50-387/388

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

December 18, 1995

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M91639 & M91640)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 155 to Facility Operating License No. NPF-14 and Amendment No. 126 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments are in response to your letter dated February 10, 1995, as supplemented by letter dated November 10, 1995.

These amendments (1) modify the Susquehanna Steam Electric Station, Unit 1 and 2 Technical Specifications to extend the allowable out-of-service times (AOTs) for maintenance and repair and the surveillance test intervals (STIs) between channel functional tests for the following groups of instruments: reactor protection systems instrumentation (TS 3.3.1), isolation actuation instrumentation (TS 3.3.2), emergency core cooling system actuation instrumentation (TS 3.3.3), ATWS (anticipated transient without scram) recirculation pump trip system instrumentation (TS 3.3.4.1), end-of-cycle recirculation pump trip system instrumentation (TS 3.3.4.2), reactor core isolation cooling system (RCIC) actuation instrumentation (TS 3.3.5), control rod block instrumentation (TS 3.3.6), radiation monitoring instrumentation (TS 3.3.7.1), and feedwater/main turbine trip system actuation instrumentation (TS 3.3.90); (2) change the required actions and AOTs for the instruments listed above to make requirements consistent with supporting analysis in General Electric topical reports and change additional actions required to prevent extended AOTs from resulting in extended loss of instrument function; (3) change the required actions and AOTs for the instruments listed above for instrumentation associated with the ADS (automatic depressurization system), recirculation pump trip, and pump suction lineup for HPCI (high pressure core injection) and RCIC; (4) change applicability requirements and required actions for the reactor vessel water level-low, level 3 function that isolates the RHR (residual heat removal) system shutdown cooling system so that the function is required to be operable in operational conditions 3,4, and 5 to prevent inadvertent loss of reactor coolant via the RHR shutdown cooling system; (5) remove notes in Table 3.3.2-1, 3.3.2-2, and 4.3.1-1 related to maintenance on leak detection temperature detectors and remove the note to

DF01

TS 3.3.6 for Unit 1 related to a previous relief from TS 3.0.4; and (6) reformat, renumber, and/or reword existing requirements to incorporate the changes listed above.

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

Sincerely,

/S/

Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/50-388

Enclosures:
1. Amendment No. 155 to License No. NPF-14
2. Amendment No. 126 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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TS 3.3.6 for Unit 1 related to a previous relief from TS 3.0.4; and (6) reformat, renumber, and/or reword existing requirements to incorporate the changes listed above.

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

Sincerely,

Chester Poslusny

Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/50-388

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cc w/encls: 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. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. NPF-14

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 February 10, 1995, as supplemented by letter dated November 10, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment ar paragraph 2.C.(2) of the Facility Operating License No. NPF-14 i. hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 155 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 18, 1995

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With one or more required channels inoperable in one trip system, place the inoperable channel(s) or the associated trip system in the tripped condition within 12 hours.

b. With one or more required channels inoperable in both trip systems, place the inoperable channel(s) in one trip system or one trip system in the tripped condition within 6 hours.*

c. With one or more RPS Functions with RPS trip capability not maintained, restore RPS trip capability within one hour.

d. If ACTION a or b or c is not met, take the ACTION required by Table 3.3.1-1 for the RPS Function.

The provisions of Specification 3.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 or 3 from OPERATIONAL CONDITION 1 for the IRMs or the Neutron Flux - Upscale, Setdown function for the APRMs.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

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

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shall be demonstrated to be within its limit at least once per 18 months**. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs or the Neutron Flux - Upscale, Setdown function of the APRMs.

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* If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause a scram to occur.

** The neutron detectors are exempt from response time testing.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,[b] S	S/U[c],W W	SA SA	2 3, 4, 5
b. Inoperative	NA	S/U[c],W	NA	2, 3, 4, 5
2. Avg. Power Range Monitor[f]				
a. Neutron Flux - Upscale, Setdown	S/U,S,[b] S	S/U[c],W W	SA SA	2 3,5[k]
b. Flow Biased Simulated Thermal Power - Upscale	S,D[g]	S/U[c],Q	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c],Q	W[d],SA	1
d. Inoperative	NA	S/U[c],Q	NA	1,2,3,5[k]
3. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q	1,2
4. Reactor Vessel Water Level - Low, Level 3	S	Q	Q	1,2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1,2[i]
7. Drywell Pressure - High	NA	Q	R	1,2

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required* by the Minimum OPERABLE Channels per Trip System in one or more Trip Systems:

1. For functions 1.a.1, 1.b, 1.e, 2.b, 3.b, 7.a, and 7.e, place the channel in the tripped condition within 12 hours; and,

2. For functions other than 1.a.1, 1.b, 1.e, 2.b, 3.b, 7.a, and 7.e, place the channel in the tripped condition within 24 hours.

   The provisions of Specification 3.0.4 are not applicable.

c. With one or more automatic functions with isolation capability not maintained, restore isolation capability within one hour.

d. If ACTIONS b or c are not met, take the ACTION required by Table 3.3.2-1 for the function.

* When a channel is placed in an inoperable status solely for performance of required Surveillances, initiation of ACTIONS may be delayed for up to 6 hours provided the associated Trip Function maintains isolation capability.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)[(a)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>				
a. RCIC Steam Line Δ Pressure - High	K	1	1,2,3	23
b. RCIC Steam Supply Pressure - Low	KB	2	1,2,3	23
c. RCIC Turbine Exhaust Diaphragm Pressure - High	K	2	1,2,3	23
d. RCIC Equipment Room Temperature - High	K	1	1,2,3	23
e. RCIC Equipment Room Δ Temperature - High	K	1	1,2,3	23
f. RCIC Emergency Area Cooler Temperature - High	K	1	1,2,3	23
g. RCIC Pipe Routing Area Δ Temperature - High	K	1	1,2,3	23
h. RCIC Pipe Routing Area Temperature - High	K	1	1,2,3	23
i. Manual Initiation	NA	1	1,2,3	24
j. Drywell Pressure - High	Z	2	1,2,3	23

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
6. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u>				
a. HPCI Steam Line Δ Pressure - High	L	1	1,2,3	23
b. HPCI Steam Supply Pressure - Low	LB	2	1,2,3	23
c. HPCI Turbine Exhaust Diaphragm Pressure - High	L	2	1,2,3	23
d. HPCI Equipment Room Temperature - High	L	1	1,2,3	23
e. HPCI Equipment Room Δ Temperature - High	L	1	1,2,3	23
f. HPCI Emergency Area Cooler Temperature - High	L	1	1,2,3	23
g. HPCI Pipe Routing Area Temperature - High	L	1	1,2,3	23
h. HPCI Pipe Routing Area Δ Temperature - High	L	1	1,2,3	23
i. Manual Initiation	NA	1	1,2,3	24
j. Drywell Pressure - High	Z	2	1,2,3	23

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	A	2*	3,4,5	27
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	UB	1	1,2,3	26
c. RHR Flow - High	M	1	1,2,3	26
d. Manual Initiation	NA	1	1,2,3	24
e. Drywell Pressure - High	Z	2	1,2,3	26

* Only one trip system required in OPERATIONAL CONDITIONS 4 and 5 when RHR Shutdown Cooling System integrity is maintained.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20	Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
ACTION 21	Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
ACTION 22	Be in at least STARTUP within 6 hours.
ACTION 23	Close the affected system isolation valves within 1 hour and declare the affected system inoperable.
ACTION 24	Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
ACTION 25	Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.
ACTION 26	Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.
ACTION 27	Initiate action to restore channel to OPERABLE status; or, initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.

NOTES

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

** Actuates dampers shown in Table 3.6.5.2-1.

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

\# When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with the potential for draining the reactor vessel. Single control rod movement, except for the purpose of SDM demonstration (TS 3.10.3), is excluded.

\#\# When handling irradiated fuel within the Railroad Access Shaft and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves which are actuated by these isolation signals.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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
<u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u> (continued)				
d. HPCI Equipment Room Temperature - High	NA	Q	Q	1,2,3
e. HPCI Equipment Room Δ Temperature - High	NA	Q	Q	1,2,3
f. HPCI Emergency Area Cooler Temperature - High	NA	Q	Q	1,2,3
g. HPCI Pipe Routing Area Temperature - High	NA	Q	Q	1,2,3
h. HPCI Pipe Routing Area Δ Temperature - High	NA	Q	Q	1,2,3
i. Manual Initiation	NA	R	NA	1,2,3
j. Drywell Pressure - High	NA	Q	R	1,2,3
7. <u>RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION</u>				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	3,4,5
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	Q	Q	1,2,3
c. RHR Flow - High	S	Q	R	1,2,3
d. Manual Initiation	NA	R	NA	1,2,3
e. Drywell Pressure - High	NA	Q	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.

\# When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with the potential for draining the reactor vessel. Single control rod movement, except for the purpose of SDM demonstration (TS 3.10.3), is excluded.

\#\# When handling irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. <u>CORE SPRAY SYSTEM</u>			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(a)]	1,2,3,4*,5*	30
b. Drywell Pressure - High	2[(a)]	1,2,3	30
c. Reactor Vessel Steam Dome Pressure - Low (Permissive)	2[(a)]	1,2,3, 4*,5*	31 30
d. Manual Initiation	1/subsystem	1,2,3,4*,5*	31
2. <u>LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM</u>			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(a)]	1,2,3,4*,5*	30
b. Drywell Pressure - High	2[(a)]	1,2,3	30
c. Reactor Vessel Steam Dome Pressure - Low (Permissive)			
1. System Initiation	2[(a)]	1,2,3, 4*,5*	31 30
2. Recirculation Discharge Valve Closure	2[(a)]	1,2,3, 4*,5*	31 30
d. Manual Initiation	1/subsystem	1,2,3,4*,5*	31
3. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM#</u>			
a. Reactor Vessel Water Level - Low Low, Level 2	2[(a)]	1,2,3	30
b. Drywell Pressure - High	2[(a)]	1,2,3	30
c. Condensate Storage Tank Level - Low	2[(a)(b)]	1,2,3	34
d. Suppression Pool Water Level - High***	2[(a)]	1,2,3	34
e. Reactor Vessel Water Level - High, Level 8	2[(f)]	1,2,3	31
f. Manual Initiation	1/system	1,2,3	31

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[a]	1,2,3	32
b. Drywell Pressure - High	2[a]	1,2,3	32
c. ADS Timer	1[a]	1,2,3	33
d. Core Spray Pump Discharge Pressure - High (Permissive)	2[d][a]	1,2,3	33
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	2[d][e][a]	1,2,3	33
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1[a]	1,2,3	32
g. ADS Drywell Pressure Bypass Timer	2[a]	1,2,3	33
h. Manual Inhibit	1	1,2,3	33
I. Manual Initiation	1/valve	1,2,3	33

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
a. 4.16 kv ESS Bus Under voltage (Loss of Voltage, < 20%)	1/bus	1/bus	1/bus	1,2,3,4**,5**	35
b. 4.16 kv ESS Bus Under voltage (Degraded Voltage, < 65%)	2/bus	2/bus	2/bus	1,2,3,4**,5**	36
c. 4.16 kv ESS Bus Under voltage (Degraded Voltage, < 93%)	2/bus	2/bus	2/bus	1,2,3,4**,5**	36
d. 480V ESS Bus 0B565 Under voltage (Degraded Voltage, < 65%)###	2/bus	1/bus	2/bus	1,2,3,4**,5**	36
e. 480V ESS Bus 0B565 Under voltage (Degraded Voltage, < 92%)###	2/bus	2/bus	2/bus	1,2,3,4**,5**	36

See footnotes on next page.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

(a) When a channel is placed in an inoperable status solely for performance of required Surveillances, initiation of ACTIONS may be delayed for up to 6 hours provided the associated Trip Function maintains trip capability.

(b) One trip system. Provides signal to HPCI pump suction valves only.

(c) Two out of two logic.

(d) Either 4d or 4e must be satisfied. The ACTION is required to be taken only if neither is satisfied. A channel is not OPERABLE unless its associated pump is OPERABLE per Specification 3.5.1.

(e) Within an ADS Trip System there are two logic subsystems, each of which contains an overall pump permissive. At least one channel associated with each of these overall pump permissives shall be OPERABLE.

(f) When a channel is placed in an inoperable status solely for performance of required Surveillances, initiation of ACTIONS may be delayed for up to 6 hours.

* When the system is required to be OPERABLE per Specification 3.5.2.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

** Required when ESF equipment is required to be OPERABLE.

\## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\### Required to be OPERABLE only when Diesel Generator E is either aligned to the Class 1E system or not aligned to the Class 1E system but operating on the Test Facility.

*** The automatic transfer of HPCI pump suction from the condensate storage tank to suppression pool on high suppression pool water level occurs only when HPCI injection valve is open.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 30 - With one or more required channel(s) inoperable in one or more Trip Systems:
a. Within 24 hours, place the inoperable channel(s) in the tripped condition or declare the associated ECCS inoperable; and,
b. Within one hour from discovery of loss of initiation capability by this trip function, declare the associated ECCS inoperable.

ACTION 31 - With one or more required channel(s) inoperable in one or more Trip Systems:
a. Within 24 hours, restore the channel(s) to OPERABLE status or declare the associated ECCS inoperable; and
b. Within one hour from discovery of loss of initiation capability by this trip function, declare the associated ECCS inoperable.*
(*ACTION 31b is not applicable to Function 3.e, Reactor Vessel Water Level - High, Level 8)

ACTION 32 - With one or more required channel(s) inoperable in one or more Trip Systems:
a. Within one hour from discovery of loss of ADS initiation capability by this trip function, declare the ADS valves inoperable; and,
b. Within 4 days from discovery of an inoperable channel(s) concurrent with HPCI or RCIC inoperable, place the inoperable channel(s) in the tripped condition; and,
c. Within 8 days from discovery of an inoperable channel(s) if both HPCI and RCIC are OPERABLE, place the inoperable channel(s) in the tripped condition.
d. If ACTION b or c is not met, declare ADS inoperable.

ACTION 33 - With one or more required channel(s) inoperable in one or more Trip Systems:
a. Within one hour from discovery of loss of ADS initiation capability by this trip function, declare the ADS valves inoperable; and,
b. Within 4 days from the discovery of an inoperable channel(s) concurrent with HPCI or RCIC inoperable, restore the channel(s) to OPERABLE status; and,
c. Within 8 days from the discovery of an inoperable channel(s) if both HPCI and RCIC are OPERABLE, restore the channel(s) to OPERABLE status.
d. If ACTION b or c is not met, declare ADS inoperable.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 34 - With one or more required channel(s) inoperable in one or more Trip Functions:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or align the HPCI pump suction to the suppression pool; and,

b. Within one hour of discovery of loss of initiation capability declare HPCI inoperable if the associated pump suction is not aligned to the suppression pool.

c. If ACTION a or b is not met, declare HPCI inoperable.

ACTION 35 - With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 36 - a) With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

b) With both channels inoperable on a 4.16Kv ESS bus, declare the associated 4.16Kv ESS bus inoperable, and take the ACTION required by Specification 3.8.3.1 or 3.8.3.2 as appropriate.

c) With both channels inoperable on the 480V ESS Bus 0B565, declare the 480V ESS Bus 0B565 not energized;

(1) For the Diesel Generator E aligned to the Class 1E system, take the ACTION required by Specification 3.8.3.1 or 3.8.3.2 as appropriate.

(2) For the Diesel Generator E not aligned to the Class 1E system, declare the Diesel Generator E 125 Volt DC distribution system load group not energized and take the ACTION required by Specification 3.8.3.1 or 3.8.3.2 as appropriate.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. <u>CORE SPRAY SYSTEM</u>				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1,2,3,4*,5*
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. Reactor Vessel Steam Dome Pressure - Low	NA	Q	Q	1,2,3,4*,5*
d. Manual Initiation	NA	R	NA	1,2,3,4*,5*
2. <u>LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM</u>				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1,2,3,4*,5*
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. Reactor Vessel Steam Dome Pressure - Low				
1) System Initiation	NA	Q	Q	1,2,3,4*,5*
2) Recirculation Discharge Valve Closure	NA	Q	Q	1,2,3,4*,5*
d. Manual Initiation	NA	R	NA	1,2,3,4*,5*
3. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM#</u>				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1,2,3
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. Condensate Storage Tank Level - Low	NA	Q	Q	1,2,3
d. Suppression Pool Water Level - High	NA	Q	Q	1,2,3
e. Reactor Vessel Water Level - High, Level 8	NA	Q	Q	1,2,3
f. Manual Initiation	NA	R	NA	1,2,3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. AUTOMATIC DEPRESSURIZATION SYSTEM##				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1,2,3
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. ADS Timer	NA	Q	Q	1,2,3
d. Core Spray Pump Discharge Pressure - High	NA	Q	Q	1,2,3
e. RHR LPCI Mode Pump Discharge Pressure - High	NA	Q	Q	1,2,3
f. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1,2,3
g. ADS Drywell Pressure Bypass Timer	NA	Q	Q	1,2,3
h. Manual Inhibit	NA	R	NA	1,2,3
i. Manual Initiation	NA	R	NA	1,2,3
5. LOSS OF POWER				
a. 4.16 kv ESS Bus Undervoltage (Loss of Voltage)	NA	NA	R	1,2,3,4**,5**
b. 4.16 kv ESS Bus Undervoltage (Degraded Voltage)	S	M	R	1,2,3,4**,5**
c. 4.16 kv ESS Bus Undervoltage (Degraded Voltage)	S	M	R	1,2,3,4**,5**
d. 480V ESS Bus 0B565 Undervoltage (Degraded Voltage < 65%)###	S	M	R	1,2,3,4**,5**
e. 480V ESS Bus 0B565 Undervoltage (Degraded Voltage < 92%)###	S	M	R	1,2,3,4**,5**

* When the system is required to be OPERABLE, after being manually realigned, as applicable, per specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

\#\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\#\#\# Required to be OPERABLE only when Diesel Generator E is either aligned to the Class 1E system or not aligned to the Class 1E system but operating on the Test Facility.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more channels required by Table 3.3.4.1-1 inoperable:

   1. Within 14 days, restore the channel(s) to OPERABLE status; or,

   2. Within 14 days, place the channel(s) in the tripped condition if the inoperable channel(s) is not the result of an inoperable breaker.

c. With one Trip Function in Table 3.3.4.1-1 with ATWS-RPT trip capability not maintained, restore ATWS-RPT trip capability within 72 hours.

d. With both Trip Functions in Table 3.3.4.1-1 with ATWS-RPT trip capability not maintained, restore ATWS-RPT trip capability for one Trip Function within 1 hour.

e. If ACTION b, c or d is not met, remove associated recirculation pump from service within 6 hours; or, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recalculation pump instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Steam Dome Pressure - High	2

[a] One channel or trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided the other trip system is OPERABLE. Upon determination that a trip setpoint cannot be restored to within its specified value during performance of the CHANNEL CALIBRATION, the appropriate ACTION shall be followed.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	Q	. R
2. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more channels required by Table 3.3.4.1-2 inoperable:

   1. Within 72 hours, restore the channel(s) to OPERABLE status; or,

   2. Within 72 hours, place the channel(s) in the tripped condition if the inoperable channel(s) is not the result of an inoperable breaker.

c. With one or more Trip Functions in Table 3.3.4.1-2 with EOC-RPT trip capability not maintained; and, with MCPR less than the limit specified in the COLR for inoperable EOC-RPT:

   1. Within 2 hours, restore EOC-RPT trip capability; or,

   2. Within 2 hours, apply the MCPR limit for inoperable EOC-RPT as specified in the COLR and take the ACTION required by Specification 3.2.3.

d. If ACTION b or c is not met:

   1. Remove associated recirculation pump from service within 4 hours; or,

   2. Reduce THERMAL POWER to less than 25% of RATED THERMAL POWER.

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 6 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 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closusre	Q	R
2. Turbine Control Valve - Fast Closure	Q	R

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	2	50
b. Reactor Vessel Water Level - High, Level 8	2[(b)]	51
c. Condensate Storage Tank Water Level - Low	2[(c)]	52
d. Manual Initiation	1/system[(d)]	53

(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 other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) One trip system with one-out-of-two logic.

(d) One trip system with one channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTION STATEMENTS

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement in one or more Trip Systems:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or declare RCIC inoperable; and,
b. Within one hour from discovery of loss of RCIC initiation capability, declare RCIC inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement:

a. Within 24 hours, restore the channel(s) to OPERABLE status or declare RCIC inoperable.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or align the RCIC pump suction to the suppression pool; and,
b. Within one hour of discovery of loss of initiation capability, declare RCIC inoperable unless RCIC pump suction is aligned to the suppression pool;
c. If ACTION a or b is not met, declare RCIC inoperable.

ACTION 53 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. Within 24 hours, restore the channel(s) to OPERABLE status or declare RCIC inoperable.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
b. Reactor Vessel Water Level - High , Level 8	S	Q	R
c. Condensate Storage Tank Water Level - Low	NA	Q	Q
d. Manual Initiation	NA	R	NA

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2; declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1. The provisions of Specification 3.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs, SRMs and the Neutron Flux - Upscale, Startup function of the APRMs.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1. The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs, SRMs and the Neutron Flux - Upscale, Startup function of the APRMs.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the trip system in the tripped condition, provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

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 12 hours.

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.
f. This function is required to be OPERABLE only prior to and during Shutdown Margin demonstrations as performed per Specification 3.10.3.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	Q	Q	1*
b. Inoperative	NA	Q	NA	1*
c. Downscale	NA	Q	Q	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	S	Q	SA	1
b. Inoperative	NA	Q	NA	1,2,5***
c. Downscale	S	Q	SA	1
d. Neutron Flux - Upscale, Startup	S	Q	SA	2,5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W	NA	2,5
b. Upscale	NA	S/U[(b)],W	Q	2,5
c. Inoperative	NA	S/U[(b)],W	NA	2,5
d. Downscale	NA	S/U[(b)],W	Q	2,5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W	NA	2,5
b. Upscale	S	S/U[(b)],W	Q	2,5
c. Inoperative	NA	S/U[(b)],W	NA	2,5
d. Downscale	S	S/U[(b)],W	Q	2,5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1,2,5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	Q	Q	1
b. Inoperative	NA	Q	NA	1
c. Comparator	NA	Q	Q	1

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Main Control Room Outside Air Intake Radiation Monitor	2/intake[c]	1,2,3,5 and *	$\leq$ 5 mR/hr	0.01 to 100 mR/hr	70[c]
2. Area Monitors					
a. Criticality Monitors					
1) New Fuel Storage Vault	2	(a)	$\leq$ 15 mR/hr	10^{-1} to 10^{3} mR/hr	71
2) Spent Fuel Storage Pool	2	(b)	$\leq$ 15 mR/hr	10^{-1} to 10^{3} mR/hr	71

* When irradiated fuel is being handled in the secondary containment.

(a) With fuel in the new fuel storage vault.

(b) With fuel in the spent fuel storage pool.

(c) When a channel is placed in an inoperable status solely for performance of required Surveillances, ACTIONS may be delayed for up to 6 hours provided control room emergency filtration system initiation capability is maintained.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Main Control Room Outside Air Intake Radiation Monitor	S	Q	R	1,2,3,5 and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	(a)
2) Spent Fuel Storage Pool	S	M	R	(b)

(a) With fuel in the new fuel storage vault.

(b) With fuel in the spent fuel storage pool.

* When irradiated fuel is being handled in the secondary containment.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE* channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE* channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

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

* When a channel is placed in an inoperable status solely for performance of required Surveillances, ACTIONS may be delayed for up to 6 hours provided feedwater/main turbine trip capability is maintained.

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Reactor Vessel Water Level - High	D	Q	R	1

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Leak detection temperature setpoints are selected to prevent a high energy line break by detecting and isolating leakage below the flow rate corresponding to critical crack size for the respective system piping. The setpoints are also set below fire suppression setpoints (HPCI and RCIC) and high enough to avoid inadvertent isolation caused by normal temperature transients or abnormal transients caused by non-leak conditions (such as loss of ventilation).

The Reactor Vessel Water Level - Low, Level 3 Function that isolates the RHR System Shutdown Cooling is only required to be OPERABLE in OPERATIONAL CONDITIONS 3, 4, and 5 to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. If an inoperable channel is not restored to OPERABLE status or placed in trip within the allowed completion time, the associated penetration flow path should be closed. However, if the shutdown cooling function is needed to provide core cooling, ACTION 27 allows the penetration flow path to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated). ACTION 27 must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated. Only one trip system is required in OPERATIONAL CONDITIONS 4 and 5 when RHR shutdown cooling system integrity is maintained meaning piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For DC. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay sensor response is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," and NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation."

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.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Valve 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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology with Demonstration for BWR ECCS Actuation Instrumentation," Parts 1 and 2, and RE-022, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Susquehanna Steam Electric Station, Units 1 and 2."

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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated July 21, 1992.

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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric report GENE-770-06-2, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated September 13, 1991.

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).

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," and GENE-770-06-2, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated September 13, 1991.

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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing for the main Control Room Outside Air Intake Radiation Monitor have been determined in accordance with General Electric report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated July 21, 1992.

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.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

DELETED

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

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

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated July 21, 1992.

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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.

REMOVE	INSERT
3/4 3-1	3/4 3-1
3/4 3-7	3/4 3-7
3/4 3-9	3/4 3-9
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
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3/4 3-29	3/4 3-29
3/4 3-29a	3/4 3-29a
3/4 3-30	3/4 3-30
---	3/4 3-30a
3/4 3-34	3/4 3-34
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3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40
3/4 3-42	3/4 3-42
3/4 3-45	3/4 3-45
3/4 3-47	3/4 3-47
3/4 3-48	3/4 3-48
3/4 3-50	3/4 3-50
3/4 3-51	3/4 3-51
3/4 3-53	3/4 3-53
3/4 3-55	3/4 3-55
3/4 3-58	3/4 3-58
3/4 3-60	3/4 3-60
3/4 3-95	3/4 3-95
3/4 3-98	3/4 3-98
B 3/4 3-2	B 3/4 3-2
---	B 3/4 3-2a
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4
---	B 3/4 3-4a
B 3/4 3-7	B 3/4 3-7

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. 126
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 February 10, 1995, as supplemented by letter dated November 10, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 126 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 18, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 126

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.

REMOVE	INSERT
3/4 3-1	3/4 3-1
3/4 3-7	3/4 3-7
3/4 3-9	3/4 3-9
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3/4 3-12	3/4 3-12
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3/4 3-58	3/4 3-58
3/4 3-60	3/4 3-60
3/4 3-96	3/4 3-96
3/4 3-99	3/4 3-99
B 3/4 3-2	B 3/4 3-2
---	B 3/4 3-2a
B 3/4 3-3	B 3/4 3-3
---	B 3/4 3-3a
B 3/4 3-4	B 3/4 3-4
---	B 3/4 3-4a
B 3/4 3-7	B 3/4 3-7

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With one or more required channels inoperable in one trip system, place the inoperable channel(s) or the associated trip system in the tripped condition within 12 hours.

b. With one or more required channels inoperable in both trip systems, place the inoperable channel(s) in one trip system or one trip system in the tripped condition within 6 hours.*

c. With one or more RPS Functions with RPS trip capability not maintained, restore RPS trip capability within one hour.

d. If ACTION a or b or c is not met, take the ACTION required by Table 3.3.1-1 for the RPS Function.

The provisions of Specification 3.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 or 3 from OPERATIONAL CONDITION 1 for the IRMs or the Neutron Flux - Upscale, Setdown function for the APRMs.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

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

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit ~~shall be~~ demonstrated to be within its limit at least once per 18 months.** Each test ~~shall~~ include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels ~~in~~ a specific reactor trip system.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs or the Neutron Flux - Upscale, Setdown function of the APRMs.

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* If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause a scram to occur.

** Neutron detectors are exempt from response time testing.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,[b]	S/U[c],W	SA	2
	S	W	SA	3, 4, 5
b. Inoperative	NA	S/U[c],W	NA	2, 3, 4, 5
2. Avg. Power Range Monitor[f]				
a. Neutron Flux - Upscale, Setdown	S/U,S,[b]	S/U[c],W	SA	2
	S	W	SA	3,5[k]
b. Flow Biased Simulated Thermal Power - Upscale	S,D[g]	S/U[c],Q	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c],Q	W[d],SA	1
d. Inoperative	NA	S/U[c],Q	NA	1,2,3,5[k]
3. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q	1,2
4. Reactor Vessel Water Level - Low, Level 3	S	Q	Q	1,2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1,2[i]
7. Drywell Pressure - High	NA	Q	R	1,2

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required* by the Minimum OPERABLE Channels per Trip System in one or more Trip Systems:

   1. For functions 1.a.1, 1.b, 1.e, 2.b, 3.b, 7.a, and 7.e, place the channel in the tripped condition within 12 hours; and,

   2. For functions other than 1.a.1, 1.b, 1.e, 2.b, 3.b, 7.a, and 7.e, place the channel in the tripped condition within 24 hours.

   The provisions of Specification 3.0.4 are not applicable.

c. With one or more automatic functions with isolation capability not maintained, restore isolation capability within one hour.

d. If ACTIONS b or c are not met, take the ACTION required by Table 3.3.2-1 for the function.

* When a channel is placed in an inoperable status solely for performance of required Surveillances, initiation of ACTIONS may be delayed for up to 6 hours provided the associated Trip Function maintains isolation capability.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

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

\# For Unit 2 Cycle 6 operation, the Non-Regenerative Heat Exchanger Discharge Temperature - High Channel shall be OPERABLE in place of RWCU Flow - High Channel 'B'.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>				
a. RCIC Steam Line Δ Pressure - High	K	1	1,2,3	23
b. RCIC Steam Supply Pressure - Low	KB	2	1,2,3	23
c. RCIC Turbine Exhaust Diaphragm Pressure - High	K	2	1,2,3	23
d. RCIC Equipment Room Temperature - High	K	1	1,2,3	23
e. RCIC Equipment Room Δ Temperature - High	K	1	1,2,3	23
f. RCIC Pipe Routing Area Temperature - High	K	1	1,2,3	23
g. RCIC Pipe Routing Area Δ Temperature - High	K	1	1,2,3	23
h. RCIC Emergency Area Cooler Temperature - High	K	1	1,2,3	23
i. Manual Initiation	NA	1	1,2,3	24
j. Drywell Pressure - High	Z	2	1,2,3	23

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
6. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u>				
a. HPCI Steam Line Δ Pressure - High	L	1	1,2,3	23
b. HPCI Steam Supply Pressure - Low	LB	2	1,2,3	23
c. HPCI Turbine Exhaust Diaphragm Pressure - High	L	2	1,2,3	23
d. HPCI Equipment Room Temperature - High	L	1	1,2,3	23
e. HPCI Equipment Room Δ Temperature - High	L	1	1,2,3	23
f. HPCI Emergency Area Cooler Temperature - High	L	1	1,2,3	23
g. HPCI Pipe Routing Area Temperature - High	L	1	1,2,3	23
h. HPCI Pipe Routing Area Δ Temperature - High	L	1	1,2,3	23
i. Manual Initiation	NA	1	1,2,3	24
j. Drywell Pressure - High	Z	2	1,2,3	23

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(S)(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
7. <u>RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION</u>				
a. Reactor Vessel Water Level - Low, Level 3	A	2*	3,4,5	27
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	UB	1	1,2,3	26
c. RHR Flow - High	M	1	1,2,3	26
d. Manual Initiation	NA	1	1,2,3	24
e. Drywell Pressure - High	Z	2	1,2,3	26

* Only one trip system required in OPERATIONAL CONDITIONS 4 and 5 when RHR Shutdown Cooling System integrity is maintained.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20	Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
ACTION 21	Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
ACTION 22	Be in at least STARTUP within 6 hours.
ACTION 23	Close the affected system isolation valves within 1 hour and declare the affected system inoperable.
ACTION 24	Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
ACTION 25	Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.
ACTION 26	Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.
ACTION 27	Initiate action to restore channel to OPERABLE status; or, initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.

NOTES

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

** Actuates dampers shown in Table 3.6.5.2-1.

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

\# When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with the potential for draining the reactor vessel. Single control rod movement, except for the purpose of SDM demonstration (TS 3.10.3), is excluded.

\#\# When handling irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves which are actuated by these isolation signals.

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

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	≤ 167°F##	≤ 174°F##
g. RCIC Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##
h. RCIC Emergency Area Cooler Temperature - High	≤ 167°F	≤ 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 Temperature - 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

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

\# For Unit 2 Cycle 6 operation, the Non-Regenerative Heat Exchanger Discharge Temperature - High Channel shall be OPERABLE in place of RWCU Flow - High Channel 'B'.

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

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
<u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u> (continued)				
d. HPCI Equipment Room Temperature - High	NA	Q	Q	1,2,3
e. HPCI Equipment Room Δ Temperature - High	NA	Q	Q	1,2,3
f. HPCI Emergency Area Cooler Temperature - High	NA	Q	Q	1,2,3
g. HPCI Pipe Routing Area Temperature - High	NA	Q	Q	1,2,3
h. HPCI Pipe Routing Area Δ Temperature - High	NA	Q	Q	1,2,3
i. Manual Initiation	NA	R	NA	1,2,3
j. Drywell Pressure - High	NA	Q	R	1,2,3
7. <u>RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION</u>				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	3,4,5
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	Q	Q	1,2,3
c. RHR Flow - High	S	Q	R	1,2,3
d. Manual Initiation	NA	R	NA	1,2,3
e. Drywell Pressure - High	NA	Q	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.

\# When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with the potential for draining the reactor vessel. Single control rod movement, except for the purpose of SDM demonstration (TS 3.10.3), is excluded.

\## When handling irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open.

TABLE 3.3.3-1
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(a)	1,2,3,4*,5*	30
b. Drywell Pressure - High	2(a)	1,2,3	30
c. Reactor Vessel Steam Dome Pressure - Low (Permissive)	2(a)	1,2,3, 4*,5*	31 30
d. Manual Initiation	1/subsystem	1,2,3,4*,5*	31
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(a)	1,2,3,4*,5*	30
b. Drywell Pressure - High	2(a)	1,2,3	30
c. Reactor Vessel Steam Dome Pressure - Low (Permissive)			
1. System Initiation	2(a)	1,2,3, 4*,5*	31 30
2. Recirculation Discharge Valve Closure	2(a)	1,2,3, 4*,5*	31 30
d. Manual Initiation	1/subsystem	1,2,3,4*,5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Water Level - Low Low, Level 2	2(a)	1,2,3	30
b. Drywell Pressure - High	2(a)	1,2,3	30
c. Condensate Storage Tank Level - Low	2(a)(b)	1,2,3	34
d. Suppression Pool Water Level - High***	2(a)	1,2,3	34
e. Reactor Vessel Water Level - High, Level 8	2(f)	1,2,3	31
f. Manual Initiation	1/system	1,2,3	31

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. <u>AUTOMATIC DEPRESSURIZATION SYSTEM</u>##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(a)]	1,2,3	32
b. Drywell Pressure - High	2[(a)]	1,2,3	32
c. ADS Timer	1[(a)]	1,2,3	33
d. Core Spray Pump Discharge Pressure - High (Permissive)	2[(d)(a)]	1,2,3	33
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	2[(d)(e)(a)]	1,2,3	33
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1[(a)]	1,2,3	32
g. ADS Drywell Pressure Bypass Timer	2[(a)]	1,2,3	33
h. Manual Inhibit	1	1,2,3	33
i. Manual Initiation	1/valve	1,2,3	33

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. <u>LOSS OF POWER</u>					
a. 4.16 kv ESS Bus Under-voltage (Loss of Voltage, < 20%)	1/bus	1/bus	1/bus	1,2,3,4**,5**	35
b. 4.16 kv ESS Bus Under-voltage (Degraded Voltage, < 65%)	2/bus	2/bus	2/bus	1,2,3,4**,5**	36
c. 4.16 kv ESS Bus Under-voltage (Degraded Voltage, < 93%)	2/bus	2/bus	2/bus	1,2,3,4**,5**	36
d. 480V ESS Bus 0B565 Under-voltage (Degraded Voltage, < 65%)###	2/bus	1/bus	2/bus	1,2,3,4**,5**	36
e. 480V ESS Bus 0B565 Under-voltage (Degraded Voltage, < 92%)###	2/bus	2/bus	2/bus	1,2,3,4**,5**	36

See footnotes on next page.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

(a) When a channel is placed in an inoperable status solely for performance of required Surveillances, initiation of ACTIONS may be delayed for up to 6 hours provided the associated Trip Function maintains trip capability.

(b) One trip system. Provides signal to HPCI pump suction valves only.

(c) Two out of two logic.

(d) Either 4d or 4e must be satisfied. The ACTION is required to be taken only if neither is satisfied. A channel is not OPERABLE unless its associated pump is OPERABLE per Specification 3.5.1.

(e) Within an ADS Trip System there are two logic subsystems, each of which contains an overall pump permissive. At least one channel associated with each of these overall pump permissives shall be OPERABLE.

(f) When a channel is placed in an inoperable status solely for performance of required Surveillances, initiation of ACTIONS may be delayed for up to 6 hours.

* When the system is required to be OPERABLE per Specification 3.5.2.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

** Required when ESF equipment is required to be OPERABLE.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

Required to be OPERABLE only when Diesel Generator E is either aligned to the Class 1E system or not aligned to the Class 1E system but operating on the Test Facility.

*** The automatic transfer of HPCI pump suction from the condensate storage tank to suppression pool on high suppression pool water level occurs only when HPCI injection valve is open.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 30 - With one or more required channel(s) inoperable in one or more Trip Systems:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or declare the associated ECCS inoperable; and,

b. Within one hour from discovery of loss of initiation capability by this trip function, declare the associated ECCS inoperable.

ACTION 31 - With one or more required channel(s) inoperable in one or more Trip Systems:

a. Within 24 hours, restore the channel(s) to OPERABLE status or declare the associated ECCS inoperable; and

b. Within one hour from discovery of loss of initiation capability by this trip function, declare the associated ECCS inoperable.*

(*ACTION 31b is not applicable to Function 3.e, Reactor Vessel Water Level - High, Level 8)

ACTION 32 - With one or more required channel(s) inoperable in one or more Trip Systems:

a. Within one hour from discovery of loss of ADS initiation capability by this trip function, declare the ADS valves inoperable; and,

b. Within 4 days from discovery of an inoperable channel(s) concurrent with HPCI or RCIC inoperable, place the inoperable channel(s) in the tripped condition; and,

c. Within 8 days from discovery of an inoperable channel(s) if both HPCI and RCIC are OPERABLE, place the inoperable channel(s) in the tripped condition.

d. If ACTION b or c is not met, declare ADS inoperable.

ACTION 33 - With one or more required channel(s) inoperable in one or more Trip Systems:

a. Within one hour from discovery of loss of ADS initiation capability by this trip function, declare the ADS valves inoperable; and,

b. Within 4 days from the discovery of an inoperable channel(s) concurrent with HPCI or RCIC inoperable, restore the channel(s) to OPERABLE status; and,

c. Within 8 days from the discovery of an inoperable channel(s) if both HPCI and RCIC are OPERABLE, restore the channel(s) to OPERABLE status.

d. If ACTION b or c is not met, declare ADS inoperable.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 34 - With one or more required channel(s) inoperable in one or more Trip Functions:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or align the HPCI pump suction to the suppression pool; and,

b. Within one hour of discovery of loss of initiation capability declare HPCI inoperable if the associated pump suction is not aligned to the suppression pool.

c. If ACTION a or b is not met, declare HPCI inoperable.

ACTION 35 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 36 -

a. With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

b. With both channels inoperable on a 4.16Kv ESS bus, declare the associated 4.16Kv ESS bus inoperable, and take the ACTION required by Specification 3.8.3.1 or 3.8.3.2 as appropriate.

c. With both channels inoperable on the 480V ESS Bus OB565, declare the 480V ESS Bus OB565 not energized;

(1) For the Diesel Generator E aligned to the Class 1E system, take the ACTION required by Specification 3.8.3.1 or 3.8.3.2 as appropriate.

