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Mr. J. A. Jones
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. 33 to Facility Operating License No. DPR-71 and Amendment No. 54 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 November 5, 1980.

The amendments revise the Technical Specifications to provide a one-time extension of certain surveillance intervals to allow the required testing to be performed during a Brunswick Unit 1 outage scheduled for Spring 1981. The request for extension of the 6 month interval for snubber visual inspections was not adequately justified on a risk assessment basis and could not be granted. This issue was discussed with members of your staff who agreed that adequate justification for extending the snubber visual inspection interval could not be developed in sufficient time to support the request. The remaining tests are on 18-month intervals and are normally performed during a refueling outage. Small extensions for these tests were justified and are hereby granted.

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

Sincerely,

Original Signed by
T. A. Ippolito

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

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

cc w/encl: See next page

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Reviewed as to form of amendment & notice only

CP 1

| OFFICE | DL:ORB#2 | DL:ORB#2 | DL:ORB#2 | DL:OR | OELD |
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Docket

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

December 23, 1980

Docket Nos. 50-325
and 50-324

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

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T A Ippolito

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

Enclosures:
1. Amendment No. 33 to DPR-71
2. Amendment No. 54 to DPR-62
3. Safety Evaluation
4. Notice

cc w/encl: See next page

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Office of Radiation Programs (ANR-460)
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Washington, D. C. 20460

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Southport, North Carolina 28461

Mr. Fred Tollison
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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. 33
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 November 5, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

   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 amendmed to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 33 , 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

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 23, 1980

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 33</u>

<u>FACILITY OPERATING LICENSE NO. DPR-71</u>

<u>DOCKET NO. 50-325</u>

Remove the following pages and replace with identically numbered pages.

<u>3/4 1-7</u> / <u>3/4 1-8</u>
3/4 1-19 / 3/4 1-20
<u>3/4 3-9</u> / <u>3/4 3-10</u>
<u>3/4 3-29</u> / 3/4 3-30
3/4 3-37 / <u>3/4 3-38</u>
3/4 3-49 / <u>3/4 3-50</u>
<u>3/5 5-7</u> / 3/4 5-8
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3/4 8-9 / 3/4 8-10

The underlined pages are overleaf pages and are provided for convenience.

## FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

| Position Inserted From Fully Withdrawn | Average Scram Insertion Time |
|---|---|
| 46 | 0.398 seconds |
| 36 | 0.954 seconds |
| 26 | 2.120 seconds |
| 6 | 3.800 seconds |

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With the average scram insertion times of the control rods exceeding the above limits, operation may continue and the provisions of Specification 3.0.4 are not applicable provided:

a. The control rods with the slower than average scram insertion times are declared inoperable,

b. The requirements of Specification 3.1.3.1 are satisfied, and

c. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 92 days when operation is continued with three or more control rods with slow scram insertion times;

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be scram time tested from the fully withdrawn position as required by Specification 4.1.3.2.

## CONTROL ROD SCRAM ACCUMULATORS

### LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2 and 5*.

ACTION:

a. In CONDITION 1 or 2 with one control rod scram accumulator inoperable, the provisions of Specification 3.0.4 are not applicable and operation may continue, provided that within 8 hours:

   1. The inoperable accumulator is restored to OPERABLE status, or

   2. The control rod associated with the inoperable accumulator is declared inoperable, and the requirements of Specification 3.1.3.1 are satisfied.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In CONDITION 5* with a withdrawn control rod scram accumulator inoperable, fully insert the affected control rod and electrically disarm the directional control valves within one hour. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.3.5 The control rod scram accumulators shall be determined OPERABLE:

a. At least once per 7 days by verifying that the pressure and leak detectors are not in the alarmed condition, and

b.** At least once per 18 months by performance of a:

   1. CHANNEL FUNCTIONAL TEST of the leak detectors (C11-LS-129-xxxx), and

   2. CHANNEL CALIBRATION of the pressure detectors (C11-PS-130-xxxx).

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**For the performance of this surveillance scheduled to be completed by February 10, 1981, a onetime-only exemption is allowed to extend this surveillance until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

## SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The volume and temperature of the sodium pentaborate solution are within the limits of Figures 3.1.5-1 and 3.1.5-2, and

2. The heat tracing circuit is OPERABLE.

b. At least once per 31 days by:

1. Starting each pump and recirculating demineralized water to the test tank,

2. Verifying the continuity of the explosive charge, and

3. Determining the concentration of boron in solution by chemical analysis. This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit established in Figure 3.1.5-2.

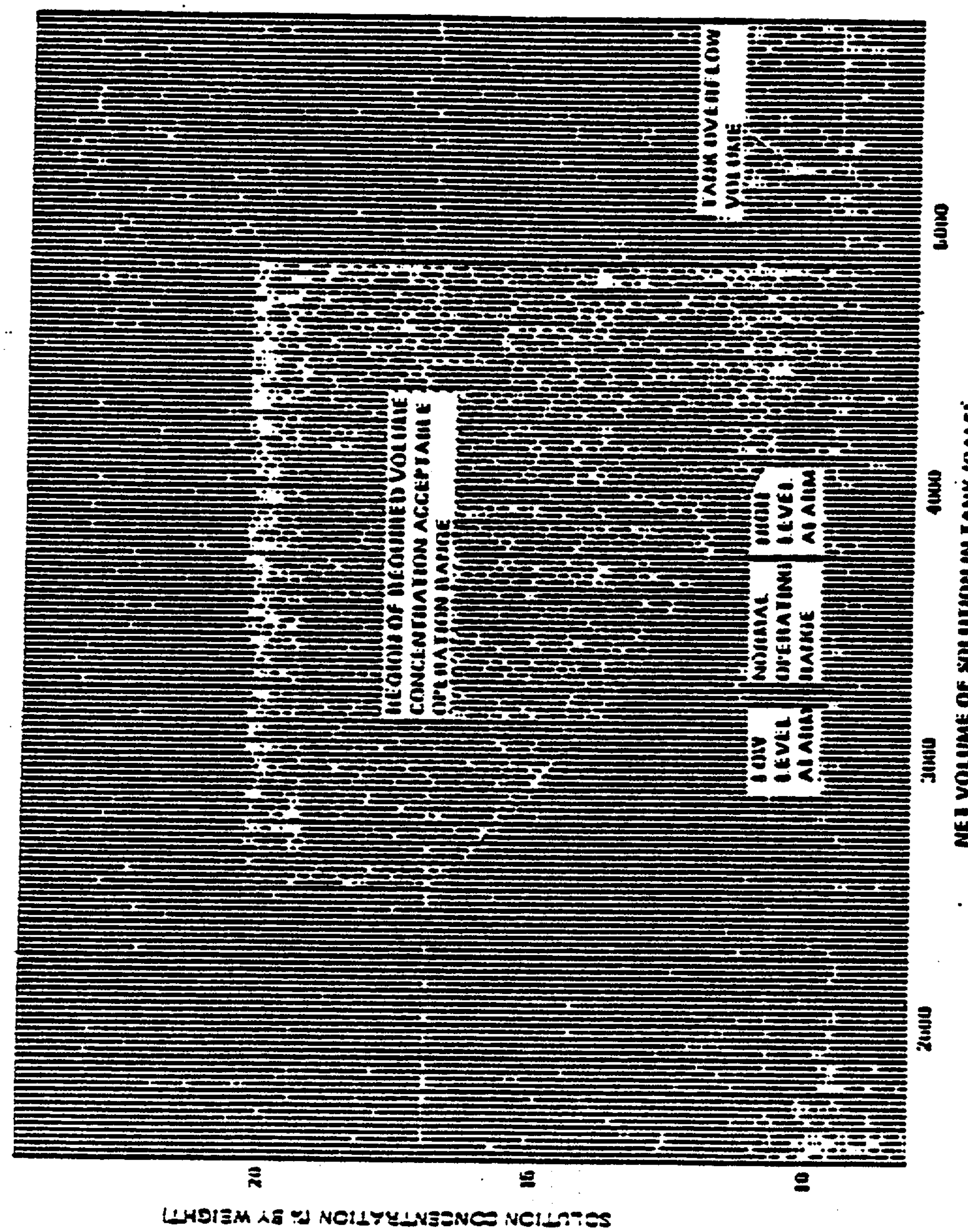
c. At least once per 18 months during shutdown by;

1.* Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection test loops shall be tested in 36 months.

2.* Demonstrating that the minimum flow requirement of 41.2 gpm at a pressure of 1190 psig is met.

3. Demonstrating that the pump relief valve setpoint is 1400 ± 50 psig.

*For the performance of this surveillance scheduled to be completed by February 25, 1981, a onetime-only exemption is allowed to extend this surveillance until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.




SODIUM PENTABORATE SOLUTION VOLUME CONCENTRATION REQUIREMENTS

Figure 3.1.6-1

## 3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

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

c. 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 3.3.2-1.

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

### SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation 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 4.3.2-1.

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

*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 would cause the Trip Function to occur.



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SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation 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 logic train such that both logic chains 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 isolation function.

*For the performance of this surveillance on item 1-e of Table 3.3.2-3, scheduled to be completed by February 6, 1981, a onetime-only exemption is allowed to extend this surveillance until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

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-LIS-N017A,B,C,D) | D | M | R | 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.

## 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 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.

**For the channel functional test scheduled to be completed by February 25, 1981, 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 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. ADS Timer (B21-TDPU-K5A,B) | NA | R | R | 1, 2, 3 |
| d. Core Spray Pump Discharge Pressure - High (E21-PS-N008A,B and E21-PS-N009A,B) | NA | M | Q | 1, 2, 3 |
| e. 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 |
| f. Bus Power Monitor (B21-K1A,B) | NA | R | NA | 1, 2, 3 |

## TABLE 4.3.5.2-1

## REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT AND INSTRUMENT NUMBER | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|
| 1. Reactor Vessel Pressure (C32-PI-3332 and C32-PT-3332) | M | Q |
| 2. Reactor Vessel Water Level (B21-LI-3331, B21-LI-R604AX, B21-LT-3331, B21-LT-N026A, B21-LT-N017D-3 and B21-LSH-N017D-3) | M | Q |
| 3. Suppression Chamber Water Level (CAC-LI-3342 and CAC-LT-3342) | M | R |
| 4. Suppression Chamber Water Temperature (CAC-TR-778-7) | M | R* |
| 5. Drywell Pressure (CAC-PI-3341 and CAC-PT-3341) | M | Q |
| 6. Drywell Temperature (CAC-TR-778-1,3,4) | M | R* |
| 7. Drywell Oxygen Concentration (CAC-AT-1259-2) | M | Q |
| 8. Residual Heat Removal Head Spray Flow (E11-FT-3339 and E11-FI-3339) | M | Q |
| 9. Residual Heat Removal System Flow (E11-FT-3338, E11-FI-3338 and E11-FY-3338) | M | Q |
| 10. Residual Heat Removal Service Water Discharge Differential Pressure (E11-PDT-N002BX and E11-PDI-3344) | M | Q |

*For verifying this channel calibration scheduled for completion by February 14, 1981, a onetime-only exemption is allowed to extend this calibration until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

INSTRUMENTATION

POST-ACCIDENT MONITORING INSTRUMENTATION

## LIMITING CONDITION FOR OPERATION

3.3.5.3 The post-accident monitoring instrumentation channels shown in Table 3.3.5.3-1 shall be OPERABLE.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3.5.3-1, either restore the inoperable channels to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.3.5.3 Each of the above required post-accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.3-1.

EMERGENCY CORE COOLING SYSTEMS

LOW PRESSURE COOLANT INJECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.3.2 Two independent Low Pressure Coolant Injection (LPCI) subsystems of the residual heat removal system shall be OPERABLE with each subsystem comprised of:

a. Two pumps,

b. An OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: CONDITIONS 1, 2, 3, 4* and 5*.

ACTION:

a. In CONDITION 1, 2 or 3;

1. With one LPCI subsystem or one LPCI pump inoperable, POWER OPERATION may continue provided both CSS subsystems are OPERABLE; restore the inoperable LPCI subsystem or pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the LPCI system cross-tie valve open or power not removed from the valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

b. In CONDITION 4* or 5* with one or more LPCI subsystems inoperable, take the ACTION required by Specification 3.5.3.1. The provisions of Specification 3.0.3 are not applicable.

*Not applicable when two CSS subsystems are OPERABLE per Specification 3.5.3.1.

SURVEILLANCE REQUIREMENTS

4.5.3.2 Each LPCI subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water,

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

3. Verifying that the subsystem cross-tie valve is closed with power removed from the valve operator.

b. At least once per 92 days by verifying each pair of LPCI pumps discharging to a common header can be started from the control room and develops a total flow of at least 17,000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 20 psig.

c. At least once per 18 months* by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

*For the performance of this system functional test scheduled to be completed by February 25, 1981, 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.

## SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve specified in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the cycling test, and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months* by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve specified in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation system isolation valve shall be demonstrated OPERABLE at least once per 18 months by cycling each valve through at least one complete cycle of full travel.

*For the performance of this surveillance scheduled to be completed by February 25, 1981, a onetime-only exemption is allowed to extend this surveillance until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

## TABLE 3.6.3-1

## PRIMARY CONTAINMENT ISOLATION VALVES

| VALVE FUNCTION | VALVE GROUP[1] | ISOLATION TIME (Seconds) |
|---|---|---|
| Main steamline isolation valves<br>B21-F022 A, B, C, D<br>B21-F028 A, B, C, D | 1 | $3 \leq t \leq 5$ |
| Main steamline drain isolation valves<br>B21-F016<br>B21-F019 | 1 | 30 |
| Reactor Water sample line isolation valves<br>B32-F019<br>B32-F020 | 1 | 5 |
| Drywell equipment drain discharge isolation valves<br>G16-F019<br>G16-F020 | 2 | 20 |
| Drywell floor drain discharge isolation valves<br>G16-F003<br>G16-F004 | 2 | 20 |
| TIP guide tube (Ball valve) | 2 | |
| Reactor water cleanup system isolation valves<br>G31-F001<br>G31-F004 | 3* | 35 |

1. See Specification 3.3.2. Table 3.3.2-1 for isolation signal that operates each valve group.

*Inboard isolation valve does not close on standby liquid control system initiation or reactor water cleanup system high temperature.

## SURVEILLANCE REQUIREMENTS

4.6.4.1 Each drywell-suppression pool vacuum breaker shall be demonstrated OPERABLE:

a. At least once per 31 days and after any discharge of steam to the suppression pool from any source, by exercising each vacuum breaker through one complete cycle and verifying that each vacuum breaker is closed as indicated by the position indication system.

b. Whenever a vacuum breaker is in the open position, as indicated by the position indication system, by conducting a test that verifies that the differential pressure is maintained > 1/2 the initial $\Delta P$ for one hour without $N_2$ makeup.

c. At least once per 18 months during shutdown by;

1.* Verifying the opening setpoint, from the closed position, to be $\leq$ 0.5 psid,

2.* Performance of a CHANNEL CALIBRATION that each position indicator indicates the vacuum breaker to be open if the vacuum breaker does not satisfy the $\Delta P$ test in 4.6.4.1.b, and

3. Conducting a leak test at an initial differential pressure of 1 psig and verifying that the differential pressure does not decrease by more than 0.25 inches of water per minute for a 10 minute period.

*For the verifying of the opening setpoint and the performance of the channel calibration scheduled for completion by January 31, 1981, a onetime-only exemption is allowed to extend these inspections until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

## SUPPRESSION POOL - REACTOR BUILDING VACUUM BREAKERS

### LIMITING CONDITION FOR OPERATION

3.6.4.2 All suppression pool-Reactor Building vacuum breakers shall be OPERABLE with an opening setpoint of $\leq$ 0.5 psid.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With one suppression pool - Reactor Building vacuum breaker inoperable for opening but known to be in the closed position, restore the inoperable vacuum breaker to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

### SURVEILLANCE REQUIREMENTS

4.6.4.2 Each suppression pool-Reactor Building vacuum breaker shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   1. Manually verifying that each vacuum breaker check valve is free to open, and

   2. Cycling each vacuum breaker butterfly valve through at least one complete cycle of full travel.

b. At least once per 18 months by:

   1. Demonstrating that the force required to open each vacuum breaker check valve does not exceed 0.5 psid.

   2. Demonstrating that the vacuum breaker butterfly valve opens at 0.45 $\pm$ 0.05 psid, drywell pressure going negative relative to Reactor Building pressure.

   3. Visual inspection.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment automtic isolation damper specified in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each automatic isolation damper testable during plant operation through at least one complete cycle of full travel.

b. Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by performance of the cycling test and verification of isolation time.

c. At least once per 18 months during COLD SHUTDOWN or REFUELING by:

1. Cycling each automatic damper through at least one complete cycle of full travel and measuring the isolation time, and

2.* Verifying that on a secondary containment isolation test signal each automatic damper actuates to its isolation position.

*For this verification scheduled to be completed by February 25, 1981, a onetime-only exemption is allowed to extend this verification until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

| DAMPER FUNCTION | ISOLATION TIME (Seconds) |
|---|---|
| 1. Reactor Building Ventilation Supply Isolation Dampers | 4 |
| 2. Reactor Building Ventilation Exhaust Isolation Dampers | 4 |

## CONTAINMENT SYSTEMS

## 3/4.6.6 CONTAINMENT ATMOSPHERE CONTROL

### STANDBY GAS TREATMENT SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Standby Gas Treatment System subsystems shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2, 3, 5 and *.

ACTION:

a. With one standby gas treatment subsystem inoperable:

   1. In CONDITION 1, 2 or 3, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In CONDITION 5 or *, restore the inoperable subsystem to OPERABLE status within 31 days or suspend irradiated fuel handling in the secondary containment, CORE ALTERATIONS or operations that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable;

   1. In CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

   2. In CONDITION 5 or *, suspend all irradiated fuel handling in the secondary containment, CORE ALTERATIONS or operations that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.6.6.1 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

   a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on automatic control.

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a., C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 3000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

3. Verifying a system flow rate of 3000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.53, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 8.5 inches Water Gauge while operating the filter train at a flow rate of 3000 cfm ± 10%.

2.* Verifying that the filter train starts on each secondary containment isolation test signal.

3. Verifying that the heaters will dissipate at least 15.2 kw when tested in accordance with ANSI N510-1975.

*For the performance of this surveillance scheduled to be completed by February 25, 1981, a onetime-only exemption is allowed to extend this surveillance until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3000 cfm $\pm$ 10%

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3000 cfm $\pm$ 10%.

## CONTAINMENT ATMOSPHERE DILUTION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.6.2 The containment atmosphere dilution (CAD) system shall be OPERABLE with:

a. An OPERABLE flow path capable of supplying nitrogen to the drywell, and

b. A minimum supply of 4350 gallons of liquid nitrogen.

APPLICABILITY: CONDITION 1*.

ACTION:

With the CAD system inoperable, restore the CAD system to OPERABLE status within 31 days or be in at least STARTUP within the next 8 hours. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.6.6.2 The CAD system shall be demonstrated to be OPERABLE;

a. At least once per 31 days by verifying that:

1. The system contains a minimum of 4350 gallons of liquid nitrogen, and

2. Each valve (manual, power operated or automatic) in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by:

1. Cycling each power operated (excluding automatic) valve in the flow path not testable during plant operation through at least one complete cycle of full travel, and

2.** Verifying that each automatic valve in the flow path actuates to its correct position on a Group 2 and 6 isolation test signal.

* When oxygen concentration is required to be < 4% per Specification 3.6.6.3.

**For the performance of this surveillance scheduled to be completed by February 25, 1981, a onetime-only exemption is allowed to extend this surveillance until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

## 3/4.7.5 HYDRAULIC SNUBBERS

### LIMITING CONDITION FOR OPERATION

3.7.5 All hydraulic snubbers shown in Table 3.7.5-1 shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2 and 3

ACTION:

With one or more hydraulic snubbers inoperable, restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

### SURVEILLANCE REQUIREMENTS

12-12-77

4.7.5.1 Hydraulic snubbers shown in Table 3.7.5-1 shall be demonstrated OPERABLE by the performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.7.5.2 Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7.5-1 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7.5-1 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

4.7.5.3 Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.5.4 At least once per 18 months* during shutdown, a representative sample of at least 10 snubbers or at least 10% of all snubbers listed in Table 3.7.5-1, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 pound capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis except snubbers identified in Table 3.7.5-1 as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

*For the inaccessible snubber functional inspection interval scheduled to end December 30, 1980, a onetime-only exemption is allowed to extend this inspection interval until "before the completion of the Spring 1981 outage," scheduled to commence in March, 1981.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the fuel level in the day fuel tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the day tank to the engine mounted tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least 514 rpm in $\leq$ 10 seconds,

5. Verifying the generator is synchronized, loaded to $\geq$ 1750 kw, and operates for $\geq$ 15 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

b. At least once per 31 days by verifying the fuel level in the plant fuel storage tank.

c. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM-D975-74 when checked for viscosity, water and sediment,

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2.* Verifying the generator capability to reject a load equal to one core spray pump without tripping,

3.* Simulating a loss of offsite power in conjunction with an emergency core cooling system test signal, and:

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected loads through the load sequence relays and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

*For the verification of this item scheduled for completion by February 20, 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.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that on the emergency core cooling system test signal, all diesel generator trips except engine overspeed, generator differential, low lube oil pressure, reverse power, loss of field and phase overcurrent with voltage restraint, are automatically bypassed.

5. Verifying the diesel generator operates for ≥ 60 minutes while loaded to ≥ 3500 kw.

6.* Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3850 kw.

7.* Verifying that the automatic load sequence relays are OPERABLE with each load sequence time within 10% of the required value.

*For the verification of this item scheduled for completion February 20, 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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 ≥ 1042.42 amperes,
2) Battery 1A-2 ≥ 1211.90 amperes,
3) Battery 1B-1 ≥ 1089.06 amperes, and
4) Battery 1B-2 ≥ 1028.67 amperes.

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

1) Battery 1A-1 ≥ 229.19 amperes,
2) Battery 1A-2 ≥ 159.10 amperes,
3) Battery 1B-1 ≥ 176.79 amperes, and
4) Battery 1B-2 ≥ 202.67 amperes.

c) During the remainder of the 8 hour test;

1) Battery 1A-1 ≥ 75.52 amperes,
2) Battery 1A-2 ≥ 50.34 amperes,
3) Battery 1B-1 ≥ 53.39 amperes, and
4) Battery 1B-2 ≥ 61.09 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 performed subsequent to the satisfactory completion of the required battery service test and after normal equalizer charge.

*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.

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. 54
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 November 5, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

   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. 54 , 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

T A Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 23, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Remove the following pages and replace with identically numbered pages

3/4 8-3 / 3/4 8-4

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the fuel level in the day fuel tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the day tank to the engine mounted tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least 514 rpm in $\leq$ 10 seconds,

5. Verifying the generator is synchronized, loaded to $\geq$ 1750 kw, and operates for $\geq$ 15 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

b. At least once per 31 days by verifying the fuel level in the plant fuel storage tank.

c. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment,

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load equal to one core spray pump without tripping,

3. Simulating a loss of offsite power in conjunction with an emergency core cooling system test signal, and:

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b)* Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected loads through the load sequence relays and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

*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.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that on the emergency core cooling system test signal, all diesel generator trips except engine overspeed, generator differential, low lube oil pressure, reverse power, loss of field and phase overcurrent with voltage restraint, are automatically bypassed.

5. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 3500 kw.

6. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3850 kw.

7.* Verifying that the automatic load sequence relays are OPERABLE with each load sequence time within 10% of the required value.

*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.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# SUPPORTING AMENDMENT NO. 33 TO FACILITY LICENSE NO. DPR-71 AND

# AMENDMENT NO. 54 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

## I. INTRODUCTION

By letter dated November 5, 1980, Carolina Power & Light Company (licensee) requested an amendment to the Brunswick Unit Nos. 1 and 2 Operating Licenses. The proposed amendment would extend certain surveillance intervals to allow required testing to be performed during a scheduled Brunswick Unit 1 shutdown. The tests involved included 18-month surveillance normally performed during a refueling outage and a visual snubber inspection required on a 6-month interval.

## II. BACKGROUND AND DISCUSSION

The licensee's request for an extension of certain surveillance intervals was submitted as a result of extended outages of both Brunswick units during spring and summer 1980 and the subsequent cancellation of the maintenance outage on Brunswick Unit 1 scheduled for September 1980. The licensee has determined that the optimum alternative is to complete all of the required surveillance during a single Unit 1 outage scheduled in March 1981.

The request has been categorized into three areas:

1. an extension of between 3% to 7% of the original interval for 18-month surveillance on engineered safety features equipment;

2. an extension of about 17% of the original interval for inaccessible snubber functional testing; and

3. an extension of about 50% of the original interval for inaccessible snubber visual inspections.

The licensee supplied information to show that the systems involved have a sustained record of satisfactory performance during initial testing and subsequent to the end of the previous test intervals. The licensee has concluded that the requested delays will not affect the safe performance of these systems.

## III. EVALUATION

The staff's requirement for 18-month surveillance intervals was set with the nominal refueling outage in mind. The refueling test frequency was intended to routinely demonstrate operability of systems over the service life of the plant. To judge the acceptability of extending the required surveillance intervals, we have taken into account the systems involved, operating history, and amount of extension requested.

The systems involved are: (1) inaccessible safety related snubbers, (2) Drywell-Torus Vaccum Breakers,(3) Main Condenser Low Vaccum, (4) Unit 1 Diesel Generators, (5) Torus-Drywell CAC Temperature Monitoring, (6) LPCI subsystem, (7) Unit 2 Diesel Generator Load Test, (8) 125 Volt Battery, (9) Core Spray System, (10) Primary & Secondary Containment System, (11) Standby Liquid Control System, and (12) CRD Accumulator Leak Detection Instrumentation. With the exception of item (7) above, all of the systems are associated with Unit 1. However, all of the above tests must be conducted with Unit 1 shutdown, including item (7).

Only two of the above systems have experienced a previous test failure. The 125 Volt Battery (LER 1-79-09) had failed the capacity test in March 1979. A preventive maintenance program was implemented and the battery was tested satisfactorily in April 1979. The most recent inaccessible snubber visual inspection found one inoperable snubber (LER 1-80-34). The two previous inspections found four and three inoperable snubbers, respectively. Thus the number of inoperable snubbers found is decreasing as the visual inspection intervals are lengthened.

On the basis of the systems involved and their operating history, a one-time extension in the surveillance intervals appears to be justifiable.

To judge the acceptability of the requested extension we performed a calculation to assess the potential impact on system unavailability (the fraction of reactor operating time during which a system is in an undetected failed state). We found that granting a 7% increase in an 18-month surveillance interval could result in a 3% increase in system unavailability.

We found that granting a 17% increase in an 18-month surveillance interval for inaccessible snubbers could result in a 10% increase in the likelihood of having an inoperable snubber identified during the next functional test. Similarly, granting a 50% increase in a 6-month surveillance interval could result in a 33% increase in likelihood of having an inoperable snubber found during the next visual inspection.

The category (1) extension requests result in a minor increase in unavailability and are acceptable. The category (2) extension request results in a slightly larger increase in unavailability, but is still judged to be acceptable for this one time. The category (3) extension request results in an unacceptable increase in unavailability and cannot be justified on a risk assessment basis. We have discussed these conclusions with the licensee in an effort to obtain further justification for the category (3) request. After some deliberation, the licensee agreed that adequate justification for extending the snubber visual inspection interval could not be developed in sufficient time to support the request. Therefore, we are denying the 6-month interval extension request. We have concluded that the 18-month interval extension requests are acceptable for this one-time and therefore, they can be granted.

## IV. 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.

## V. 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: December 23, 1980

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. 33 and 54 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, Unit Nos. 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to provide a one-time extension of certain surveillance intervals to allow the required testing to be performed during a Brunswick Unit 1 outage scheduled for Spring 1981.

The application for amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's 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 November 5, 1980, (2) Amendment Nos. 33 and 54 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 Licensing.

Dated at Bethesda, Maryland, this 23rd day of December, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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