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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated February 3, 1992, as supplemented by letter dated June 16, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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 Facility Operating License No. NPF-57 is hereby amended to read as follows:

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PDR ADOCK 05000354
P PDR

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 53 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James C. Stone

For Charles L. Miller, 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: August 17, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages provided to maintain document completeness.*

Remove	Insert
2-5	2-5
-	-
B 2-7	B 2-7*
B 2-8	B 2-8
3/4 3-3	3/4 3-3
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5*
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-11	3/4 3-11*
3/4 3-12	3/4 3-12
3/4 3-17	3/4 3-17*
3/4 3-18	3/4 3-18
3/4 3-25	3/4 3-25*
3/4 3-26	3/4 3-26

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS
(continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. This item intentionally blank		
7. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	Elevation 110' 10.5"	Elevation 111' 0.5"
b. Level Transmitter/Trip Unit	Elevation 110' 10.5"*	Elevation 111' 4.5"*
9. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 530 psig	$\geq$ 465 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*80.5" above instrument zero EL 104' 2" for Level Transmitter/Trip Unit A&B (South Header) 83.25" above instrument zero EL 103' 11.25" for Level Transmitter/Trip Unit C&D (North Header)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 0.6 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when CMFLPD is greater than or equal to FRTP.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine control valve fast closure and turbine stop valve closure trip are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, low reactor water level, high steam tunnel temperature, and the low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. This item intentionally blank

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and the primary containment. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of approximately 35 gallons of water.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - This ACTION is deleted

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

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

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 4 APRMS, 6 IRMS and 2 SRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10. 9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is < 159.7 psig equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of ≤ 135.7 psig is used.

(k) Also actuates the EOC-RPT system.

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

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

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

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Not including simulated thermal power time constant, 6 ± 0.6 seconds.
#Measured from start of turbine control valve fast closure.

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[(b)],S	S/U[(c)], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[(f)]:				
a. Neutron Flux - Upscale, Setdown	S/U[(b)],S	S/U[(c)], W	SA	2
	S	W	SA	3, 4, 5
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	Q	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q[(k)]	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q[(k)]	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. This item intentionally blank				
7. Drywell Pressure - High	S	Q[(k)]	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter/Trip Unit	S	Q(k)	R	1, 5(j)
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

(i) This item intentionally blank

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

(k) Verify the tripset point of the trip unit at least once per 92 days.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL[d]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, Level 2	1, 2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low Low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	1, 8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	1, 8, 9, 10 11, 12, 13, 14, 15, 16, 17, 18	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	19[c]	2	1, 2, 3 and *	26
b. Drywell Pressure - High	19[c]	2	1, 2, 3	26
c. Refueling Floor Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	29
d. Reactor Building Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	28
e. Manual Initiation	19[c]	1	1, 2, 3 and *	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION		VALVE ACTUATION GROUPS OPERATED BY SIGNAL[(d)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
3.	MAIN STEAM LINE ISOLATION				
a.	Reactor Vessel Water Level - Low Low Low, Level 1	1	2	1, 2, 3	21
b.	Main Steam Line Radiation - High, High	2[(b)]	2	1, 2, 3##	28
c.	Main Steam Line Pressure - Low	1	2	1	22
d.	Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e.	Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f.	Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g.	Manual Initiation	1, 2, 17	2	1, 2, 3	25
4.	REACTOR WATER CLEANUP SYSTEM ISOLATION				
a.	RWCU Δ Flow - High	7	1/Valve[(e)]	1, 2, 3	23
b.	RWCU Δ Flow - High, Timer	7	1/Valve[(e)]	1, 2, 3	23
c.	RWCU Area Temperature - High	7	6/Valve[(e)]	1, 2, 3	23
d.	RWCU Area Ventilation Δ Temperature-High	7	6/Valve[(e)]	1, 2, 3	23
e.	SLCS Initiation	7[(f)]	1/Valve[(e)]	1, 2, 5#	23
f.	Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve[(e)]	1, 2, 3	23
g.	Manual Initiation	7	1/Valve[(e)]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

This table notation identifies which valves, in an actuation group, are closed by a particular trip signal. If all valves in the group are closed by the trip signal, only the valve group number will be listed. If only certain valves in the group are closed by the trip signal, the valve group number will be listed followed by, in parentheses, a listing of which valves are closed by the trip signal.

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level -	
1) Low Low, Level 2	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 2, 8, 9, 12, 13, 14, 15 (HV-5154, HV-5155), 17, 18
2) Low Low Low, Level 1	10, 11, 15(HV-5126 A&B, HV-5152 A&B, HV-5147, HV-5148 HV-5162), 16
b. Drywell Pressure - High	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18
c. Reactor Building Exhaust Radiation - High	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 12, 13, 14, 15, 17 (HV-5161), 18
d. Manual Initiation	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 10. 11, 12, 13, 14, 15, 16, 17 (HV-5161), 18
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	19
b. Drywell Pressure - High	19
c. Refueling Floor Exhaust Radiation - High	19
d. Reactor Building Exhaust Radiation - High	19
e. Manual Initiation	19

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	1 (HV-F022A, B, C & D, HV-F028A, B, C & D, HV-F067A, B, C & D, HV-F016, HV-F019)
b. Main Steam Line Radiation - High, High	2
c. Main Steam Line Presure - Low	1 (as above)
d. Main Steam Line Flow - High	1 (as above)
e. Condenser Vacuum - Low	1 (as above)
f. Main Steam Line Tunnel Temperature - High	1 (as above)
g. Manual Initiation	1 (as above), 2, 17 (SV-J004A-1, 2, 3, 4 & 5)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	7
b. RWCU Δ Flow - High, Timer	7
c. RWCU Area Temperature - High	7

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches*	$\geq$ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 82.0 psig	$\leq$ 102.0 psig
c. Manual Initiation	NA	NA

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

***These setpoints are as follows:

160°F - RWCU pipe chase room 4402

140°F - RWCU pump room and heat exchanger rooms

135°F - RWCU pipe chase room 4505

#30 minute time delay.

##15 minute time delay.

###The hydrogen water chemistry (HWC) system shall not be placed in service until reactor power reaches 20% of RATED THERMAL POWER. After reaching 20% of RATED THERMAL POWER, and prior to operating the HWC system, the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER and after the HWC system ha been shutoff, the background level and associated setpoint shall be returned to the normal full power values. If a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low Low, Level 2	NA
2) Low Low Low, Level 1	NA
b. Drywell Pressure - High	NA
c. Reactor Building Exhaust Radiation - High	NA
d. Manual Initiation	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level-Low Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Refueling Floor Exhaust Radiation - High(b)	$\leq$ 4.0
d. Reactor Building Exhaust Radiation - High(b)	$\leq$ 4.0
e. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\leq$ 1.0*/$\leq$ 13(a)**
b. Main Steam Line Radiation - High, High(a)(b)	$\leq$ 13(a)**
c. Main Steam Line Pressure - Low	$\leq$ 1.0*/$\leq$ 13(a)**
d. Main Steam Line Flow-High	$\leq$ 0.5*/$\leq$ 13(a)**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	NA
b. RWCU Δ Flow - High, Timer	NA
c. RWCU Area Temperature - High	NA
d. RWCU Area Ventilation Δ Temperature - High	NA
e. SLCS Initiation	NA
f. Reactor Vessel Water Level - Low Low, Level 2	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure (Flow) - High	NA
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA
c. RCIC Steam Supply Pressure - Low	NA
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA