DMB-016

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U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-325
and 50-324

Mr. J. A. Jones
Senior Executive Vice President
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 38 to Facility Operating License No. DPR-71 and Amendment No. 60 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications in response to your application dated June 11, 1981.

Amendment No. 38 for Brunswick Unit 1 revises the Technical Specifications to (1) establish new vessel level setpoints that are consistent with the installation of a common reference level required by TMI Action Item II.K.3.27 of NUREG-0737, (2) incorporate the Low Level #1 instrument into the Automatic Depressurization System (ADS) instrumentation tables, and (3) reflect instrument number designation changes resulting from the replacement of certain digital instrumentation with analog equipment.

Amendment No. 60 for Brunswick Unit 2 changes the Technical Specifications to (1) incorporate the Low Level #1 instrument into the ADS instrumentation table, (2) include reactor vessel shroud level instrumentation inadvertently omitted in a previous amendment, and (3) standardize instrumentation table terminology as previously discussed with members of your staff.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

CP
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ORIGINAL SIGNED BY

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 38 to DPR-71
2. Amendment No. 60 to DPR-62
3. Safety Evaluation
4. Notice

FR NOTICE
AMENDMENT ONLY

cc w/enclosures: See next page

OFFICE	ORB#2	ORB#2	OELD	ADOR		
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DATE	6/11/81	6/19/81	6/20/81	6/19/81		

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

June 25, 1981

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and 50-324

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Senior Executive Vice President
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

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Amendment No. 60 for Brunswick Unit 2 changes the Technical Specifications to (1) incorporate the Low Level #1 instrument into the ADS instrumentation table, (2) include reactor vessel shroud level instrumentation inadvertently omitted in a previous amendment, and (3) standardize instrumentation table terminology as previously discussed with members of your staff.

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Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

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3. Safety Evaluation
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cc w/enclosures: See next page

Mr. J. A. Jones
Carolina Power & Light Company

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Southport - Brunswick County Library
109 W. Moore Street
Southport, North Carolina 28461

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N. W.
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
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Southport, North Carolina 28461

Mr. Charles R. Dietz
Plant Manager
P. O. Box 458
Southport, North Carolina 28461

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-71

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

   A. The application for amendment by Carolina Power & Light Company dated June 11, 1981 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;

   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. DPR-71 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 25, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Remove the following pages and replace with identically numbered pages.

2-3/2-4
2-5/2-6
3/4 3-1/ 3/4 3-2
3/4 3-3/ 3/4 3-4
3/4 3-5/ 3/4 3-6
3/4 3-6A
3/4 3-7/ 3/4 3-8
3/4 3-8A
3/4 3-11
3/4 3-11A/ 3/4 3-12
3/4 3-15/ 3/4 3-16
3/4 3-17/ 3/4 3-18
3/4 3-21/ 3/4 3-22
3/4 3-22A
3/4 3-23/ 3/4 3-24
3/4 3-25/ 3/4 3-25A
3/4 3-26/ 3/4 3-26A
3/4 3-29/ 3/4 3-30
3/4 3-31/ 3/4 3-31A
3/4 3-32/ 3/4 3-32A
3/4 3-33/ 3/4 3-34
3/4 3-34A
3/4 3-35/ 3/4 3-35A
3/4 3-37/ 3/4 3-37A
3/4 3-38/ 3/4 3-38A
3/4 3-38B
3/4 3-51/ 3/4 3-52
3/4 3-63/ 3/4 3-64
3/4 3-65
3/4 8-9/ 3/4 8-10

The underlined page is an overleaf page and is provided for convenience.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown for each channel in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High[1] (C51-IRM-K601 A,B,C,D,E,F,G,H)	≤120 divisions of full scale	≤120 divisions of full scale
2. Average Power Range Monitor (C51-APRM-CH.A,B,C,D,E,F)		
a. Neutron Flux - High, 15%[2]	≤15% of RATED THERMAL POWER	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High[3][4]	≤(0.66 W + 54%)	≤(0.66 W + 54%)
c. Fixed Neutron Flux - High[4]	≤120% of RATED THERMAL POWER	≤120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (B21-PTM-N023A-1,B-1,C-1,D-1)	≤1045 psig	≤1045 psig
4. Reactor Vessel Water Level - Low, Level #1 (B21-LTM-N017A-1,B-1,C-1,D-1)	≥+162.5 inches above top fuel guide	≥+162.5 inches above top fuel guide
5. Main Steam Line Isolation Valve - Closure[5] (B21-F022 A,B,C,D; B21-F028 A,B,C,D)	≤10% closed	≤10% closed
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	≤3 x full power background	≤3.5 x full power background

TABLE 2.2.1-1 (continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
7. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤2 psig	≤2psig
8. Scram Discharge Volume Water Level - High (C11-LSH-N013 A,B,C,D)	≤109 gallons	≤109 gallons
9. Turbine Stop Valve - Closure(6) (EHC-SVOS-1X,2X,3X,4X)	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low(6) (EHC-PSL-1756,1757,1758,1759)	≥ 500 psig	≥ 500 psig

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

TABLE NOTATION

(1) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(2) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(3) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

(4) The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(5) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(6) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

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 with REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2. Set points and interlocks are given in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function of Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function 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 function.

*If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this could cause the Trip Function to occur.

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TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)(b)	ACTION
1. Intermediate Range Monitors: (C51-IRM-K601A,B,C,D,E,F,G,H)			
a. Neutron Flux - High	2, 5[b]	3	1
	3, 4	2	2
b. Inoperative	2, 5	3	1
	3, 4	2	2
2. Average Power Range Monitor: (C51-APRM-CH.A,B,C,D,E,F,)			
a. Neutron Flux - High, 15%	2, 5	2	3
b. Flow Biased Neutron Flux - High	1	2	4
c. Fixed Neutron Flux-High, 120%	1	2	4
d. Inoperative	1, 2, 5	2	5
e. Downscale	1	2	4
f. LPRM	1, 2, 5	(c)	NA
3. Reactor Vessel Steam Dome Pressure - High (B21-PT-N023A,B,C,D) (B21-PTM-N023A-1,B-1,C-1,D-1)	1, 2[d]	2	6
4. Reactor Vessel Water Level Low, Level #1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	1, 2	2	6
5. Main Steam Line Isolation Valve - Closure (B21-F022A,B,C,D, and B21-F028A,B,C,D)	1	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)(b)	ACTION
6. Main Steam Line Radiation - High (D12-RM-K603A,B,C,D)	1, 2[d]	2	7
7. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	1, 2[e]	2	6
8. Scram Discharge Volume Water Level - High (C11-LSH-N013A,B,C,D)	1, 2, 5[f]	2	5
9. Turbine Stop Valve - Closure (EHC-SVOS-1X,2X,3X,4X)	1[g]	4	8
10. Turbine Control Valve Fast Closure Control Oil Pressure - Low (EHC-PSL-1756,1757,1758,1759	1[g]	2	8
11. Reactor Mode Switch in Shutdown Position (C71A-S1)	1, 2, 3, 4, 5	1	9
12. Manual Scram (C71A-S3A,B)	1, 2, 3, 4, 5	1	10

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In CONDITION 5, suspend all operations involving CORE ALTERATIONS or postive reactivity changes and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 2 hours.

ACTION 5 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

ACTION 6 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 8 - Initiate a reduction in THERMAL POWER within 15 minutes and be at less than 30% of RATED THERMAL POWER within 2 hours.

ACTION 9 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, immediately and at least once per 12 hours verify that all control rods are fully inserted.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION 10 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

TABLE NOTATIONS

a. A channel may be placed in an inoperable status for up to 2 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. The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations.

c. An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than eleven LPRM inputs to an APRM channel.

d. These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

e. This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

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

g. These functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

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

JUN 11 1980

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors (C51-IRM-K601A,B,C,D,E,F,G,H):	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor* (C51-APRM-CH.A,B,C,D,E,F):	
a. Neutron Flux - High, 15%	≤0.09
b. Flow Biased Neutron Flux - High	NA
c. Neutron Flux - High, 120%	≤0.09
d. Inoperative	NA
e. Downscale	NA
f. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High (B21-PT-N023A,B,C,D) (B21-PTM-N023A-1,B-1,C-1,D-1)	≤0.55
4. Reactor Vessel Water Level - Level #1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	≤1.05
5. Main Steam Line Isolation Valve-Closure (B21-F022A,B,C,D and B21-F028A,B,C,D)	≤0.06
6. Main Steam Line Radiation - High (D12-RM-K603A,B,C,D)	NA
7. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	NA
8. Scram Discharge Volume Water Level - High (C11-LSH-N013A,B,C,D)	NA
9. Turbine Stop Valve - Closure (EHC-SV03-1X,2X,3X,4X)	≤0.06

TABLE 3.3.1-2 (Continued)

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low (EHC-PSL-1756,1757,1758,1759)	≤0.08
11. Reactor Mode Switch in Shutdown Position (C71A-S1)	NA
12. Manual Scram (C71A-S3A,B)	NA

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

TABLE 4.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors: (C51-IRM-K601 A,B,C,D,E,F,G,H)				
a. Neutron Flux - High	D	S/U[b][c]	R	2
	D	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor: (C51-APRM-CH.A,B,C,D,E,F)				
a. Neutron Flux - High 15%	S	S/U[b], W[d]	Q	2
	S	W	Q	5
b. Flow Biased Neutron Flux-High	S	S/U[b], W	W[e][f], Q	1
c. Fixed Neutron Flux - High, 120%	S	S/U[b], W	W[e], Q	1
d. Inoperative	NA	W	NA	1, 2, 5
e. Downscale	NA	W	NA	1
f. LPRM	D	NA	(g)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High				
(B21-PT-N023A, B, C, D)	NA[k]	NA	R[l]	
(B21-PTM-N023A-1, B-1, C-1, D-1)	D	M	M	1, 2
4. Reactor Vessel Water Level - Low Level #1				
(B21-LT-N017A-1, B-1, C-1, D-1)	NA[k]	NA	R[l]	
(B21-LTM-N017A-1, B-1, C-1, D-1)	D	M	M	1, 2
5. Main Steam Line Isolation Valve - Closure (B21-F022 A,B,C,D and B21-F028 A,B,C,D)	NA	M	R[h]	1

TABLE 4.3.1-1 (continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. Main Steam Line Radiation-High (D12-RM-K603 A,B,C,D)	S	M[i]	R[j]	1, 2
7. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	NA	M	Q	1, 2
8. Scram Discharge Volume Water Level - High (C11-LSH-N013 A, B, C, D)	NA	Q	R	1, 2, 5
9. Turbine Stop Valve - Closure (EHC-SVOS-1X, 2X, 3X, 4X)	NA	M	R[h]	1
10. Turbine Control Valve Fast Closure, Control Oil Pressure-Low (EHC-PSL-1756,1757,1758, 1759)	NA	M	R	1
11. Reactor Mode Switch in Shutdown Position (C71A-S1)	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram (C71A-S3A,B)	NA	Q	NA	1, 2, 3, 4, 5

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.
b. Within 24 hours prior to startup, if not performed within the previous 7 days.
c. The IRM channels shall be compared to the APRM channels and the SRM instruments for overlap during each startup, if not performed within the previous 7 days.
d. When changing from CONDITION 1 to CONDITION 2, perform the required surveillance within 12 hours after entering CONDITION 2.
e. This calibration shall consist of the adjustment of the APRM readout to conform to the power values calculated by a heat balance during CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.

TABLE 4.3.1-1 (continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

f. This calibration shall consist of the adjustment of the APRM flow biased setpoint to conform to a calibrated flow signal.

g. The LPRM's shall be calibrated at least once per effective full power month (EFPM) using the TIP system.

h. This calibration shall consist of a physical inspection and actuation of these position switches.

i. Instrument alignment using a standard current source.

j. Calibration using a standard radiation source.

k. The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

l. Transmitters are exempted from the monthly channel calibration.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low				
1. Level #1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	2, 6, 7, 8	2	1, 2, 3	20
2. Level #2 (B21-LT-N024A-1,B-1,and B21-LT-N025A-1,B-1 (B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	1, 3	2	1, 2, 3	20
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	2, 6, 7	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (d) (D12-RM-K603A,B,C,D)	1	2	1, 2, 3	21
2. Pressure - Low (B21-PS-N015 A,B,C,D)	1	2	1	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION (Continued)				
c. Main Steam Line (Continued)				
3. Flow - High (B21-dPIS-N006 A,B,C,D; B21-dPIS-N007 A,B,C,D; B21-dPIS-N008 A,B,C,D; and B21-dPIS-N009 A,B,C,D)	1	2/line	1	22
d. Main Steam Line Tunnel Temperature - High (B21-TS-N010A,B,C,D; B21-TS-N011A,B,C,D; B21-TS-N012A,B,C,D and B21-TS-N013A,B,C,D)	1	2(e)	1, 2, 3	21
e. Condenser Vacuum - Low (B21-PS-N056 A,B,C,D)	1	2	1, 2(f)	21
f. Turbine Building Area Temperature - High (B21-TS-3225A,B,C,D; B21-TS-3226A,B,C,D; B21-TS-3227A,B,C,D; B21-TS-3228A,B,C,D; B21-TS-3229A,B,C,D; B21-TS-3230A,B,C,D; B21-TS-3231A,B,C,D; and B21-TS-3232A,B,C,D)	1	4(e)	1, 2, 3	21

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High (D12-RM-N010A,B)	6	1	1, 2, 3, 5 and *	23
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	2, 6, 7	2	1, 2, 3	23
c. Reactor Vessel Water Level - Low, Level #2 (B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1)				
(B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	1, 3	2	1, 2, 3	23
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High (G31-dFS-N603-1A,1B)	3	1	1, 2, 3	24
b. Area Temperature - High (G31-TS-N600A,B,C,D,E,F)	3	2	1, 2, 3	24
c. Area Ventilation Δ Temp. - High (G31-TS-N602A,B,C,D,E,F)	3	2	1, 2, 3	24
d. SLCS Initiation (C41A-S1)	3(B)	NA	1, 2, 3	24
e. Reactor Vessel Water Level - Low, Level #2 (B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1)				
(B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	1, 3	2	1, 2, 3	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water - Low, Level #1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	2, 6, 7, 8	2	1, 2, 3	27
b. Reactor Steam Dome Pressure - High (B32-PS-N018A,B)	7, 8	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTIONS

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 24 - Isolate the reactor water cleanup system.

ACTION 25 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 26 - Verify power availability to the bus at least once per 12 hours.

ACTION 27 - Deactivate the shutdown cooling supply and reactor vessel head spray isolation valves in the closed position until the reactor steam dome pressure is within the specified limits.

NOTES

* When handling irradiated fuel in the secondary containment.

a. See Specification 3.6.3.1, Table 3.6.3.1-1 for valves in each valve group.

b. A channel may be placed in an inoperable status for up to 2 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.

c. With only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

d. Trips the mechanical vacuum pumps.

e. A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

f. With reactor steam pressure $\geq$ 500 psig.

g. Closes only RWCU outlet isolation valve.

h. Alarm only.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low		
1. Level #1 (B21-LTM-N017A-1, B-1, C-1, D-1)	≥ +162.5 inches	≥ +162.5 inches
2. Level #2 (B21-LTM-N024A-1, B-1 and B21-LTM-N025A-1, B-1)	≥ +112 inches	≥ +112 inches
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Main Steam Line		
1. Radiation - High (D12-RM-K603 A, B, C, D)	≤ 3 x full power background	≤ 3.5 x full power background
2. Pressure - Low (B21-PS-N015 A,B,C,D)	≥ 825 psig	≥ 825 psig
3. Flow - High (B21-dPIS-N006 A,B,C,D; B21-dPIS-N007 A,B,C,D; B21-dPIS-N008 A,B,C,D; and B21-dPIS-N009 A,B,C,D)	≤ 140% of rated flow	≤ 140% of rated
d. Main Steam Line Tunnel Temperature - High (B21-TS-N010 A, B, C, D; B21-TS-N011 A, B, C, D; B21-TS-N012 A, B, C, D; and B21-TS-N013 A, B, C, D)	≤ 200° F	≤ 200° F
e. Condenser Vacuum - Low (B21-PS-N056 A,B,C,D)	≥ 7 inches Hg Vacuum	≥ 7 inches Hg vacuum
f. Turbine Building Area Temp. - High (B21-TS-3225 A, B, C, D; B21-TS-3226 A, B, C, D; B21-TS-3227 A, B, C, D; B21-TS-3228 A, B, C, D; B21-TS-3229 A, B, C, D; B21-TS-3230 A, B, C, D; B21-TS-3231 A, B, C, D and B21-TS-3232 A, B, C, D)	≤ 200° F	≤ 200° F

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High (D12-RM-N010 A, B)	≤ 11 mr/hr	≤ 11 mr/hr
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Reactor Vessel Water Level - Low, Level #2 (B21-LTM-N024A-1, B-1 and B21-LTM-N025A-1, B-1)	≥ +112 inches	≥ +112 inches
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High (G31-dFS-N603-1A, 1B)	≤ 53 gal/min	≤ 53 gal/min
b. Area Temperature - High (G31-TS-N600A, B, C, D, E, F)	≤ 150° F	≤ 150° F
c. Area Ventilation Temperature Δ Temp - High (G31-TS-N602A, B, C, D, E, F)	≤ 50° F	≤ 50° F
d. SLCS Initiation (C41A-S1)	NA	NA
e. Reactor Vessel Water - Low, Level #2 (B21-LTM-N024A-1, B-1 and B21-LTM-N025A-1, B-1)	≥ +112 inches	≥ +112 inches

TABLE 3.3.2-2 (continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
5. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water - Low, Level #1 (B21-LTM-N017A-1, B-1, C-1, D-1)	≥ +162.5 inches	≥ +162.5 inches
b. Reactor Steam Dome Pressure - High (B32-PS-N018A, B)	≤ 140 psig	≤ 140 psig

TABLE 3.3.2.-3

ISOLATION SYSTEM RESPONSE TIME

TRIP FUNCTION AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low	
1. Level #1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	≤13
2. Level #2 (B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1) (B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	≤1.0**
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤13
c. Main Steam Line	
1. Radiation - High* (D12-RM-K603A,B,C,D)	≤1.0**
2. Pressure - Low (B21-PS-N015 A,B,C,D)	≤13
3. Flow - High (B21-dPIS-N006 A,B,C,D; B21-dPIS-N007 A,B,C,D; B21-dPIS-N008 A,B,C,D and B21-dPIS-N009 A,B,C,D)	≤0.5**
d. Main Steam Line Tunnel Temperature - High (B21-TS-N010A,B,C,D; B21-TS-N011A,B,C,D; B21-TS-N012A,B,C,D; and B21-TS-N013A,B,C,D)	≤13
e. Condenser Vacuum - Low (B21-PS-N056 A,B,C,D)	≤13

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM RESPONSE TIME

TRIP FUNCTION AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
1. PRIMARY CONTAINMENT ISOLATION (Continued)	
f. Turbine Building Area Temperature - High (B21-TS-3225A,B,C,D; B21-TS-3226A,B,C,D; B21-TS-3227A,B,C,D; B21-TS-3228A,B,C,D; B21-TS-3229A,B,C,D; B21-TS-3230A,B,C,D; B21-TS-3231A,B,C,D; B21-TS-3232A,B,C,D)	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Exhaust Radiation - High * (D12-RM-N010A,B)	≤13
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤13
c. Reactor Vessel Water Level - Low, Level #2 (B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1) (B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	≤1.0**

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM RESPONSE TIME

TRIP FUNCTION AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. △ Flow - High (G31-dTS-N603-1A,1B)	≤13
b. Area Temperature - High (G31-TS-N600A,B,C,D,E,F)	≤13
c. Area Ventilation Temperature △ T - High (G31-TS-N602A,B,C,D,E,F)	≤13
d. SLCS Initiation (C41A-S1)	NA
e. Reactor Vessel Water - Low, Level #2 (B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1) B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	≤1.0**
4. CORE STANDBY COOLING SYSTEMS ISOLATION	
a. High Pressure Coolant Injection Isolation	
1. HPCI Steam Line Flow - High (E41-dPIS-N004 and E41-dPIS-N005)	≤13
2. HPCI Steam Supply Pressure - Low (E41-PSL-N001A,B,C,D)	≤13
3. HPCI Steam Line Tunnel Temperature - High (E41-TS-3314; E41-TS-3315; E41-TS-3316; E41-TS-3317; E41-TS-3318; E41-TS-3354; E41-TS-3488 and E41-TS-3489)	≤13
4. Bus Power Monitor (E41-K55 and E41-K56)	NA
5. HPCI Turbine Exhaust Diaphragm Pressure - High (E41-PSH-N012A,B,C,D)	NA
6. HPCI Steam Line Ambient Temperature - High (E51-TS-N603C,D)	NA
7. HPCI Steam Line Area △ Temp - High (E51-dTS-N604C,D)	NA
8. Emergency Area Cooler Temperature - High (E41-TS-602A,B)	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM RESPONSE TIME

TRIP FUNCTION AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
b. Reactor Core Isolation Cooling System Isolation	
1. RCIC Steam Line Flow - High (E51-dPIS-N017 and E51-dPIS-N018)	NA
2. RCIC Steam Supply Pressure - Low (E51-PS-N019A,B,C,D)	NA
3. RCIC Steam Line Tunnel Temp - High (E51-TS-3319; E51-TS-3320; E51-TS-3321; E51-TS-3322; E51-TS-3323; E51-TS-3355 and E51-TS-3487)	NA
4. Bus Power Monitor (E51-K42 and E51-K43)	NA
5. RCIC Turbine Exhaust Diaphram Pressure - High (E51-PS-N012A,B,C,D)	NA
6. RCIC Steam Line Ambient Temperature - High (E51-TS-N603A,B)	NA
7. RCIC Steam Line Area Δ Temp - High (E51-dTS-N604A,B)	NA
8. Emergency Area Cooler Temperature - High (E51-TS-N602A,B)	NA
9. RCIC Equipment Room Δ Temp - High (E51-dTS-N601A,B)	NA
5. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low, Level #1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	NA
b. Reactor Steam Dome Pressure - High (B32-PS-N018A,B)	NA

* Radiation monitors 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 actuation instrumentation response time only.

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low				
1. Level #1				
(B21-LT-N017A-1,B-1,C-1,D-1)	NA[a]	NA	R[b]	
(B21-LTM-N017A-1,B-1,C-1,D-1)	D	M	M	1, 2, 3
2. Level #2				
(B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1)	NA[a]	NA	R[b]	
(B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	D	M	M	1, 2, 3
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	NA	M	Q	1, 2, 3
c. Main Steam Line				
1. Radiation - High (D12-RM-K603A,B,C,D)	D	W	R	1, 2, 3
2. Pressure - Low (B21-PS-N015 A,B,C,D)	NA	M	Q	1
3. Flow - High (B21-dPIS-N006 A,B,C,D; B21-dPIS-N007 A,B,C,D; B21-dPIS-N008 A,B,C,D; and B21-dPIS-N009 A,B,C,D)	NA	M	Q	1

TABLE 4.3.2.-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION (Continued)				
d. Main Steam Line Tunnel Temperature - High (B21-TS-N010A,B,C,D; B21-TS-N011A,B,C,D; B21-TS-N012A,B,C,D and B21-TS-N013A,B,C,D)	NA	M	R	1, 2, 3
e. Condenser Vacuum - Low (B21-PS-N056 A,B,C,D)	NA	M	R	1, 2#
f. Turbine Building Area Temp - High (B21-TS-3225A,B,C,D; B21-TS-3226A,B,C,D; B21-TS-3227A,B,C,D; B21-TS-3228A,B,C,D; B21-TS-3229A,B,C,D; B21-TS-3230A,B,C,D; B21-TS-3231A,B,C,D and B21-TS-3232A,B,C,D)	NA	M	R	1, 2, 3

#When reactor steam pressure ≥500 psig.

TABLE 4.3.2.-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High (D12-RM-N010A,B)	D	M	R	1, 2, 3, 5 and *
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low, Level #2 (B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1)	NA[a]	NA	R[b]	
(B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	D	M	M	1, 2, 3
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High (G31-dFS-N603-1A,1B)	D	M	R	1, 2, 3
b. Area Temperature - High (G31-TS-N604A,B,C,D,E,F)	NA	M	R	1, 2, 3
c. Area Ventilation Δ Temp - High (G31-TS-N602A,B,C,D,E,F)	NA	M	R	1, 2, 3

*When handling irradiated fuel in the secondary containment.

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION (continued)				
d. SLCS Initiation (C41A-S1)	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low, Level #2				
(B21-LT-N024A-1,B-1 and B21-LT-N025A-1,B-1)	NA[a]	NA	R[b]	
(B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	D	M	M	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water - Low, Level #1				
(B21-LT-N017A-1,B-1,C-1,D-1)	NA[a]	NA	R[b]	
(B21-LTM-N017A-1,B-1,C-1,D-1)	D	M	M	1, 2, 3
b. Reactor Steam Dome Pressure High (B32-PS-N018A,B)	NA	S/U*, M	R	1, 2, 3

* If not performed within the previous 31 days.

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

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 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 and 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.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS 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 logic train such that both logic trains are tested at least once per 36 months and one channel per function 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 function.

*For the ECCS response time test for Items 1 and 2 of Table 3.3.3-3 scheduled for completion by February 21, 1981, and February 19, 1981, respectively, a onetime-only exemption is allowed to extend this test until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low, Level #3 (B21-LT-N031A, B, C, D) (B21-LTS-N031A-4, B-4, C-4, D-4)	2	1, 2, 3, 4, 5	30
b. Reactor Steam Dome Pressure - Low (Injection Permissive) (B21-PT-N021A, B, C, D) (B21-PTM-N021A-1, B-1, C-1, D-1) (B21-PTS-N021A-2, B-2, C-2, D-2)	2	1, 2, 3, 4, 5	31
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	2	1, 2, 3	30
d. Time Delay Relay	1	1, 2, 3, 4, 5	31
e. Bus Power Monitor #(E21-K1A, B)	1/bus	1, 2, 3, 4, 5	32
2. LPCI MODE OF RHR SYSTEM			
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	2	1, 2, 3	30
b. Reactor Vessel Water Level - Low, Level #3 (B21-LT-N031A, B, C, D) (B21-LTS-N031A-4, B-4, C-4, D-4)	2	1, 2, 3, 4*, 5*	30

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.
#Alarm Only.

TABLE 3.3.3-1 (continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTIO
2. LPCI MODE OF RHR SYSTEM (continued)			
c. Reactor Vessel Shroud Level (Drywell Spray Permissive) (B21-LT-N036 and B21-LT-N037) (B21-LTM-N036-1 and B21-LTM-N037-1)	1	1, 2, 3, 4*, 5*	31
d. Reactor Steam Dome Pressure - Low (Injection Permissive) (B21-PT-N021A, B, C, D) (B21-PTM-N021A-1, B-1, C-1, D-1) (B21-PTS-N021A-2, B-2, C-2, D-2)			
1. RHR Pump Start and LPCI Injection Valve Actuation	2	1, 2, 3, 4*, 5*	31
2. Recirculation Loop Pump Discharge Valve Actuation	2	1, 2, 3, 4*, 5*	31
e. RHR Pump Start - Time Delay Relay	1	1, 2, 3, 4*, 5*	31
f. Bus Power Monitor# (E11-K106A, B)	1/bus	1, 2, 3, 4*, 5*	32

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.
#Alarm only.

TABLE 3.3.3-1 (continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
3. HPCI SYSTEM			
a. Reactor Vessel Water Level - Low, Level #2 (B21-LT-N031A, B, C, D) (B21-LTM-N031A-1, B-1, C-1, D-1; B21-LTS-N031A-2, B-2, C-2, D-2)	2	1, 2, 3	30
b. Drywell Pressure - High (E11-PS-N011A,B,C,D)	2	1, 2, 3	30
c. Condensate Storage Tank Level-Low (E41-LS-N002, E41-LS-N003)	2**	1, 2, 3	33
d. Suppression Chamber Water Level-High (E41-LSH-N015A, B)	2**	1, 2, 3	33
e. Bus Power Monitor # (E41-K55 and E41-K56)	1/bus	1, 2, 3	32
4. ADS			
a. Drywell Pressure - High, coincident with (E11-PS-N010A,B,C,D)	2	1, 2, 3	30
b. Reactor Vessel Water Level - Low, Level #3 (B21-LT-N031A, B, C, D) (B21-LTS-N031A-3, B-3, C-3, D-3)	2	1, 2, 3	30

Alarm only.

**Provides signal to HPCI pump suction valves only.

TABLE 3.3.3-1 (continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. ADS (continued)			
c. Reactor Vessel Water Level - Low Level #1 (B21-LT-N042A, B; B21-LTM-N042A-1, B-1)	1	1, 2, 3	30
d. ADS Timer (B21-TDPU-K5A, B)	1	1, 2, 3	31
e. Core Spray Pump Discharge Pressure - High (Permissive) (E21-PS-N008A, B and E21-PS-N009A, B)	2	1, 2, 3	31
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (E11-PS-N016A,B,C,D and E11-PS-N020A,B,C,D)	2/pump	1, 2, 3	31
g. Bus Power Monitor # (B21-K1A, B)	1/bus	1, 2, 3	32

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	1/bus	1/bus	1/bus	1, 2, 3, 4##, 5##	34
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	3/bus	2/bus	2/bus	1, 2, 3, 4##, 5##	35

#Alarm only.
##Required when ESF equipment is required to be OPERABLE.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place at least one inoperable channel in the tripped condition within one hour or declare the associated ECCS inoperable.

b. For both trip systems, declare the associated ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, declare the associated ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, verify bus power availability at least once per 12 hours or declare the associated ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the HPCS system inoperable.

ACTION 34 - 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 35 - 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.

JUN 11 1980

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low, Level #3 (B21-LTS-N031A-4, B-4, C-4, D-4)	≥ +2.5 inches above top fuel guide	≥ +2.5 inches above top fuel guide
b. Reactor Steam Dome Pressure - Low (B21-PTS-N021A-2, B-2, C-2, D-2)	410 ± 15 psig	410 ± 15 psig
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	≤ 2 psig	≤ 2 psig
d. Time Delay Relay	14 ≤ t ≤ 16 secs	14 ≤ t ≤ 16 secs
e. Bus Power Monitor (E21-K1A, B)	NA	NA
2. LPCI MODE OF RHR SYSTEM		
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	≤ 2 psig	≤ 2 psig
b. Reactor Vessel Water Level - Low, Level #3 (B21-LTS-N031A-4, B-4, C-4, D-4)	≥ +2.5 inches above top fuel guide	≥ +2.5 inches above top fuel guide
c. Reactor Vessel Shroud Level (B21-LTM-N036-1 and B21-LTM-N037-1)	≥ 53" below top fuel guide	≥ 53" below top fuel guide

TABLE 3.3.3-2 (continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
d. Reactor Steam Dome Pressure - Low		
1. RHR Pump Start and LCPI Injection Valve Actuation (B21-PTS-N021A-2, B-2, C-2, D-2)	410 ± 15 psig	410 ± 15 psig
2. Recirculation Pump Discharge Valve Actuation (B21-PTM-N021A-1, B-1, C-1, D-1)	310 ± 15 psig	310 ± 15 psig
e. RHR Pump Start - Time Delay Relay	$9 \leq t \leq 11$ seconds	$9 \leq t \leq 11$ seconds
f. Bus Power Monitor (E11-K106A, B)	NA	NA

TABLE 3.3.3-2 (continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
3. HPCI SYSTEM		
a. Reactor Vessel Water Level - Low, Level #2 (B21-LTS-N031A-2, B-2, C-2, D-2)	≥ +112 inches above top fuel guide	≥ +112 inches above top fuel guide
b. Drywell Pressure - High (E11-PS-N011A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Condensate Storage Tank Level - Low (E41-LS-N002, E41-LS-N003)	≥ 23'4"	≥ 23'4"
d. Suppression Chamber Water Level - High* (E41-LSH-N015A, B)	≤ -2 feet	≤ -2 feet
e. Bus Power Monitor (E41-K55 and E41-K56)	N/A	N/A
4. ADS		
a. Drywell Pressure - High (E11-PS-N010A,B,C,D)	≤ 2 psig	≤ 2 psig
b. Reactor Vessel Water Level - Low, Level #3 (B21-LTS-N031A-3, B-3, C-3, D-3)	≥ +2.5 inches	≥ +2.5 inches
c. Reactor Vessel Water Level - Low, Level #1 (B21-LTM-N042A-1, B-1)	≥ +162.5 inches	≥ +162.5 inches
d. ADS Timer (B21-TDPU-K5A, B)	≤ 120 seconds	≤ 120 seconds
e. Core Spray Pump Discharge Pressure - High (E21-PS-N008A, B and E21-PS-N009A, B)	≥ 100 psig	≥ 100 psig
f. RHR (LPCI Mode) Pump Discharge Pressure- High (E11-PS-N016A, B, C, D and E11-PS-N020A, B, C, D)	≥ 100 psig	≥ 100 psig

*Suppression chamber water level zero is the torus centerline minus 1 inch.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
4. ADS (continued)		
g. Bus Power Monitor (B21-K1A, B)	NA	NA
5. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)*	a. 4.16 kv Basis - 2940 ± 161 volts b. 120 v Basis - 84 ± 4.6 volts c. ≤ 10 sec. time delay	2940 ± 315 volts 84 ± 9 volts ≤ 10 sec. time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3727 ± 9 volts b. 120 v Basis - 106.5 ± 0.25 volts c. 10 ± 0.5 sec. time delay	3727 ± 21 volts 106.5 ± 0.60 volts 10 ± 1.0 sec. time delay

*This is an inverse time delay voltage relay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 4.3.3-1

ENERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low, Level #3				
(B21-LT-N031A,B,C,D)	NA[a]	NA	R[b]	
(B21-LTS-N031A-4,B-4,C-4,D-4)	D	M	M	1, 2, 3, 4, 5
b. Reactor Steam Dome Pressure - Low				
(B21-PT-N021A,B,C,D)	NA[a]	NA	R[b]	
(B21-PTM-N021A-1,B-1,C-1,D-1)	D	M	M	1, 2, 3, 4, 5
(B21-PTS-N021A-2,B-2,C-2,D-2)	D	M	M	
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	NA	M	Q	1, 2, 3
d. Time Delay Relay	NA	R	R	1, 2, 3, 4, 5
e. Bus Power Monitor (E21-K1A,B)	NA	R	NA	1, 2, 3, 4, 5
2. LPCI MODE OF RHR SYSTEM				
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	NA	M	Q	1, 2, 3

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

TABLE 4.3.3.-1 (Continued)

ENERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
2. LPCI MODE OF RHR SYSTEM (Continued)				
b. Reactor Vessel Water Level - Low, Level #3				
(B21-LT-N031A,B,C,D)	NA[a]	NA	R[b]	
(B21-LTS-N031A-4,B-4,C-4,D-4)	D	M	M	1, 2, 3, 4*, 5*
c. Reactor Vessel Shroud Level				
(B21-LT-N036; B21-LT-N037)	NA[a]	NA	R[b]	
(B21-LTM-N036-1 and B21-LTM-N037-1)	D	M	M	1, 2, 3, 4*, 5*
d. Reactor Steam Dome Pressure - Low (B21-PT-N021A,B,C,D)	NA[a]	NA	R[b]	
1. RHR Pump Start and LPCI Injection Valve Actuation (B21-PTS-N021A-2,B-2,C-2,D-2)	D	M	M	1, 2, 3, 4*, 5*
2. Recirculation Loop Pump Discharge Valve Actuation (B21-PTM-N021A-1,B-1,C-1,D-1)	D	M	M	1, 2, 3, 4*, 5*
e. RHR Pump Start-Time Delay Relay	NA	R	R	1, 2, 3, 4*, 5*
f. Bus Power Monitor (E11-K106A,B)	NA	R	NA	1, 2, 3, 4*, 5*

* Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

TABLE 4.3.3.-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. HPCI SYSTEM				
a. Reactor Vessel Water Level - Low Level #2				
(B21-LT-N031A,B,C,D)	NA[a]	NA	R[b]	
(B21-LTM-N031A-1,B-1,C-1,D-1)	D	M	M	1, 2, 3
(B21-LTS-N031A-2,B-2,C-2,D-2)	D	M	M	
b. Drywell Pressure - High (E11-PS-N011A,B,C,D)	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level Low (E41-LS-N002, E41-LS-N003)	NA	M	Q	1, 2, 3
d. Suppression Chamber Water Level - High (E41-LSH-N015A,B)	NA	M	Q	1, 2, 3
e. Bus Power Monitor (E41-K55 and E41-K56)	NA	R	NA	1, 2, 3
4. ADS				
a. Drywell Pressure - High (E11-PS-N010A,B,C,D)	NA	M	Q	1, 2, 3

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

TABLE 4.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. ADS (Continued)				
b. Reactor Vessel Water Level - Low, Level #3				
(B21-LT-N031A,B,C,D)	NA[a]	NA	R[b]	
(B21-LTS-N031A-3,B-3,C-3,D-3)	D	M	M	1, 2, 3
c. Reactor Vessel Water Level - Low Level #1				
(B21-LT-N042A,B)	NA[a]	NA	R[b]	1, 2, 3
(B21-LTM-N042A-1,B-1)	D	M	M	
d. ADS Timer (B21-TDPU-K5A,B)	NA	R	R	1, 2, 3
e. Core Spray Pump Discharge Pressure - High (E21-PS-N008A,B and E21-PS-N009A,B)	NA	M	Q	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High (E11-PS-N016A,B,C,D and E11-PS-N020A,B,C,D)	NA	M	Q	1, 2, 3
g. Bus Power Monitor (B21-K1A,B)	NA	R	NA	1, 2, 3

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

TABLE 4.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4*, 5*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	S	M	R	1, 2, 3, 4*, 5*

*Required when ESF equipment is required to be OPERABLE.

TABLE 3.3.5.3-1

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT AND INSTRUMENT NUMBER	MINIMUM NO. OF OPERABLE INSTRUMENT CHANNELS
1. Reactor Vessel Water Level (B21-LITS-N026A,B; B21-LR-615; B21-LI-R604A,B; B21-LT-N037 and B21-LTM-N037-1)	2
2. Reactor Vessel Pressure (B21-PI-R004A,B; C32-LPR-R608 and C32-PT-N005A,B)	2
3. Containment Pressure (CAC-PI-2599; CAC-PT-2559; CAC-PR-1257-1 and CAC-PT-1257-1)	2
4. Containment Pressure (CAC-TR-1258-1 thru 13,22,23,24 and C91-P602)	2
5. Suppression Chamber Atmosphere Temperature (CAC-TR-1258-17 thru 20 and C91-P602)	2
6. Suppression Chamber Water Level (CAC-LI-2601-3; CAC-LA-2602; CAC-LT-2601; CAC-LT-2602 and CAC-LY-2601-1)	2
7. Suppression Chamber Water Temperature (CAC-TR-1258-14, 21 and C91-P602)	2
8. Containment Radiation (CAC-AR-1260; CAC-AQH-1260-1,2,3; CAC-AR-1261; CAC-AQH-1261-1,2,3; CAC-AR-1262 and CAC-AQH-1262-1,2,3)	2
9. Containment Oxygen (CAC-AT-1259-2; CAC-AR-1259; CAC-AT-1263-2 and CAC-AR-1263)	2
10. Containment Hydrogen (CAC-AT-1959-1; CAC-AR-1259; CAC-AT-1263-1 and CAC-AR-1263)	2
11. Safety relief valve position indication: Primary - Sonic (B21-FY-4157 through 4167 Secondary - Temp. (B21-TR-R614 points 1-11)	1/valve

TABLE 4.3.5.3-1

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Water Level (B21-LITS-N026A,B; B21-LR-R615; B21-LI-R604A,B; B21-LT-N037 and B21-LTM-N037-1)	M	R
2. Reactor Vessel Pressure (B21-PI-R004A,B; C32-LPR-R608 and C32-PT-N005A,B)	M	R
3. Containment Pressure (CAC-PI-2599; CAC-PT-2599; CAC-PR-1257-1 and CAC-PT-1257-1)	M	R
4. Containment Temperature (CAC-TR-1258-1 thru 13,22,23,24 and C91-P602)	M	R
5. Suppression Chamber Atmosphere Temperature (CAC-TR-1258-17 thru 20 and C91-P602)	M	R
6. Suppression Chamber Water Level (CAC-LI-2601-3; CAC-LR-2602; CAC-LT-2601; CAC-LT-2602 and CAC-LY-2601-1)	M	R
7. Suppression Chamber Water Temperature (CAC-TR-1258-14, 21 and C91-P602)	M	R
8. Containment Radiation (CAC-AR-1260; CAC-AQH-1260-1,2,3; CAC-AR-1261; CAC-AQH-1261-1,2,3; CAC-AR-1262 and CAC-AQH-1262-1,2,3)	M	R
9. Containment Oxygen Concentration (CAC-AT-1259-2; CAC-AR-1259; CAC-AT-1263-2 and CAC-AR-1263)	M	R
10. Containment Hydrogen Concentration (CAC-AT-1259-1; CAC-AR-1259; CAC-AT-1263-1 and CAC-AR-1263)	M	R
11. Safety relief valve position indication (Primary-Sonic) (B21-FY-4157 thru 4167)	M	R
12. Safety relief valve position indication (Secondary-Temperature) (B21-TR-R614 points 1 thru 11)	M	R

TABLE 3.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE TRIP SYSTEMS PER OPERATING PUMP
1. Reactor Vessel Water Level - Low Low, Level 2 (B21-LT-N024A-2,B-2 and B21-LT-N025A-2,B-2) B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	1
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	1

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low low, Level 2 (B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	≥ +112 inches above top fuel guide	≥ +112 inches above top fuel guide
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	≤ 1120 psig	≤ 1120 psig

TABLE 4.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2			
(B21-LT-N024A-2,B-2; B21-LT-N025A-2,B-2)	NA[a]	NA	R[b]
B21-LTM-N024A-2,B-2; B21-LTM-N025A-2,B-2)	D	M	M
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	NA	M	R

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

4.8.2.3.2 Each 125-volt battery and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

2. The pilot cell specific gravity, corrected to 77°F, is ≥ 1.18,

3. The pilot cell voltage is ≥ 2.0 volts, and

4. The overall battery voltage is ≥ 120 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is ≥ 2.0 volts under float charge and has not decreased more than 0.3 volts from the value observed during the original acceptance test,

2. The specific gravity, corrected to 77°F, of each connected cell is ≥ 1.18 and has not decreased more than 0.04 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1.* The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of abnormal corrosion and coated with anti-corrosion material, and

3.* The battery charger will supply at least 250 amperes at a minimum of 135 volts for at least 4 hours.

*For the verification of this item scheduled for completion by February 23, 1981, a onetime-only exemption is allowed to extend this inspection until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 8 hours when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage ≥ 105 volts.

a) During the initial 60 seconds of the test;

1) Battery 1A-1 ≥ 1056.42 amperes,
2) Battery 1A-2 ≥ 1211.90 amperes,
3) Battery 1B-1 ≥ 1089.06 amperes, and
4) Battery 1B-2 ≥ 1042.67 amperes.

b) During the remainder of the first 30 minutes of the test;

1) Battery 1A-1 ≥ 243.19 amperes,
2) Battery 1A-2 ≥ 159.10 amperes,
3) Battery 1B-1 ≥ 176.79 amperes, and
4) Battery 1B-2 ≥ 216.67 amperes.

c) During the remainder of the 8 hour test;

1) Battery 1A-1 ≥ 89.52 amperes,
2) Battery 1A-2 ≥ 50.34 amperes,
3) Battery 1B-1 ≥ 53.39 amperes, and
4) Battery 1B-2 ≥ 75.09 amperes.

3. At the completion of either of the above tests, the battery charger shall be demonstrated capable of recharging its battery at a rate of at least 200 amperes while supplying normal D.C. loads. The battery shall be charged to at least 95% capacity in ≤ 24 hours.

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be perfor as subsequent to the satisfactory completion of the required battery service test and after normal equalizer charge.

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company dated June 11, 1981 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;

   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. DPR-62 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 60 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 25, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Remove the following pages and replace with identically numbered pages.

2-3/2-4
3/4 3-31/3/4 3-32
3/4 3-33/3/4 3-34
3/4 3-35/3/4 3-35A
3/4 3-37/3/4 3-38

The underlined page is an overleaf page and is provided for convenience.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown for each channel in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High[1] (C51-IRM-K601 A,B,C,D,E,F,G,H)	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor (C51-APRM-CH.A,B,C,D,E,F)		
a. Neutron Flux - High, 15%[2]	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High[3][4]	≤ (0.66 W + 54%)	≤ (0.66 W + 54%)
c. Fixed Neutron Flux - High[4]	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (B21-PS-N023 A,B,C,D)	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1 (B21-LIS-N017 A,B,C,D)	≥ 162.5 inches above instrument zero	≥ 162.5 inches above top fuel guide
5. Main Steam Line Isolation Valve - Closure[5] (B21-F022 A,B,C,D; B21-F028 A,B,C,D)	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	≤ 3 x full power background	≤ 3.5 x full power background
7. Drywell Pressure - High (C72-PS-N002 A,B,C,D)	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High (C12-LSH-N013 A,B,C,D)	≤ 109 gallons	≤ 109 gallons

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	2	1, 2, 3, 4, 5	30
b. Reactor Steam Dome Pressure - Low (Injection Permissive) (B21-PS-N021A,B,C,D)	2	1, 2, 3, 4, 5	31
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	2	1, 2, 3	30
d. Time Delay Relay	1	1, 2, 3, 4, 5	31
e. Bus Power Monitor# (E21-K1A,B)	1/bus	1, 2, 3, 4, 5	32
2. LPCI MODE OF RHR SYSTEM			
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	2	1, 2, 3	30
b. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	2	1, 2, 3, 4*, 5*	30
c. Reactor Vessel Shroud Level (Drywell Spray Permissive) (B21-LITS-N036 and B21-LITS-N037)	1	1, 2, 3, 4*, 5*	31
d. Reactor Steam Dome Pressure - Low (Injection Permissive) (B21-PS-N021A,B,C,D)			
1. RHR Pump Start and LPCI Injection Valve Actuation	2	1, 2, 3, 4*, 5*	31
2. Recirculation Loop Pump Discharge Valve Actuation	2	1, 2, 3, 4*, 5*	31
e. RHR Pump Start - Time Delay Relay	1	1, 2, 3, 4*, 5*	31
f. Bus Power Monitor# (E11-K106A,B)	1/bus	1, 2, 3, 4*, 5*	32
3. HPCI SYSTEM			
a. Reactor Vessel Water Level - Low, Level #2 (B21-LIS-N031A,B,C,D)	2	1, 2, 3	30
b. Drywell Pressure - High (E11-PS-N011A,B,C,D)	2	1, 2, 3	30
c. Condensate Storage Tank Level-Low (E41-LS-N002, E41-LS-N003)	2**	1, 2, 3	33
d. Suppression Chamber Water Level-High (E41-LSH-N015A,B)	2**	1, 2, 3	33
e. Bus Power Monitor# (E41-K55 and E41-K56)	1/bus	1, 2, 3	32

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. ADS			
a. Drywell Pressure - High, coincident with (E11-PS-N010A,B,C,D)	2	1, 2, 3	30
b. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	2	1, 2, 3	30
c. Reactor Vessel Water Level - Low Level #1 (B21-LIS-N042 A, B)	1	1, 2, 3	30
d. ADS Timer (B21-TDPU-K5A,B)	1	1, 2, 3	31
e. Core Spray Pump Discharge Pressure - High (Permissive) (E21-PS-N008A,B and E21-PS-N009A,B)	2	1, 2, 3	31
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (E11-PS-N016A,B,C,D and E11-PS-N020A,B,C,D)	2/pump	1, 2, 3	31
g. Bus Power Monitor# (B21-K1A,B)	1/bus	1, 2, 3	32

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	1/bus	1/bus	1/bus	1, 2, 3, 4##, 5##	34
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	3/bus	2/bus	2/bus	1, 2, 3, 4##, 5##	35

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.
**Provides signal to HPCI pump suction valves only.
#Alarm only.
##Required when ESF equipment is required to be OPERABLE.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place at least one inoperable channel in the tripped condition within one hour or declare the associated ECCS inoperable.

b. For both trip systems, declare the associated ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, declare the associated ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, verify bus power availability at least once per 12 hours or declare the associated ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the HPCS system inoperable.

ACTION 34 - 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 35 - 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.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	≥ 2.5 inches	≥ 2.5 inches above top fuel guide
b. Reactor Steam Dome Pressure - Low (B21-PS-N021A,B,C,D)	410 ± 15 psig	410 ± 16 psig
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	≤ 2 psig	≤ 2 psig
d. Time Delay Relay	14 ≤ t ≤ 16 secs	14 ≤ t ≤ 16 secs
e. Bus Power Monitor (E21-K1A,B)	NA	NA
2. LPCI MODE OF RHR SYSTEM		
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	≤ 2 psig	≤ 2 psig
b. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	≥ 2.5 inches	≥ 2.5 inches above top fuel guide
c. Reactor Vessel Shroud Level (B21-LITS-N036 and B21-LITS-N037)	≥ 39" below TAF*	≥ 53" below top fuel guide
d. Reactor Steam Dome Pressure - Low (B21-PS-N021A,B,C,D)		
1. RHR Pump Start and LCPI Valve Actuation	410 ± 15 psig	410 ± 15 psig
2. Recirculation Pump Discharge Valve Actuation	310 ± 15 psig	310 ± 15 psig
e. RHR Pump Start - Time Delay Relay	9 ≤ t ≤ 11 seconds	9 ≤ t ≤ 11 seconds
f. Bus Power Monitor (E11-K106A,B)	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
3. HPCI SYSTEM		
a. Reactor Vessel Water Level - Low, Level #2 (B21-LIS-N031A,B,C,D)	≥ 112 inches	≥ 112 inches
b. Drywell Pressure-High (E11-PS-N011A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Condensate Storage Tank Level - Low (E41-LS-N002, E41-LS-N003)	≥ 23'4"	≥ 23'4"
d. Suppression Chamber Water Level - High* (E41-LSH-N015A,B,)	≤ -2 feet	≤ -2 feet
e. Bus Power Monitor (E41-K55 and E41-K56)	NA	NA
4. ADS		
a. Drywell Pressure-High (E11-PS-N010A,B,C,D)	≤ 2 psig	≤ 2 psig
b. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	≥ 2.5 inches	≥ 2.5 inches
c. Reactor Vessel Water Level - Low Level #1 (B21-LIS-N042 A, B)	≥ + 162.5 inches	≥ + 162.5 inches
d. ADS Timer (B21-TDPU-K5A,B)	≤ 120 seconds	≤ 120 seconds
e. Core Spray Pump Discharge Pressure - High (E21-PS-N008A,B and E21-PS-N009A,B)	≥ 100 psig	≥ 100 psig
f. RHR (LPCI MODE) Pump Discharge Pressure - High (E11-PS-N016A,B,C,D and E11-PS-N020A,B,C,D)	≥ 100 psig	≥ 100 psig
g. Bus Power Monitor (B21-K1A,B)	NA	NA

*Suppression chamber water level zero is the torus centerline minus 1 inch.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)*	a. 4.16 kv Basis - 2940 ± 161 volts b. 120 v Basis - 84 ± 4.6 volts c. ≤ 10 sec. time delay	2940 ± 315 volts 84 ± 9 volts ≤ 10 sec. time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3727 ± 9 volts b. 120 v Basis - 106.5 ± 0.25 volts c. 10 ± 0.5 sec. time delay	3727 ± 21 volts 106.5 ± 0.60 volts 10 ± 1.0 sec. time delay

*This is an inverse time delay voltage relay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 4.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	D	M	R	1, 2, 3, 4, 5
b. Reactor Steam Dome Pressure - Low (B21-PS-N021A,B,C,D)	NA	M	Q	1, 2, 3, 4, 5
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	NA	M	Q	1, 2, 3
d. Time Delay Relay	NA	R	R	1, 2, 3, 4, 5
e. Bus Power Monitor (E21-K1A,B)	NA	R	NA	1, 2, 3, 4, 5
2. LPCI MODE OF RHR SYSTEM				
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	NA	M	Q	1, 2, 3
b. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	D	M	R	1, 2, 3, 4*, 5*
c. Reactor Vessel Shroud Level B21-LITS-N036 and B21-LITS-N037)	NA	M	Q	1, 2, 3, 4*, 5*
d. Reactor Steam Dome Pressure - Low (B21-PS-N021A,B,C,D)				
1. RHR Pump Start and LPCI Injection Valve Actuation	NA	M	Q	1, 2, 3, 4*, 5*
2. Recirculation Loop Pump Discharge Valve Actuation	NA	M	Q	1, 2, 3, 4*, 5*
e. RHR Pump Start-Time Delay Relay	NA	R	R	1, 2, 3, 4*, 5*
f. Bus Power Monitor (E11-K106A,B)	NA	R	NA	1, 2, 3, 4*, 5*

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

TABLE 4.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. HPCI SYSTEM				
a. Reactor Vessel Water Level - Low Level #2 (B21-LIS-N031A,B,C,D)	D	M	R	1, 2, 3
b. Drywell Pressure-High (E11-PS-N011A,B,C,D)	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low (E41-LS-N002, E41-LS-N003)	NA	M	Q	1, 2, 3
d. Suppression Chamber Water Level - High (E41-LSH-N015A,B)	NA	M	Q	1, 2, 3
e. Bus Power Monitor (E41-K55 and E41-K56)	NA	R	NA	1, 2, 3
4. ADS				
a. Drywell Pressure-High (E11-PS-N010A,B,C,D)	NA	M	Q	1, 2, 3
b. Reactor Vessel Water Level - Low, Level #3 (B21-LIS-N031A,B,C,D)	D	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Level #1 (B21-LIS-No. 42, A, B)	D	M	R	1, 2, 3
d. ADS Timer (B21-TDPU-K5A,B)	NA	R	R	1, 2, 3
e. Core Spray Pump Discharge Pressure - High (E21-PS-N008A,B and E21-PS-N009A,B)	NA	M	Q	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High (E11-PS-N016A,B,C,D and E11-PS-N020A,B,C,D)	NA	M	Q	1, 2, 3
g. Bus Power Monitor (B21-K1A,B)	NA	R	NA	1, 2, 3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 38 TO FACILITY LICENSE NO. DPR-71

AND

AMENDMENT NO. 60 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 Introduction

By letter dated June 11, 1981 Carolina Power and Light Company (the licensee) forwarded proposed changes to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP) Unit Nos. 1 and 2. The proposed changes to the BSEP Unit No. 1 Technical Specifications (1) establish revised vessel level setpoints that are consistent with a new common instrument zero level, (2) include the previously omitted Low Level #1 instrument into the Automatic Depressurization System (ADS) instrument tables, and (3) incorporate instrument number designation changes resulting from replacement of certain digital instrumentation with analog loops.

The proposed changes to the BSEP Unit No. 2 Technical Specifications include the Low Level #1 instrument in the ADS tables. (The revised vessel level setpoints were previously included in the BSEP Unit No. 2 Technical Specifications by License Amendment No. 56.) Additional changes proposed by the staff that are acceptable to the licensee have also been included in the Unit No. 2 amendment. These changes correct the listed Reactor Vessel Shroud Level instrumentation setpoints and change level setpoint terminology to be consistent with Unit No. 1 terminology.

2.0 Evaluation

2.1 Revised Vessel Level Setpoints for BSEP Unit No. 1

The proposed Technical Specification change establishes revised vessel level setpoints that are consistent with a new common instrument zero level. The proposed common instrument level is 367" above the vessel bottom. This is identical to the common level already implemented for BSEP Unit No. 2. A common zero level for all reactor vessel level instrumentation is called for in TMI Action Item II.K.3.27 in NUREG-0737.

We have reviewed each of the proposed revised setpoints and find them to be consistent with the previously established safety settings. We have considered the potential for operator error given the change in level setpoints for Unit 1. Since the proposed revisions will bring the Unit No. 1 setpoints into conformance with the setpoints of Unit No. 2, and since the licensee has committed, by letter dated March 18, 1981, to train all operators on the new level setpoints, we find that the proposed revised setpoints do not create an increased potential for operator error, and are acceptable from a human engineering standpoint. In addition, the proposed Technical Specification revisions do not involve a change in actual water level for any function and no instrumentation is being changed.

For these reasons we find the proposed vessel level setpoint Technical Specification revisions acceptable for use.

2.2 Inclusion of Low Level #1 Instrument for BSEP Unit Nos. 1 and 2

Through administrative oversight the Low Level #1 instrument (B21--N042--) has not previously been included in the Automatic Depressurization System (ADS) instrumentation tables. This instrument does provide an input signal to ADS and, therefore, it should be included in the ADS instrumentation tables.

2.3 Instrument Number Designation Changes for Unit No. 1

On November 19, 1979 the licensee submitted a request for Technical Specification changes to replace certain digital instrumentation with analog equipment. In response, the staff issued License Amendment Nos. 26 and 50 on March 14, 1980. The accompanying Safety Evaluation Report (SER) found the proposed modifications to be technically acceptable and established a procedure whereby the licensee would periodically submit administrative Technical Specification changes to revise instrument designations for those systems that had been modified. The licensee is proposing certain instrument designation Technical Specification changes in accordance with the previously issued SER. Therefore, we find the proposed designation changes to be acceptable.

2.4 Reactor Vessel Shroud Level Instrumentation Setpoint for BSEP Unit No. 2

License Amendment No. 56 for BSEP Unit No. 2 revised the Technical Specifications to reflect new vessel level setpoints as required by TMI Action Item II.K.3.27 of NUREG-0737. The Reactor Vessel Shroud Level instrumentation setpoints were unintentionally omitted from that amendment. Therefore, the staff has requested, and the licensee has agreed, that the subject setpoints should be changed to reflect the common reference level.

2.5 Level Setpoint Terminology Change for BSEP Unit No. 2

The Technical Specification changes issued under License Amendment No. 56 were worded such that the setpoints were based on the number of inches above or below instrument zero (e.g. Reactor Vessel Low Water Level is +162.5 inches above instrument zero). The proposed Technical Specifications accompanying the licensee's corresponding submittal for Unit 1 were worded such that the setpoints were based on the number of inches above or below the top fuel guide (e.g. Reactor Vessel Low Water level is +162.5 inches above top fuel guide). Since the referenced top fuel guide and instrument zero heights are both 367" above the vessel bottom, there is no technical question involved. However, the staff has requested, and the licensee has agreed that, to maintain conformity, the Unit No. 2 Technical Specifications should be revised to state "inches above top fuel guide," where appropriate.

3.0 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 25, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-325 AND 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 38 and 60 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the Technical Specifications for operation of the Brunswick Steam Electric Plant, Units Nos. 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendment for Brunswick Unit No. 1 revises the Technical Specifications to (1) establish new vessel level setpoints that are consistent with a common instrument zero level, (2) incorporate the Low Level #1 instrument into the Automatic Depressurization System (ADS) instrumentation tables, and (3) reflect instrument number designation changes resulting from the replacement of certain digital instrumentation with analog loops. The Amendment for Brunswick Unit No. 2 revises the Technical Specifications to (1) incorporate the Low Level #1 instrument into the ADS instrumentation tables, (2) include a level instrument inadvertently omitted from a previous amendment, and (3) standardize instrumentation table terminology.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendments dated June 11, 1981, (2) Amendment Nos. 38 and 60 to License Nos. DPR-71 and DPR-62, and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C. and at the Southport-Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 25th day of June 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing