psig (reactor design pressure). This pressure is chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Increasing the leakage rate pressure to 1035 psig is consistent with the expected uprated operating pressure. Increasing the reactor steam dome pressure has been analyzed and found to be within allowable limits. Keeping the current 1 gpm leakage rate limit at the increased reactor system pressure is conservative and is acceptable.

25. In Specification 3.4.6-2 and 4.4.6.2, Reactor Coolant System, Reactor Steam Dome, the reactor steam dome pressure limits have been increased from 1040 psig to 1050 psig. Operating pressure for uprated power is increased by a minimum amount necessary to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1032 psig is expected and is being assumed in the transient analysis. The 1050 psig limit was chosen to maintain an adequate level of operating flexibility while maintaining an adequate distance from the high pressure scram for trip avoidance. This limit is the initial pressure value used in the overpressure protection safety analysis for power uprate, for which all licensing criteria have been met. The 10 psig increase in the steam dome pressure limit was discussed in the staff's safety evaluation of November 30, 1993, and is acceptable.

26. Specification 4.5.1.b.3, Emergency Core Cooling Systems, has been revised to specify a test line pressure for the flow surveillance of the HPCI system of greater than or equal to 1140 psig at nominal reactor operating

conditions. The staff's assessment of the HPCI system under power uprate conditions was discussed in Section 3.3.2 of the November 30, 1993 safety evaluation. As noted in item 25 above, the steam dome pressure at the uprated power is expected to be 1032 psig. The upper pressure limit is being set at 1050 psig. The licensee has proposed that HPCI test acceptance pressure be set at 1140 psig, approximately 100 psig above the expected steam dome pressure. The staff concludes that this test criteria will assure that the HPCI system will be able to inject the required 5000 gpm at the higher reactor operating pressures associated with power uprate. The proposed HPCI pump test criteria is acceptable.

27. In Bases Table B 3/4 4.6-1, the characteristics of the limiting plate material were revised per R.G. 1.99, Revision 2. The change is in accordance with Generic Letter 92-01 and is acceptable.

28. Specification 5.4.2 on Design Features, Reactor Coolant System, Volume, was revised to show that the nominal T_{ave} is being changed from 528 °F to 532 °F. This change is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure. The staff's assessment of the effect of 4 °F increase in average primary coolant temperature on stresses and fatigue usage factors was discussed in Sections 3.2.3, 3.2.4, and 3.2.5 of the November 30, 1993, safety evaluation. The effects of power uprate have been evaluated to ensure that the increase in system temperatures causes minor increases in thermal loadings on pipe supports, equipment nozzles, and in-line components. The results of the analyses show that at uprated conditions, all ASME components will satisfy design specification requirements and code limits when evaluated to the rules of Subsection NB-3600 of the ASME Boiler and Pressure Vessel Code Section III. The effects of thermal expansion as a result of power uprate were found to be insignificant. The slight increase in average coolant temperature is a consequence of the increase in reactor pressure. The increase in temperature results in no significant increase in thermal stresses and is acceptable.

29. In Table 5.7.1-1, Component Cyclic or Transient Limits, the design cycle or transient limit for the reactor was changed to raise the upper limit for a heat cycle from 546 °F to 551 °F. This change is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure. The purpose of this specification is to limit the number of heatup and cooldown cycles. The effects of power uprate have been evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. The analyses were performed for the design, normal, upset, emergency, and faulted conditions. The increase in the temperature limitation is not significant with respect to the affect it has upon the RPV and associated components. The staff's assessment of stresses and fatigue usage factor for the reactor vessel were discussed in Section 3.2.3 of the November 30, 1993 safety evaluation. The 5 °F increase in the upper transient limit was determined to not be significant and is acceptable.

30. Administrative Control Section 6.9.3.2 describes and lists topical reports that are used to determine core operating limits. Topical reports 15 through 19 are LOCA methodology reports and are being deleted. These reports describe Siemens LOCA methodology. The GE SAFER/GESTR LOCA methodology is being used for this uprated cycle. In addition, other minor methodology changes were made for power uprate transient analysis. GE topical report NEDC-32071P, PP&L topical report NE-092-001, and the NRC Safety Evaluation Report on PP&L power uprate licensing topical are added as Topical Reports No. 15, 16, 17, respectively. The referenced reports and safety evaluations have been previously approved by the NRC staff and are an acceptable basis for the Core Operating Limits Report.

The licensee's application was submitted on November 24, 1993. The Commission's safety evaluation was issued a week later on November 30, 1993. One of the TS changes proposed by the licensee was a revision to the list of topical reports on TS Page 6-20b approved by the NRC and which are the basis for the "Core Operating Limits Report." The licensee's proposed wording for Reference 17 was: "NRC SER on PP&L Power Uprate Ltr (later)." The NRC's safety evaluation was issued on November 30, 1993. The staff substituted "November 30, 1993," in place of "(later)." This change updates the TS submittal and is acceptable.

In the original license, NPF-22, issued on March 23, 1984, there was a typographical error in the first line of paragraph 2.C.(1) in that the "L" was omitted from PP&L. This error was corrected by this amendment with the licensee's concurrence and did not change the original no significant hazards consideration determination.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on March 18, 1994 (59 FR 12990). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission 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, (2) such

activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment:
Letter dated November 30, 1993, to R. G. Byram, PP&L, from T. Murley, NRC, transmitting safety evaluation.

Principal Contributors: W. Lefave
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Date: April 11, 1994

ATTACHMENT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 30, 1993

Docket Nos. 50-387
and 50-388

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

Dear Mr. Byram:

SUBJECT: LICENSING TOPICAL REPORT FOR POWER UPRATE WITH INCREASED CORE FLOW, REVISION 0, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (PLA-3788) (TAC NOS. M83426 AND M83427)

Your letter of June 15, 1992, submitted "Licensing Topical Report NE-092-001, Revision 0, for Power Uprate With Increased Core Flow," for Susquehanna Steam Electric Station (SSES), Units 1 and 2. The report was submitted to support future proposed amendments to the Units 1 and 2 licenses to permit a 4.5-percent increase in reactor thermal power and an 8-percent increase in core flow for each unit. Your initial submittal was revised and supplemented by letters of July 24, September 17, and December 18, 1992, and January 8, January 25, April 2, August 5, August 12, and September 29, 1993.

As discussed in the enclosed safety evaluation, we have concluded that the revised (Revision 2) licensing topical report adequately supports your proposed power uprate. We have also concluded that SSES, Units 1 and 2, can operate safely with the proposed 8-percent increase in core flow, the proposed 4.5-percent increase in reactor thermal power, the corresponding 5-percent increase in main turbine inlet steam flow, and the corresponding increases in flows, temperatures, pressures, and capacities required in supporting systems and components at these uprated conditions. However, authorization for any increase in reactor thermal power will be based on our review of the technical specifications you will submit when you submit the amendment application.

Sincerely,

Frank J. Miraglia

for Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING "LICENSING TOPICAL REPORT NE-092-001,

REVISION 0, POWER UPRATE WITH INCREASED CORE FLOW"

PENNSYLVANIA POWER & LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter of June 15, 1992, as revised and supplemented by letters of July 24, September 17, and December 18, 1992, and January 8, January 25, April 2, August 5, August 12, and September 29, 1993, the Pennsylvania Power & Light Company (PP&L or the licensee) requested approval of "Licensing Topical Report NE-092-001, Revision 0, Power Uprate With Increased Core Flow," for the Susquehanna Steam Electric Station, Units 1 and 2. The topical report describes the licensee's intention to change the licensed thermal power level of the reactor from the current limit of 3293 megawatts thermal (MWt) to an increased limit of 3441 MWt. This request is made in accordance with the generic boiling-water reactor (BWR) power uprate program established by General Electric Nuclear Energy (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter of September 30, 1991. This request is similar to a request made on September 24, 1991, by the Detroit Edison Company for the Fermi-2 facility.

2.0 DISCUSSION

By letter of June 10, 1991, GE submitted Revision 1 to "Licensing Topical Report (LTR) NEDC-31897P, Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" (Reference 1). In this LTR, GE proposed to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent. The LTR contained a proposed outline for individual license amendment submittals, as well as discussions of the scope and depth of reviews that would need to be performed and the methodologies that would be used in these reviews. By letter of September 30, 1991, the NRC issued a staff position concerning the LTR (Reference 2), which approved the proposed program, provided that individual power uprate amendment requests meet certain requirements contained in the document.

The generic BWR power uprate program was created to provide a consistent means for individual licensees to recover additional generating capacity beyond their current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level was generally based on the vendor-guaranteed power level for the reactor. The difference between the guaranteed power level and the

design power level is often referred to as "stretch power." Since the design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling system (ECCS), increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment, nor does it significantly impact the reliability of this equipment.

The licensee's topical report proposes to increase the current licensed power level of 3293 MWt to a new limit of 3441 MWt which represents an approximate 4.5-percent increase in thermal power with a corresponding 5-percent increase in rated steam and feedwater flows. The planned approach to achieving the higher power level consists of (1) an increase in the core thermal power level to increase steam production in the reactor; (2) an increase in feedwater flow corresponding to the increase in steam flow; (3) an 8-percent increase in maximum allowable core flow; and (4) operation of the reactor along extensions of current rod position/flow rate control lines. This approach is consistent with the generic guidelines for BWR power uprate presented in Reference 1 and approved by the staff. The increased core power will be achieved by utilizing a slightly flatter radial power distribution while maintaining the most limiting fuel bundles within their operating constraints. The operating pressure of the reactor will be increased approximately 30 psi to ensure satisfactory turbine pressure control and pressure drop characteristics with the increased steam flow.

3.0 EVALUATION

In its review of the Susquehanna power uprate topical report, the NRC staff used applicable rules, regulatory guides, Standard Review Plan (SRP) sections, and NRC staff positions regarding the topics being evaluated. Additionally, the staff evaluated the Susquehanna submittal for compliance with the generic BWR power uprate program as defined in Reference 1. Detailed discussions of individual review topics follow.

3.1 Reactor Core and Fuel Performance

The effect of power uprate was evaluated for potential impact on various areas related to reactor thermohydraulic and neutronic performance. These included changes to the power/flow operating map, core stability, reactivity control, fuel design, control rod drives, and scram performance. Additionally, the staff considered the impact of power uprate on reactor transients, anticipated transients without scram (ATWS), emergency core cooling system (ECCS) performance, and peak cladding temperature (PCT) for design-basis-accident (DBA) break spectra.

3.1.1 Fuel Design and Operation

The licensee has stated that no new fuel designs would be needed to achieve power uprate. This statement is consistent with information submitted by GE in LTR NEDC-31984P (Reference 3). Fuel operating limits, such as the maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR) for future fuel reloads will continue to

be met after power uprate. The methods used for calculating MAPLHGR and OLMCPR limits will not be changed by power uprate, although the actual thermal limits may vary between cycles. Cycle-specific thermal limits will be included in the plant Core Operating Limits Report (COLR).

3.1.2 Power/Flow Operating Map

The power/flow operating map described in the topical report includes operating domain changes for both uprated power and increased core flow operations. Specifically, the licensee proposed to permit plant operations within an operating domain consisting of an increased core flow (ICF) range and a revised Extended Load Line Limit Analysis (ELLLA). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100-percent rated power. The staff has concluded that the proposed extension of the power/flow operating map will not degrade plant operations.

3.1.3 Stability

The BWR Owners Group (BWROG) and the NRC continue to address methods to minimize the occurrence and potential effects of core power oscillations which have occasionally been observed for certain boiling-water reactor (BWR) operating conditions. Until this issue is resolved, the licensee has adopted the generic interim operating constraints proposed by GE. Existing plant procedures have been incorporated in accordance with NRC Bulletin 88-07 and Supplement 1 to that bulletin which restrict plant operation in the high-power/low-flow region of the power/flow operating map. Since plant operation after power uprate will primarily extend the power/flow map to a higher power level (with corresponding higher flow), the current restricted operation regions of the power/flow map will remain essentially unchanged, and operator actions upon entry into these regions will likewise remain the same. This is consistent with information presented in the generic evaluations provided by GE in Reference 3.

The restrictions recommended by NRC Bulletin 88-07 and Supplement 1 to that bulletin will continue to be followed by the licensee for uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWROG and the NRC. The staff considers this approach acceptable.

3.1.4 Control Rod Drives and Scram Performance

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated steam flow and dome pressure.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rods will insert more slowly because of the high pressure. As the scram continues, the reactor pressure

will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Therefore, an increase in the reactor pressure has little effect on scram time. The licensee stated that CRD performance during power uprate will meet current technical specifications requirements. The licensee will continue to monitor, by various surveillance requirements, the scram time performance as required in the plant technical specifications to ensure that the original licensing basis for the scram system is preserved.

For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The minimum drive water pressure for power uprate conditions is, therefore, 1325 psig. Recent operating data show a range of CRD pump discharge pressures from 1435 to 1455 psig. The licensee's calculations indicate that the CRD system insert and withdraw operations will be satisfactory with these discharge pressures.

The staff concludes that the CRD system will continue to perform all its safety-related functions at uprated power with increased core flow, and will function adequately during insert and withdraw modes.

The licensee assured the adequacy of the control rod drive mechanisms (CRDMs) in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 1971 Edition, through Winter 1972 Addenda (Reference 9). The limiting components of the CRDM were identified to be the indicator tubes. The maximum stress and fatigue usage factors are below the allowable limits, and provide safety factors of about 1.5 and 6.6, respectively, for the design-basis conditions.

The increase in the reactor dome pressure, operating temperature, and steam flow rate as a result of the power uprate are bounded by the conditions assumed in the GE generic guidelines for the power uprate. The increase in core flow rate has no adverse effects on the control rod drive mechanism (CRDM). The CRDM was originally evaluated for a normal maximum reactor dome pressure of 1045 psig, which is higher than the power uprate dome pressure of 1035 psig.

On the basis of its review, the staff concludes that the CRDM will continue to meet its design-basis and performance requirements at uprated power conditions.

3.2 Reactor Coolant System and Connected Systems

In reviewing the mechanical engineering portions of the Susquehanna power uprate topical report, the staff focused on the effects of power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components.

GE based its generic guidelines for BWR power uprate effects on a 5-percent higher steam flow, an operating temperature increase of 5 °F, and an operating

pressure increase of 40 psig or less. For Susquehanna, the maximum reactor vessel dome pressure increases from 1005 psig to 1035 psig (30 psi), the dome temperature increases from 547 °F to 550.5 °F (3.5 °F) and the steam flow rate increases from 13.483 million pounds-mass per hour (lbm/hr) to 14.139 million lbm/hr (approximately 5%). The maximum core flow rate increases from 100 million lbm/hr to 108 million lbm/hr (8%) for the Susquehanna power uprate conditions, while GE generic guidelines assumed no change in core flow.

3.2.1 Nuclear Steam Pressure Relief

The nuclear boiler pressure/relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) and the high-pressure reactor scram offer this protection. The changes in the nuclear system pressure relief for power uprate are increases in the SRV setpoints (as described below), and a decrease in the number of valve groups from five to three.

The operating steam dome pressure is defined in order to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to uprated power. The uprate dome pressure increase will require an increase in the SRV setpoints. The appropriate increase in the SRV setpoints also ensures that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in the number of unnecessary SRV actuations.

3.2.2 Reactor Overpressure Protection

The results of the overpressure protection analysis are cycle specific and will be incorporated in the Core Operating Limits Report. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The ASME Code-allowable peak pressure for the reactor vessel is 1375 psig (110% of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is an MSIV closure with a failure of the valve position scram. The MSIV closure will be analyzed by the licensee using the NRC-approved methods, with the following exceptions: (1) the MSIV closure event will be analyzed at 102 percent of the uprated core power and 108 million lbm/hr core flow and (2) the maximum initial reactor pressure will be assumed to be the technical specifications maximum value.

The number of SRVs which will be assumed to be out of service is based on the maximum allowed by the technical specifications. Uprated conditions will produce a higher peak pressure in the reactor pressure vessel (RPV), and with reduced valve grouping, the cycle-specific analysis must show that it remains below 1375 psig, which is the ASME Code limit.

3.2.3 Reactor Vessel and Internal Components

The licensee evaluated the reactor vessel and internal components considering load combinations that include reactor internal pressure difference (RIPD), loss-of-coolant accident (LOCA), safety-relief valve (SRV), seismic, annulus pressurization (AP), jet reaction (JR), and fuel lift loads.

The licensee evaluated LOCA loads such as pool swell, condensation oscillation (CO), and chugging for the Susquehanna power uprate with increased core flow, and found that the original LOCA analyses did not change because the LOCA dynamic loads for Susquehanna were defined on the basis of the Kraftwerk Union (KWU) test conditions which bound the power uprate blowdown conditions with respect to vent mass and energy flow rate, and suppression pool water temperature. The design-basis SRV containment dynamic loads that affect the reactor vessel and piping systems are defined in accordance with the original KWU-specified SRV load boundary specification. The licensee stated that there is adequate conservatism in the design-basis loads to accommodate the slight increase in reactor pressure due to the power uprate. The licensee's review of the original analyses for the AP and jet loads indicated that the assumptions, analysis methodologies, and input parameters are conservative for the power uprate. On this basis, the staff concurs with the licensee's evaluation that the LOCA, SRV, AP, and jet design-basis loads remain bounding for power uprate with increased core flow.

In analyzing the potential for lifting of fuel as a result of the power uprate with increased core flow, the licensee considered load combinations that include SRV, LOCA, AP, JR, pipe restraint, seismic loads, and scram loads for the power uprate conditions. On the basis of its review, the staff concludes that the potential increase in fuel lift due to the power uprate is negligible.

The licensee evaluated stresses and fatigue usage factors for reactor vessel components in accordance with the requirements of the 1968 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB, 1968 Edition through Summer 1970 Addenda (Reference 8), to ensure compliance with the original code of record for Susquehanna, Units 1 and 2. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. A limiting fatigue usage factor of 0.92 was calculated for the reactor vessel head flange region for 40 years of operation based upon the uprated power level. There were no new assumptions used in the analysis for the power uprate conditions from those utilized by the licensee in previous evaluations. On the basis of the staff's review, the maximum stresses and fatigue usage factor, as provided by the licensee, are within the code's allowable limits and are, therefore, acceptable.

The licensee assessed the effects of increased core flow on flow-induced vibration by reviewing the startup test data for the valid prototype plant in comparison with the Susquehanna power uprate condition. The licensee stated that 113 percent of rated core flow, versus 108 percent of core flow for the power uprate, was tested and that the measured data do not show any indicat

of potential fluid-elastic instability. Therefore, the staff concurs with the licensee's assessment that the reactor internal response to flow-induced vibration will remain within acceptable limits.

3.2.4 Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with allowance for increased core flow (ICF). The cycle-specific core reload analyses in the Core Operating Limits Report will consider the full core flow range, up to 108 million lbm/hr. In evaluating the performance of the reactor recirculation system at uprated power with ICF, the licensee determined that the core flow can be maintained.

The cavitation protection interlock will remain the same in absolute thermal power, since it is based on the feedwater flow rate. These interlocks are based on subcooling in the external recirculation loop and thus are a function of absolute thermal power. With power uprate, slightly more subcooling occurs in the external recirculation loop as a result of the higher RPV dome pressure. It would, therefore, be possible to lower the cavitation interlock setpoint slightly, but this change would be small and is not necessary.

An evaluation by the licensee of recirculation pump net positive suction head (NPSH) found that at full power, power uprate alone (i.e., without an increase in core flow) does not increase NPSH required (NPSHr), and that the secondary effect of the 30-psi increase in RPV pressure increases NPSH available (NPSHa), so that power uprate alone increases the NPSH margin.

Increased core flow both increases NPSHr and reduces NPSHa, and thereby reduces the NPSH margin. Despite this reduction, NPSHa will remain at least three times the NPSHr with uprated power, with power uprate and increased core flow, or with increased core flow alone.

The licensee reviewed the recirculation drive flow stops for application to uprated power and ICF conditions. Because of the increase in core flow (up to 108 percent of rated), the recirculation pump motor-generator set scoop tube electrical and mechanical stops will be adjusted upward from 102.5 percent and 105 percent of 100 million lbm/hr, respectively, to 109.5 percent and 110.5 percent of 100 million lbm/hr.

An estimate by the licensee of the required pump head and pump flow indicates that the power demand of the recirculation motors increases up to 2.5 percent with power uprate, and up to 30 percent with both increased core flow and power uprate. These increases are within the capability of the recirculation system. The licensee has committed to provide a startup test plan with the proposed license amendment application. The staff expects that the recirculation flow control system will be tested as part of this startup test plan and will review the test program in conjunction with review of the amendment application.

3.2.5 Reactor Coolant Piping

The licensee evaluated the effects of the power uprate conditions, including higher flow rate, temperature and pressure for thermal expansion, dynamic loads, and fluid transient loads on the Class 1 reactor coolant pressure boundary (RCPB) piping systems, including such in-line components as equipment nozzles, valves and flange connections, and pipe supports. The licensee performed the evaluation in accordance with requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB-3600, 1971 Edition through Winter 1972 Addenda (Reference 9).

The licensee stated that stresses and fatigue usage factors were calculated for the power uprate, based on Equations 9 through 14 of the ASME Code (Reference 9), for the design, normal, upset, emergency, and faulted conditions. The revised stresses caused by power uprate were compared with the code allowables for acceptability. The licensee concluded that the code requirements are satisfied for the evaluated piping systems and that power uprate will not have an adverse effect on the Class 1 piping system design.

The licensee evaluated pipe supports, equipment nozzles, and in-line components by comparing the increased piping interface loads on the system components due to the power uprate thermal expansion, with the margin in the original design-basis calculation. The licensee concluded that sufficient margin exists between the original design stresses and the code limits to accommodate the stress increase caused by the power uprate. The licensee also evaluated the effect of power uprate conditions on thermal and vibration displacement limits for struts, springs, and pipe snubbers, and found them acceptable. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified.

On the basis of its review of the licensee's submittal, the staff concludes that the design of piping, components, and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop in the power uprate conditions.

3.2.6 Main Steam Isolation Valves

The performance of the main steam isolation valves (MSIVs) with regard to reactor coolant pressure boundary requirements, such as closure time and leakage, could potentially be impacted by the increased reactor operating pressure. However, the pressure increase is relatively small (less than 3%) and MSIV performance will be monitored by surveillance requirements in the plant technical specifications to ensure that the original licensing basis for the MSIVs is preserved.

Increased core flow alone does not change the conditions within the main steam lines, and thus cannot affect the MSIVs. Performance will be monitored by surveillance requirements in the technical specifications to ensure that the original licensing basis for MSIVs is preserved.

3.2.7 Balance-of-Plant Piping

The licensee evaluated the balance-of-plant (BOP) piping systems by comparing the original design-basis conditions with those for the proposed uprated conditions and by performing stress analyses in accordance with requirements of the code and the code addenda of record under the power uprate conditions. The BOP piping systems were determined from the uprated reactor and BOP heat balances. These systems include lines that are affected by power uprate, but not evaluated in Section 3.5.1 of the letter (Reference 13), such as main steam bypass lines, reactor feed pump turbine lines, and SRV discharge lines.

On the basis of a review of the existing design-basis calculation, the licensee determined that a majority of the BOP systems were originally designed to maximum temperatures and pressures that bounded the increased operating temperature and pressure due to the power uprate, and, therefore, are acceptable.

For the other portions of systems whose design temperature and pressure did not envelope the conditions of uprated power, the licensee performed stress analyses based on the power uprate conditions, and concluded that the actual calculated pipe stresses and support loads remained within the code-allowable limits.

The licensee evaluated the original pipe break analyses in accordance with the Standard Review Plan Section 3.6 guidance based on the revised fatigue analysis and concluded that the existing postulated break configurations and locations in these systems were not affected. No new postulated pipe break locations were identified in any system evaluated.

On the basis of its review of information submitted by the licensee, the staff concurs with the licensee's evaluations and concludes that the BOP systems will operate at the proposed power uprate conditions without adverse effects on the piping system and pipe supports.

3.2.8 Reactor Core Isolation Cooling System

The reactor core isolation cooling system (RCIC) provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low-pressure core cooling system. The licensee evaluated the RCIC system, and it is consistent with the bases and conclusions of the generic evaluation. In response to a staff request, the licensee indicated in a letter of January 25, 1993, that the recommendations of GE Service Information Letter (SIL) No. 377 have been implemented on the RCIC system at each Susquehanna unit. The staff noted that instead of adding a startup bypass line, the licensee chose to modify the control circuit of the RCIC steam admission valve. This modification is intended to achieve the turbine speed control/system reliability desired by SIL 377, and is consistent with the requirements in the staff's safety evaluation report (SER) of the generic topical report. The purpose of the modification is to mitigate the concern that a slightly higher

steam pressure and flow rate at the RCIC turbine inlet will challenge the system trip functions, such as turbine overspeed, high steam flow isolation, low pump suction pressure, and high turbine exhaust pressure. The licensee also plans to perform startup testing on the RCIC system during the initial startup after being licensed at uprated power. Further details of the startup testing plan will be submitted with the proposed license amendment. The staff requires that the licensee provide assurance that the RCIC system will be capable of injecting the design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, the licensee must also provide assurance that the reliability of this system will not be decreased by the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads. This may be done during startup testing following implementation of the power uprate.

3.2.9 Residual Heat Removal System

The residual heat removal (RHR) system is designed to restore and maintain the coolant inventory in the reactor vessel and to perform primary system decay heat removal following reactor shutdown for both normal and postaccident conditions. The RHR system is designed to operate in the low-pressure coolant-injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on these operating modes (except for LPCI which is discussed in 3.3.2.2) are discussed in the paragraphs that follow:

(1) Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. This increased time is judged to be insignificant.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," requires demonstration of cold-shutdown capability (200 °F reactor fluid temperature) within 36 hours. The Final Safety Analysis Report (FSAR), Section 15.2.9, indicates that cold shutdown can be reached in a much shorter time even considering the availability of only one RHR heat exchanger. For power uprate, licensee analysis of the alternate path for shutdown cooling based on the criteria of Regulatory Guide 1.139 shows that the reactor can be cooled to 200 °F in 28 hours, which meets the 36-hour criterion.

(2) Suppression Pool Cooling Mode

The functional design basis for suppression pool cooling mode (SPCM) stated in the FSAR is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate, since the peak suppression pool temperature analysis by the licensee confirms that the pool temperature will stay below its design limit at uprated conditions.

(3) Containment Spray Cooling Mode

In the containment spray cooling mode, water is pumped from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during postaccident conditions. Power uprate increases the containment spray temperature by only a few degrees. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure since these reach peak values before the actuation of the containment spray.

3.2.10 Reactor Water Cleanup System

The operating pressure and temperature of the reactor water cleanup (RWCU) system will increase slightly as a result of power uprate. The licensee has evaluated the impact of these increases and has concluded that the power uprate will not impair the integrity of the RWCU system. The cleanup effectiveness of the RWCU system may be slightly diminished as a result of increased feedwater flow to the reactor; however, current technical specifications limits for reactor water chemistry will not be changed as a result of a power uprate. Therefore, the power uprate will not significantly impact the operation or coolant boundary integrity of the RWCU system.

3.3 Engineered Safety Features

The staff reviewed the impact of the power uprate on containment system performance, the standby gas treatment system, (due to increased iodine loading), post-LOCA combustible gas control, the main steam isolation valve leakage control system, the control room atmosphere control system, and the emergency cooling water system. This review was performed to ensure that the ability of these systems to perform their safety function when responding to or mitigating the effects of design-basis accidents was not impaired by the approval of power uprate. Additionally, the effects of power uprate on high-energy line breaks, fire protection, and station blackout were considered.

3.3.1 Containment System

Primary containment temperature and pressure response following a postulated LOCA is important when determining the potential for offsite release of radioactive material, in determining ECCS pump NPSH requirements, and in determining environmental qualification requirements for safety-related equipment located inside the primary containment. Short-term and long-term containment analyses results are reported in the FSAR following a large break inside the drywell. The short-term analysis is directed primarily at determining the peak drywell pressure responses during the initial blowdown of the reactor vessel inventory to the containment following a DBA LOCA. The long-term analysis is directed primarily at determining the peak pool temperature response. The licensee indicated that the analyses were performed in accordance with Regulatory Guide 1.49 and Reference 1.

The effect of power uprate on the events which yield the limiting containment pressure and temperature response is evaluated in the following sections:

(1) Long-Term Suppression Pool Temperature Response

Bulk Pool Temperature

The licensee stated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA at 102 percent of the uprated power by using the SHEX computer code. The SHEX code utilizes more refined models than are used by the M3CPT/HXSIZ code in the original analysis to determine the suppression pool temperature. The SHEX code is capable of modeling containment response to more accident scenarios than the HXSIZ code. Many of the models used in the SHEX code are the same as, or very similar to, those used in the M3CPT code to calculate the short-term containment temperature and pressure response following a LOCA.

In a July 11, 1992, safety evaluation regarding GE Licensing Topical Report NEDC-31984P (Reference 3), the staff stated that although the SHEX code was not yet formally approved on a generic basis, use of the SHEX code in place of the M3CPT/HXSIZ code would be acceptable on a plant-specific basis, if adequate information is given to justify its use. In a letter of July 13, 1993 (Reference 12), the staff confirmed and clarified its position regarding the use of SHEX and the ANSI/ANS 5.1-1979 decay heat source term in containment response analyses for BWRs.

The licensee has submitted the results of three analyses performed using the SHEX code.

One analysis was performed to show how the current analysis methods compare with the analysis methods used in the FSAR. This case was evaluated with input parameters that match those used in the FSAR analysis as closely as possible. This analysis predicted a peak suppression pool temperature of 209 °F, while the original FSAR analysis had predicted 208.2 °F. Since SHEX and M3CPT/HXSIZ predicted essentially identical peak suppression pool temperatures, the use of SHEX for the analysis of long-term suppression pool response at power uprate is acceptable for Susquehanna.

The second analysis was performed to demonstrate the effects of power uprate with no other changes. This analysis predicted a long-term peak pool temperature of 211 °F.

The third analysis was performed at uprated power and included updated plant parameters. In this analysis, the licensee has updated the long-term cooling parameters for RHR service water temperature, RHR heat exchanger K-factor (Susquehanna design factor in lieu of generic BWR-4 factor), and ECCS pump heat (does not include HPCI system heat, because HPCI does not operate long term for a DBA LOCA). This analysis predicted a peak long-term suppression pool temperature of 203 °F, which remains below the suppression pool design temperature of 220 °F.

The licensee has also analyzed the highest bulk pool temperature response from an alternate shutdown cooling event according to Regulatory Guide 1.139 for power uprate assuming only one RHR heat-exchanger. This analysis predicted a peak bulk pool temperature of 208 °F, which remains below the design limit of 220 °F.

On the basis of its review (as discussed above), the staff concludes that the use of the SHEX code for calculating containment long-term peak bulk suppression pool temperature response is acceptable, and that the long-term peak bulk suppression pool temperature will remain acceptable after power uprate.

Local Pool Temperature Response With SRV Discharge

The licensee stated that the maximum local pool temperature with SRV discharge was previously calculated at 104.3 percent of current rated power to demonstrate compliance with NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments." This analysis predicted a peak local pool temperature of 212.2 °F. The above event was reanalyzed at 102 percent of uprated power as reported in the Susquehanna Design Assessment Report. This analysis predicted a peak local pool temperature of 214 °F, which remains below the NUREG-0783 local pool temperature design limit of 216 °F. Both analyses were performed for 90 °F initial pool temperature.

On the basis of these results, the staff concludes that the local pool temperature will remain acceptable after the power is uprated.

Containment Gas Temperature Response

The licensee stated that the design temperatures for the containment drywell and wetwell will not be affected by the power uprate. The containment drywell design temperature of 340 °F was based on a bounding analysis of the superheated gas temperature which can be reached with blowdown of steam to the drywell during a LOCA and predicted a maximum temperature of 318 °F. The licensee stated that since the vessel dome pressure of 1055 psia (1040 psig) assumed for the FSAR containment analysis bounds the power uprate vessel dome pressure of 1053 psia (1038 psig), the initial break flow rate for this event will not change and therefore, power uprate will have no impact on the containment drywell design temperature.

The licensee also stated that the wetwell gas space peak temperature response is calculated assuming thermal equilibrium between the pool and wetwell gas space. Since the power uprate analysis has not changed the wetwell temperature response, it will have no effect on the wetwell space design temperature.

On the basis of this review, the staff concludes that the containment gas temperature response will remain acceptable after power uprate.

(2) Short-Term Containment Pressure Response

The licensee stated that the short-term containment response analyses were performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation line to demonstrate that operation with power uprate will not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and differential pressure between the drywell and wetwell occur. This analysis was performed at 102 percent of the uprated power level using the GE M3CPT computer code. The calculated maximum containment pressure at uprated conditions is 44.6 psig. This code was also used in the original FSAR case which predicted a maximum containment pressure of 40.5 psig. The Susquehanna containment was designed for a maximum pressure of 53 psig.

The licensee also performed three analyses using the updated methods described above (see "Bulk Pool Temperature"). One was performed at current power level and predicted a maximum pressure of 43.4 psig. The updated methods show an increase of 2.9 psi in peak drywell pressure at current power due to the different assumptions (shorter MSIV closure time and use of the Moody slip flow model with different subcooled flow assumptions) used in performing the evaluations. The second analysis was performed at uprated power which predicted a maximum pressure of 44.6 psig. The third analysis was also performed at uprated power level in which the long-term cooling parameters were also updated. This analysis predicted no change in the maximum pressure of 44.6 psig because the additional parameter changes only affect the long-term results, and do not affect the peak containment pressure which is a short-term response.

On the basis of its review, the staff concludes that the pressure response following a postulated LOCA will remain acceptable after power uprate.

(3) Containment Dynamic Loads

LOCA Containment Dynamic Loads

NEDC-31897 (Reference 1) requires that the power uprate applicant determine if the containment pressure, temperature, and vent flow conditions calculated with the M3CPT code for power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads were based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by power uprate, and thus do not require further analysis.

The licensee stated that the results of the short-term LOCA containment pressure and temperature analysis were used to evaluate the LOCA dynamic loads such as pool swell, vent thrust, condensation oscillation, and chugging. The change in the short-term containment response with power uprate is small, and the loads remain bounded by the test conditions used to define the original loads except the pool swell loads. The licensee reported that a detailed evaluation of the wetwell components within the pool swell zone has been

completed. The power uprate loads and stresses were compared to the component's allowable values to determine component qualification. This comparison showed that all wetwell components within the pool swell zone are qualified for the power uprate pool swell loads. On the basis of its review, the staff finds that the LOCA dynamic loads and their effect on the components qualification at power uprate are acceptable. The staff is currently reviewing the effect of LOCA dynamic loads on spent fuel pool cooling components in response to the November 27, 1992 10 CFR Part 21 report described in section 3.5.1. NRC conclusions on the acceptability of LOCA dynamic loads on SFP cooling systems will be documented in separate correspondence.

SRV Containment Dynamic Loads

Safety-relief valve (SRV) containment dynamic loads include loads on the SRV discharge lines, quenchers and quencher supports, pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by SRV opening setpoints, discharge line configuration, and suppression pool configuration. Of these parameters, only the SRV setpoint is affected by power uprate. NEDC-31897 states that if the SRV setpoints are increased, the power uprate applicant will show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

The licensee stated that the original SRV load specification was based on a maximum reactor pressure of 1276 psig, and that the original SRV load specification has adequate conservatism to accommodate the slight increase in reactor pressure. The results of the reanalysis indicate that the loads remain below their design-allowable values and are not affected by power uprate.

Subcompartment Pressurization

A postulated pipe break in the annulus region between the reactor vessel and biological shield wall produces asymmetric pressure loads on the vessel, attached piping, and biological shield wall. The licensee stated that the original pressure loads were calculated using conservative mass and energy release rates and a computer code that predicted conservative pressure responses within the annulus and that a review of the original pressure load analysis has verified that adequate margin exists to accommodate the slight increase in reactor pressure due to power uprate. The staff agrees with the licensee's position that subcompartment pressurization effects will remain acceptable for uprated power.

(4) Containment Isolation

The licensee stated that the containment isolation capability is not affected by power uprate. The peak drywell pressure resulting from power uprate remains bounded by the original design conditions. Therefore, containment isolation valves and actuators will meet closure and leakage requirements at uprated containment pressure, temperature, and flows. On the basis of its

review, the staff agrees with the licensee that the operation of the plant at the uprated power level will not impact the containment isolation system.

(5) Post-LOCA Combustible Gas Control System

The licensee stated that the Susquehanna units have nitrogen-inerted containments even though worst-case hydrogen concentrations for the original power level did not require inerting. The worst-case concentration at the original power level is 3.5 volume percent. The design-basis hydrogen will increase by about 4.5 percent with power uprate. The staff has verified that the hydrogen recombiners have sufficient capacity to accommodate this increased load.

On the basis of its review, the staff concludes that the power uprate will not impact the post-LOCA combustible gas control system.

3.3.2 Emergency Core Cooling Systems

The effect of power uprate and the increase in RPV dome pressure on each emergency core cooling system (ECCS) is addressed below.

As discussed in the FSAR, compliance with the NPSH requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig (no containment overpressure) and the maximum expected temperature of pumped fluids. The pumps are assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit. Assuming a LOCA occurs during operation at the uprated power, the suppression pool temperature will remain below its NPSH limit. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

(1) High-Pressure Coolant Injection (HPCI) System

The licensee evaluated the HPCI system and determined that operation of this system at uprated conditions will be consistent with the bases and conclusions of the generic evaluation. In response to a staff request, the licensee has reported, in a letter of January 25, 1993, that it had installed the modifications on the HPCI system on each unit in response to GE SIL 480. These modifications were performed during the Unit 1 fourth and Unit 2 third refueling outages. As discussed in Section 4.2 of the GE letter (Reference 3), the modifications will avoid the possibility of turbine overspeed trips at the higher reactor pressure associated with power uprate. The purpose of this modification is similar to that of the RCIC system as discussed in Section 3.8. The licensee also plans to perform startup testing on HPCI during the initial startup after being licensed at uprated power. Further details of the startup testing plan will be submitted with the proposed license amendment. The staff requires that the licensee provide assurance that the HPCI system will be capable of injecting the design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, the licensee must also provide assurance that the reliability of the HPCI system will not be decreased by the higher loads placed on the system

or because of any modifications made to this system to compensate for these increased loads. This may be accomplished during startup testing following implementation of power uprate.

(2) RHR System (Low-Pressure Coolant Injection)

The hardware for the low-pressure portions of the RHR is not affected by power uprate. The upper limit of the low-pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low-pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low-pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system does not experience different operating conditions upon power uprate, there is no impact from power uprate.

(3) Core Spray System

The hardware for the low-pressure core spray (CS) system is not affected by power uprate. The upper limit of the low-pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low-pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low-pressure ECCS will not be increased. Therefore, since these systems do not experience different operating conditions upon power uprate, there is no impact from power uprate. Also, the impact of power uprate on the long-term response to a LOCA will continue to be bounded by the short-term response.

(4) Automatic Depressurization System

The automatic depressurization system (ADS) uses safety/relief valves to reduce reactor pressure following a small-break LOCA and failure of the HPCI system to maintain reactor water level. This function allows low-pressure coolant injection (LPCI) and core spray (CS) to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for power uprate. Plant design requires a minimum flow capacity for the SRVs, and that ADS initiate after a time delay on either low water level plus high drywell pressure, or on low water level alone. The ability to perform either of these functions is not affected by power uprate. This assessment is based on the analysis of system response under various LOCA conditions presented in the GE report NEDC-32071P, "SAFER/GESTR-LOCA Report," which was submitted in the licensee's June 15, 1992, submittal (Reference 13).

3.3.3 Emergency Core Cooling System Performance

The emergency core cooling systems (ECCSs) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCAs) caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50 (Appendix K). The licensee analyzed the Siemens Nuclear Power (SNP) 9x9-2 fuel, used in Susquehanna Units 1 and 2, using NRC-approved methods.

The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

The licensee used the staff-approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-LOCA analysis for Susquehanna Units 1 and 2 was performed by the licensee with SNP 9x9-2 fuel in accordance with NRC requirements and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 and 10 CFR Part 50 (Appendix K). A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K peak cladding temperature (PCT) as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases. The Susquehanna-specific analysis was performed with a conservatively high peak linear heat generation rate (PLHGR) and a conservatively low minimum critical power ratio (MCPR). In addition, some of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The nominal (expected) PCT is below 1050 °F. The statistical upper bound PCT is below 1320 °F. The licensing basis PCT for Susquehanna is 1510 °F, which is well below the 10 CFR 50.46 PCT limit of 2200 °F. The analysis also meets the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in the PP&L Licensing Topical Report. Therefore, the ECCS/LOCA analysis contained in the topical report for Susquehanna, Units 1 and 2, meets the NRC S/G-LOCA licensing analysis requirements.

The licensee also reevaluated the ECCS performance for single-loop operation (SLO) using the S/G-LOCA methodology. The DBA size break is also limiting for SLO. Using the same assumptions in the S/G-LOCA calculation with no MAPLHGR reduction, yields a calculated nominal and Appendix K PCT of 1160 °F and 1661 °F, respectively. Since the PCT was below the 10 CFR 50.46 limit of 2200 °F, the licensee claimed that no MAPLHGR reduction is required for SLO. The staff asked the licensee to reconcile the fact that the S/G-LOCA analysis PCT results for SLO were higher than those presented for two-loop operation, and no statistical analysis of the upper bound PCT had been provided for this case. The licensee reviewed this staff question, and has proposed in a letter of April 2, 1993, to impose an LHGR reduction (multiplier) of 0.70 during SLO. On the basis of this reduction, the calculated SLO licensing basis PCT and upper bound PCT are lower than their respective values for two-loop operation. The proposed technical specification markup reflecting the LHGR reduction (multiplier) has been transmitted to the NRC in Reference 19 and will be incorporated in the licensee's proposed amendment application.

An S/G-LOCA analysis for the ELLLA region was performed by the licensee at a core flow of 87 million lbm/hr and uprated power for Susquehanna with SNP 9x9-2 fuel. A DBA recirculation suction-line break coincident with a false LOCA signal from the opposite unit was assumed. The results of the analysis show that early dryout of the high-power node would not occur and the MAPLHGR multipliers as a function of flow are not required. Consistent with the Appendix K licensing basis calculations performed by the licensee, the high-power node is assumed to experience early dryout for the Appendix K Extended

Load Line Limit Analysis (ELLLA). The nominal and Appendix K results both show a small increase in the PCT when compared to the base 100 million lbm/hr core flow cases; however, the PCT is still well below the 10 CFR 50.46 limit. The nominal and Appendix K values for the base case are 916 °F and 1499 °F, respectively, and for the ELLLA case they are 937 °F and 1514 °F, respectively. The increase in PCT for the ELLLA case is due to (1) the lower heat transfer rate during flow coastdown from the lower initial core flow; and (2) more subcooling in the downcomer which results in increased break flow and earlier core uncovery. No statistical upper bound PCT was provided for the ELLLA case. In response to a staff question to give an explanation for not providing the upper bound PCT for the ELLLA case, the licensee presented additional clarifying information in a letter of August 5, 1993 (Reference 20). The licensee stated that the upper bound PCT documented in NEDC-32071P is not based on ELLLA. If it were, the event would begin at a slightly lower core flow, but would otherwise be essentially the same. The licensee reported that the nominal PCT is only 21 °F higher when ELLLA is taken into account. The statistical uncertainties between the two cases do not change. Therefore, on the basis of the results reported in the submittal, the ELLLA case will not impact the 1600 °F limit on the upper bound PCT, nor the 2200 °F limit on the licensing basis PCT, and the licensing basis PCT will continue to be greater than the upper bound PCT. This explanation is acceptable to the staff.

The licensee also evaluated the applicability of the S/G-LOCA methodology to Susquehanna, Units 1 and 2, which operates with Siemens Nuclear Power (SNP) 9x9-2 fuel. The dimensions and characteristics of the SNP fuel are similar to those of GE fuels. The reactor and core response during a LOCA are not strongly dependent on fuel design. This is because for most BWRs, including BWR/4s (Susquehanna is a BWR/4), the core heatup, and corresponding PCT, occurs late in the event, well after the stored energy in the fuel is released. Hence, the PCT is more dependent on the decay heat power level and the heat transfer coefficient in the core. The maximum cladding temperature (or PCT) occurs during a period that is governed predominantly by steam cooling and eventually by core reflooding, both of which are well understood in fuel bundle geometries. The fuel-specific input geometry and characteristics for the SNP fuel were input directly into S/G-LOCA following the same procedures used for GE fuel. The results of the break spectrum analysis show that the large-break PCT was second-peak limited, i.e., late in the event following core uncovery, and that the PCT was similar to the second-peak PCT for the generic BWR/4 with GE fuel.

Since the geometry and characteristics of the SNP fuel used at Susquehanna are similar to GE fuels, and since the S/G-LOCA results for Susquehanna are similar to those of the generic BWR/4 S/G-LOCA analysis and also similar to those for a typical GE BWR/4 plant, the S/G-LOCA methodology is applicable to Susquehanna with SNP fuel.

3.3.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to ensure controlled and filtered release of particulates and halogens from primary and secondary

containment to the environment during abnormal and accident situations in order to maintain offsite thyroid doses within the 10 CFR Part 100 limits. The SGTS consists of two 100-percent-capacity, parallel, redundant flow trains. Each flow train consists of a mist eliminator, an electric air heater, a bank of prefilters, a high-efficiency particulate air (HEPA) (pre)filter, an upstream and downstream charcoal adsorber, a HEPA (after)filter, a vertical 8-inch-deep charcoal adsorber bed with fire-detection temperature sensors, a water spray system for fire protection, and one 100-percent-capacity exhaust fan. Each train is sized to change one secondary containment (SC) air volume per day while maintaining the SC at a slight negative pressure of 0.25-inch water gauge with respect to the outside atmosphere. Maintaining this negative pressure serves to prevent unfiltered release of radioactive material from the SC to the environment. The staff agrees with the licensee that the proposed slight increase in power (4.5%) by itself will not impair the capability of the SGTS to meet the design objective as stated above, since it does not change the ventilation design aspect of the SGTS.

The licensee stated that the proposed power uprate will increase the loss-of-coolant accident source term by 4.5 percent which will increase the loading on the SGTS filter trains by 4.5 percent. The staff recognizes that iodine loading in the filters will increase marginally (4.5%) due to the proposed power uprate. The SGTS design utilizes filters that meet the intent of Regulatory Guide (RG) 1.52 guidelines with respect to the design, testing, and maintenance criteria of enginered safety features (ESFs) grade filters. The staff notes that one of the criteria deals with the filter loading capability. Since the two SGTS filter trains have more than 500-percent excess capacity at the original power level, the licensee has determined that the slight increase (4.5%) in iodine loading will remain well below the original design capacity of the filters. The licensee stated and the staff agrees that even with a slight increase in the previously calculated limiting offsite thyroid dose due to the uprated power, filter design capacity will sustain the thyroid dose well below the 10 CFR Part 100 limit.

On the basis of these findings, the staff concludes that the uprated power level operation will not have any impact on the ability of the SGTS to meet its design objectives.

3.3.5 Other Engineered Safety Features Systems

(1) <u>Main Steam Isolation Valve Leakage Control System</u>

The licensee's containment analysis calculated that the peak post-LOCA pressures at uprated power conditions do not increase beyond the original design basis. On the basis of its review of those calculations, the staff agrees with the licensee's assertion that the operation of the MSIV leakage control system will not be affected by power uprate.

(2) Post-LOCA Combustible Gas Control System

In its submittal, the licensee confirmed the ability of the combustible gas control system (CGCS) to maintain oxygen and hydrogen concentrations within acceptable levels following a LOCA. This conclusion is consistent with that reached by GE in Reference 3. The licensee stated that although the amount of oxygen liberated by radiolytic decomposition of water is expected to increase slightly because of the power uprate, the expected concentrations are well within the capacity of the CGCS. The licensee also stated that hydrogen recombiners may need to be started sooner following a postulated LOCA after uprate; however, current procedures which direct control room operators to initiate the recombiners are based on combustible gas concentrations, not on a fixed time following a LOCA.

Additionally, the revised hydrogen generation calculations submitted by the licensee indicate that less hydrogen will be liberated due to corewide metal-water reactions than previously predicted. This slight decrease is primarily due to significantly lower predicted fuel cladding temperatures during a postulated LOCA. The decrease in expected PCT is a result of the use of more realistic calculational methods in the ECCS/LOCA analysis (see Section 3.3.3). On the basis of its review of the licensee's submittals, the staff concludes that the existing post-LOCA combustible gas control systems will continue to perform their design function after power uprate.

(3) Main Control Room Atmosphere Control System

The control room atmosphere control system (CRACS) is one of the control room habitability systems. The CRACS includes an emergency filtration system which in turn contains an emergency makeup air filter train and an emergency recirculation filter train. The emergency makeup air filter train filters the radioiodine and radioactive material in particulate form present in the outside makeup air intake during an emergency situation such as a design-basis accident (DBA). The emergency recirculation filter train filters a mixture of the control room recirculated air and already once-filtered outside makeup air.

The emergency filtration system is designed to maintain the control room envelope at a slightly positive pressure (1/8-inch water gauge) relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room during an accident situation. The system accomplishes this design objective by bringing in controlled and filtered outside air and filtering the recirculated air to keep the control room operator doses within the General Design Criteria (GDC) 19 limits during an accident. Since power uprate does not change the design aspect of the control room emergency filtration system, the staff concludes that the proposed uprate in power (4.5%) by itself will not cause a significant increase in unfiltered inleakage of contaminated outside air into the control room during an accident.

The staff recognizes that iodine loading in the makeup air filters and recirculation air filters will increase marginally (4.5%) under uprate conditions and has concluded that the control room emergency filtration system filters are designed, tested, and maintained in accordance with RG 1.52 guidelines. On this basis, the staff concludes that the filters will continue to be valid for the CRACS at uprated power operation.

On the basis of these findings, the staff concludes that the uprated power level will have little or no impact on CRACS meeting its design objectives.

3.4 Instrumentation and Control

The staff's evaluation of setpoint changes associated with power uprate was limited to those setpoint changes for instrumentation identified in the licensee's submittals. The staff has completed its review of GE Topical Report NEDC-31336P, "General Electric Instrument Setpoint Methodology," October 1986 (Reference 6) and has approved the application of these methods to plant-specific data within the limits stated in the topical report.

A review of the licensee's submittals indicates that GE performed plant-specific calculations for PP&L using methods recommended by the Instrument Society of America (ISA) as outlined in GE Topical Report NEDC-31336P.

The licensee is considering the following setpoint changes:

(1) Flow-biased simulated thermal power for two-Loop operation

Change trip from (0.58W + 59%) to (0.555W + 56.5%).

Change allowable value from (0.58W + 62%) to (0.555W + 59.4%).

(2) Flow-biased simulated thermal power for one-loop operation

Change trip from (0.58W + 54%) to (0.555W + 51.7%).

Change allowable value from (0.58W + 57%) to (0.555W + 54.6%).

(3) Reactor vessel steam dome pressure high

Change trip from 1037 psig to 1087 psig.

Change allowable value from 1057 psig to 1093 psig.

(4) Main steam high flow

Change trip from 107 psid to 113 psid.

Change allowable value from 110 psid to 121 psid.

(5) Rod block for two-loop operation

Change trip from (0.58W + 50%) to (0.555W + 47.9%).

Change allowable value from (0.58W + 53%) to (0.555W + 50.8%).

(6) Rod block for one-loop operation

Change trip from (0.58W + 45%) to (0.555W + 43%).

Change allowable value from (0.58W + 48%) to (0.555W + 46%).

(7) Turbine stop valve and turbine control valve fast closure scram bypass

The turbine first-stage pressure setpoint will be changed to reflect the expected pressure at the new 30-percent power point.

(8) Average power range monitor (APRM) rod block, APRM simulated thermal power high-power clamps, and APRM neutron flux scram

These setpoints will not be physically changed. However, the change in the definition of rated thermal power (from 3293 MWt to 3441 MWt) will result in an increase of approximately 148 MWt to each of these points.

To verify the results of licensee-sponsored calculations and to better understand the quantitative effects of the assumed instrument errors, the staff audited the calculations for the reactor vessel steam dome high-pressure trip, the main steam high-flow trip, and the APRM trips (both fixed and flow-biased). The review demonstrated that the instrumentation errors assumed in the analyses were conservative with respect to the manufacturers' ratings and that the methods of analysis generally conform to those described in Reference 6. Exceptions to the methods described in Reference 6 are based on plant-specific data and instrumentation calibration procedures. The staff also acknowledges that these changes represent more current knowledge than was available when the topical report was issued in 1986.

The proposed setpoint changes are designed to maintain the existing margins between the proposed operating conditions and the new trip points. The same margins to the new safety limits are also maintained. These new setpoints do not significantly increase the likelihood of a false trip or a failure to trip upon demand. Therefore, the staff finds the setpoint changes, as described in the licensee's submittals, to be acceptable for power uprate. The licensee has stated that some of the setpoints described in the topical report may be changed; any such changes will be evaluated with the licensee's technical specification amendment submittal.

3.5 Auxiliary Systems

3.5.1 Spent Fuel Pool Cooling

The fuel pool cooling and cleanup system (FPCCS) consists of fuel pool cooling pumps, heat exchangers, skimmer surge tanks, filter demineralizers, associated piping, valves, and instrumentation. The system is designed to cool the fuel storage pool water by transferring the decay heat of the irradiated fuel through heat exchangers to the service water system.

The licensee stated that the fuel pool storage capacity will not be changed for power uprate, and that the fuel pool cooling and cleanup system, its filter demineralizer system, the service water system, and the fuel pool cooling assist mode of RHR will not require modification. The licensee stated that cycle-specific calculations will ensure that cooling loads on the normal pool cooling system and fuel pool cooling assist mode of RHR will remain within their design capacities. The condensate system and the emergency service water (ESW) system will provide the necessary makeup flow to the fuel pool to maintain level if required, and normal makeup requirements are not significant. The licensee stated that power uprate will not adversely affect fuel pool water chemistry, and that the fuel pool cooling and cleanup system will be adequate for all required functions after power uprate.

On November 27, 1992, two former contract engineers for PP&L filed a 10 CFR Part 21 report contending that significant deficiencies exist in the design of the spent fuel pool cooling and cleanup system for Susquehanna, Units 1 and 2. The individuals asserted that the design of the FPCCS fails to meet numerous regulatory requirements, and that, following a design-basis LOCA, assuming the RG 1.3 source term for 100-percent core damage, or following a LOCA with an extended loss-of-offsite-power event, spent fuel pool cooling could not be restored and would lead to boiling of the water in the spent fuel pool. The NRC staff is reviewing the issues raised by the contract engineers, including the potential decay heat loads associated with the 4.5-percent increase in thermal power described in the topical report. Any issues raised by the contract engineers will be resolved separately from the staff's assessment of the power uprate amendment application.

3.5.2 Water Systems

The licensee evaluated the impact of power uprate on the various plant water systems, including the safety-related and non-safety-related service water systems, closed-loop cooling systems, circulating water system, and the plant ultimate heat sink. The licensee's evaluations considered increased heat loads, temperatures, pressures, and flow rates. The staff's review of these evaluations is discussed below.

(1) <u>Safety-Related Service Water Systems</u>

These systems include the emergency service water (ESW) system and the residual heat removal service water (RHRSW) system. All heat removed by these systems is rejected to the atmosphere via the ultimate heat sink (UHS). The staff's evaluation of the effects of uprated power level operation on each of these systems appears below.

<u>Emergency Service Water System</u>

The emergency service water (ESW) system removes heat from HVAC coolers, diesel generators, emergency core cooling (ECCS) and engineered safety feature (ESF) components, and other equipment required to operate under normal and accident conditions, including loss of offsite power (LOOP) and loss-of-

coolant accident (LOCA) conditions. The licensee revised the rated heat-removal capacities and flow requirements based on the effect of the increased design temperature on the system heat exchangers due to power uprate.

On the basis of its review, the staff finds that the ESW system heat exchangers can satisfy the uprated power cooling requirements at the new design temperature resulting from uprated power operation. The ESW system piping and components meet all their safety and design objectives at the uprated design temperature. Therefore, the staff concludes that the uprated power level operation has no impact on the ESW system operation.

Residual Heat Removal Service Water System

The residual heat removal service water (RHRSW) system provides a safety-related cooling water source for the RHR system under normal or postaccident conditions. The system pumps water from the ultimate heat sink (UHS) spray pond through the RHR heat exchangers and returns it to the pond via a spray network. The system may also be used to flood the reactor core or the primary containment following an accident.

The licensee stated that power uprate increases the heat loads on the RHRSW system proportional to the increase in reactor power level. The effect of higher UHS design temperature on the RHRSW system heat removal capacities has been considered in the RHR system and containment safety analysis and reviews and has been found to be acceptable.

On the basis of its review, the staff finds that the RHRSW system piping and components meet their safety and design objectives. In addition, the post-accident reactor core and containment flooding functions of the RHRSW are not affected by power uprate. Therefore, the staff concludes that the uprated power level has no significant impact on the RHRSW system design.

Ultimate Heat Sink

The ultimate heat sink (UHS) provides a safety-related cooling water source for the emergency service water (ESW) system and the residual heat removal service water (RHRSW) system during testing, normal shutdown, and accident conditions. The UHS consists of an 8-acre, 25-million-gallon concrete-lined spray pond. Following a design-basis accident (DBA), the UHS provides enough cooling water at or below the ESW and RHRSW design temperature for a minimum of 30 days without makeup.

The licensee stated that as a result of operation at the uprated power level, the post-LOCA UHS water temperature for minimum heat transfer (MHT) meteorology and system alignment will increase. The licensee performed an updated power uprate spray pond analysis for maximum water loss (MWL) conditions which revealed that the UHS contains sufficient water inventory to sustain a DBA for 30 days without makeup. The licensee determined that the revised UHS MHT and MWL analyses shows that the current technical specifications for normal operation maximum pond temperature and minimum water

level remain adequate to ensure that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions.

On the basis of its review, the staff agrees with the licensee's conclusion that the UHS design is adequate for the uprated power operation and no modification to the UHS system is required.

(2) Non-Safety-Related Service Water System

The service water (SW) system is designed to continuously supply cooling water to various heat exchangers in the turbine, reactor, and radwaste buildings during normal plant operation, and has no safety-related function.

The licensee stated that the service water system will support power uprate with no equipment or setpoint changes. The design SW heat load bounds the power uprate conditions. Therefore, the cooling tower is able to dissipate the service water heat load at uprated conditions without affecting the existing design service water temperature.

Since the SW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level operation to the SW system design and performance.

Reactor Building Closed Cooling Water System

The reactor building closed cooling water (RBCCW) system cools various auxiliary plant components in the reactor and radwaste buildings during normal and loss-of-offsite power (LOOP) conditions, and has no safety-related function. The licensee stated that the increase in heat load due to uprated power operation is insignificant to the RBCCW system design.

Since this system does not perform any safety function, the staff has not reviewed the impact of the uprated power level operation to the RBCCW system.

Turbine Building Closed Cooling Water System

The turbine building closed cooling water (TBCCW) system supplies cooling water to auxiliary plant equipment in the turbine building. The licensee stated that the increase in heat load from the equipment due to the uprated power level operation is insignificant and that the TBCCW system design cooling capacity will not be exceeded.

Since the TBCCW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level to the TBCCW system design and performance.

Gaseous Radwaste Recombiner Closed Cooling Water System

The licensee stated that the 4.5-percent power uprate will increase the heat loads from the offgas recombiner condenser, the steam jet condenser, and condensate cooler of the offgas system by the same percentage. The licensee performed an evaluation of the offgas system and determined that it will remain within its original design capacities. Therefore, the licensee concluded that the increase in offgas system heat load on the gaseous radwaste recombiner closed cooling water (GRRCCW) system is also within the original GRRCCW system capacity.

On the basis of its review, the staff concludes that the effect of uprated power operation on the GRRCCW system is negligible and that there is adequate operating margin for this system to perform at uprated power operation.

River Water Makeup

The river water makeup system consists of four river water pumps and their screens, the intake structure and pump house, piping, valves, and controls. It supplies raw water to compensate for cooling tower and spray pond blowdown and evaporation, and for makeup to the plant water treatment and storage systems.

The licensee determined that during the peak demand periods, power uprate will increase the maximum system design above the original system design capacity with three of the four river water pumps running. However, the licensee performed a preliminary evaluation which concluded that the fourth makeup pump can be operated to maintain sufficient margin without adversely affecting intake structure HVAC performance, electrical distribution, system piping, or traveling screen operation. The licensee intends to further evaluate and test the need for four-pump operation after power uprate.

Since the river water makeup system does not perform any safety function, the staff has not reviewed the impact of the uprated power level on the river water makeup system design and performance.

(3) Chilled Water Systems

Reactor Building Chilled Water (RBCW) System

The RBCW system supplies chilled water to various reactor building and drywell heating, ventilation, and air conditioning (HVAC) systems and equipment loads during normal plant operation. The RBCW system does not perform any safety function.

The licensee stated that the actual peak loads during hot weather conditions can exceed the original calculated design load on the RBCW system. In order to meet the peak heat load demands (even for current power rating), the licensee has developed and implemented system operating strategies, such as tandem chiller operation to address this situation. The licensee stated that

these system operating strategies will be implemented as necessary for power uprate conditions.

Since the RBCW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level to the RBCW system design and performance.

Control Structure Chilled Water System

This system serves the control structure HVAC system and is addressed in Section 3.5.4.

Radwaste Building Chilled Water System

This system serves the radwaste building HVAC system and is addressed in Section 3.5.4.

Turbine Building Chilled Water System

This system serves the turbine building HVAC system and is addressed in Section 3.5.4.

3.5.3 Standby Liquid Control System (SLCS)

The ability of the SLCS to achieve and maintain safe shutdown is not directly affected by core thermal power; rather, it is a function of amount of excess reactivity present in the core; and as such, is dependent upon fuel loading techniques and uranium enrichment. The SLCS system is designed to inject at a maximum pressure equal to that of the lowest safety/relief valve setpoint. The SLCS pumps are positive displacement pumps, and the small (29 psig) increase in the lowest safety/relief valve setting as a result of uprate will not impair the performance of the pumps. The staff concludes that the ability of the SLCS system to inject to the reactor will not be impaired by the power uprate.

The SLCS shutdown requirements for future operating cycles will be evaluated by the licensee on a cycle-specific basis.

3.5.4 Heating, Ventilation and Air-Conditioning (HVAC) Systems

The licensee evaluated the impact of higher process fluid temperature in piping for all HVAC systems including the drywell cooling system, reactor building HVAC system, control structure HVAC system, radwaste building HVAC system, turbine building ventilation system, engineered safeguards service water (ESSW) pump house heating and ventilation system, and the diesel generator building ventilation system.

The licensee indicated that adequate margin exists in the drywell cooling system and the reactor building HVAC system capacities to meet the additional heat loads imposed by power uprate and increased core flow. The licensee

performed an evaluation which confirmed that power uprate has no significant effects on the control structure HVAC, the radwaste building HVAC system, and the turbine building ventilation system. The licensee confirmed that the system functions and performance of the ESSW pump house heating and ventilation system and the diesel generator building ventilation and their interfacing and supporting systems will not be affected as a result of power uprate. The licensee stated that the uprated heat loads would have no impact on maintaining the design environmental temperature parameters for these systems.

On the basis of its review, the staff agrees with the licensee that uprated power level operation will have no impact to the plant HVAC systems.

3.5.5 Fire Protection

The licensee stated that the operation of the plant at the uprated power level does not affect the fire-suppression or fire-detection systems. There are no physical plant configuration or combustible load changes resulting from the uprate. The safe-shutdown systems and equipment used to achieve and maintain cold-shutdown conditions do not change, and are adequate for the uprated conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire-protection systems and analyses are not affected by power uprate.

On the basis of its review, the staff finds that the fire-suppression and fire-detection systems and their associated analyses are not affected by power uprate.

3.6 Power Conversion Systems

The steam and power conversion systems and associated components (e.g., the turbine/generator, condenser and steam jet air ejectors, turbine steam bypass, feedwater and condensate systems) were originally designed to utilize the energy available from the nuclear steam supply system and to accept the system and equipment flows resulting from continuous operation at 105 percent of the currently licensed rated power. Therefore, these systems will not be affected by power uprate.

On the basis of its review, the staff agrees that operation at uprated power should not have a significant impact on the steam and power conversion systems and associated components.

3.7 Radwaste Systems and Radiation Sources

The licensee evaluated the radiological impact of the proposed uprate to show that the applicable regulatory acceptance criteria continue to be satisfied for the uprated power conditions. In conducting this evaluation, the licensee considered the effect of the higher power levels on source terms, onsite and offsite doses, and control room habitability during both normal and accident conditions.

3.7.1 Liquid Waste Management

The licensee stated that influent to the liquid radwaste processing system would increase approximately 4.7 percent due to uprate. On the basis of plant experience obtained in 1991, the licensee has determined that the liquid radwaste system has sufficient capacity to handle the increased influent.

The licensee also noted that a 10-percent increase in activated corrosion products would be expected because of the power uprate, but that the total volume of processed waste would not be expected to increase appreciably. The licensee concluded, from reviewing plant operating effluent reports and after considering the expected slight increase in effluents as a result of power uprate, that the requirements related to 10 CFR Part 20 and 10 CFR Part 50 (Appendix I) will continue to be satisfied. Having reviewed available plant data and experience with previous power uprates, the staff concludes that the pwoer uprate will have no significant adverse effect on liquid effluents.

3.7.2 Gaseous Waste Management

The licensee noted that gaseous wastes generated during both normal and abnormal operation are collected, controlled, processed, stored, and disposed of by means of the gaseous waste processing treatment systems. These systems include the standby gas treatment system, the offgas recombiner system, and the ambient temperature charcoal treatment system, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airborne radioactive gases. On the basis of its review of available plant data and previous experience with other power uprates, the staff concludes that no significant adverse effect on airborne effluents will occur as a result of power uprate.

3.7.3 Radiation Sources in the Core and Coolant

Radioactive materials in the reactor core are produced in direct proportion to the fission rate. Thus, the expected increase in the levels of radioactive materials (for both fission and neutron activation products) produced are expected to increase by a maximum of 4.5 percent. The licensee noted that experience to date with operation of Susquehanna Units 1 and 2 indicates that concentrations of fission and activation products in the reactor coolant will not increase significantly above those currently experienced. Current experience with operation of the Susquehanna units indicates that both units operate well below the 0.1 Curie/sec design basis and that current offsite radiological release rates are well below the original design basis. On the basis of its review of available plant data and experience with previous power uprates, the staff concludes that no significant adverse effect on radiation sources in either the core or reactor coolant will occur due to power uprate.

3.7.4 Radiation Levels

The licensee considered the effects of power uprate on radiation levels in the Susquehanna facility during normal operation as well as during postaccident conditions. The licensee concluded that radiation levels from both normal operation and accident conditions could increase slightly. However, any such increase would be small and would be bounded by conservatisms in the original plant design and analysis. Further, the licensee noted that the calculated offsite radiological consequences are well below the regulatory limits given in 10 CFR Part 20 and 10 CFR Part 50 (Appendix I). On the basis of its review of plant data and previous experience with other power uprates, the staff finds that no significant adverse effect on radiation levels (either onsite or offsite) will result from the proposed power uprate.

3.8 Reactor Safety Performance Evaluations

3.8.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances in the plant, caused by a malfunction, a single failure of equipment, or a personnel error, are investigated according to the type of initiating event. The licensee will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients. The limiting events for the Susquehanna units were identified. The relatively small changes in rated power and maximum allowed core flow are not expected to affect the selection of limiting events. The following events will be explicitly evaluated for cycle-specific reload analyses:

(1) loss of feedwater heating
(2) feedwater controller failure (FWCF)
(3) generator load rejection without bypass (GLRWOB)
(4) turbine trip without bypass (TTWOB)
(5) rod withdrawal error
(6) recirculation flow controller failure/increase (RFCF)
(7) fuel loading error

The limiting events which establish the minimum critical power ratio (MCPR) operating limits are currently GLRWOB, FWCF, and RFCF. These events are expected to remain limiting. The licensing analyses will be performed by the licensee up to a maximum power level of 102 percent of the uprated power level, or 3510 MWt, to account for power uncertainty.

Parametric studies were conducted as part of developing the licensee's licensing methods. These studies lead to the following expectations. The GLRWOB delta CPR (critical power ratio) is determined on the basis of a parametric analysis up to the maximum power level, and the FWCF is analyzed as a function of power. Thus, the increase in core power only changes the maximum power level considered. The increased flow rate for the GLRWOB and the FWCF is expected to produce slightly higher delta CPRs. This expectation will be confirmed as part of the reload licensing analyses. The RFCF is

analyzed as a function of core flow. The effect of increased core flow on the RFCF event will be evaluated as part of the reload licensing analyses. In response to a question from the staff, the licensee, in a letter of August 5, 1993, has indicated that it has decided not to take credit for the flow-biased simulated thermal power trip in the RFCF analysis for power uprate.

The safety limit minimum critical power ratio (SLMCPR) is calculated by the licensee as part of the reload licensing analyses using the NRC-approved Siemens Nuclear Power (SNP) methodology. No change will be made to this methodology as a result of power uprate or increased core flow. The analysis plan proposed by the licensee is acceptable.

3.8.2 Design-Basis Accidents

The licensee reanalyzed a number of events to determine the whole-body and thyroid doses at the exclusion area boundary and in the low population zone. In evaluating the effects of power uprate on accident consequences, the licensee reanalyzed the loss-of-coolant accident, the main steamline break accident, the fuel handling accident, and the control rod drop accident. The analysis was performed based upon operation at 105 percent of uprated power, using current NRC-approved methodologies. The staff has reviewed the information submitted by the licensee, and concludes that the analyzed consequences of postulated accidents will remain well within the staff acceptance criteria and are, therefore, acceptable.

3.8.3 Anticipated Transients Without Scram (ATWS)

Although General Electric has performed generic bounding ATWS analyses, these analyses cannot be used for Susquehanna because the licensee: (1) uses non-GE fuel and (2) has taken exceptions to Revision 4 of the Emergency Procedure Guidelines (EPGs) for responding to ATWS, which are assumed in the GE generic analyses.

The licensee is currently performing a plant-specific analysis of the plant response to an ATWS under uprated conditions. The licensee will submit the results of this analysis with the proposed license amendment application in support of power uprate implementation. The results will also be included in the ongoing project to upgrade Susquehanna Emergency Operating Procedures. The staff will evaluate the plant-specific analysis when the licensee submits it.

3.8.4 Station Blackout

Per the NUMARC 87-00 methodology, Susquehanna is classified as a 4-hour-duration station blackout (SBO) plant based on an offsite power design characteristic group of "P1," an emergency AC power configuration group of "D", and a target emergency diesel generator reliability of 0.975. Power uprate conditions will not affect this 4-hour-duration classification.

The limiting parameters for SBO events lasting longer than 4 hours are water inventory for decay heat removal, class 1E battery capacity, compressed-air capacity, and the effects of loss of ventilation. Power uprate will result in more decay heat which will require a slightly larger water inventory. However, the current SBO analysis provides for adequate water inventory to meet the additional requirements of power uprate.

Class 1E battery capacity and the compressed-air system are unaffected by power uprate, and power uprate will not increase demand on these systems for SBO scenarios. The capacity of these systems will, therefore, remain adequate.

Power uprate will have a slight effect on loss of ventilation, since slightly more heat will be transferred to the containment. This will result in slightly higher compartment temperatures. The Compartment Transient Temperature Analysis Program (COTTAP) computer code developed by the licensee was run for the station blackout scenarios using revised heat inputs from major equipment affected by power uprate. It simulates the control room and reactor building thermal response under loss-of-HVAC conditions. The licensee stated that the results of this calculation show that the compartment temperatures only rise 2 or 3 °F as a result of power uprate, and that the temperatures during an SBO event will not exceed the 180 °F limit identified in Appendix F of NUMARC 87-00, Revision 1.

The equipment with revised heat inputs used for the power uprate SBO evaluations includes motors, electrical cabinets, piping, and such miscellaneous mechanical equipment as heat exchangers. The rest of the equipment whose heat load changes with power uprate, but that was not included in these calculations, adds very little to the heat loads already considered, and will not contribute significantly to the increase in compartment temperatures.

3.9 Additional Aspects of Power Uprate

3.9.1 High-Energy Line Break

The slight increase in the operating pressure and temperature caused by the uprated power condition results in a small increase in the mass and energy release rates following a high-energy line break (HELB). This results in a small increase in the subcompartment pressure and temperature profiles and a negligible change in the humidity profile. The licensee performed a reanalysis of high-energy line breaks for all systems currently evaluated in the FSAR. The licensee has reevaluated the HELB for the main steam system, high-pressure coolant injection system, reactor core isolation cooling system, reactor water cleanup system, and the residual heat removal system. As a result of this reevaluation, the licensee has concluded that the affected compartments that support the safety-related functions are designed to withstand the resulting pressure and thermal loading following an HELB at uprated power conditions. The staff has reviewed the results of the licensee's reanalysis and finds them acceptable.

The licensee is currently evaluating the calculations supporting the disposition of potential targets of pipe whip and jet impingement from the postulated HELBs to determine the effects of power uprate. The licensee expects the evaluation to yield results that confirm the adequacy of the existing design under power uprate conditions.

Since the licensee has not completed calculations supporting the disposition of potential targets of pipe whip and jet impingement from postulated HELBs to confirm the adequacy of the existing design under power uprate conditions, the staff has not reviewed the impact of the uprated power level operation on HELBs. The licensee stated that the results of its evaluation will be included with the proposed license amendment.

3.9.2 Moderate-Energy Line Break and Internal Flooding

The licensee determined that the existing moderate-energy piping experiences no appreciable pressure or temperature increases due to power uprate. The high-pressure, moderate-energy HPCI and RCIC pump discharge piping does experience a small increase in core injection mode pressure, but it is within the existing design pressure and their status as moderate-energy lines is not affected.

On the basis of its review, the staff concludes that the moderate-energy line break (MELB) water spray and flooding evaluation of the plant is not affected by the uprated conditions and is acceptable for uprated power operation.

3.9.3 Equipment Qualification

(1) Environmental Qualification of Electrical Equipment

The licensee evaluated safety-related electrical equipment to ensure qualification for the normal and accident conditions expected in the areas in which the equipment is located. For equipment located inside the containment, the licensee indicated that current accident and normal design conditions for temperature, pressure, and humidity are unchanged for power uprate. Accident and normal radiation levels increase in proportion to the increase in power. For equipment outside the containment, normal operational temperature, pressure, and humidity conditions are unchanged. However, accident temperatures increase less than 5 °F and pressures increase less than 1 psi. Normal operational and accident radiation levels increase in relationship to the increase in power.

On the basis of the evaluation, the licensee determined that no safety-related equipment was identified as unqualified for power uprate environmental conditions. The qualified life of certain equipment may be reduced, but a revised aging analysis will assure replacement before the equipment exceeds qualified life.

On the basis of its review, the staff finds the licensee's approach to qualification of safety-related electrical equipment for power uprate conditions acceptable.

(2) Environmental Qualification of Mechanical Equipment With Non-Metallic Components

The licensee stated that operation at the uprated power level is expected to increase the normal process temperatures by less than 6 °F. As in the case of electrical equipment, normal operational and accident radiation levels also increase slightly due to uprate.

The licensee stated that its reevaluation is not expected to identify any components which are unqualified for the uprated environmental conditions. The qualified life of certain equipment may be reduced, but a revised aging analysis will assure replacement before the equipment exceeds its qualified life.

On the basis of its review, the staff finds the licensee's approach to qualifying mechanical equipment with non-metallic components for power uprate conditions acceptable.

(3) Mechanical Component Design Qualification

Having reviewed the licensee's submittals, the staff finds that the original seismic and dynamic qualification of the safety-related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons:

(a) Seismic loads are unchanged by power uprate.

(b) The original LOCA load conditions bound the power uprate conditions as stated in Section 3.2.3.

(c) The slight increase (about 1 to 2 percent) in AP, JR and SRV loads as delineated in Section 3.2.3 has a negligible effect on equipment dynamic response.

(d) No new pipe break locations resulted from the uprated conditions.

3.10 Evaluation of Impact on Responses to Generic Communications

In Reference 3, GE assessed the impact of power uprate on licensee responses to generic NRC and industry communications. GE reviewed both NRC and industry communications to determine whether parameter changes associated with power uprate could potentially affect previous licensee commitments or responses. Of the large number of documents reviewed (more than 3000 items), GE found that only a small number were potentially affected by power uprate. The list of affected topics was then divided into those that could be bounded generically by GE, and those that would require plant-specific reevaluation.

The NRC staff audited the GE assessment in December 1991, and approved the assessment in Reference 4.

In addition to assessing those items requiring a plant-specific reevaluation, the licensee also reviewed the potential effects of uprate on pending licensing actions and internal commitments, such as nonconformance reports and engineering deficiency reports. The licensee found no additional commitments that require modification to accommodate power uprate.

4.0 CONCLUSION

The staff has completed its review of PP&L's "NE-092-001, Revision 0, Licensing Topical Report for Power Uprate with Increased Core Flow," (Reference 13) and subsequent submittals, and has concluded that operation of the Susquehanna Steam Electric Station, Units 1 and 2, in the manner described in the topical report will continue to comply with all applicable regulations and is, therefore, acceptable. As discussed in this safety evaluation, there are four open items that PP&L will address when it submits the proposed license amendment application. These four items are: (1) the startup test plan (3.2.4), (2) the ATWS analysis (3.8.3), (3) the pipe whip and jet impingement evaluation (3.9.1), and (4) upgrading the Emergency Operating Procedures (3.8.3).

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