(2) For the Diesel Generator E not aligned to the Class 1E system, declare the Diesel Generator E 125 Volt DC distribution system load group not energized and take the ACTION required by Specification 3.8.3.1 or 3.8.3.2 as appropriate.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. <u>CORE SPRAY SYSTEM</u>				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1,2,3,4*,5*
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. Reactor Vessel Steam Dome Pressure - Low	NA	Q	Q	1,2,3,4*,5*
d. Manual Initiation	NA	R	NA	1,2,3,4*,5*
2. <u>LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM</u>				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1,2,3,4*,5*
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. Reactor Vessel Steam Dome Pressure - Low				
1) System Initiation	NA	Q	Q	1,2,3,4*,5*
2) Recirculation Discharge Valve Closure	NA	Q	Q	1,2,3,4*,5*
d. Manual Initiation	NA	R	NA	1,2,3,4*,5*
3. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM</u>#				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1,2,3
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. Condensate Storage Tank Level - Low	NA	Q	Q	1,2,3
d. Suppression Pool Water Level - High	NA	Q	Q	1,2,3
e. Reactor Vessel Water Level - High, Level 8	NA	Q	Q	1,2,3
f. Manual Initiation	NA	R	NA	1,2,3

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. <u>AUTOMATIC DEPRESSURIZATION SYSTEM</u>##				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1,2,3
b. Drywell Pressure - High	NA	Q	Q	1,2,3
c. ADS Timer	NA	Q	Q	1,2,3
d. Core Spray Pump Discharge Pressure - High	NA	Q	Q	1,2,3
e. RHR LPCI Mode Pump Discharge Pressure - High	NA	Q	Q	1,2,3
f. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1,2,3
g. ADS Drywell Pressure Bypass Timer	NA	Q	Q	1,2,3
h. Manual Inhibit	NA	R	NA	1,2,3
i. Manual Initiation	NA	R	NA	1,2,3
5. <u>LOSS OF POWER</u>				
a. 4.16 kv ESS Bus Undervoltage (Loss of Voltage)	NA	NA	R	1,2,3,4**,5**
b. 4.16 kv ESS Bus Undervoltage (Degraded Voltage)	S	M	R	1,2,3,4**,5**
c. 4.16 kv ESS Bus Undervoltage (Degraded Voltage)	S	M	R	1,2,3,4**,5**
d. 480V ESS Bus 0B565 Undervoltage (Degraded Voltage < 65%)###	S	M	R	1,2,3,4**,5**
e. 480V ESS Bus 0B565 Undervoltage (Degraded Voltage < 92%)###	S	M	R	1,2,3,4**,5**

* When the system is required to be OPERABLE, after being manually realigned, as applicable, per Specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

\## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\### Required to be OPERABLE only when Diesel Generator E is either aligned to the Class 1E system or not aligned to the Class 1E system but operating on the Test Facility.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more channels required by Table 3.3.4.1-1 inoperable:

   1. Within 14 days, restore the channel(s) to OPERABLE status; or,

   2. Within 14 days, place the channel(s) in the tripped condition if the inoperable channel(s) is not the result of an inoperable breaker.

c. With one Trip Function in Table 3.3.4.1-1 with ATWS-RPT trip capability not maintained, restore ATWS-RPT trip capability within 72 hours.

d. With both Trip Functions in Table 3.3.4.1-1 with ATWS-RPT trip capability not maintained, restore ATWS-RPT trip capability for one Trip Function within 1 hour.

e. If ACTION b, c or d is not met, remove associated recirculation pump from service within 6 hours; or, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Steam Dome Pressure - High	2

(a) One channel or trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided the other trip system is OPERABLE. Upon determination that a trip setpoint cannot be restored to within its specified value during performance of the CHANNEL CALIBRATION, the appropriate ACTION shall be followed.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
2. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more channels required by Table 3.3.4.1-2 inoperable:

   1. Within 72 hours, restore the channel(s) to OPERABLE status; or,

   2. Within 72 hours, place the channel(s) in the tripped condition if the inoperable channel(s) is not the result of an inoperable breaker.

c. With one or more Trip Functions in Table 3.3.4.1-2 with EOC-RPT trip capability not maintained; and, with MCPR less than the limit specified in the COLR for inoperable EOC-RPT:

   1. Within 2 hours, restore EOC-RPT trip capability; or,

   2. Within 2 hours, apply the MCPR limit for inoperable EOC-RPT as specified in the COLR and take the ACTION required by Specification 3.2.3.

d. If ACTION b or c is not met:

   1. Remove associated recirculation pump from service within 4 hours; or,

   2. Reduce THERMAL POWER to less than 25% of RATED THERMAL POWER.

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 6 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 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closusre	Q	R
2. Turbine Control Valve - Fast Closure	Q	R

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	2	50
b. Reactor Vessel Water Level - High, Level 8	2[b]	51
c. Condensate Storage Tank Water Level - Low	2[c]	52
d. Manual Initiation	1/system[d]	53

(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 other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) One trip system with one-out-of-two logic.

(d) One trip system with one channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTION STATEMENTS

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement in one or more Trip Systems:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or declare RCIC inoperable; and,
b. Within one hour from discovery of loss of RCIC initiation capability, declare RCIC inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement:

a. Within 24 hours, restore the channel(s) to OPERABLE status or declare RCIC inoperable.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. Within 24 hours, place the inoperable channel(s) in the tripped condition or align the RCIC pump suction to the suppression pool; and,
b. Within one hour of discovery of loss of initiation capability, declare RCIC inoperable unless RCIC pump suction is aligned to the suppression pool;
c. If ACTION a or b is not met, declare RCIC inoperable.

ACTION 53 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. Within 24 hours, restore the channel(s) to OPERABLE status or declare RCIC inoperable.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
b. Reactor Vessel Water Level - High , Level 8	S	Q	R
c. Condensate Storage Tank Water Level - Low	NA	Q	Q
d. Manual Initiation	NA	R	NA

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1. The provisions of Specification 3.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs, SRMs, and the Neutron Flux - Upscale, Startup function of the APRMs.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1. The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs, SRMs, and the Neutron Flux - Upscale, Startup function of the APRMs.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the trip system in the tripped condition, provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

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 12 hours.

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.
(f) This function is required to be OPERABLE only prior to and during Shutdown Margin demonstrations as performed per Specification 3.10.3.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	Q	SA	1*
b. Inoperative	NA	Q	NA	1*
c. Downscale	NA	Q	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	S	Q	SA	1
b. Inoperative	NA	Q	NA	1,2,5***
c. Downscale	S	Q	SA	1
d. Neutron Flux - Upscale, Startup	S	Q	SA	2,5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W	NA	2,5
b. Upscale	NA	S/U[(b)],W	SA	2,5
c. Inoperative	NA	S/U[(b)],W	NA	2,5
d. Downscale	NA	S/U[(b)],W	SA	2,5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W	NA	2,5
b. Upscale	S	S/U[(b)],W	SA	2,5
c. Inoperative	NA	S/U[(b)],W	NA	2,5
d. Downscale	S	S/U[(b)],W	SA	2,5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1,2,5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	Q	Q	1
b. Inoperative	NA	Q	NA	1
c. Comparator	NA	Q	Q	1

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Main Control Room Outside Air Intake Radiation Monitor	2/intake[(c)]	1,2,3,5 and *	≤ 5 mR/hr	0.01 to 100 mR/hr	70[(c)]
2. Area Monitors					
a. Criticality Monitors					
1) New Fuel Storage Vault	2	(a)	≤ 15 mR/hr	10^{-1} to 10^{3} mR/hr	71
2) Spent Fuel Storage Pool	2	(b)	≤ 15 mR/hr	10^{-1} to 10^{3} mR/hr	71

* When irradiated fuel is being handled in the secondary containment.

(a) With fuel in the new fuel storage vault.

(b) With fuel in the spent fuel storage pool.

(c) When a channel is placed in an inoperable status solely for performance of required Surveillances, ACTIONS may be delayed for up to 6 hours provided control room emergency filtration system initiation capability is maintained.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Main Control Room Outside Air Intake Radiation Monitor	S	Q	R	1,2,3,5 and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	(a)
2) Spent Fuel Storage Pool	S	M	R	(b)

(a) With fuel in the new fuel storage vault.
(b) With fuel in the spent fuel storage pool.
* When irradiated fuel is being handled in the secondary containment.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE* channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE* channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

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

* When a channel is placed in an inoperable status solely for performance of required Surveillances, ACTIONS may be delayed for up to 6 hours provided feedwater/main turbine trip capability is maintained.

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Reactor Vessel Water Level - High	D	Q	R	1

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inopeable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Leak detection temperature setpoints are selected to prevent a high energy line break by detecting and isolating leakage below the flow rate corresponding to critical crack size for the respective system piping. The setpoints are also set below fire suppression setpoints (HPCI and RCIC) and high enough to avoid inadvertent isolation caused by normal temperature transients or abnormal transients caused by non-leak conditions (such as loss of ventilation).

The Reactor Vessel Water Level - Low, Level 3 Function that isolates the RHR System Shutdown Cooling is only required to be OPERABLE in OPERATIONAL CONDITIONS 3, 4, and 5 to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. If an inoperable channel is not restored to OPERABLE status or placed in trip within the allowed completion time, the associated penetration flow path should be closed. However, if the shutdown cooling function is needed to provide core cooling, ACTION 27 allows the penetration flow path to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated). ACTION 27 must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated. Only one trip system is required in OPERATIONAL CONDITIONS 4 and 5 when RHR shutdown cooling system integrity is maintained meaning piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For DC. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay sensor response is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," and NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation."

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Valve 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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology with Demonstration for BWR ECCS Actuation Instrumentation," Parts 1 and 2, and RE-022, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Susquehanna Steam Electric Station, Units 1 and 2."

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 or 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 Operation 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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated July 21, 1992.

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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric report GENE-770-06-2, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated September 13, 1991.

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).

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," and GENE-770-06-2, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated September 13, 1991.

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.

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing for the main Control Room Outside Air Intake Radiation Monitor have been determined in accordance with General Electric report GENE-770-06-1, 'Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated July 21, 1992.

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.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

DELETED

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

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

CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," and the associated NRC Safety Evaluation Report dated July 21, 1992.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letter dated February 10, 1995, as supplemented by letter dated November 10, 1995, the Pennsylvania Power and Light Company (the licensee) submitted a request for changes to the Susquehanna Steam Electric Station, Units 1 and 2, Technical Specifications (TSs). The supplemental letter provided corrected TSs and did not change the original proposed no significant hazards consideration nor the Federal Register notice. The requested changes would modify the Susquehanna Steam Electric Station, Unit 1 and 2 Technical Specifications to extend the allowable out-of-service times (AOTs) for maintenance and repair and the surveillance test intervals (STIs) between channel functional tests for the following groups of instruments: reactor protection systems instrumentation (TS 3.3.1), isolation actuation instrumentation (TS 3.3.2), emergency core cooling system actuation instrumentation (TS 3.3.3), ATWS (anticipated transient without scram) recirculation pump trip system instrumentation (TS 3.3.4.1), end-of-cycle recirculation pump trip system instrumentation (TS 3.3.4.2), reactor core isolation cooling system (RCIC) actuation instrumentation (TS 3.3.5), control rod block instrumentation (TS 3.3.6), radiation monitoring instrumentation (TS 3.3.7.1), and feedwater/main turbine trip system actuation instrumentation (TS 3.3.90); (2) change the required actions and AOTs for the instruments listed above to make requirements consistent with supporting analysis in General Electric topical reports and change additional actions required to prevent extended AOTs from resulting in extended loss of instrument function; (3) change the required actions and AOTs for the instruments listed above for instrumentation associated with the ADS (automatic depressurization system), recirculation pump trip, and pump suction lineup for HPCI (high pressure core injection) and RCIC; (4) change applicability requirements and required actions for the reactor vessel water level-low, level 3 function that isolates the RHR (residual heat removal) system shutdown cooling system so that the function is required to be operable in operational conditions 3,4, and 5 to prevent inadvertent loss of reactor coolant via the RHR shutdown cooling system; (5) remove notes in Table 3.3.2-1, 3.3.2-2, and 4.3.1-1 related to maintenance on leak detection temperature detectors and remove the note to TS 3.3.6 for Unit 1 related to a previous relief from TS 3.0.4; and (6) reformat, renumber, and/or reword existing requirements to incorporate the changes listed above. The licensee stated in its request that the proposed changes are consistent with the NRC staff's previous approvals of several

General Electric Company (GE) Licensing Topical Reports (LTRs). The licensee's submittal also stated that the proposed AOTs are consistent with the guidance provided in NUREG-1433, Standard Technical Specifications for General Electric Plants, BWR/4. The proposed changes would permit specified instrument channel functional tests to be performed quarterly rather than once per week or once per month. During the review of the licensee's TS page submittals, the staff identified a number of minor editorial errors in the camera ready versions and in the original TS page markups. These items have been discussed with the licensee and TS pages with corrections for each item have been included in the amendment package.

2.0 EVALUATION

The licensee has proposed changes to TS sections listed above based on the NRC staff's previous approvals of the following GE LTRs:

1. S. Visweswaran, et al., "BWR Owners' Group Response to NRC Generic Letter 83-28, Item 4.5.3," General Electric Company, NEDC-30844A, March 1988.

2. W. P. Sullivan, et al., "Technical Specification Improvement Analyses for BWR Reactor Protection System," General Electric Company, NEDC-30851P-A, March 1988.

3. D. B. Atcheson, et al., "BWR Owners' Group Technical Specification Improvement Methodology with Demonstration for BWR ECCS Actuation Instrumentation," Parts 1 and 2, General Electric Company, NEDC-30936P-A, December 1988.

4. S. Visweswaran, et al., "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," General Electric Company, NEDC-30851P-A, Supplement 1, October 1988.

5. L. G. Frederick, et al., "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," General Electric Company, NEDC-30851P, Supplement 2, July 1986.

6. W. P. Sullivan, et al., "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation," General Electric Company, NEDC-31677P-A, July 1990.

7. "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," General Electric Company, GENE-770-06-1A, December 1992.

8. "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," General Electric Company, GENE-770-06-2A, December 1992.

Each of the above letters was prepared and approved on a generic basis with requirements for individual licensees to perform plant-specific evaluations to demonstrate that the LTRs are applicable to plant-specific license amendment requests. PP&L has performed the required plant-specific evaluations for SSES. These evaluations are discussed below:

References 1 and 2:

Appendix L of Reference 2 identifies Susquehanna, Units 1 and 2, relay type BWR4s, as participants in the development of the Technical Specification Improvement Analysis discussed in Reference 2. Verification of applicability of References 1 and 2 to a specific plant is based on verification that the specific design of the plant and the reactor protection system are bounded by the assumptions and conditions used in the analyses. Appendix K to Reference 2 provides for the Reactor Protection System a "step-by-step procedure used in the plant specific application of the generic results ... of this report." Furthermore, PP&L's submittal included a copy of GE Report MDE-79-0485, April 1985 (Proprietary), "Technical Specification Improvement Analysis for the Reactor Protection system for Susquehanna Steam Electric Station, Units 1 and 2," which concludes in that the generic analysis in Reference 1 is applicable to Susquehanna and that any differences between the generic model and Susquehanna would not significantly affect the improvements in plant safety resulting from the proposed changes to AOTs and STIs for the Reactor Protection System.

Reference 3:

Reference 3 (NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology with Demonstration for BWR ECCS Actuation Instrumentation," Parts 1 and 2) provides justification for extending AOTs and STIs for Emergency Core Cooling System actuation instrumentation for a generic BWR 4. Appendix N of Part 1 and Appendix B of Part 2 of Reference 3 identifies Susquehanna, Units 1 and 2, relay type BWR4s, as participants in the development of the Technical Specification Improvement Analysis discussed in Reference 3. Part 2 of Reference 3 applied the generic analysis in Part 1 of Reference 3 to six "envelope cases" intended to ensure that all plants of each GE product line are bounded by the analysis and conclusions for the generic model. The SER supporting Reference 3 (Part 2) states that the review of Reference 3 resulted in "full confidence that the envelope models do, in fact, bound all parts of a particular product line..." The SER further states that "In order for a licensee to use the generic analysis as justification for ECCS actuation instrumentation STI and AOT changes, the licensee should provide verification that either the BWR product line generic model or one of the envelope cases provides an accurate representation of its plant ..."

"Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Susquehanna Steam Electric Station, Units 1 and 2," General Electric Company, RE-022 (DRF A00-02558E), dated January 1987) was prepared in accordance with the requirements of Appendix F to Reference 2 and constitutes a plant specific verification that Susquehanna, Units 1 and 2, are bounded by the analyses, conditions, assumptions and results of Reference 3. This document concluded that the differences between Susquehanna, Units 1 and 2, and the generic BWR are enveloped by a combination of the analyses for BWR 3/4 case 4A and BWR 5/6 case 5C described in Reference 3. This conclusion demonstrates the impact of the proposed changes to the ECCS actuation instrumentation Technical Specifications on ECCS water injection function failure meets the acceptance criteria in Reference 3, Parts 1 and 2. Therefore, the generic analysis is applicable to Susquehanna.

Reference 4:

Reference 4 (NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation") provides justification for extending AOTs and STIs for instrumentation that initiate Control Rod Blocks. Reference 4 identifies Susquehanna, Units 1 and 2, BWR4s which utilize a solid-state Reactor manual Control System (RMCS), as participants in the development of the Technical Specification Improvement Analysis discussed in Reference 4.

Although the SER associated with Reference 4 requires confirmation of the applicability of the generic analyses to a specific plant, no guidance is provided in Reference 4 or the SER for performing this verification. Therefore, PP&L's verification of applicability of Reference 4 to Susquehanna, Units 1 and 2, is based on verification that the Susquehanna is consistent with the design, conditions and any other assumptions used in the generic analysis contained in Reference 4. The results of this review are presented below:

The generic analysis in Reference 4 is based on the assumption that the analyses and conclusions of References 1 and 2 which justify extensions to RPS AOTs and STIs are applicable. As determined in Reference 9 and confirmed above in Section II.1 of this evaluation, the analyses and conclusions of References 1 and 2 are applicable to Susquehanna Units 1 and 2.

The generic analysis is based on Control Rod Block (CRB) function designs as described in Section 3 of Reference 4. The design of the each of CRB functions at Susquehanna is consistent to the level of detail presented with the design as described in Section 3 of Reference 4. Susquehanna Units 1 and 2 utilize a flow biased APRM Rod Block Monitor system which does not incorporate the APRM/RBM/Technical Specification (ARTS) improvement program modifications.

The generic analysis in Reference 4 assumes that the component failure data used in the RPS study in References 1 and 2 are applicable to instruments common to RPS and CRB and the failure rates for other instruments are assumed to be in the same range as the common instruments based on their physical similarities. The same assumptions are applicable to the CRB systems at Susquehanna.

The generic analysis in Reference 4 (Section 5) justifies extending AOTs and STIs based on a qualitative review of the consequences of a failure of each of the CRB functions. All aspects of the discussions in Reference 4, Section 5, Justification for Extending Surveillance Test Intervals, are applicable to Susquehanna, Units 1 and 2.

The PP&L review of Reference 4 and supporting documentation, did not identify any discrepancies that would invalidate the conclusion that the results of the generic analyses in Reference 4 are applicable to Susquehanna, Units 1 and 2.

Reference 5:

Reference 5 (NEDC-30851P, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation, Supplement 2 (SER dated January 6, 1994)) provides justification for extending AOTs and STIs for isolation actuation instrumentation that utilize instruments that are common to RPS or ECCS. Appendix B of Reference 5 identifies Susquehanna, Units 1 and 2, relay type BWR4s, as participants in the development of the Technical Specification Improvement Analysis discussed in Reference 5.

Although the SER associated with Reference 5 requires confirmation of the applicability of the generic analyses to a specific plant, no guidance is provided in Reference 5 or the SER for performing this verification. Therefore, PP&L's verification of applicability of Reference 5 to Susquehanna is based on verification that Susquehanna is consistent with the design, conditions and any other assumptions used in the generic analysis contained in Reference 5. The results of this review are presented below:

The generic analysis in Reference 5 is based on the assumption that the analyses and conclusions of References 1, 2 and 3 that justify extensions to RPS and ECCS AOTs and STIs are applicable. As discussed above, the analyses and conclusions show that References 1, 2 and 3 are applicable to Susquehanna, Units 1 and 2.

The generic analysis in Reference 5 assumes that the component failure data used in the RPS and ECCS reliability studies in References 1, 2 and 3 is applicable to instruments common to RPS and ECCS. This assumption is applicable to Susquehanna. No verification of this assumption was performed because the conclusion is generic to all BWR4 plants. Additionally, the results of the generic analysis were generally insensitive to instrumentation reliability.

The generic analysis in Reference 5 included a bounding analysis using a conservatively assumed single sensor controlling a single isolation valve. The results of extending AOTs and STIs on this simple instrument configuration were acceptable based on the observation that the impact of actuation device reliability, which is not affected by the proposed changes to AOTs and STIs, was dominant. Since the simple model determined the proposed AOT/STI extensions were acceptable, penetrations protected by multiple actuation devices (valves) and multiple instruments channels are bounded by the conclusions.

The PP&L review of Reference 4 and supporting documentation, did not identify any discrepancies that would invalidate the conclusion that the results of the generic analyses in Reference 5 are applicable to Susquehanna, Units 1 and 2.

Reference 6:

Reference 6 (NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation") provides justification for extending AOTs and STIs for isolation actuation instrumentation that is not common to either RPS or ECCS. Appendix E of Reference 6 identifies Susquehanna, Units 1 and 2, relay type BWR4s, as a participant in the development of the Technical Specification Improvement Analysis discussed in Reference 6. The isolation instrumentation on Table 3.3.2-1 not common to ECCS and RPS include functions: 1.a.2; 1.c; 1.d; 2.a; 2.c; 2.d; 2.e; 2.f; 3.a; 3.c; 3.d; 3.e; 3.f; 3.g; 3h; 3i; 4.a through 4.g; 5.a through 5.i; 6.a through 6.i; 7b; 7.c; and 7d..

Although the SER associated with Reference 6 requires confirmation of the applicability of the generic analyses to a specific plant, no guidance is provided in Reference 6 or the SER for performing this verification. Therefore, PP&L's verification of applicability of Reference 6 to Susquehanna is based on verification that Susquehanna is consistent with the design, conditions and any other assumptions used in the generic analysis contained in Reference 6. The results of this review are presented below:

The generic analysis in Reference 6 assumes that the component failure data used in the RPS and ECCS reliability studies in References 1, 2 and 3 is also applicable to instruments not common to RPS and ECCS. This assumption is applicable to Susquehanna without verification for the following reasons:

a) The generic analysis in Reference 6 assumes that the component failure data used in the RPS study in References 1 and 2 is applicable to instruments common to RPS and Control Rod Block and the failure rates for other instruments are assumed to be in the same range as the common instruments based on their physical similarities.

b) As stated in the SER for Reference 6, "to 'envelope' the effect of the variations in failure rates and number of components within a logic channel, GE increased the sensor and relay failure rates by a factor of 3...;" and,

c) The results of the analysis were determined to be insensitive to instrument reliability and uncertainty in component failure rates does not significantly affect the results of the analysis.

The conclusions in Reference 6 regarding the impact of common cause failure rates, component wear caused by testing, reduced redundancy during testing, and sensitivity to human error rates during testing are generic and are assumed to be applicable to Susquehanna. These same assumptions are applicable to the Control Rod Block systems at Susquehanna.

Reference 6, Section 5.5, Application of Results to Other Plants, discusses the instrumentation configurations including variations in number of sensors and logic designs which are enveloped by the generic analysis. This discussion demonstrates that the generic analysis envelopes all isolation instrumentation not common to RPS or ECCS at Susquehanna, Units 1 and 2.

The PP&L review of Reference 6 and supporting documentation, did not identify any discrepancies that would invalidate the conclusion that the results of the generic analyses in Reference 6 are applicable to Susquehanna, Units 1 and 2.

Reference 7:

Reference 7 (GENE-770-06-1A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications" (SER dated July 21, 1992)) provides justification for extending AOTs and STIs for miscellaneous actuation instrumentation (except RCIC) for the following plant systems:

Feedwater System/Main Turbine Trip;
ATWS/RPT and ARI/RPT;
Refueling Floor Radiation Monitoring;
Control Room Inlet Radiation Monitoring; and,
Control Rod Block Instrumentation not common to RPS.

Although the SER associated with Reference 7 requires confirmation of the applicability of the generic analyses to a specific plant, no guidance is provided in Reference 7 or the SER for performing this verification. Therefore, PP&L's verification of applicability of Reference 7 to Susquehanna is based on verification that the Susquehanna is consistent with the design, conditions and any other assumptions used in the generic analysis contained in Reference 7. The results of this review are presented below:

The generic analysis in Reference 7 assumes that the component failure data used in the RPS and ECCS reliability studies in References 1, 2 and 3 are also applicable to instruments not common to RPS and ECCS. This assumption is applicable to Susquehanna without verification. The generic analysis in Reference 6 assumes that the component failure data used in the RPS study in References 1 and 2 is applicable to instruments common to RPS and ECCS and the failure rates for other instruments are assumed to be in the same range as the common instruments based on their physical similarities.

The approach used in Reference 7 to justify the AOT/STI extensions for the selected instrumentation listed above differed from the approach used in References 1 through 6. Instead of applying criteria consisting of specific percent limits on changes in system unavailability or failure frequency due to AOT and STI extensions, the results of References 1 through 6 were determined to be applicable based on the assumption of the similarity (components, configurations, redundancy, and required actions) between the instruments covered in Reference 7 and those previously analyzed in References 1 through 6. Additionally, the SER associated with Reference 7 justified extending AOTs and STIs based on a qualitative review of the consequences of a failure of each of the instrument functions covered by Reference 7.

Based on the approach used in Reference 7, PP&L confirmed the applicability of Reference 7 to Susquehanna, Units 1 and 2, as follows:

a) PP&L confirmed the similarity (components, configurations, redundancy, and required actions) between the instruments covered in Reference 7 and those previously analyzed in References 1 through 6.

b) PP&L confirmed that the qualitative evaluations of the consequences of a failure of each of the instrument functions covered by Reference 7 were consistent with the discussions in Section IV of Reference 7 and were also applicable to Susquehanna Units 1 and 2.

The PP&L review of Reference 7 and supporting documentation, did not identify any discrepancies that would invalidate the conclusion that the results of the generic analyses in Reference 7 are applicable to Susquehanna, Units 1 and 2.

Reference 8:

Reference 8 (GENE-770-06-2A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications" (SER dated July 21, 1992)) provides justification for extending AOTs and STIs for RCIC actuation instrumentation. The specific instruments and corresponding Technical Specification line items covered by this review of Reference 8 are also used to actuate HPCI and were included in the evaluation of Reference 3. Therefore, the applicability of both References 3 and 8 to Susquehanna must be confirmed to use either as the justification for extending STIs and AOTs for HPCI or RCIC.

The SER associated with Reference 8 requires confirmation of the applicability of the generic analyses to a specific plant by verification that either the appropriate BWR product line generic model or one of the enveloped cases provides an accurate or conservative representation of the plant. The results of PP&L's verification of applicability of Reference 8 to Susquehanna Units 1 and 2 are presented below:

The generic analysis in Reference 8 assumes that the plant is enveloped by either the appropriate BWR product line generic model or one of the enveloped cases provides an accurate or conservative representation of the plant. Susquehanna Units 1 and 2 satisfy this assumption as discussed in the evaluation of the applicability of Reference 3 for the ECCS actuation instrumentation.

The generic analysis in Reference 8 assumes RCIC actuation instrumentation is seismically and environmentally qualified. Susquehanna satisfies this assumption.

The generic analysis in Reference 8 assumes RCIC actuation instrumentation is directly comparable to the HPCI actuation instrumentation. These assumptions include: that transmitters are all located in the reactor building outside the drywell and not subjected to harsh environments; initiation is based on a one out-of-two twice logic for reactor vessel low-low water level; and, redundant instrumentation is physically separated and meets single failure criteria up to the final actuated device. Susquehanna satisfies these assumptions.

The generic analysis in Reference 8 assumes that the component failure data used in the RPS and ECCS reliability studies in References 1, 2 and 3 are also applicable to RCIC actuation instrumentation. Susquehanna meets this criteria because the HPCI and RCIC share common instrumentation.

The PP&L review of Reference 8 and supporting documentation did not identify any discrepancies that would invalidate the conclusion that the results of the generic analyses in Reference 8 are applicable to Susquehanna, Units 1 and 2.

The PP&L review of References 1 through 8 and supporting documentation, did not identify any discrepancies that would invalidate the conclusion that the results of the generic analyses in Reference 1 through 8 are applicable to Susquehanna, Units 1 and 2 for the following: the Reactor Protection System Instrumentation, the Isolation Actuation Instrumentation, the Emergency Core Cooling System Actuation Instrumentation, the ATWS Recirculation Pump Trip System Instrumentation, the End-of-Cycle Recirculation Pump Trip System Initiation, the Reactor Core Isolation Cooling System Actuation Instrumentation, the Control Rod Block Instrumentation, the Radiation Monitoring Instrumentation, the Feedwater/Main Turbine Trip System Actuation Instrumentation. The staff concludes that the licensee has adequately evaluated the differences in its design from that discussed in the generic evaluations included in the LTRs referenced above and finds the evaluations acceptable.

In addition, each of the above LTRs also contains requirements for licensees to demonstrate that the drift characteristics for the applicable instrumentation are bounded by the assumptions used in the LTRs when the functional test interval is extended from monthly to quarterly. The licensee has reviewed current drift information provided by the equipment vendors and the applicable setpoint calculations for SSES instruments in response to these requirements. The SSES setpoint calculation methodology assumed 18-month trip unit calibration intervals and therefore is not affected by the changes proposed in the licensee's amendment request. In addition, sensor calibration intervals for the SSES instrumentation addressed by the LTRs were verified by SSES to be equal to or longer than once per quarter and are therefore unaffected by the proposed changes. The licensee has concluded that the drift characteristics of the involved instrumentation are bounded by the assumptions used in the LTRs when the functional test interval is extended from monthly to quarterly. The NRC staff agrees with this SSES conclusion since it is consistent with the clarification regarding instrument drift allowances provided in a letter dated April 27, 1988 from C. C. Rossi (NRC) to R. F. Janecek (BWR Owners Group).

NRC staff evaluations of specific proposed changes are as follows:

3.3.1: Reactor Protection System Instrumentation

a. LCO 3.3.1, Actions a. and b. (page 3/4 3-1):

The required actions and AOTs for the condition of one or more inoperable RPS instrument channels have been modified by Amendments 115 (Unit 1) and 84 (Unit 2) to extend allowable out of service times. This proposed change will add specific provisions that ensure the extended AOT will not permit an extended loss of scram function. The proposed change will replace the LCO 3.3.1, ACTIONS a. and b. with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.1.1, Reactor Protection System Instrumentation, Conditions A, B, C, and D. The proposed change involves no technical changes to existing Technical Specifications except for the additional requirement: "With one or more RPS Functions with RPS trip capability not maintained, restore RPS trip capability within one hour." Existing exceptions regarding the applicability of Specification 3.0.4 are maintained.

The proposed change has a positive impact on the margin of safety because operation with loss of scram function is prohibited. The proposed wording is consistent with NUREG-1433 and has been determined to improve the clarity and usability of the specification.

This staff finds this change to provide additional safety margin by minimizing the time allowed for loss of scram function. This change is therefore acceptable.

b. Table 4.3.1.1-1 (page 3/4 3-7):

The proposed change will modify Table 4.3.1.1-1, Function 4, Reactor Vessel Water Level - Low, Level 3, to identify the required frequency for CHANNEL FUNCTIONAL TESTING as quarterly by marking the appropriate column with the letter "Q." The frequency is currently shown as "NA." This change will not affect the frequency for the performance of the CHANNEL FUNCTIONAL TEST because the CHANNEL CALIBRATION is performed quarterly and a CHANNEL FUNCTIONAL TEST is performed as part of the quarterly calibration. This change will make the identification of the testing requirements for this function consistent with similar functions in Table 4.3.1.1-1. This is an administrative change with no impact on margin of safety or operator performance.

The staff agrees that this change is administrative in nature, provides clarification and is therefore acceptable.

3.3.2: Isolation Actuation Instrumentation

a. LCO 3.3.2, ACTIONS b. and c. (page 3/4 3-9):

The proposed change will extend AOTs for isolation actuation instrument channels by replacing LCO 3.3.2, ACTIONS b. and c. with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.6.1 (and 3.3.6.2), Primary (and Secondary) Containment Isolation Function, Conditions A, B, and C. Existing exceptions regarding the applicability of Specification 3.0.4 are maintained. The proposed change makes the following technical changes to the Technical Specifications:

i. The AOT before an inoperable channel must be placed in trip is increased from one hour to 12 hours for isolation instruments common to the reactor protection system (Table 3.3.2-1, functions 1.a.1, 1.b, 1.e, 2.b, 3.b, 7.a, and 7.e) and from one hour to 24 hours for instruments not common to RPS (functions other than 1.a.1, 1.b, 1.e, 2.b, 3.b, 7.a, and 7.e). Justification for extending the AOTs, including the positive impact on the margin of safety and operator performance, is provided in References 5 and 6.

ii. The proposed change adds a requirement that specifically will not permit the extended AOTs to result in an extended loss of isolation function.

The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in References 5 and 6 and because operation with extended loss of any isolation function is prohibited. The proposed wording is consistent with NUREG-1433 and has been determined to improve the clarity and usability of the specification.

The staff finds that these changes are consistent with NEDC-30851P and NEDC-31677P and are therefore acceptable.

b. LCO 3.3.2, Notes for ACTIONS b. and c. (page 3/4 3-9) and Table 3.3.2-1, Notes (page 3/4 3-16):

The proposed change adds a Note to the proposed actions for LCO 3.3.2 that specifies when a channel is placed in an inoperable status solely for performance of required surveillances, initiation of actions may be delayed for up to 6 hours (instead of 2 hours currently in Table 3.3.2-1, Note (b)) provided the associated trip function maintains isolation capability. Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in References 5 and 6.

The proposed change deletes two existing Notes associated with the existing required actions for LCO 3.3.2 and deletes Note (b) to Table 3.3.2-1. Justification for deletion of these three notes is as follows:

i. The first note associated with LCO 3.3.2 and Note (b) to Table 3.3.2-1 both provide extended AOTs for special situations including: HPCI and RCIC isolation instrument channels when trip capability is maintained by the redundant function; inoperability caused by surveillance testing; and/or, situations where placing a channel in trip will result in an actuation. These notes are deleted because AOTs in the proposed required actions for LCO 3.3.2 are longer than those permitted by the existing notes. Therefore, removal of these notes is an administrative change with no effect on the margin of safety or operator performance.

ii. The second note associated with LCO 3.3.2 requires that the trip system with the most inoperable channels be placed in trip when there are multiple inoperable channels. This note is deleted because, in conjunction with the proposed changes to required actions, it would have no effect on operator action. This change is consistent with NUREG-1433. Deletion of this note has no effect on margin of safety of operator performance.

The staff finds that these changes are consistent with NEDC-30851P and NEDC-31677P and are therefore acceptable.

c. <u>Table 3.3.2.1-1 (page 3/4 3-25) and Table 4.3.2.1-1 (page 3/4 3-26)</u>:

The proposed change changes Applicability requirements and required actions for the Isolation Trip Function 7.a, Reactor Vessel Water Level - Low, Level 3. This function isolates the RHR System Shutdown Cooling System and is intended to isolate a potential leakage path in the event of a loss of reactor coolant during operation of the Shutdown Cooling (SDC) System. The Reactor Vessel Water Level - Low, Level 3 Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected to two two-out-of-two trip systems. Each of the two trip systems is connected to one of the two valves on each shutdown cooling penetration.

Currently, the Reactor Vessel Water Level - Low, Level 3 function is required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2 and 3. In OPERATIONAL CONDITIONS 1, 2 and 3 another isolation (i.e., Reactor Steam Dome Pressure - High; setpoint £ 98 psig) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path. The proposed change will require that the Reactor Vessel Water Level - Low, Level 3 Function be OPERABLE in OPERATIONAL CONDITIONS 3, 4, and 5 when it is possible that the SDC isolation valves are open and the safety function provided by this isolation function is needed to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System. The Reactor Vessel Water Level - Low, Level 3. function will no longer be required to be OPERABLE when RHR SDC isolation is already maintained by the Reactor Steam Dome Pressure - High Function. This change is more conservative than existing SSES Technical Specifications because the safety function provided by the Reactor Vessel Water Level - Low, Level 3 Function is extended to include OPERATIONAL CONDITIONS 4 and 5. In conjunction with this change, Table 4.3.2.1-1 will be changed to identify the "OPERATIONAL CONDITIONS for which Surveillance Required" consistent with the change in Applicability requirements.

The staff also finds that the combination of the operability requirements for the isolation for the reactor steam dome pressure and the reactor vessel water level isolation as proposed will provide additional requirements for the maintenance of the isolation capability for OPERATIONAL CONDITIONS 4 and 5 and will result in a more conservative TS. Therefore, the change is acceptable.

A new footnote to Table 3.3.2-1 will be added that "Only one trip system required in OPERATIONAL CONDITIONS 4 and 5 when RHR Shutdown Cooling System integrity maintained." System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. The requirement to have two OPERABLE Channels in one trip system ensures that the isolation function will be available in OPERATIONAL CONDITIONS 4 and 5 although redundancy is reduced. This change is more conservative than existing SSES Technical Specifications because currently there are no OPERABILITY requirements for this function in OPERATIONAL CONDITIONS 4 and 5. The staff notes that this change is consistent with NUREG-1433.

The staff agrees with the licensee that this change will be more conservative than the current TS and therefore, the change is acceptable.

Required actions for an inoperable Reactor Vessel Water Level - Low, Level 3 instrument channel will be revised by adding ACTION 27 to Table 3.3.2-1. If the number of OPERABLE channels of Reactor Vessel Water Level - Low, Level 3 Function is less than required and cannot be restored within the AOTs specified in proposed Required ACTIONS b or c of Technical Specification 3.3.2, Table 3.3.2-1 will require entering ACTION 27. ACTION 27 will require that plant personnel initiate action to restore channel(s) to OPERABLE status; or, initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System. To ensure proper interpretation of proposed ACTION 27, the Bases for Technical Specification 3.3.2 will be revised to provide the following guidance:

If an inoperable channel(s) is not restored to OPERABLE status or placed in trip within the allowed completion time, the associated penetration flow path should be closed. However, if the shutdown cooling function is needed to provide core cooling, ACTION 27 allows the penetration flow path to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling

System (i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated). ACTION 27 must continue until the channel(s) is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.

The staff notes that these changes are consistent with Section 3.3.6.1 of NUREG-1433.

The staff finds that this change to the Required Actions for the Isolation Trip Function 7.a, Reactor Vessel Water Level - Low, Level 3 and the bases revision are more conservative than the current TS and are therefore acceptable.

d. Table 4.3.2.1-1 (pages 3/4 3-23 through 3/4 3-26):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for all isolation actuation instrumentation, except manual initiation, from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in References 5 and 6.

The staff agrees that this change is consistent with NEDC-30851P and NEDC-31677P and therefore is acceptable.

e. Table 3.3.2-1 (pages 3/4 3-11 through 3/4 3-15);
Table 3.3.2-2 (pages 3/4 3-18 through 3/4 3-20); and,
Table 4.3.1-1 (pages 3/4 3-23 through 3/4 3-26):

The proposed change will delete notes to Table 3.3.2-1, Table 3.3.2-2, and Table 4.3.1-1 that were added by Amendment 94 (Unit 1) and 61 (Unit 2) to permit modifications to temperature instruments associated with leak detection during the period between October 19, 1989 and January 19, 1990. The time period specified by these notes has expired and the associated modifications are complete. This is an administrative change with no effect on margin of safety or operator performance.

The staff agrees that these changes are administrative in nature and are therefore acceptable.

3.3.3: Emergency Core Cooling System Actuation Instrumentation

a. Table 3.3.3-1 (pages 3/4 3-28 and 29) and Action Statements (page 3/4 3-30):

The proposed change will modify required actions associated with inoperable emergency core cooling system actuation instrumentation that will:

1) extend AOTs before an inoperable channel must be placed in trip based on the analysis reported in Reference 3;
2) extend AOTs before an inoperable channel must be placed in trip to be

consistent with NUREG-1433, Section 3.3.5.1 and based on the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design; and,

3) ensure that the extended AOTs do not permit an extended loss of actuation capability.

Each of the proposed changes is consistent with NUREG-1433, Section 3.3.5.1, Conditions A, B, C, D, E, F, G and H, as appropriate. In some cases, the action statement assigned to specific functions are changed, consistent with NUREG-1433, to ensure that appropriate required actions are applied to each function.

The proposed change makes the following specific changes to the Technical Specifications:

The proposed change will replace the LCO 3.3.3, Table 3.3.3-1, ACTION 30 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.1, Conditions A, B and H (except HPCI). Existing ACTION 30 requires that: the inoperable channel be placed in trip or the associated ECCS be declared inoperable within one hour whenever the number of OPERABLE channels in one trip system is less than the required minimum number per trip system. The proposed change results in the following technical changes:

i. The AOT before an inoperable channel must be placed in trip is increased from one hour to 24 hours. Justification for extending the AOTs, including the positive impact on the margin of safety and operator performance, is provided in References 5 and 6.

ii. The proposed change adds a requirement that specifically prohibits the extended AOTs from resulting in an extended loss of actuation function.

In conjunction with this change, the required action designated in Table 3.3.3-1 for an inoperable channel is changed from ACTION 32 to proposed ACTION 30 for the following functions:

Function 1.c: Reactor Vessel Steam Dome Pressure Low (Permissive for System Initiation in OPERATIONAL CONDITIONS 4 and 5);

Function 2.c. Reactor Vessel Steam Dome Pressure Low (Permissive for System Initiation and Recirculation Discharge Valve Closure in OPERATIONAL CONDITIONS 4 and 5).

Other than extending the AOT from within one hour to 24 hours and preventing extended loss of function, assigning proposed ACTION 30 to these functions results in no technical change to the required actions and the assignment of proposed ACTION 30 to these functions is consistent with NUREG-1433, Section 3.3.5.1.

The proposed changes to ACTION 30, in conjunction with other proposed changes that extend AOTs and STIs, have a positive impact on the margin of safety and operator performance for reasons described in Reference 3 and because operation with extended loss of any isolation function is prohibited. The proposed wording is consistent with NUREG-1433 and has been determined to improve the clarity and usability of the specification.

The staff finds the proposed changes to be acceptable because of consistency with NEDC-30851P, NEDC-31667P, and NEDC-30936P.

ACTION 31:

The proposed change will replace the LCO 3.3.3, Table 3.3.3-1, ACTION 31 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.1, Conditions A, C and H. Existing ACTION 31 requires that the associated ECCS be declared inoperable within whenever the number of operable channels is less than the required minimum number per trip system. The proposed change results in the following technical changes:

i. The AOT before an inoperable channel must be restored to OPERABLE status or the ECCS declared inoperable is increased from immediately to within 24 hours. Justification for extending the AOTs, including the positive impact on the margin of safety and operator performance, is provided in References 5 and 6.

ii. The proposed change adds a requirement that specifically prohibits the extended AOTs from resulting in an extended loss of actuation capability.

The staff finds these changes to be consistent with NEDC-30851P and NEDC-31667P. The added requirement addressing the extended loss of actuation capability provides additional safety margin. Therefore, the staff finds these TS changes acceptable.

In conjunction with this change, the required ACTION designated in Table 3.3.3-1 for an inoperable channel is changed from ACTION 33 to proposed ACTION 31 for manual initiation of Core Spray (Table 3.3.3-1, Function 1.d), LPCI Mode of RHR (Table 3.3.3-1, Function 2.d) and High Pressure Coolant Injection (Table 3.3.3-1, Function 3.f). Existing ACTION 33 requires that an inoperable channel be restored to OPERABLE within 8 hours or the associated ECCS declared inoperable. Other than extending the AOT from within 8 hours to 24 hours and preventing extended loss of function, assigning proposed ACTION 31 to these functions results in no technical change to the required ACTIONS and the assignment of proposed ACTION 31 to these functions is consistent with NUREG-1433, Section 3.3.5.1.

This change is also consistent with NEDC-30851P and NEDC-31667P and is acceptable.

Proposed ACTION 31 will be modified by a footnote stating that ACTION 31 b is not applicable to Function 3.e, Reactor Vessel Water Level - High, Level 8. ACTION 31 b requires that HPCI be declared inoperable within one hour from discovery of loss of initiation capability by this trip function. This requirement is not applicable to Function 3.e, HPCI Reactor Vessel Water Level High, Level 8, because this function is for equipment protection and is not assumed in the SSES safety analysis. ACTION 31 a will require that HPCI be declared inoperable if trip capability for this function is not restored within 24 hours. This change is consistent with NUREG-1433, Section 3.3.5.1, Condition C.

The staff accepts this proposed change because the trip is not associated with a safety function.

As noted above, the staff has found the proposed changes to ACTION 31, in conjunction with other proposed changes that extend AOTs and STIs, have a positive impact on the margin of safety and operator performance for reasons described in the referenced LTRs and because operation with extended loss of any isolation function is prohibited. The changes are found to be acceptable. The proposed wording is also consistent with NUREG-1433.

<u>ACTION 32:</u>

The proposed change will replace the LCO 3.3.3, Table 3.3.3-1, ACTION 32 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.1, Conditions A, F and H. Existing ACTION 32 requires that an inoperable channel be placed in trip within one hour whenever the number of operable channels is less than the required minimum number per trip system. The proposed change results in the following technical changes:

i. The proposed change adds a requirement that specifically prohibits extended AOTs from resulting in an extended loss of ADS actuation capability. The change requires that ADS be declared inoperable within one hour from discovery of loss of ADS initiation capability.

ii. The AOT for an inoperable ADS initiation channel that does not result in the loss of ADS initiation capability is extended from within one hour to: within 4 days with HPCI or RCIC inoperable; and, within 8 days from discovery of inoperable channel if both HPCI and RCIC are OPERABLE. The extension of the AOTs is justified by: the redundancy and independence of sensors available to provide ADS initiation signals; the redundancy of the ECCS design; the requirement to declare ADS inoperable within one hour of the determination of loss of ADS initiation capability; and, the analysis in Reference 3. Proposed ACTION 32 is worded so that the AOT limits the total time for an inoperable, untripped channel to less than 8 days even if the status of HPCI or RCIC changes following the discovery of the inoperable ADS channel. This change adds the 24 hour AOT extension justified by Reference 3 to the AOTs in the BWR Standard Technical Specifications. This combination of conditional AOTs and the length of the AOTs for

ADS initiation is consistent with NUREG-1433, Section 3.3.5.1, Condition F.

The staff finds that because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, the proposed extension to the AOT is acceptable. In addition, the added requirement to declare the ADS inoperable one hour after initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

In conjunction with this change, the required action designated in Table 3.3.3-1 for an inoperable channel is changed from ACTION 30 or ACTION 31 to proposed ACTION 32 for the following functions:

Function 4.a:	ADS Reactor Vessel Water Level Low Low Low, Level 1;
Function 4.b:	ADS Drywell High Pressure; and
Function 4.f:	ADS Reactor Vessel Water Level Low, Level 1 (Permissive).

Existing ACTION 30, which was the required action for Functions 4.a and 4.b, and existing ACTION 31, which was the required action for Function 4.f, both required that an inoperable channel be placed in trip within one hour. The proposed change maintains this limit and adds an extended AOT for an inoperable channel that does not result in loss of initiation capability for the reasons justified above. This change is consistent with NUREG 1433, Section 3.3.5.1, Table 3.3.5.1-1.

The staff finds this change to be acceptable for the same reasons discussed above for item ii.

The proposed changes to ACTION 32 to extend AOTs for inoperable ADS initiation channels that do not result in loss of ADS initiation capability are consistent with BWR Standard Technical Specifications for BWR4s with ADS initiation logic identical to SSES, Units 1 and 2. These AOT extensions, in conjunction with other proposed changes that extend AOTs and STIs, have a positive impact on the margin of safety and operator performance for reasons described in Reference 3 and because operation with extended loss of any isolation function is prohibited. The proposed wording is consistent with NUREG-1433 and has been determined to improve the clarity and usability of the specification.

As noted above, the staff approves the proposed changes to Action 32 which are consistent with the ITS.

<u>ACTION 33:</u>

The proposed change will replace the LCO 3.3.3, Table 3.3.3-1, ACTION 33 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.1, Conditions A, G and H. Existing ACTION 33 requires that an inoperable channel be restored to OPERABLE within 8 hours or the associated ECCS (ADS) be declared inoperable.

The difference between proposed ACTION 32 and proposed ACTION 33 is that ACTION 32 allows inoperable channels to be placed in trip within the AOT while proposed ACTION 33 requires inoperable channels to be restored to OPERABLE status within the AOT. This difference recognizes that placing an inoperable channel in trip has two consequences: it provides greater assurance that an actuation will occur when required; and, it increases the potential for an inadvertent actuation. Therefore, proposed ACTION 33 does not permit continued operation with an inoperable channel in trip for an instrument channel that provides an interlock or permissive because both the failure to actuate and an inadvertent actuation are undesirable. Other than this difference, which is already recognized in existing ACTION 33, the proposed changes and the justification for proposed ACTION 33 are identical to the proposed changes and justification for proposed ACTION 32 above.

In conjunction with this change, the required action designated in Table 3.3.3-1 for an inoperable channel is changed from existing ACTION 31 to proposed ACTION 32 for the following functions:

Function 4.c: ADS Timer;
Function 4.d: Core Spray Pump Discharge Pressure - High (Permissive);
Function 4.e: RHR LPCI Mode Pump Discharge Pressure - High (Permissive); and,
Function 4.g: ADS Drywell pressure Bypass Timer.

Existing ACTION 31, which was the required action for the trip functions listed above, required that the associated ECCS (ADS) be declared inoperable within one hour following discovery of an inoperable channel. The proposed change, other than extending the AOT from one hour to 24 hours which is justified by Reference 3, is not different because it requires that the channel be restored to OPERABLE or ADS declared inoperable within the AOT. This change is consistent with NUREG-1433, Section 3.3.5.1, Table 3.3.5.1-1.

Justification for the proposed changes to ACTION 33 and the impact on margin of safety and operator performance are the same as for proposed ACTION 32 and are discussed above.

For the reasons discussed above for Action 32, the staff finds the changes to Action 33 to be acceptable.

ACTION 34:

The proposed change will replace the LCO 3.3.3, Table 3.3.3-1, ACTION 34 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.1, Conditions A, D and H. Existing ACTION 34 requires whenever the number of OPERABLE channels is less than the required minimum number per trip system that one inoperable channel be placed in trip within one hour (which results in an actuation, i.e., HPCI pump suction transfer to the suppression pool) or that HPCI be declared inoperable. The proposed change makes the following technical changes to the required action:

i. The AOT before an inoperable channel must be placed in trip or the HPCI declared inoperable is increased from within one hour to within 24 hours. Justification for extending the AOTs, including the positive impact on the margin of safety and operator performance, is provided in Reference 3.

ii. The proposed change adds a requirement that specifically prohibits the extended AOTs from resulting in an extended loss of actuation capability.

iii. The proposed change provides the option of transferring pump suction to the suppression pool. Aligning the pump suction from the Condensate Storage Tank to the suppression pool is an acceptable alternative to placing the channel in trip or declaring HPCI inoperable because it completes the intended function of inoperable instrument.

The proposed changes to ACTION 34, in conjunction with other proposed changes that extend AOTs and STIs, have a positive impact on the margin of safety and operator performance for reasons described in Reference 3 and because operation with extended loss of any isolation function is prohibited. The proposed wording is consistent with NUREG-1433.

The staff finds the proposed TS changes to be consistent with NEDC-30936P and also provides additional assurance that extended loss of actuation capability will be minimized. Therefore the changes are found to be acceptable.

b. Table 3.3.3-1, Notes (page 3/4 3-29a):
The proposed change modifies Notes (a) and (f) for Table 3.3.3-1 to extend from 2 hours to 6 hours the amount of time that initiation of required actions may be delayed when a channel is placed in an inoperable status solely for performance of required surveillances. Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 3. In conjunction with this change, the following technical changes were made to proposed Note (a) and proposed Note (f):

i. The wording of Note (a) was changed to provide an unambiguous requirement that the 6 hour AOT for surveillance testing was applicable only if the associated trip function maintains trip capability. The change in wording to unambiguously prohibit loss of function as a condition of the AOT permitted Table 3.3.3-1 to be modified to make Note (a) (instead of Note (f)) applicable to the ADS trip functions. This change is consistent with NUREG-1433.

ii. The proposed change described above results in Note (f) no longer being applicable to the ADS functions in Table 3.3.3-1. Note (f) was modified to apply the 6 hour AOT for surveillance testing to those functions that do not depend on the condition that trip capability be maintained. Consistent with NUREG-1433, Table 3.3.3-1 was modified to make proposed Note (f) applicable to Function 3.e, HPCI Reactor Vessel Water Level High, Level 8, because this function is for equipment protection and is not assumed in the SSES safety analysis.

The staff finds that the extension in the AOT while in a surveillance mode from 2 to 6 hours is acceptable based on its consistency with NEDC-30936P. Further the change i above for clarification results in a more effective TS and is acceptable, and the application of the 6 hour AOT for HPCI Reactor Vessel Water Level High, Level 8 acceptable based on guidance in the ITS and because the relaxation is justified since the function is not assumed in the plants' safety analysis.

c. <u>Table 4.3.3.1-1 (page 3/4 3-34 and 3/4 3-34)</u>

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for the following ECCS instrumentation:

<u>Core Spray System</u>

1.a Reactor Vessel Water Level - Low Low Low, Level 1
1.b Drywell Pressure - High
1.c Reactor Pressure Stream Dome Pressure - Low (Permissive)

<u>Low Pressure Coolant Injection Mode of RHR System</u>

2.a Reactor Vessel Water Level - Low Low Low, Level 1
2.b Drywell Pressure - High
2.c Reactor Pressure Stream Dome Pressure - Low (Permissive)
2.c.1) System Initiation
2.c.2) Recirculation Discharge Valve Closure

<u>High Pressure Coolant Injection System</u>

3.a Reactor Vessel Water Level - Low Low, Level 2
3.b Drywell Pressure - High
3.c Condensate Storage Tank Level - Low
3.d Suppression Pool Water Level - High
3.e Reactor Vessel Water Level - High, Level 8

<u>Automatic Depressurization System</u>

4.a Reactor Vessel Water Level - Low Low Low, Level 1
4.b Drywell Pressure - High
4.c ADS Timer
4.d Core Spray Pump Discharge Pressure - High (Permissive)

4.e RHR LPCI Mode Pump Discharge Pressure - High (Permissive)
4.f Reactor Vessel Water Level - Low, Level 3 (Permissive)
4.g ADS Drywell Pressure Bypass Timer

The proposed change modifies Table 4.3.3.1-1 to change the required frequency for the instruments listed above from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in Reference 3. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 3.

The staff finds this increase in the maximum interval between functional tests for the identified ECCS instrumentation to be consistent with NEDC-30936P and therefore these TS changes are acceptable.

3.3.4.1: ATWS Recirculation Pump Trip System Instrumentation

a. LCO 3.3.4.1, ACTIONS b, c, d, and e (page 3/4 3-36):

The proposed change will modify required actions and extend AOTs for ATWS Recirculation Pump Trip Instrumentation by replacing LCO 3.3.4.1, ACTIONS b, c, d, and e with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.4.2, ATWS-RPT Instrumentation, Conditions A, B, C and D. The proposed change makes the following specific changes to the Technical Specifications:

i. The proposed change adds a requirement that specifically prohibits extended AOTs from resulting in an extended loss of ATWS-RPT actuation capability. The change requires that ATWS-RPT be declared inoperable within one hour from discovery of loss of initiation capability for both the reactor steam dome pressure and reactor vessel water level trip function. Additionally, the proposed change prohibits satisfying required actions for an inoperable channel by placing the channel in trip if the inoperability is the result of an inoperable breaker.

ii. The AOT for one or more inoperable ATWS-RPT initiation channels that do not result in the loss of initiation capability is extended from within one hour to within 14 days. This change is consistent with BWR Standard Technical Specifications, NUREG-1433, and is justified based on the ATWS-RPT design and function. ATWS-RPT consists of two independent trip systems, with two channels of reactor steam dome pressure and two channels of reactor vessel water level in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each of these functions. Either two Reactor Water Level or two Reactor Pressure signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps. The 14 day AOT for an inoperable channel is justified because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse functions, and the low probability of an event requiring the initiation of ATWS-RPT.

The staff notes that this change includes the 24 hour AOT extension justified by GENE-770-06-1A and the AOTs in the BWR Standard Technical Specifications. Each of the proposed changes is consistent with NUREG-1433, Section 3.3.4.2, Conditions A, B, C and D. The proposed change, extending the AOT for an inoperable ATWS-RPT channel that does not result in a loss of function, does not have a significant impact on margin of safety because of diversity of sensors, the low probability of multiple inoperabilities and the low probability of an event requiring the initiation of ATWS-RPT. Based on the above, the staff has found the proposed TS changes to be acceptable.

b. Table 3.3.4.1-1 (page 3/4 3-37):

The proposed change modifies the Note to Table 3.3.4.1-1 that specifies when a channel is placed in an inoperable status solely for performance of required surveillances, initiation of actions may be delayed for up to 6 hours (instead of 2 hours currently in Table 3.3.4.1-1) provided the associated trip function maintains ATWS-RPT capability.

The staff finds that this TS change is consistent with GENE-770-06-1A and is therefore acceptable.

c. Table 4.3.4.1-1 (page 3/4 3-39):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for ATWS-RPT trip functions (reactor vessel level and reactor vessel pressure). This change modifies Table 4.3.4.1-1 to change the CHANNEL FUNCTION TEST frequency from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in Reference 7.

The staff finds that the proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance as discussed and justified in GENE-770-06-1A. Therefore the changes are approved.

3.3.4.2: End-of-Cycle Recirculation Pump Trip Initiation

a. LCO 3.3.4.2, ACTIONS b., c., d., and e. (page 3/4 3-40):

The proposed change will modify required actions and extend AOTs for End-of-Cycle Recirculation Pump Trip (EOC-RPT) by replacing LCO 3.3.4.2, ACTIONS b, c, d, and e with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.4.1, EOC-RPT Instrumentation, Conditions

B, C and D. The proposed change makes the following specific changes to the Technical Specifications:

i. The AOT for placing one or more inoperable EOC-RPT channels in trip is increased from within one hour to within 72 hours. This change adds the 24 hour AOT extension justified by Reference 7 to the AOTs allowed for in BWR Standard Technical Specifications for an inoperable EOC-RPT function. This change is justified because with one or more channels inoperable, but with EOC-RPT trip capability maintained, EOC-RPT is capable of performing the intended function. However, the reliability and redundancy of the EOC-RPT instrumentation is reduced such that a single failure in the remaining trip system could result in the inability of the EOC-RPT to perform the intended function. Therefore, only a limited time is allowed to restore compliance with the LCO. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse functions, and the low probability of an event requiring the initiation of an EOC-RPT, 72 hours is provided to restore the inoperable channels or apply the EOC-RPT inoperable MCPR limit. Alternately, the inoperable channels may be placed in trip since this would restore capability to accommodate a single failure. Loss of function is prohibited by specifying that the LCO cannot be satisfied by placing a channel in trip if the inoperable channel is the result of an inoperable breaker.

ii. The proposed change prohibits the extended AOT from resulting in a loss of EOC-RPT function by specifically establishing an AOT of 2 hours following the loss of EOC-RPT trip capability in one or both trip systems.

The staff finds that the proposed change is consistent with NUREG-1433, Section 3.3.4.1, Conditions A, B, C and D. The proposed change, extending the AOT for an inoperable EOC-RPT channel that does not result in a loss of function is consistent with GENE-770-06-1A, and does not have a significant impact on margin of safety because of diversity of sensors, the low probability of multiple inoperabilities and the low probability of an event requiring the initiation of EOC-RPT actuation. Based on the above, the staff finds the proposed changes to be acceptable.

b. <u>Table 3.3.4.2-1 (page 3/4 3-42)</u>:

The proposed change modifies the Note to Table 3.3.4.2-1 that specifies when a channel is placed in an inoperable status solely for performance of required surveillances, initiation of actions may be delayed for up to 6 hours (instead of 2 hours currently in Table 3.3.4.2-1) provided the other trip system is OPERABLE. Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 7.

The staff notes that this change is consistent with GENE-770-06-1A and therefore it is acceptable.

c. Table 4.3.4.2-1 (page 3/4 3-45):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for EOC-RPT trip functions (Turbine Stop Valve-Closure and Turbine Control Valve-Fast Closure). This change modifies Table 4.3.4.2-1 to change the CHANNEL FUNCTION TEST frequency from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in Reference 7. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 7.

The staff notes that this change is consistent with GENE-770-06-1A and therefore it is acceptable.

3.3.5: Reactor Core Isolation Cooling System Actuation Instrumentation

a. Table 3.3.5-1 (page 3/4 3-47):

The proposed change modifies Note (a) for Table 3.3.5-1 to extend from 2 hours to 6 hours the amount of time that initiation of required actions may be delayed when a channel is placed in an inoperable status solely for performance of required surveillances. Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 7.

The staff notes that this change is consistent with GENE-770-06-1A and therefore it is acceptable.

b. Table 3.3.5-1, Action Statements (page 3/4 3-48):

ACTION 50:

The proposed change will replace LCO 3.3.5, Table 3.3.5-1, ACTION 50 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.2, Conditions B and E. Existing ACTION 50 requires that: the inoperable channel be placed in trip or RCIC declared inoperable within one hour whenever the number of operable channels in one trip system is less than the required minimum number per trip system. The proposed change results in the following technical changes:

i. The AOT before an inoperable channel must be placed in trip is increased from one hour to 24 hours. Justification for extending the AOTs, including the positive impact on the margin of safety and operator performance, is provided in Reference 8.

ii. The proposed change adds a requirement that specifically prohibits the extended AOTs from resulting in an extended loss of actuation function.

The staff finds that the proposed changes are consistent with GENE-770-06-2A and are therefore acceptable.

ACTION 51:

The proposed change will replace LCO 3.3.5, Table 3.3.5-1, ACTION 51 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.2, Conditions C and E. Existing ACTION 51 requires that RCIC be declared inoperable within one hour whenever the number of OPERABLE channels in one trip system is less than the required minimum number per trip system. The proposed change results in the following technical change:

i. The AOT before an inoperable channel must be restored to OPERABLE or RCIC declared inoperable is increased from one hour to 24 hours. Justification for extending the AOTs, including the positive impact on the margin of safety and operator performance, is provided in Reference 8.

The staff finds that the proposed changes are consistent with GENE-770-06-2A and are therefore acceptable.

ACTION 52:

The proposed change will replace LCO 3.3.5, Table 3.3.5-1, ACTION 52 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.2, Conditions D and E. Existing ACTION 52 requires whenever the number of OPERABLE channels is less than the required minimum number per trip system that one inoperable channel be placed in trip within one hour (which results in an actuation, i.e., RCIC pump suction transfer to the suppression pool) or that RCIC be declared inoperable. The proposed change makes the following technical changes to the required action:

i. The AOT before an inoperable channel must be placed in trip or the RCIC declared inoperable is increased from within one hour to within 24 hours. Justification for extending the AOT, including the positive impact on the margin of safety and operator performance, is provided in Reference 8.

ii. The proposed change adds a requirement that specifically prohibits the extended AOTs from resulting in an extended loss of actuation capability.

iii. The proposed change provides the option of transferring pump suction to the suppression pool. Aligning the pump suction from the Condensate Storage Tank to the suppression pool is an acceptable alternative to placing the channel in trip or declaring RCIC inoperable because it completes the intended function of inoperable instrument.

The staff finds that the proposed changes are consistent with GENE-770-06-2A and are therefore acceptable.

ACTION 53:

The proposed change will replace LCO 3.3.5, Table 3.3.5-1, ACTION 53 with required actions and AOTs that are consistent with the NUREG-1433, Section 3.3.5.2, Conditions C and E. Existing ACTION 53 requires whenever the number of OPERABLE channels is less than the required minimum number per trip system that the inoperable channel be restored to OPERABLE within 8 hours or that RCIC be declared inoperable. The proposed change makes the following technical changes to the required action:

i. The AOT before an inoperable channel must be restored to OPERABLE or the RCIC declared inoperable is increased from within 8 hours to within 24 hours. Justification for extending the AOT, including the positive impact on the margin of safety and operator performance, is provided in Reference 8. There is no requirement to prevent the extended AOT from causing an extended loss of actuation capability because this function is for equipment protection and is not assumed in the SSES safety analysis.

The staff finds that the proposed changes are consistent with GENE-770-06-2A and are therefore acceptable.

c. Table 4.3.5-1 (page 3/4 3-50):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for the following RCIC instrumentation:

Reactor Vessel Water Level - Low Low, Level 2
Reactor Vessel Water Level - High, Level 8
Condensate Storage Tank Water Level - Low

The proposed change modifies Table 4.3.5.1-1 to change the required frequency for the instruments listed above from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in Reference 8. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 8.

The staff finds that the proposed changes are consistent with GENE-770-06-2A and are therefore acceptable.

D.7 3.3.6: Control Rod Block Instrumentation

a. LCO 4.3.6, Footnote (page 3/4 3-51) (Unit 1 only):

The proposed change eliminates the footnote for LCO 4.3.6, ACTION a that states: "For the Intermediate Range Monitors the provisions of Specification 3.0.4 are not applicable for the purposes of entering OPERATIONAL CONDITION 5 from OPERATIONAL CONDITION 4 on September 14, 1987." This is an administrative change with no effect on margin of safety or operator performance.

The staff agrees with the licensee that the proposed change is administrative in nature and is therefore acceptable.

b. SR 4.3.6, Note (page 3/4 3-51):

The proposed change adds a Note to the proposed actions for Surveillance Requirement 4.3.6 that specifies when a channel is placed in an inoperable status solely for performance of required surveillances, initiation of actions may be delayed for up to 6 hours provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 4.

The staff finds that this change is consistent with NEDC-30851P, Supplement 1, and that the increase in the AOT is acceptable.

c. Table 3.3.6-1, ACTION Statements (page 3/4 3-53):

The proposed change modifies ACTION 62 to extend the AOT before an inoperable channel must be placed in trip from one hour to within 12 hours. ACTION 62 is applicable to the rod blocks associated with the scram discharge instrument volume high function and the Reactor Coolant System Recirculation Flow function. Justification for extending this AOT, including the impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 4.

The staff finds that this change is consistent with NEDC-30851P, Supplement 1, and that the increase in the AOT is acceptable.

d. Table 4.3.6-1, (page 3/4 3-55):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for the following Control Rod Block Instrumentation:

Rod Block Monitor

1.a Upscale
1.b Inoperative
1.c Downscale

The proposed change modifies Table 4.3.6-1 to change the required frequency for the instruments listed above from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Additionally, the requirement to perform CHANNEL FUNCTIONAL TEST "within 24 hours prior to startup, if not performed within the previous 7 days" is being eliminated. This change is consistent with NUREG-1433. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in Reference 4. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 4.

The staff finds that the proposed changes are consistent with NEDC-30851P, Supplement 1, and the increase in the frequency between tests, and the elimination of the functional test recurrent post start-up are acceptable.

e. Table 4.3.6-1, (page 3/4 3-55):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for the following Control Rod Block Instrumentation:

APRM

2.a Flow Biased Neutron Flux - Upscale
2.b Inoperative
2.c Downscale
2.d Neutron Flux - Upscale, Startup

Scram Discharge Volume

5.a Water Level - High

Reactor Coolant System Recirculation Flow

6.a Upscale
6.b Inoperative
6.c Comparator

The proposed change modifies Table 4.3.6-1 to change the required frequency for CHANNEL FUNCTIONAL TESTS for the instruments listed above from the current requirement for either weekly (W) or monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in Reference 4. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 4.

The staff finds that the proposed changes are consistent with NEDC-30851P, Supplement 1, and the increase in the frequency between tests are acceptable.

3.3.7.1: Radiation Monitoring Instrumentation

a. Table 3.3.7.1-1 (page 3/4 3-58):

The proposed change adds new Note (c) to Table 3.3.7.1-1 for ACTION 70 which is associated with the Main Control Room Outside Air Radiation Monitor. This note specifies when a channel is placed in an inoperable status solely for performance of required surveillances, initiation of actions may be delayed for up to 6 hours provided control room emergency ventilation capability is maintained. Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 7.

The staff finds that the proposed changes to increase the AOT for radiation monitoring instrumentation are consistent with GENE-770-06-1A and are therefore acceptable.

b. Table 4.3.7.1-1 (page 3/4 3-60):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for the Main Control Room Outside Air Radiation Monitor from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in References 5 and 6. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 4.

The staff finds that the proposed increase to quarterly for the tests of the MCR Outside Air Radiation Monitor to be consistent with NEDC-30851P Supplements 1 and 2, and NEDC-31677P, and are therefore acceptable.

3.3.9: Feedwater/Main Turbine Trip System Actuation Instrumentation

a. LCO 3.3.9, Notes (page 3/4 3-95):

The proposed change adds a new Note to LCO 3.3.9, ACTIONS b and c that specifies that when a channel is placed in an inoperable status solely for performance of required Surveillances, actions may be delayed for up to 6 hours provided feedwater/main turbine trip capability is maintained.

Justification for extending this AOT, including the positive impact on the margin of safety when considered in conjunction with other proposed changes, is provided in Reference 7.

The staff finds that this increase in AOT for the Feedwater/Main Turbine Trip System Actuation Instrumentation is consistent with GENE-770-06-1A and is acceptable.

b. Table 4.3.9.1-1 (page 3/4 3-98):

The proposed change increases the maximum interval between required performances of CHANNEL FUNCTIONAL TESTS for the Reactor Vessel Water level - High function of the Feedwater/Main Turbine Trip System from the current requirement for monthly (M) performance to a proposed requirement of quarterly (Q) performance. Justification for extending the CHANNEL FUNCTIONAL TEST frequency, including the positive impact on the margin of safety and operator performance, is provided in References 5 and 6. The proposed change, in conjunction with other proposed changes that extend AOTs and STIs, has a positive impact on the margin of safety and operator performance for reasons described in Reference 7.

The staff finds that this proposed TS change is consistent with GENE-770-06-1A and is acceptable.

Bases for 3.3.2, 3.3.3, 3.3.4, 3.3.5, 3.3.6, 3.3.7, 3.3.8 (pages B 3/4 3-2 to B 3/4 3-4)

The Bases for each of the Technical Specifications listed above were modified to identify that CHANNEL FUNCTIONAL TEST frequencies and allowed out of service times for repair and surveillance testing have been determined in accordance with General Electric reports used to justify the changes in this Safety Assessment. As discussed in Section D.2.c, the Bases for Technical Specification 3.3.2, Isolation Actuation Instrumentation, were revised to provide guidance regarding implementation of ACTION 27 for inoperable instrument channels of the Reactor Vessel Water Level - Low, Level 3 Function.

Based on the discussion above, the staff finds this TS Bases change to be acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR

Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 16194). Accordingly, the amendments meet eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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