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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
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 October 18, 1993, and supplement dated March 7, 1994, 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:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 70 , 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.

9406080347 940525
PDR ADOCK 05000354
P PDR

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

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: May 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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 page(s) provided to maintain document completeness.*

Remove	Insert
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10*
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12*
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16*
3/4 3-16a	3/4 3-16a
-	-
3/4 3-27	3/4 3-27*
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
B 3/4 3-1	B 3/4 3-1*
B 3/4 3-2	B 3/4 3-2
-	B 3/4 3-2a
-	-

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 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

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 requirement for one trip system, either

1) place the inoperable channel(s) in the tripped condition within

a) 1 hour for trip functions without an OPERABLE channel,
b) 12 hours for trip functions common to RPS instrumentation, and
c) 24 hours for trip functions not common to RPS instrumentation,

or

2) take the ACTION required by Table 3.3.2-1.

The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for both trip systems,

1) place the inoperable channel(s) in one trip system in the tripped condition within one hour, and

2) a) place the inoperable channel(s) in the remaining trip system in the tripped condition within

1) 1 hour for trip functions without an OPERABLE channel,
2) 12 hours for trip functions common to RPS instrumentation, and
3) 24 hours for trip functions not common to RPS instrumentation,

or

b) take the ACTION required by Table 3.3.2-1.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation 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 4.3.2.1-1.

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 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 isolation trip system.

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[j]	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[j]	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

Amendment No. 53
AUG 17 1992

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
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	3[j]	2/Valve[e]	1, 2, 3	27
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	3[j]	2/Valve[e]	1, 2, 3	27
c. Manual Initiation	3	1/Valve[e]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

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 one hour and declare the affected system inoperable.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - 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.

ACTION 26 - Establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation and Ventilation System (FRVS) operating within one hour. The action of operating FRVS is not required when the Reactor Vessel Water Level - Low Low, Level 2 instrumentation is inoperable as long as the following conditions are met:

a) the reactor water level is maintained at least 22 feet 2 inches over the top of the reactor pressure vessel flange,

b) the suppression pool level is maintained at greater than or equal to 5 inches indicated level,

c) at least one channel of the suppression pool high level alarm is operable, and

d) the spent fuel pool gates are removed.

ACTION 27 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 28 - Place the inoperable channel in the tripped condition or close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 29 - Place the inoperable channel in the tripped condition or establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation, and Ventilation System (FRVS) operating within one hour.

TABLE 3.3.2-1 (Continued)

NOTES

* 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 greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

Refer to Specification 3.1.5 for applicability.

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 has 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.

(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) Also trips and isolates the mechanical vacuum pumps.

(c) Also starts the Filtration, Recirculation and Ventilation System (FRVS).

(d) Refer to Table 3.3.2-1 table notation for the listing of which valves in an actuation group are closed by a particular isolation signal. Refer to Tables 3.6.3-1 and 3.6.5.2-1 for the listings of all valves within an actuation group.

(e) Sensors arranged per valve group, not per trip system.

(f) Closes only RWCU system isolation valve(s) HV-F001 and HV-F004.

(g) Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.

(h) Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.

(i) Manual isolation closes HV-F003 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.

(j) Trip functions common to RPS instrumentation.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
e.	RCIC Pump Room Temperature - High	NA
f.	RCIC Pump Room Ventilation Ducts Δ Temperature - High	NA
g.	RCIC Pipe Routing Area Temperature - High	NA
h.	RCIC Torus Compartment Temperature - High	NA
i.	Drywell Pressure - High	NA
j.	Manual Initiation	NA
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a.	HPCI Steam Line Δ Pressure (Flow) - High	NA
b.	HPCI Steam Line Δ Pressure (Flow) - High, Timer	NA
c.	HPCI Steam Supply Pressure - Low	NA
d.	HPCI Turbine Exhaust Diaphragm Pressure - High	NA
e.	HPCI Pump Room Temperature - High	NA
f.	HPCI Pump Room Ventilation Ducts Δ Temperature - High	NA
g.	HPCI Pipe Routing Area Temperature - High	NA
h.	HPCI Torus Compartment Temperature - High	NA
i.	Drywell Pressure - High	NA
j.	Manual Initiation	NA
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a.	Reactor Vessel Water Level - Low, Level 3	NA
b.	Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
c.	Manual Initiation	NA

(a) Isolation system instrumentation response time specified includes diesel generator starting and sequence loading delays.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed for MSIVs.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

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 Low, Level 2	S	Q	R	1, 2, 3
2) Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Building Exhaust Radiation - High	S	Q	R	1, 2, 3
d. Manual Initiation	NA	Q(a)	NA	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	S	Q	R	1, 2, 3
c. Refueling Floor Exhaust Radiation - High	S	Q	R	1, 2, 3 and *
d. Reactor Building Exhaust Radiation - High	S	Q	R	1, 2, 3 and *
e. Manual Initiation	NA	Q(a)	NA	1, 2, 3 and *
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, High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	Q	R	1
d. Main Steam Line Flow - High	S	Q	R	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
MAIN STEAM LINE ISOLATION (Continued)				
e. Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
g. Manual Initiation	NA	Q(a)	NA	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	S	Q	R	1, 2, 3
b. RWCU Δ Flow - High, Timer	NA	Q	R	1, 2, 3
c. RWCU Area Temperature - High	NA	Q	R	1, 2, 3
d. RWCU Area Ventilation Δ Temperature - High	NA	Q	R	1, 2, 3
e. SLCS Initiation	NA	Q(b)	NA	1, 2, 5#
f Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3
g. Manual Initiation	NA	Q(a)	NA	1, 2, 3
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure (Flow) - High	NA	Q	R	1, 2, 3
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA	Q	R	1, 2, 3
c. RCIC Steam Supply Pressure - Low	NA	Q	R	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	Q	R	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
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)				
e. RCIC Pump Room Temperature - High	NA	Q	R	1, 2, 3
f. RCIC Pump Room Ventilation Ducts Δ Temperature - High	NA	Q	R	1, 2, 3
g. RCIC Pipe Routing Area Temperature - High	NA	Q	R	1, 2, 3
h. RCIC Torus Compartment Temperature -High	NA	Q	R	1, 2, 3
i. Drywell Pressure - High	S	Q	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure (Flow) - High	NA	Q	R	1, 2, 3
b. HPCI Steam Line Δ Pressure (Flow) - High, Timer	NA	Q	R	1, 2, 3
c. HPCI Steam Supply Pressure - Low	NA	Q	R	1, 2, 3
d. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	Q	R	1, 2, 3
e. HPCI Pump Room Temperature - High	NA	Q	R	1, 2, 3
f. HPCI Pump Room Ventilation Ducts Δ Temperature - High	NA	Q	R	1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA	Q	R	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
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
h. HPCI Torus Compartment Temperature - High	NA	Q	R	1, 2, 3
i. Drywell Pressure - High	NA	Q	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	Q	R	1, 2, 3
c. Manual Initiation	NA	Q(a)	NA	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 greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

Refer to Specification 3.1.5 for applicability.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 92 days.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, 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 ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS 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 4.3.3.1-1.

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the 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 ECCS trip system.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter to T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

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. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation Common to RPS and ECCS Instrumentation," and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation." The safety evaluation reports documenting NRC approval of NEDC-30851P-A, Supplement 2 and NEDC-31677P-A are contained in letters to D. N. Grace from C. E. Rossi dated January 6, 1989 and to S. D. Floyd from C. E. Rossi dated June 18, 1990. 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.

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 D.C. 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 13 second delay. It follows that checking the valve speeds and the 13 second time for emergency power establishment will establish the response time for the isolation functions.

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 an allowance for instrument drift specifically allocated 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. Specified

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (Continued)

surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2. The safety evaluation reports documenting NRC approval of NEDC-30936P-A are contained in letters to D. N. Grace from A. C. Thadani (Part 1) and C. E. Rossi (Part 2) dated December 9, 1988. 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 Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.