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AUG 0 3 2007

U. S. Nuclear Regulatory Commission
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**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED LICENSE AMENDMENT NO. 285
FOR UNIT 1 OPERATING LICENSE NO. NPF-14
AND PROPOSED LICENSE AMENDMENT NO. 253
FOR UNIT 2 OPERATING LICENSE NO. NPF-22
EXTENDED POWER UPRATE APPLICATION
REGARDING WITHDRAWAL OF CHANGE TO
TECHNICAL SPECIFICATION SR 3.3.1.1.8 –
CALIBRATION FREQUENCY FOR LPRMS
PLA-6254**

**Docket Nos. 50-387
and 50-388**

Reference: PPL Letter PLA-6076, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 for Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Constant Pressure Power Uprate," dated October 11, 2006.

Pursuant to 10 CFR 50.90, PPL Susquehanna LLC (PPL) requested in the above Reference approval of amendments to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Operating Licenses (OLs) and Technical Specifications (TS) to increase the maximum power level authorized from 3489 megawatts thermal (MWt) to 3952 MWt, an approximate 13% increase in thermal power. The proposed Constant Pressure Power Uprate (CPPU) represents an increase of approximately 20% above the Original Licensed Thermal Power (OLTP).

The purpose of this letter is to withdraw the proposed change to Technical Specification 3.3.1.1 Surveillance Requirement SR 3.3.1.1.8 which requested a longer frequency between calibrations of the Local Power Range Monitors (LPRMs). The proposed change would have changed the calibration frequency from 1000 megawatt days per metric ton (MWD/MT) to 2000 MWD/MT. In order to support this change, NRC requested additional analysis be performed. The analysis will take several months to develop. Therefore, so as not to extend the approval of the CPPU submittal, PPL is withdrawing the proposed change to SR 3.3.1.1.8.

A001
NRR

Attachment 1 contains the marked-up Technical Specification pages for Section 3.3.1.1 for Units 1 & 2, which supersede those pages that were transmitted in the above reference. Attachment 2 contains the revised marked-up Technical Specification Bases pages for Section B3.3.1.1 for Units 1 & 2 for information only.

There are no new regulatory commitments associated with this submittal.

PPL has reviewed the "No Significant Hazards Consideration" and the "Environmental Consideration" submitted with the Reference relative to the Enclosure. We have determined that there are no changes required to the "Environmental Consideration." The original "No Significant Hazards Consideration" contained statements on the LPRM calibration frequency. The "No Significant Hazards Consideration" in Attachment 3 to this letter has been revised to delete any reference to the LPRM calibration frequency. The removal of the reference to the LPRM calibration frequency does not invalidate the original conclusion.

If you have any questions or require additional information, please contact Mr. Michael H. Crowthers at (610) 774-7766.

I declare under perjury that the foregoing is true and correct.

Executed on: 8-3-07

B. T. McKinney

Attachment 1 - Revised Technical Specification Pages for Section 3.3.1.1 Units 1 & 2. (Mark-ups)
Attachment 2 - Revised Technical Specification Bases Pages for Section B3.3.1.1 Units 1 & 2. (Mark-ups – For Information Only)
Attachment 3 – Revised No Significant Hazards Consideration.

Copy: NRC Region I
Mr. R. V. Guzman, NRC Sr. Project Manager
Mr. R. R. Janati, DEP/BRP
Mr. F. W. Jaxheimer, NRC Sr. Resident Inspector

Attachment 1 to PLA-6254

Revised Technical Specification Pages for Section 3.3.1.1 Units 1 & 2 (Mark-ups)

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channels.	Immediately
E.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours 26
F.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

Retype to reflect PRNMS and ARTS/MELLLA

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1	I.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations. <u>AND</u> I.2 Restore require channels to OPERABLE.	12 hours 120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

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Retype to reflect PRNMS and ARTS/MELLLA

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.3	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	23 7 days
SR 3.3.1.1.4	------------------------------NOTE------------------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

Retype to reflect PRNMS and ARTS/MELLLA

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.11	------------------------------NOTES------------------------------ 1. Neutron detectors are excluded. 2. For Function 1.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.12	------------------------------NOTES------------------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~% RTP.	24 months

(continued)

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Retype to reflect PRNMS and ARTS/MELLLA

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.17	------NOTES------ 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency . 3. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and OPRM outputs shall alternate. ------------ Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	------NOTES------ 1. Neutron detectors are excluded. 2. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. ------------ Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.19	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 20% and recirculation drive flow is ≤ value equivalent to the core flow value defined in the COLR.	24 months 25
SR 3.3.1.1.20	Adjust recirculation drive flow to conform to reactor core flow.	24 months

Retype to reflect PRNMS and ARTS/MELLLA

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. *Intermediate Range* Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5[(a)]	3	H	SR 3.3.1.1.5 SR 3.3.2.2.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—High (Setdown)	2	3[(c)]	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 20% RTP
b. Simulated Thermal Power—High	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18 SR 3.3.1.1.20	≤ ~~0.62 W + 64.3~~ 0.55W + 60.7 % RTP[(b)] and ≤ 115.5% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.62(W - ΔW) + 64.3~~ 0.55(W - ΔW) + 60.7 % RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems

~~Retype~~ to ~~reflect~~ PR~~NMS~~ and A~~BTS/M~~ELLLA

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux—High	1	$3^{(c)}$	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 120% RTP
d. Inop	1, 2	$3^{(c)}$	G	SR 3.3.1.1.12	NA
e. 2-Out-Of-4 Voter	1, 2	2	G	SR 3.3.1.2 SR 3.3.1.12 SR 3.3.1.15 SR 3.3.1.17	NA
f. OPRM Trip	≥ 25% RTP 23	$3^{(c)}$	I	SR 3.3.1.2 SR 3.3.1.8 SR 3.3.1.12 SR 3.3.1.18 SR 3.3.1.19 SR 3.3.1.20	(d)
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≤ 1093 psig
4. Reactor Vessel Water Level—Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≥ 11.5 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 11% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≤ 1.88 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.

(d) See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

Retype to reflect PRNMS and ARTS/MELLLA

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level—High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 66 gallons
	5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 66 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 62 gallons
	5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 62 gallons
8. Turbine Stop Valve—Closure	≥ 30% RTP 26	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 7% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low	≥ 30% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ 460 psig
10 Reactor Mode Switch— Shutdown Position	1,2	2	G	SR 3.3.1.1.14 SR 3.3.1.1.15	NA
	5(a)	2	H	SR 3.3.1.1.14 SR 3.3.1.1.15	NA
11. Manual Scram	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Retype to reflect
PRNMS and ARTS/MELLLA

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channels.	Immediately
E.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours 26
F.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

Retype to reflect PRNMS and ARTS/MELLLA

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1	I.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
	<u>AND</u>	
	I.2 Restore require channels to OPERABLE.	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

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Retype to reflect PRNMS and ARTS/MELLLA

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.3	-----------------NOTE----------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -------------------------------------- 23 Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.4	-----------------NOTE----------------- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -------------------------------------- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

Retype to reflect PRNMS and ARTS/MELLLA

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.11	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 1.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.12	-----NOTES----- 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. ---------- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~ 26% RTP.	24 months

(continued)

Retype to reflect PRNMS and ARTS/MELLLA

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.17	---NOTES--- 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. 3. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and OPRM outputs shall alternate. ------ Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	---NOTES--- 1. Neutron detectors are excluded. 2. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. ------ Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.19	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 30% and recirculation drive flow is ≤ value equivalent to the core flow value defined in the COLR.	24 months 25
SR 3.3.1.1.20	Adjust recirculation drive flow to conform to reactor core flow.	24 months

~~Retype to reflect PRNMS and ARTS/MELLLA~~

Table 3.3.1.1-1 (page 1 of 3)

Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5[a]	3	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—High (Setdown)	2	3[c]	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 20% RTP
b. Simulated Thermal Power—High	1	3[c]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18 SR 3.3.1.1.20	0.55W + 60.7 ≤ 0.62 W + 64.2% RTP[b] and ≤ 115.5% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.62(W - ΔW) + 64.2% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides ~~inputs to both~~ trip systems.

0.55(W - ΔW) + 60.7

~~Retype to reflect~~
PRNMS and ARTS/MELLLA

Table 3.3.1.1-1 (page 2 of 3)

Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux—High	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 120% RTP
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.12	NA
e. 2-Out-Of-4 Voter	1, 2	2	G	SR 3.3.1.2 SR 3.3.1.12 SR 3.3.1.15 SR 3.3.1.17	NA
f. OPRM Trip	≥ 25% RTP 23	3[(c)]	I	SR 3.3.1.2 SR 3.3.1.8 SR 3.3.1.12 SR 3.3.1.18 SR 3.3.1.19 SR 3.3.1.20	(d)
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≤ 1093 psig
4. Reactor Vessel Water Level—Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≥ 11.5 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 11% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≤ 1.88 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.

(d) See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

Retype to reflect PRNMS and ARTS/MELLLA

Table 3.3.1.1-1 (page 3 of 3)

Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level—High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 66 gallons
	5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 66 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 62 gallons
	5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 62 gallons
8. Turbine Stop Valve—Closure	≥ 30% RTP 26	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 7% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low	≥ 30% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ 460 psig
10. Reactor Mode Switch—Shutdown Position	1,2	2	G	SR 3.3.1.1.14 SR 3.3.1.1.15	NA
	5(a)	2	H	SR 3.3.1.1.14 SR 3.3.1.1.15	NA
11. Manual Scram	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Retype to reflect PRNMS and ARTS/MELLLA

Attachment 2 to PLA-6254

Revised Technical Specification Bases Pages for Section B3.3.1.1 Units 1 & 2 (Mark-ups – For Information Only)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.a. Average Power Range Monitor Neutron Flux - High (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

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No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM protects against control rod withdrawal error events.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Trip

The OPRM Trip Function provides compliance with GDC 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations" thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

References 17, 18 and 19 describe three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm (confirmation count and cell amplitude), the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Trip Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Trip Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Trip Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into "cells" for evaluation by the OPRM algorithms. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. Three of the four channels are required to be OPERABLE.

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The OPRM Trip is automatically enabled (bypass removed) when THERMAL POWER is ≥ 30% RTP, as indicated by the APRM Simulated Thermal Power, and reactor core flow is ≤ the value defined in the COLR, as indicated by APRM measured recirculation drive flow. This is the operating region where actual thermal-hydraulic instability and related neutron flux oscillations are expected to occur. Reference 21 includes additional discussion of OPRM Trip enable region limits.

These setpoints, which are sometimes referred to as the "auto-bypass" setpoints, establish the boundaries of the OPRM Trip enabled region. The APRM Simulated Thermal Power auto-enable setpoint has 1% deadband while the drive flow setpoint has a 2% deadband. The deadband for these setpoints is established so that it increases the enabled region once the region is entered.

(continued)

23 25

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Trip (continued)

The OPRM Trip Function is required to be OPERABLE when the plant is at ≥25% RTP. The 25% RTP level is selected to provide margin in the unlikely event that a reactor power increase transient occurring without operator action while the plant is operating below 30% RTP causes a power increase to or beyond the 30% APRM Simulated Thermal Power OPRM Trip auto-enable setpoint. This OPERABILITY requirement assures that the OPRM Trip auto-enable function will be OPERABLE when required.

An APRM channel is also required to have a minimum number of OPRM cells OPERABLE for the Upscale Function 2.f to be OPERABLE. The OPRM cell operability requirements are documented in the Technical Requirements Manual, TRO 3.3.9, and are established as necessary to support the trip setpoint calculations performed in accordance with methodologies in Reference 19.

An OPRM Trip is issued from an APRM channel when the period based detection algorithm in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with period confirmations and relative cell amplitude exceeding specified setpoints. One or more cells in a channel exceeding the trip conditions will result in a channel OPRM Trip from that channel. An OPRM Trip is also issued from the channel if either the growth rate or amplitude based algorithms detect oscillatory changes in the neutron flux for one or more cells in that channel. (Note: To facilitate placing the OPRM Trip Function 2.f in one APRM channel in a "tripped" state, if necessary to satisfy a Required Action, the APRM equipment is conservatively designed to force an OPRM Trip output from the APRM channel if an APRM Inop condition occurs, such as when the APRM chassis keylock switch is placed in the Inop position.)

There are three "sets" of OPRM related setpoints or adjustment parameters: a) OPRM Trip auto-enable region setpoints for STP and drive flow; b) period based detection algorithm (PBDA) confirmation count and amplitude setpoints; and c) period based detection algorithm tuning parameters.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7.a, 7.b. Scram Discharge Volume Water Level - High (continued)

Four channels of each type of Scram Discharge Volume Water Level - High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve - Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded. Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure instruments sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this function non-conservatively, THERMAL POWER is derived from first stage pressure. The main turbine bypass valves must not cause the trip Function to be bypassed when THERMAL POWER is ≥ 30% RTP.

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The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

8. Turbine Stop Valve - Closure (continued)

Eight channels (arranged in pairs) of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

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

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure instrument is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure instruments sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this function non-conservatively, THERMAL POWER is derived from first stage pressure. The main turbine bypass valves must not cause the trip Function to be bypassed when THERMAL POWER is ≥ 30% RTP.

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The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

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10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, to each of the four RPS logic channels, which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Four channels of Reactor Mode Switch - Shutdown Position. Function, with two channels in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch - Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.3

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based ~~on minor changes in~~ LPRM sensitivity, which could affect the APRM reading ~~between performances of SR 3.3.1.1.8.~~

23

A restriction to satisfying this SR when ≥ 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. ~~A Note is provided which~~ allows an increase in THERMALPOWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and inconsideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

As noted, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.15

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM and OPRM trip conditions at the 2-out-of-4 Voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

The 24 month Frequency is based on the need to perform portions of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.16

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This is performed by a Functional check that ensures the scram feature is not bypassed at ≥ 30% RTP. Because main turbine bypass flow can affect this function nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the opening of the main turbine bypass valves must not cause the trip Function to be bypassed when Thermal Power is ≥ 30% RTP.

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If any bypass channel's trip function is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

~~Retyped to include PRNMS and ARTS/MELLLA~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.17 (continued)

After 8 cycles, the sequence repeats.

Each test of an OPRM or APRM output tests each of the redundant outputs from the 2-Out-Of-4 Voter channel for that Function and each of the corresponding relays in the RPS. Consequently, each of the RPS relays is tested every fourth cycle. The RPS relay testing frequency is twice the frequency justified by References 15 and 16.

SR 3.3.1.1.19

This surveillance involves confirming the OPRM Trip auto-enable setpoints. The auto-enable setpoint values are considered to be nominal values as discussed in Reference 21. This surveillance ensures that the OPRM Trip is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation drive flow properly correlate with THERMAL POWER (SR 3.3.1.1.2) and core flow (SR 3.3.1.1.20), respectively.

If any auto-enable setpoint is nonconservative (i.e., the OPRM Trip is bypassed when APRM Simulated Thermal Power ≥ ~~30~~ 25% and recirculation drive flow ≤ value equivalent to the core flow value defined in the COLR, then the affected channel is considered inoperable for the OPRM Trip Function. Alternatively, the OPRM Trip auto-enable setpoint(s) may be adjusted to place the channel in a conservative condition (not bypassed). If the OPRM Trip is placed in the not-bypassed condition, this SR is met and the channel is considered OPERABLE.

For purposes of this surveillance, consistent with Reference 21, the conversion from core flow values defined in the COLR to drive flow values used for this SR can be conservatively determined by a linear scaling assuming that 100% drive flow corresponds to 100 Mlb/hr core flow, with no adjustment made for expected deviations between core flow and drive flow below 100%.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor (APRM) (continued)

Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no signal failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core consistent with the design bases for the APRM Functions 2.a, 2.b, and 2.c, at least (20) LPRM inputs with at least three LPRM inputs from each of the four axial levels at which the LPRMs are located must be OPERABLE for each APRM channel, with no more than (9), LPRM detectors declared inoperable since the most recent APRM gain calibration. Per Reference 23, the minimum input requirement for an APRM channel with 43 LPRM inputs is determined given that the total number of LPRM outputs used as inputs to an APRM channel that may be bypassed shall not exceed twenty-three (23). Hence, (20) LPRM inputs needed to be operable. For the OPRM Trip Function 2.f, each LPRM in an APRM channel is further associated in a pattern of OPRM "cells," as described in References 17 and 18. Each OPRM cell is capable of producing a channel trip signal.

2.a. Average Power Range Monitor Neutron Flux—High (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

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No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Trip (continued)

References 17, 18 and 19 describe three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm (confirmation count and cell amplitude), the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Trip Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Trip Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Trip Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into "cells" for evaluation by the OPRM algorithms. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. Three of the four channels are required to be OPERABLE.

The OPRM Trip is automatically enabled (bypass removed) when THERMAL POWER is ≥ ~~30~~ 25% RTP, as indicated by the APRM Simulated Thermal Power, and reactor core flow is ≤ the value defined in the COLR, as indicated by APRM measured recirculation drive flow. This is the operating region where actual thermal-hydraulic instability and related neutron flux oscillations are expected to occur. Reference 21 includes additional discussion of OPRM Trip enable region limits.

These setpoints, which are sometimes referred to as the "auto-bypass" setpoints, establish the boundaries of the OPRM Trip enabled region. The APRM Simulated Thermal Power auto-enable setpoint has 1% deadband while the drive flow setpoint has a 2% deadband. The deadband for these setpoints is established so that it increases the enabled region once the region is entered.

The OPRM Trip Function is required to be OPERABLE when the plant is at ≥ ~~25~~ 23% RTP. The ~~25~~ 23% RTP level is selected to provide margin in the unlikely event that a reactor power increase transient occurring without operator action while the plant is operating below ~~30~~ 25% RTP causes a power increase to or beyond the ~~30~~ 25% APRM Simulated Thermal Power OPRM Trip auto-enable setpoint. This OPERABILITY requirement assures that the OPRM Trip auto-enable function will be OPERABLE when required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

8. Turbine Stop Valve—Closure (continued)

valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded. Turbine Stop Valve—Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure instruments sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this function non-conservatively, THERMAL POWER is derived from first stage pressure. The main turbine bypass valves must not cause the trip Function to be bypassed when THERMAL POWER is ≥ 30% RTP.

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The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels (arranged in pairs) of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

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(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure instrument is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure instruments sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this function non-conservatively, THERMAL POWER is derived from first stage pressure. The main turbine bypass valves must not cause the trip Function to be bypassed when THERMAL POWER is ≥ 30% RTP.

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The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.1 and SR 3.3.1.1.2 (continued)

Agreement criteria which are determined by the plant staff based on an investigation of a combination of the channel instrument uncertainties, may be used to support this parameter comparison and include indication and readability. If a channel is outside the criteria, it may be an indication ~~that the instrument has drifted outside its limit,~~ and does not necessarily indicate the channel is Inoperable.

The Frequency of once every 12 hours for SR 3.3.1.1.1 is based upon operating experience that demonstrates that channel failure is rare. The Frequency of once every 24 hours for SR 3.3.1.1.2 is based upon operating experience that demonstrates that channel failure is rare and the evaluation in References 15 and 16. The CHANNEL CHECK supplements less formal checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.3

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

23

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

Retyped to Include PRNMS and ARTS/MELLA

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.15 (continued)

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM and OPRM trip conditions at the 2-out-of-4 Voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

The 24 month Frequency is based on the need to perform portions of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.16

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This is performed by a Functional check that ensures the scram feature is not bypassed at ≥ 30% RTP. Because main turbine bypass flow can affect this function nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the opening of the main turbine bypass valves must not cause the trip Function to be bypassed when Thermal Power is ≥ 30% RTP.

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If any bypass channel's trip function is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

Retyped to Include PRNMS and ARTS/MELLA

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.19

This surveillance involves confirming the OPRM Trip auto-enable setpoints. The auto-enable setpoint values are considered to be nominal values as discussed in Reference 21. This surveillance ensures that the OPRM Trip is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation drive flow properly correlate with THERMAL POWER (SR 3.3.1.1.2) and core flow (SR 3.3.1.1.20), respectively.

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If any auto-enable setpoint is nonconservative (i.e., the OPRM Trip is bypassed when APRM Simulated Thermal Power ≥ 20% and recirculation drive flow ≤ value equivalent to the core flow value defined in the COLR, then the affected channel is considered inoperable for the OPRM Trip Function. Alternatively, the OPRM Trip auto-enable setpoint(s) may be adjusted to place the channel in a conservative condition (not bypassed). If the OPRM Trip is placed in the not-bypassed condition, this SR is met and the channel is considered OPERABLE.

For purposes of this surveillance, consistent with Reference 21, the conversion from core flow values defined in the COLR to drive flow values used for this SR can be conservatively determined by a linear scaling assuming that 100% drive flow corresponds to 100 Mlb/hr core flow, with no adjustment made for expected deviations between core flow and drive flow below 100%.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.20

The APRM Simulated Thermal Power-High Function (Function 2.b) uses drive flow to vary the trip setpoint. The OPRM Trip Function (Function 2.f) uses drive flow to automatically enable or bypass the OPRM Trip output to RPS. Both of these Functions use drive flow as a representation of reactor core flow. SR 3.3.1.1.18 ensures that the drive flow transmitters and processing electronics are calibrated. This SR adjusts the recirculation drive flow scaling factors in each APRM channel to provide the appropriate drive flow/core flow alignment.

(continued)

Attachment 3 to PLA-6254

Revised No Significant Hazards Consideration

5.1 No Significant Hazards Consideration

PPL Susquehanna has evaluated whether or not a significant hazards consideration is involved with the proposed change, by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Extended Power Uprate

Response: No.

The probability (frequency of occurrence) of a Design Basis Accident occurring is not affected by the increased power level, because Susquehanna continues to comply with the regulatory and design basis criteria established for plant equipment. A probabilistic risk assessment demonstrates that the calculated core damage frequencies do not significantly change due to Constant Pressure Power Uprate (CPPU). Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to CPPU. No new challenges to safety-related equipment result from CPPU.

The changes in consequences of postulated accidents, which would occur from 102% of the CPPU rated thermal power (RTP) compared to those previously evaluated, are acceptable. The results of CPPU accident evaluations do not exceed the NRC-approved acceptance limits. The spectrum of postulated accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of fuel and core design, for example, the Safety Limit Minimum Critical Power Ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDLs) are still met. Continued compliance with the SLMCPR and other SAFDLs will be confirmed on a cycle-specific basis consistent with the criteria accepted by the NRC.

Challenges to the Reactor Coolant Pressure Boundary were evaluated at CPPU conditions (pressure, temperature, flow, and radiation) were found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet 10 CFR 50, Appendix A, Criterion 16, Containment Design; Criterion 38, Containment Heat Removal; and Criterion 50, Containment Design Basis. The increase in the calculated post-LOCA suppression pool temperature above the currently assumed peak temperature was evaluated and determined to be acceptable.

Radiological release events (accidents) have been evaluated, and meet the guidelines of 10 CFR 50.67.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>RHR Service Water System and Ultimate Heat Sink Technical Specification and Methods Change</u>

Response: No.

The proposed changes do not involve any new initiators for any accidents nor do they increase the likelihood of a malfunction of any Structures, Systems or Components (SSCs). Implementation of the subject changes reduces the probability of adverse consequences of accidents previously evaluated, because inclusion of the manual spray array bypass isolation valves and the small spray array isolation valves in the Technical Specifications (TS) increases their reliability to function for safe shutdown.

The use of the ANS/ANSI-5.1-1979 decay heat model in the UHS performance analysis is not relevant to accident initiation, but rather, pertains to the method used to evaluate currently postulated accidents. Its use does not, in any way, alter existing fission product boundaries, and provides a conservative prediction of decay heat. Therefore, the change in decay heat calculational method does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>Containment Analysis Methods Change</u>

Response: No.

The use of passive heat sinks and the ANS/ANSI-5.1-1979 decay heat model are not relevant to accident initiation, but rather, pertain to the method used to evaluate postulated accidents. The use of these elements does not, in any way, alter existing fission product boundaries, and provides a conservative prediction of the containment response to DBA-LOCAs. Therefore, the Containment Analysis Method Change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Feedwater Pump / Condensate Pump Trip Change

Response: No

Feedwater pump trips and condensate pump trips rarely occur. A low water level SCRAM on loss of one feedwater pump or one condensate pump is bounded by the loss of all feedwater transient in Final Safety Analysis Report (FSAR) Appendix 15E. A trip of one feedwater pump or a trip of one condensate pump does not result in the loss of all feedwater. The Feedwater Pump / Condensate Pump Trip Change is included in the CPPU Probabilistic Risk Assessment (PRA). The best estimate for the Susquehanna Steam Electric Station (SSES) Core Damage Frequency (CDF) risk increase due to the CPPU is 6E-08 for Unit 1 and 7E-08 for Unit 2 which are in the lower left corner of Region III of Regulatory Guide Regulatory (Reference 15) (i.e., very small risk changes). The best estimate for the Large Early Release Frequency (LERF) increase is 1.0E-09/yr for both units, which is also in the lower left corner of the Region III range of Regulatory Guide 1.174. Therefore, the Feedwater Pump / Condensate Pump Trip Change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Main Turbine Pressure Regulation System

Response: No.

Technical Specification 3.7.8 does not directly or indirectly affect any plant system, equipment, component, or change the process used to operate the plant. Technical Specification 3.7.8 would ensure acceptable performance, since it would establish requirements for adhering to the appropriate thermal limits, depending on the operability of the main turbine pressure regulation system. Use of the appropriate limits assures that the appropriate safety limits will not be exceeded during normal or anticipated operational occurrences. Thus, Technical Specification 3.7.8 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Extended Power Uprate

Response: No.

Equipment that could be affected by EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. CPPU uses developed technology and applies it within capabilities of existing or modified plant safety related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). No new accidents or event precursors have been identified.

The SSES TS require revision to implement EPU. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>RHR Service Water System and Ultimate Heat Sink Technical Specification and Methods Change</u>

Response: No.

The subject changes apply Technical Specification controls to new UHS manual bypass isolation valves and the existing small spray array isolation valves. The design functions of the systems are not affected.

The addition of manually operated valves in the system, operational changes and the Technical Specification changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The use of the ANS/ANSI-5.1-1979 decay heat model is not relevant to accident initiation, but rather pertains to the method used to evaluate currently postulated accidents. The use of this analytical tool does not involve any physical changes to plant structures or systems, and does not create a new initiating event for the spectrum of events currently postulated in the FSAR. Further, it does not result in the need to postulate any new accident scenarios. Therefore, the decay heat calculational method change does not create the possibility of a new or different kind of accident from any accident previously evaluated

<u>Containment Analysis Methods Change</u>

Response: No.

The use of passive heat sinks and the ANS/ANSI-5.1-1979 decay heat model are not relevant to accident initiation, but pertain to the method used to evaluate currently postulated accidents. The use of these analytical tools does not involve any physical changes to plant structures or systems, and does not create a new initiating event for the spectrum of events currently postulated in the FSAR. Further, they do not result in the need to postulate any new accident scenarios.

Therefore, the Containment Analysis Method Change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Feedwater Pump / Condensate Pump Trip Change

Response: No

The occurrence of a reactor SCRAM is already considered in the current licensing basis and is not an accident. A SCRAM resulting from the trip of a feedwater pump or a condensate pump is bounded by a loss of all feedwater event. The loss of all feedwater transient is already considered in the plant licensing basis. The SCRAM due to the feedwater or condensate pump trip does not change the results of the loss of all feedwater transient in any way. Therefore, the Feedwater Pump / Condensate Pump Trip Change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Main Turbine Pressure Regulation System

Response: No.

Technical Specification 3.7.8 will not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items. Thus, Technical Specification 3.7.8 does not create the possibility of a previously unevaluated operator error or a new single failure.

Therefore, Technical Specification 3.7.8 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Extended Power Uprate

Response: No.

The CPPU affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were evaluated for CPPU conditions. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on affected structures, systems and components, including the reactor coolant pressure boundary, will remain within their design allowables for design basis event categories. No NRC acceptance criterion is exceeded. Because the SSES configuration and responses to transients and postulated accidents do not result in exceeding the presently approved NRC acceptance limits, the proposed changes do not involve a significant reduction in a margin of safety.

RHR Service Water System and Ultimate Heat Sink Technical Specification and Methods Change

Response: No.

Implementation of the subject changes does not significantly reduce the margin of safety since these changes add components and Technical Specification controls for the components not currently addressed in the Technical Specifications. These changes increase the reliability of the affected components/systems to function for safe shutdown.

Therefore, these changes do not involve a significant reduction in margin of safety.

The ANS/ANSI-5.1-1979 model provides a conservative prediction of decay heat. The use of this element is consistent with current industry standards, and has been previously accepted by the staff for use in containment analysis by other licensees, as described in GE Nuclear Energy. "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A, Revision 4, dated July 2003; and the letter to Gary L. Sozzi (GE) from Ashok Thandani (NRC) on the Use of the SHEX Computer Program and ANSI/ANS 5.1-1979, "Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993. Therefore, the decay heat calculational method change does not involve a significant reduction in the margin of safety.

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks and the ANS/ANSI-5.1-1979 decay heat model are realistic phenomena, and provide a conservative prediction of the plant response to DBA-LOCAs. The use of these elements is consistent with current industry standards, and has been previously accepted by the staff for other licensees, as described in GE Nuclear Energy: "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A, Revision 4, dated July 2003; the letter to Gary L. Sozzi (GE) from Ashok Thandani (NRC) on the Use of the SHEX Computer Program; and ANSI/ANS 5.1-1979, "Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993. Therefore, the Containment Analysis Method Change does not involve a significant reduction in the margin of safety.

Feedwater Pump / Condensate Pump Trip Change

Response: No

A low water level SCRAM on loss of one feedwater pump or one condensate pump is bounded by the loss of all feedwater transient in FSAR Appendix 15E. The loss of all feedwater transient is a non-limiting event that does not contribute to the setting of the fuel safety limits. Consequently, a SCRAM resulting from a feedwater pump or condensate pump trip does not reduce the margin to fuel safety limits. Therefore, the potential for a SCRAM resulting from a feedwater pump trip or a condensate pump trip does not involve a significant reduction in the margin of safety.

Main Turbine Pressure Regulation System

Since Technical Specification 3.7.8 does not alter any plant system, equipment, component, or processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. Technical Specification 3.7.8 preserves the margin of safety by establishing requirements for adhering to the appropriate thermal limits.

Conclusion for All Changes

Based upon the above, PPL Susquehanna concludes that the proposed amendment presents no significant hazards consideration, under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements / Criteria

5.2.1 Analysis

Extended Power Uprate

10 CFR 50.36 (c)(2)(ii) Criterion 2, requires that TS LCOs include process variables, design features, and operating restrictions that are initial conditions of design basis accident analysis. The Technical Specifications ensure that the SSES system performance parameters are maintained within the values assumed in the safety analyses. The Technical Specification changes are supported by the safety analyses that were performed consistent with NRC approved methodology approved for SSES and continue to provide a comparable level of protection as the current Technical Specifications. Applicable regulatory requirements and

significant safety evaluations performed in support of the proposed changes are described in Attachment 4.

<u>RHR Service Water System and Ultimate Heat Sink Technical Specification and Methods Change</u>

GDC-5 requires SSCs important to safety not to be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The proposed changes do not affect compliance with GDC-5 as described in FSAR Section 3.1. The RHRSW system and UHS continue to be designed such that no single active failure will prevent their safety function from being achieved.

GDC-44 requires that a system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. The proposed change does not affect compliance with GDC-44 as described in FSAR Section 3.1. The RHRSW system and the UHS are designed to Seismic Category I requirements. Redundant safety related components served by RHRSW and the UHS are supplied through redundant supply headers and returned through redundant discharge or return lines. Electric power for operation of redundant safety related components of RHRSW is supplied from separate independent offsite and redundant onsite standby power sources. No single active failure renders RHRSW or the UHS incapable of performing its safety function.

Regulatory Guide 1.27 Revision 2 applies to nuclear power plants that use water as the ultimate heat sink. The proposed change does not affect compliance with Regulatory Guide 1.27 Revision 2 as described in Sections 3.13 and 9.2.7 of the FSAR. The UHS continues to be capable of providing sufficient cooling for 30 days to permit simultaneous safe shutdown and cooldown of both SSES units and maintain them in a safe shutdown condition.

10 CFR 50.36(c)(3), requires that Technical Specification LCOs include surveillance requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The Technical Specification changes are supported by the safety analyses that were performed consistent with NRC approved methodology and continue to provide a comparable level of protection as current Technical Specifications.

Containment Analysis Methods Change

10 CFR 50, Appendix A, GDC-16 requires that a reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity, and that it be assured that containment design parameters important to safety not be exceeded for as long as postulated accident conditions require. The evaluations described in Attachment 4, Section 4.1 demonstrate that containment parameters stay within their design limits.

10 CFR 50, Appendix A, GDC-50 requires that the reactor containment structure be designed so that the structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any loss of coolant accident. The evaluations described in Attachment 4, Section 4.1 demonstrate that containment parameters stay within their design limits.

Feedwater Pump / Condensate Pump Trip Change

General Design Criterion 10 (GDC 10), "Reactor Design," in Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50 states, in part, that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded. General Design Criterion 20 (GDC 20), Protection System Functions, in Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50 states, in part that the protection system shall be designed to initiate automatically the operation of appropriate systems including reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a results of anticipated operational occurrences.

During the feedwater pump and condensate pump trip transients for CPPU, the water level may reduce to the level at which a SCRAM is initiated. Fuel design limits are not exceeded during this transient.

The Feedwater Pump / Condensate Pump Trip Change will not cause the MCPR safety limit to be violated nor the fuel cladding strain to exceed 1%. Therefore, the requirements of GDC-10 and GDC-20 regarding acceptable fuel design limits is satisfied.

Main Turbine Pressure Regulation System

Title 10 of the Code of Federal Regulations (10 CFR) establishes the fundamental regulatory requirements with respect to reactivity control systems. Specifically, General Design Criterion 10 (GDC 10), "Reactor design," in Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50 states, in part, that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.

Technical Specification 3.7.8 will ensure that the MCPR Safety Limit will not be violated and that fuel cladding strain will not exceed 1%. This satisfies the requirement of GDC-10 regarding acceptable fuel design limits.

5.2.2 Conclusion

Based on the analyses provided in Section 4, Technical Analysis, the proposed change is consistent with applicable regulatory requirements and criteria. In conclusion, based on the considerations discussed above, (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.