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Docket
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BGrimes
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OI&E (5)
BJones (8)
BScharf (10)
DBrinkman
BHarless
ACRS (16)
RDiggs
BJRuchanan
TERA

Docket Nos. 50-325
and 50-324

APRIL 6 1979

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. 23 to Facility Operating License No. DPR-71 and Amendment No. 47 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 operating licenses and Technical Specifications in response to your applications dated December 29, 1978, February 19, 1979, and March 6, 1979. The December 29, 1978 reload license application was supplemented by letters dated January 17, 1979, March 16 and March 27, 1979. The March 6, 1979 fire protection application was supplemented by letters dated March 7, March 15, March 22, and March 29, 1979.

The ECCS reevaluation submitted for the reload fully meets the requirements of 10 CFR 50.46 and satisfies the conditions of our Order for Modification of License dated March 11, 1977.

The amendment for BSEP Unit No. 1 changes the Technical Specifications to establish revised safety and operating limits for operation in fuel Cycle No. 2.

The amendments for BSEP Unit Nos. 1 and 2 change the Technical Specifications to allow implementation of permanent modifications to the suppression pool-reactor building vacuum breaker lines. In addition, these amendments change the operating licenses for both units to allow revised implementation dates for certain modifications intended to improve the level of fire protection. To support these changes Supplement No. 1 to the Fire Protection Safety Evaluation report for this facility has been prepared.

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GD

OFFICE▶						
SURNAME▶						
DATE▶						

APRIL 6 1979

Copies of the Safety Evaluation, Supplement No. 1 to the Fire Protection Safety Evaluation, and the notice of issuance are also enclosed.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 23 to DPR-71
2. Amendment No. 47 to DPR-62
3. Safety Evaluation (see reports for encl.)
4. Supp. #1 to the Fire Protection
   SE for Brunswick Steam Electric
   Plant Units 1 & 2
5. Notice

cc w/enclosures:
See next page

ORB-2
T Wambach
TVW 4/6/79

Subject to changes agreed to by John Hannon

OFFICE	ORB #3	ORB #1	AD:E&P	OELD	ORB #3	
SURNAME	SSheppard	JHannon:mjf	BGrimes	S.H. Lewis	TIppolito	
DATE	4/4/79	4/4/79	4/4/79	4/5/79	4/4/79	

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

April 6, 1979

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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The amendments for BSEP Unit Nos. 1 and 2 change the Technical Specifications to allow implementation of permanent modifications to the suppression pool-reactor building vacuum breaker lines. In addition, these amendments change the operating licenses for both units to allow revised implementation dates for certain modifications intended to improve the level of fire protection. To support these changes Supplement No. 1 to the Fire Protection Safety Evaluation report for this facility has been prepared.

Copies of the Safety Evaluation, Supplement No. 1 to the Fire Protection Safety Evaluation, and the notice of issuance are also enclosed.

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 23 to DPR-71
2. Amendment No. 47 to DPR-62
3. Safety Evaluation
4. Supp. #1 to the Fire Protection SE for Brunswick Steam Electric Plant Units 1 & 2
5. Notice

cc w/enclosures:
See next page

cc: Richard E. Jones, Esquire
Carolina Power & Light Company
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Raleigh, North Carolina 27602

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Mr. Steve J. Varnam
Chairman, Board of County
Commissioners of Brunswick County
Southport, North Carolina 28461

Denny McGuire (Ms)
State Clearinghouse
Division of Policy Development
116 West Jones Street
Raleigh, North Carolina 27603

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

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

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

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. 23
License No. DPR-71

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

   A. The applications for amendments by Carolina Power & Light Company (the licensee) dated December 29, 1978, February 19, 1979, and March 6, 1979, as supplemented, comply 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 applications, 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 revising paragraph 2.B(6) to read as indicated below:

   2.B(6) - The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.35 of the NRC's Fire Protection Safety Evaluation Report on the Brunswick facility dated November 22, 1977 and supplements thereto. These modifications shall be completed by the dates identified in the Safety Evaluation Report or Table 3.1 in supplements thereto. In addition, the licensee

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may proceed with and is required to complete the modifications identified in Section B.2.1 of Supplement 1 to the Fire Protection Safety Evaluation Report, and any future supplements. These modifications shall be completed by the dates identified in Table B.2.1 of the supplement.

The license is further amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) is hereby amended to read as follows:

2.C(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 23, 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 #3
Division of Operating Reactors

Date of Issuance: April 6, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 23

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Remove	Insert
iv	iv
v	v
viii	viii
2-1	2-1
B 2-1	B 2-1
B 2-9	B 2-9
3/4 1-17	3/4 1-17
3/4 2-1 thru 2-8	3/4 2-1 thru 2-9
3/4 3-39	3/4 3-39
3/4 3-42	3/4 3-42
3/4 3-48	3/4 3-48
3/4 3-49	3/4 3-49
3/4 3-60	3/4 3-60
3/4 3-61	3/4 3-61
3/4 6-20	3/4 6-20
3/4 7-38	3/4 7-38
3/4 7-42 thru 7-43	3/4 7-42 thru 7-46
3/4 10-3	3/4 10-3
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1 thru 2-3	B 3/4 2-1 thru 2-3
B 3/4 2-6	B 3/4 2-6
5-1	5-1
5-4	5-4
6-5	6-5

Oversides included for convenience

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER (Low Pressure or Low Flow)

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER (High Pressure and High Flow)

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure $\leq$ 1325 psig within 2 hours.

2.1 SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: CONDITIONS 3, 4 and 5

ACTION:

With the reactor water level at or below the top of the active irradiated fuel, manually initiate the low pressure ECCS to restore the reactor vessel water level, after depressurizing the reactor vessel, if required.

2.1 SAFETY LIMITS

BASES

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MINIMUM CRITICAL POWER RATIO (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER (Low Pressure or Low Flow)

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 800 psia or core flows less than 10% of rated flow. Therefore the fuel cladding integrity limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 800 psia is conservative.

2.1.2 THERMAL POWER (High Pressure and High Flow)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power, result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), (GEXL), correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia
Mass Flux:	0.1 to 1.25 10^6 lb/hr-ft^2
Inlet Subcooling:	0 to 100 Btu/lb
Local Peaking:	1.61 at a corner rod to
	1.47 at an interior rod

Reference

1. "General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design application," NEDO-10958 and NEDE-10958.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. Range 10 allows the IRM instruments to remain on scale at higher power levels to provide for additional overlap and also permits calibration at these higher powers.

The most significant source of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed, Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM's are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above 1.07. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. This margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup, is not much colder than that already in the system, temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all

2. Average Power Range Monitor (Continued)

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

The APRM flow biased trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation; i.e., the thermal power of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer. Analyses demonstrate that with only the 120% trip setting, none of the abnormal operational transients analyzed violates the fuel safety limit and there is substantial margin from fuel damage. Therefore the use of the flow referenced trip setpoint, with the 120% fixed setpoint as backup, provides adequate margins of safety.

The APRM trip setpoint was selected to provide adequate margin for the Safety Limits and yet allows operating margin that reduces the possibility of unnecessary shutdowns. The flow referenced trip setpoint must be adjusted by the specified formula in order to maintain these margins.

3. Reactor Vessel Steam Dome Pressure-High

High Pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating, will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: CONDITION 1, when THERMAL POWER is greater than the preset power level of the RWM and RSCS.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that either:

   1. The inoperable RBM channel is restored to OPERABLE status within 24 hours, or

   2. The redundant RBM is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable RBM is restored to OPERABLE status, and the inoperable RBM is restored to OPERABLE status within 7 days, or

   3. THERMAL POWER is limited such that MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

Otherwise, trip at least one rod block monitor channel.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and during the OPERATIONAL CONDITIONS specified in Table 4.3.4-1.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE with:

a. An OPERABLE flow path from the storage tank to the reactor core, containing two pumps and and two inline explosive injection valves,

b. The contained solution volume-concentration within the limits of Figure 3.1.5-1, and

c. The solution temperature above the limit of Figure 3.1.5-2.

APPLICABILITY: CONDITIONS 1, 2, and 5.

ACTION:

a. In CONDITION 1 or 2:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In CONDITION 5:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 31 days or suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

2. With the standby liquid control system inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

3. The provisions of Specification 3.0.3 are not applicable.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR's) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, or 3.2.1-4.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3 or 3.2.1-4, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

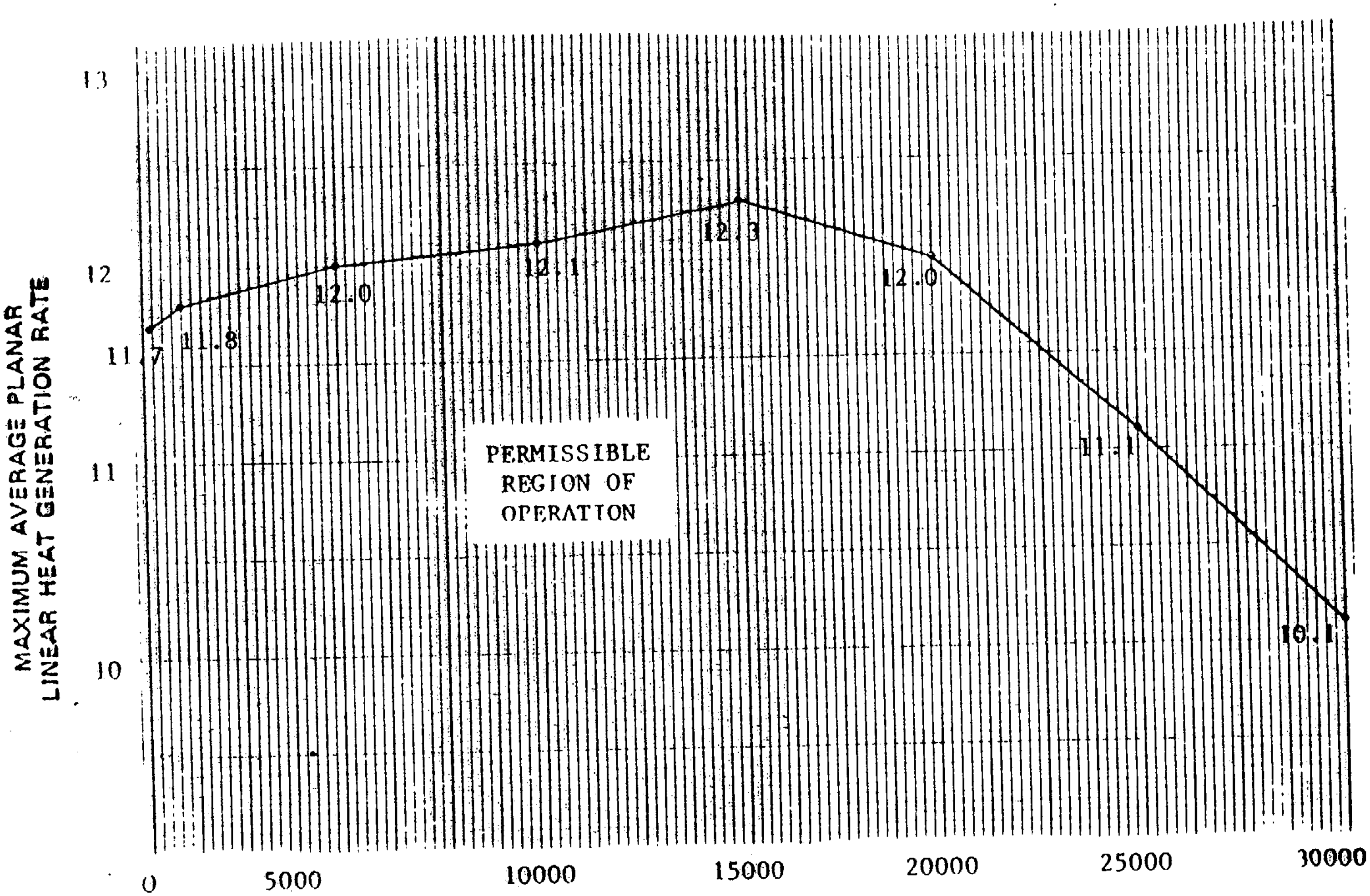
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGR's shall be verified to be equal to or less than the applicable limit determined from Figure 3.2.1-1, 3.2.1-2, 3.2.1-3 or 3.2.1-4:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

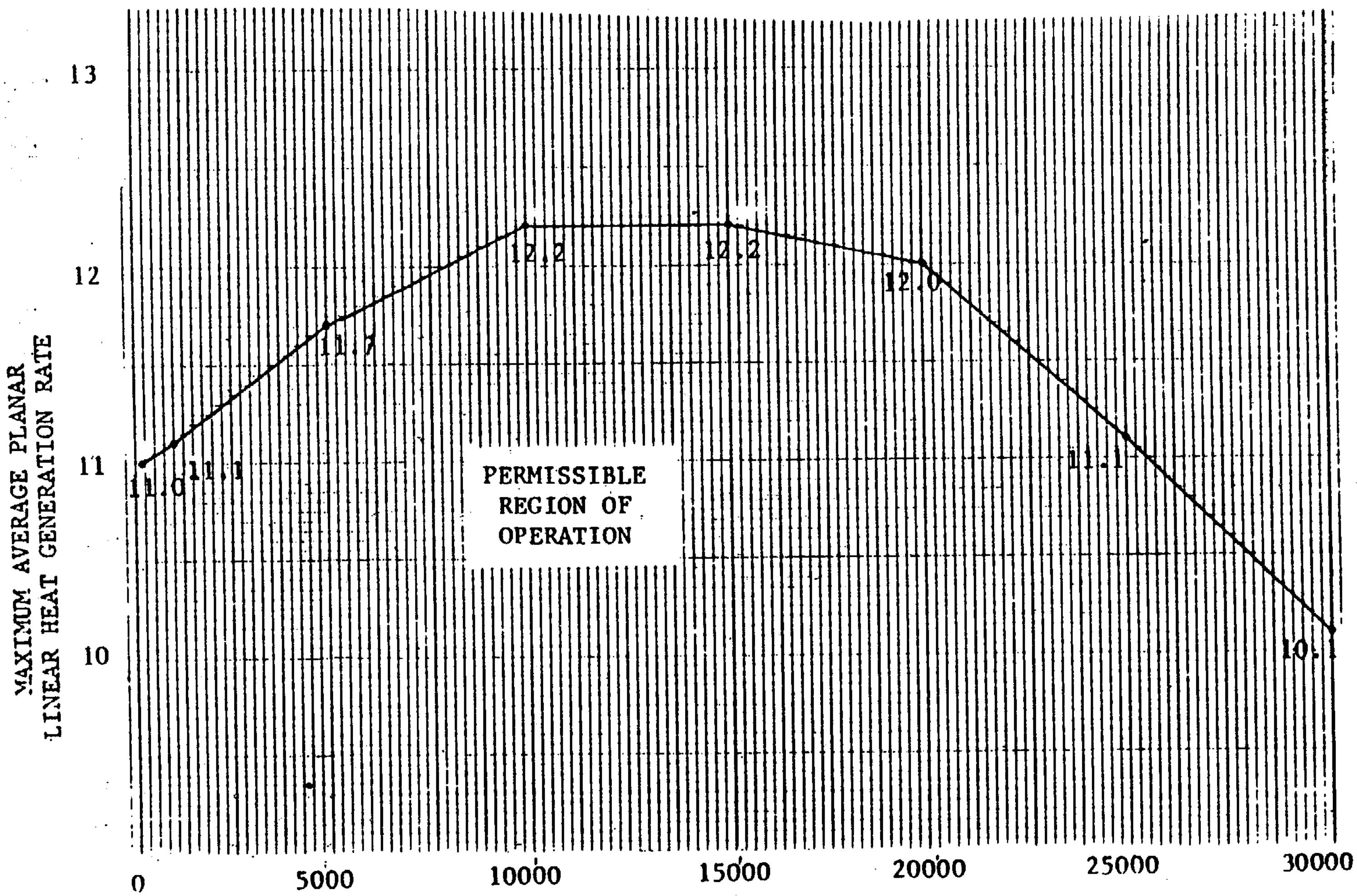
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



AVERAGE PLANAR EXPOSURE (MWd/t)
FUEL TYPE 1 (8 x 8)

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

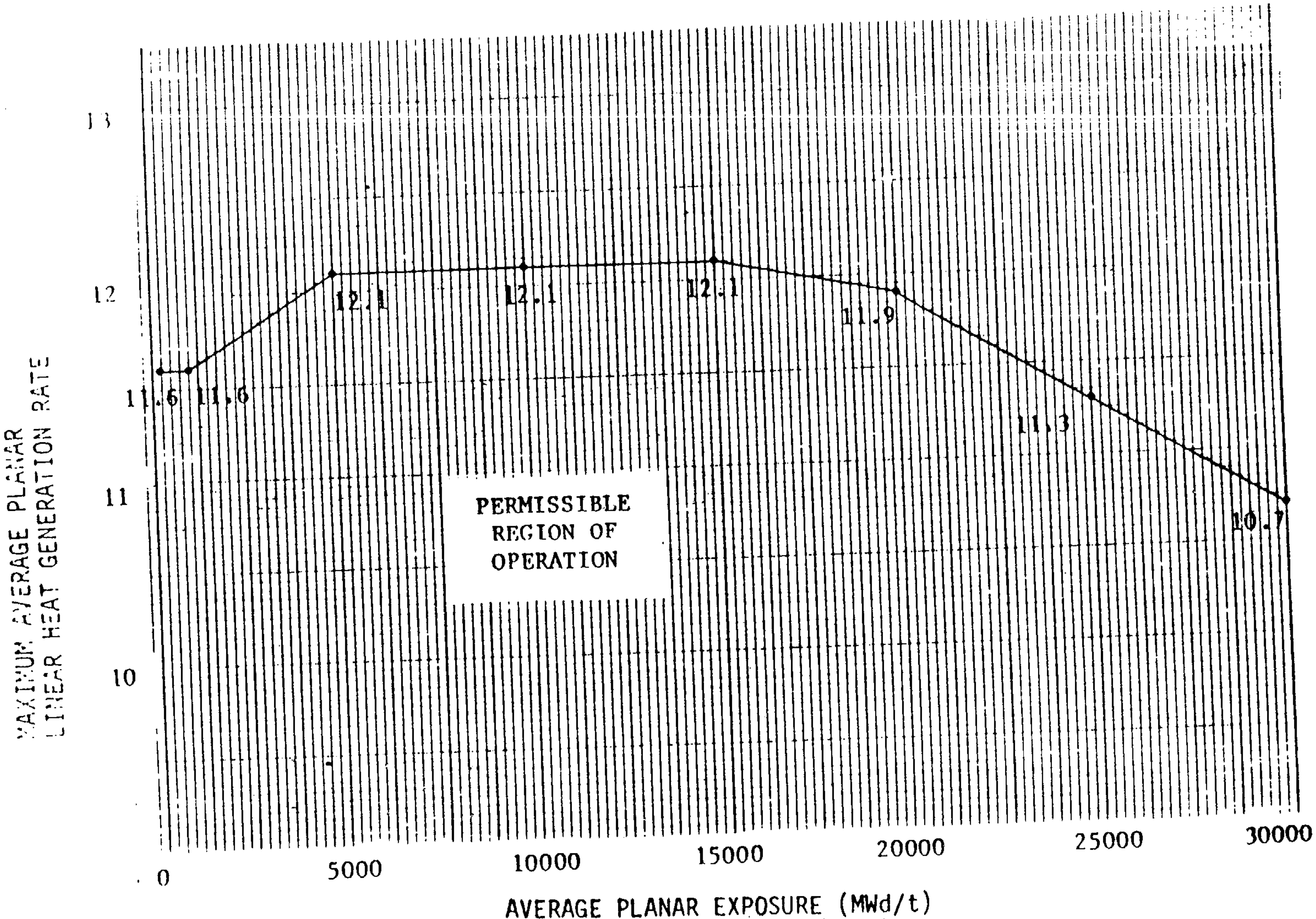
Figure 3.2.1-1



AVERAGE PLANAR EXPOSURE (MWd/t)
FUEL TYPE 2 (8 x 8)

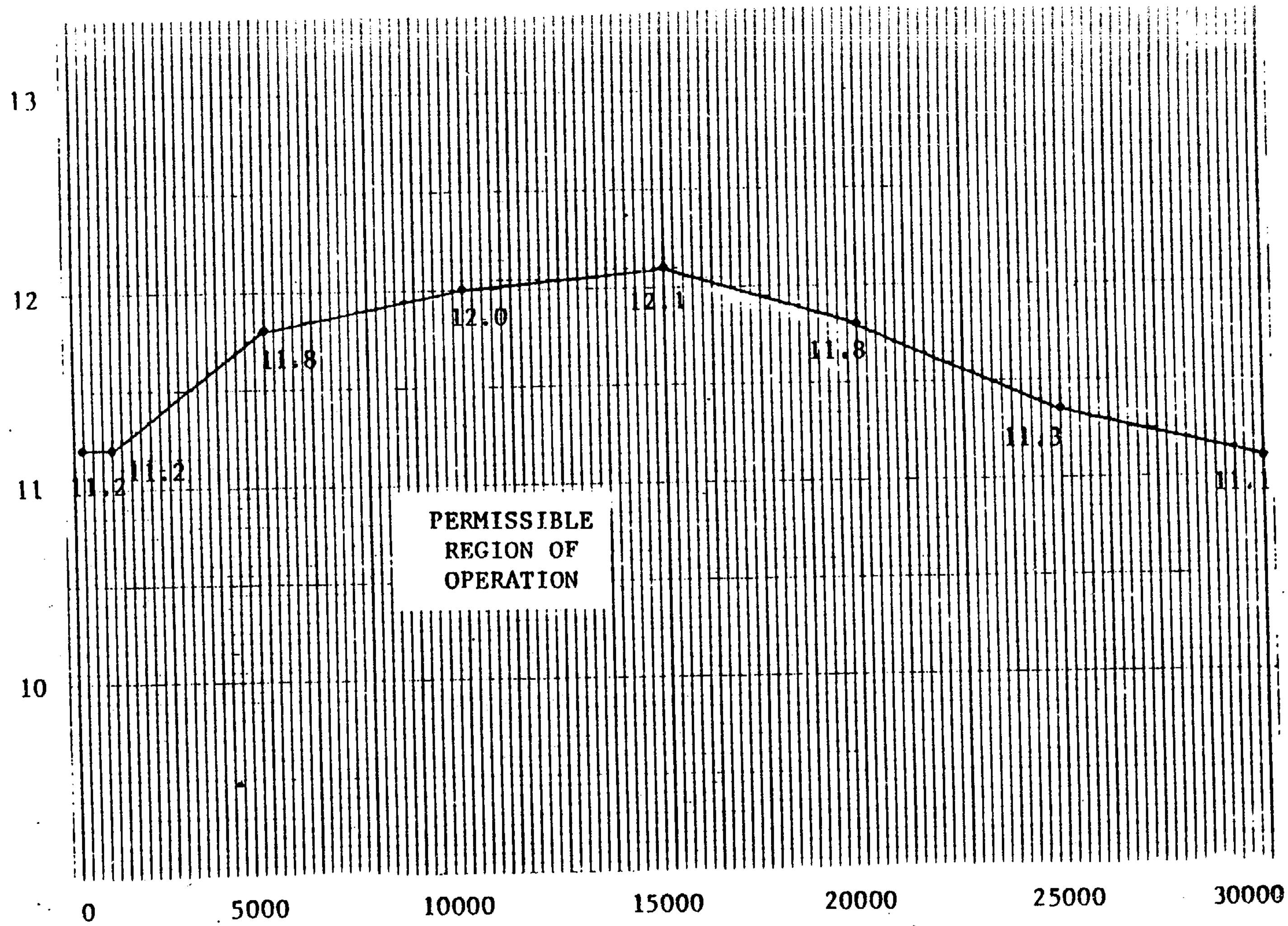
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-2



FUEL TYPE 8DRB265L (8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-3



PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE 8DRB283H (8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-4

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The flow biased APRM scram trip setpoint (S) and rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

$$S \leq (0.66W + 54\%)\ T$$

$$S_{RB} \leq (0.66W + 42\%)\ T$$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow in percent of rated flow,
T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any class of fuel in the core ($T \leq 1.0$), and

Design TPF for 8 x 8 fuel = 2.45.
Design TPF for 8 x 8R fuel = 2.48.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The MTPF for each class of fuel shall be determined, the value of T calculated, and the flow biased APRM trip setpoint adjusted, as required:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MTPF.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than MCPR x the K_f shown in Figure 3.2.3-1 where:

a. MCPR = 1.22 from BOC2* to (EOC2** - 2000 MWD/t).

b. MCPR = 1.23 from (EOC2 - 2000 MWD/t) to (EOC2 - 1000 MWD/t).

c. MCPR = 1.28 from (EOC2 - 1000 MWD/t) to EOC2.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% RATED THERMAL POWER

ACTION:

With MCPR less than the applicable limit determined from Figure 3.2.3-1, initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

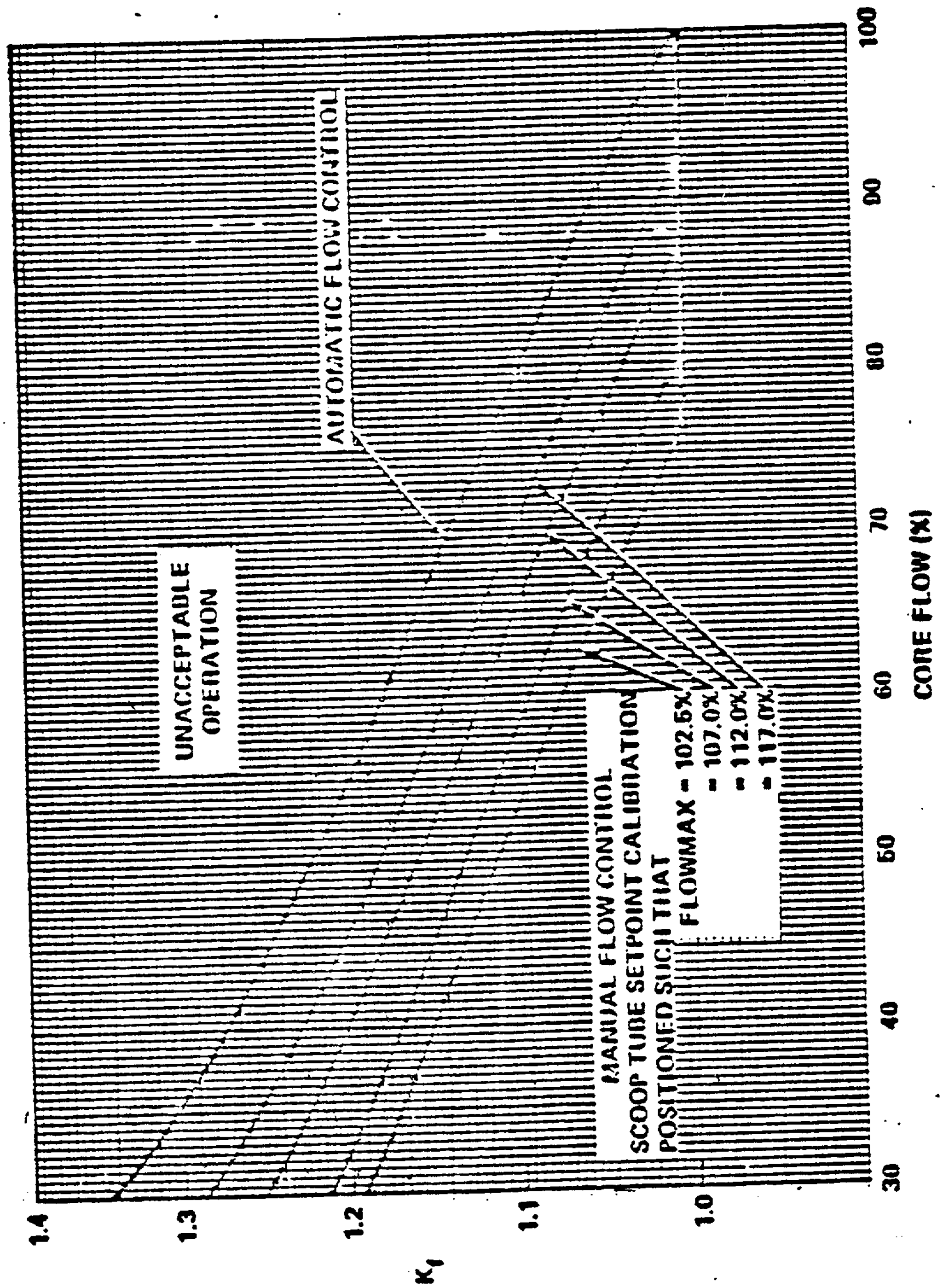
4.2.3 MCPR shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

*Beginning of Cycle 2.
**End of Cycle 2.



K_f FACTOR

FIGURE 3.2.3-1

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 All LINEAR HEAT GENERATION RATES (LHGR's), shall not exceed 13.4 kw/ft.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER

ACTION:

With the LHGR of any fuel rod exceeding 13.4 kw/ft., initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than 13.4 kw/ft:

a. At least once per 24 hours,

b. When THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

INSTRUMENTATION

3/4.3.4 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The control rod withdrawal block instrumentation shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

APPLICABILITY: As shown in Table 3.3.4-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4-2, declare the channel inoperable until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, POWER OPERATION may continue provided that either:

   1. The inoperable channel(s) is restored to OPERABLE status within 24 hours, or

   2. The redundant trip system is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable channel is restored to OPERABLE status, and the inoperable channel is restored to OPERABLE status within 7 days, or

   3. For the Rod Block Monitor only, THERMAL POWER is limited such that MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   4. Otherwise, place at least one trip system in the tripped condition within the next hour.

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

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

SURVEILLANCE REQUIREMENTS

4.3.4 Each of the above required control rod withdrawal block instrumentation channels shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK, CHANNEL CALIBRATION and a CHANNEL FUNCTIONAL TEST during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.4-1.

TABLE 3.3.4-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OF OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS
1. APRM (C51-APRM-CH.A,B,C,D,E,F)		
a. Upscale (Flow Biased)	2	1
b. Inoperative	2	1, 2, 5
c. Downscale	2	1
d. Upscale (Fixed)	2	2, 5
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)		
a. Upscale	1	1*
b. Inoperative	1	1*
c. Downscale	1	1*
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in(b)	1	2, 5
b. Upscale(c)	1	2, 5
c. Inoperative(c)	1	2, 5
d. Downscale(b)	1	2, 5
4. INTERMEDIATE RANGE MONITORS(d) (C51-IRM-K601A,B,C,D,E,F,G,H)		
a. Detector not full in(e)	2	2, 5
b. Upscale	2	2, 5
c. Inoperable	2	2, 5
d. Downscale(e)	2	2

TABLE 3.3.4-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTE

* When THERMAL POWER exceeds the preset power level of the RWM and RSCS.

a. The minimum number of OPERABLE CHANNELS may be reduced by one for up to 2 hours in one of the trip systems for maintenance and/or testing except for Rod Block Monitor function.

b. This function is bypassed if detector is reading > 100 cps or the IRM channels are on range 3 or higher.

c. This function is bypassed when the associated IRM channels are on range 8 or higher.

d. A total of 6 IRM instruments must be OPERABLE.

e. This function is bypassed when the IRM channels are on range 1.

TABLE 3.3.4-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM (C51-APRM-CH.A,B,C,D,E,F)		
a. Upscale (Flow Biased)	$\le (0.66\ W + 42\%) \frac{T^*}{MTPF}$	$\le (0.66\ W + 42\%) \frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\ge$ 3/125 of full scale	$\ge$ 3/125 of full scale
d. Upscale (Fixed)	$\le$ 12% of RATED THERMAL POWER	$\le$ 12% of RATED THERMAL POWER
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)		
a. Upscale	$\le (0.66W + 40\%) \frac{T^*}{MTPF}$	$\le (0.66\ W + 40\%) \frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\ge$ 3/125 of full scale	$\ge$ 3/125 of full scale
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in	NA	NA
b. Upscale	$\le 1 \times 10^5$ cps	$\le 1 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\ge$ 3 cps	$\ge$ 3 cps
4. INTERMEDIATE RANGE MONITORS (C51-IRM-K601A,B,C,D,E,F,G,H)		
a. Detector not full in	NA	NA
b. Upscale	$\le$ 108/125 of full scale	$\le$ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\ge$ 3/125 of full scale	$\ge$ 3/125 of full scale

*T=2.45 for 8 x 8 fuel.
T=2.48 for 8 x 8 R fuel.

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.2 The remote shutdown monitoring instrumentation channels shown in Table 3.3.5.2-1 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the requirements of Table 3.3.5.2-1, either restore the inoperable channel to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.2 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.2-1.

TABLE 3.3.5.2-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	READOUT LOCATION	MINIMUM CHANNELS OPERABLE
1. Reactor Vessel Pressure (C32-PI-3332 and C32-PT-3332)	RSP*	1
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)	RSP*	1
3. Suppression Chamber Water Level (CAC-LI-3342 and CAC-LT-3342)	RSP*	1
4. Suppression Chamber Water Temperature (CAC-TR-778-7)	RSP*	1
5. Drywell Pressure (CAC-PI-3341 and CAC-PT-3341)	RSP*	1
6. Drywell Temperature (CAC-TR-778-1,3,4)	RSP*	1
7. Drywell Oxygen Concentration (CAC-AT-1259-2)	Local Panel	1
8. Residual Heat Removal Head Spray Flow (E11-FT-3339 and E11-FI-3339)	RSP*	1
9. Residual Heat Removal System Flow (E11-FT-3338, E11-FI-3338 and E11-FY-3338)	RSP*	1
10. Residual Heat Removal Service Water Discharge Differential Pressure (E11-PDT-N002BX and E11-PDI-3344)	RSP*	1

*Remote Shutdown Panel, Reactor Building 20' Elevation

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

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.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.5.7-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to the OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3.5.7-1 inoperable:

a. Within 1 hour, increase the inspection frequency for the zone(s) with the inoperable instrument(s) to at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.7.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.5.7.2 The non-supervised circuits between the local panels associated with the detector alarms of each of the above required fire detection instruments and the control room shall be demonstrated OPERABLE at least once per 31 days in accordance with approved procedures.

TABLE 3.3.5.7-1

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION		MINIMUM INSTRUMENTS OPERABLE FLAME	HEAT	SMOKE
1. Reactor Building #1				
Zone 1	-17'	0	0	0
Zone 2	-17'	0	0	1
Zone 3	-17'	0	0	6
Zone 4	-17'	0	0	6
Zone 5	20'	0	0	7
Zone 6	20'	0	0	9
Zone 7	20'	0	0	6
Zone 8	50	0	0	5
Zone 9	50	0	0	7
Zone 10	80'	0	0	6
Zone 11	80'	0	0	6
Zone 12	98'	0	0	3
Zone 13	117'	0	0	1
Zone 14	117'	0	0	35
Zone 15	77'	0	0	3
2. Control Building				
Zone 1	70'	0	0	7
Zone 2	49'	0	0	5
Zone 3	49'	0	0	5
Zone 4	49'	0	0	12
Zone 5	49'	0	0	14
Zone 6	49'	0	0	1
Zone 7	23'	0	0	1
Zone 8	23'	0	0	1
Zone 9	23'	0	0	15
Zone 10	23'	0	0	14
Zone 11	23'	0	0	1
Zone 12	23'	0	0	1
Zone 13	49'	0	0	10
Zone 14	49'	0	0	10
3. Diesel Generator Building				
Zone 1	2'	0	0	7
Zone 2	2'	0	0	7
Zone 3	50'	0	0	6
Zone 4	23'	0	0	3
Zone 5	23'	0	0	1
Zone 6	23'	0	0	1
Zone 7	23'	0	0	1
Zone 8	23'	0	0	1
Zone 9	23'	0	0	1
Zone 10	50'	0	0	6

TABLE 3.3.5.7-1 (Continued)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE FLAME	HEAT	SMOKE
4. Service Water Building			
Zone 1 4'	0	0	6
Zone 2 20	0	0	5
5. AOG Building			
Zone 1 20'	1	0	0
Zone 2 20'	1	0	0
Zone 3 20'	1	5	1
Zone 4 37' - 49'	1	6	0

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 ΔP 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 ΔP 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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

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

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

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

LIMITING CONDITION FOR OPERATION

3.7.7.2 The following spray and/or sprinkler systems shall be OPERABLE:

a.* Diesel Generator #1 Preaction System - Diesel Generator Building

b.* Diesel Generator #2 Preaction System - Diesel Generator Building

c.* Diesel Generator #3 Preaction System - Diesel Generator Building

d.* Diesel Generator #4 Preaction System - Diesel Generator Building

e.* South Cable Spread Area Sprinkler System - Diesel Generator Building

f.* North Cable Spread Area Sprinkler System - Diesel Generator Building

g. Two Standby Gas Treatment Train 1A Deluge Systems - Reactor Building #1.

h. Two Standby Gas Treatment Train 1B Deluge Systems - Reactor Building #1.

i.* Area Sprinkler System - Reactor Building #1.

j.* Service Water Pump Area Sprinkler System - Service Water Building

k.* Service Water Cable Spread Area Sprinkler System - Service Water Building

l.* Drumming Room Sprinkler System - Radwaste Building

m. Makeup Water Treatment Area Sprinkler System - Makeup Water Treatment Building

APPLICABILITY: Whenever equipment in the areas protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

*Effective July 27, 1979

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.7.4 The fire hose stations shown in Table 3.7.7.4-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7.7.4-1 inoperable, within one hour:

   1. Provide an alternate means of fire suppression for the unprotected area(s) or

   2. Route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.4 Each of the fire hose stations shown in Table 3.7.7.4-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

TABLE 3.7.7.4-1

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Unit No. 1 Reactor Bldg.	-17'	1-RB-19[a]
	-17'	1-RB-20[a]
	-17'	1-RB-24[a]
	-17'	1-RB-25[a]
	-17'	1-RB-26[a]
	20'	1-RB-21[a]
	20'	1-RB-22[a]
	20'	1-RB-23[a]
	20'	1-RB-27[a]
	20'	1-RB-28[a]
	20'	1-RB-29[a]
	50'	1-RB-30[a]
	50'	1-RB-31[a]
	50'	1-RB-32[a]
	50'	1-RB-33[a]
	50'	1-RB-34[a]
	50'	1-RB-35[a]
	67'	1-RB-48A[a]
	80'	1-RB-36[a]
	80'	1-RB-39[a]
	80'	1-RB-41[a]
	80'	1-RB-43[a]
	80'	1-RB-44[a]
	80'	1-RB-45[a]
	98'	1-RB-37[a]
	117'	1-RB-38[a]
	117'	1-RB-40[a]
	117'	1-RB-42[a]
	117'	1-RB-46[a]
	117'	1-RB-47[a]
	117'	1-RB-48[a]
AOG Building	23'	2-AOG-57
	23'	2-AOG-58
	23'	2-AOG-59
	23'	2-AOG-60
	37'	2-AOG-62
	49'	2-AOG-61
Radwaste Building	-3'	RW-49
	-3'	RW-50
	-3'	RW-51
	23'	RW-52
	23'	RW-53
	23'	RW-54
	23'	RW-55
	23'	RW-56

[a] Effective May 31, 1979.

TABLE 3.7.7.4-1 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Diesel Generator Building	2'	DGB-1[b]
	2'	DGB-2[b]
	2'	DGB-3[b]
	23'	DGB-4[b]
	23'	DGB-5[b]
	23'	DGB-6[b]
	23'	DGB-7[b]
	23'	DGB-8[b]
	23'	DGB-9[b]
	50'	DGB-10[b]
	50'	DGB-11[b]
	50'	DGB-12[b]
	50'	DGB-13[b]
	50'	AFFF HR-2[c]
	50'	AFFF HR-3[c]
Service Water Building	4'	SW-1[d]
	20'	SW-2[d]
	20'	SW-3[d]
Control Building	23'	1-CB-1[e]
	49'	1-CB-2[e]
	70'	2-CB-3[e]
Diesel Generator Tank Area	NA	AFFF HR-1[c]

(b) Effective May 11, 1979

(c) Effective July 27, 1979

(d) Effective April 11, 1979

(e) Effective April 23, 1979

FOAM SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.5 The following foam systems shall be OPERABLE:

a. Diesel Generator Fuel Oil Tank Area Foam System with:

1. The concentrate proportioning and storage subsystem OPERABLE with 240 gallons of concentrate.

2. Each tank room subsystem OPERABLE.

b. Diesel Generator Air Filter Foam System with:

1. The concentrate proportioning and storage subsystem OPERABLE with 40 gallons of concentrate.

2. Each air filter subsystem OPERABLE.

APPLICABILITY: Whenever the diesel generators are required to be OPERABLE.*

ACTION:

a. With one tank room subsystem inoperable, verify the OPERABILITY of the backup foam hose reel within one hour.

b. With one air filter subsystem inoperable, verify the OPERABILITY of two backup foam hose reels within one hour.

c. With any inoperability other than as provided in a and b, above, verify the availability of backup fire suppression equipment for the unprotected area(s) within one hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE stauts.

d. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

*Effective July 27, 1979

SURVEILLANCE REQUIREMENTS

4.7.7.5 Each of the above required foam systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. A visual inspection of the spray headers to verify their integrity.

3. A visual inspection of each nozzle's spray area to verify that the spray pattern is not obstructed.

4. Conducting a performance evaluation of the concentrate.

3/4.7.8 PENETRATION FIRE BARRIERS

LIMITING CONDITIONS FOR OPERATION

3.7.8 All penetration fire barriers protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required penetration fire barriers non-functional, within one hour:

   1. Establish a continuous fire watch on at least one side of the affected penetration, or

   2. Verify the OPERABILITY of the fire detection instruments providing coverage for the fire detection zones on each side of the non-functional barrier(s) by performance of the surveillance requirements of Specifications 4.3.5.7.1 and 4.3.5.7.2, as applicable.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 Each of the above required penetration fire barriers:

a. Shall be verified to be functional by a visual inspection;

   1. At least once per 18 months, and

   2. Prior to declaring a penetration fire barrier functional following repairs or maintenance.

b. That performs a pressure sealing function shall be verified to be functional by performance of a local leakage test prior to declaring a penetration fire barrier functional following repairs or maintenance.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The requirements of Specifications 3.9.1 and 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be locked in the Startup position and to allow up to three control rods to be withdrawn for shutdown margin demonstrations provided at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry shorting links removed per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, and

c. The "notch-override" control shall not be used during movement of the control rods.

APPLICABILITY: CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately restore the reactor mode switch to the Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to the performance of a shutdown margin demonstration verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2, and

b. The rod worth minimizer is OPERABLE with the required program, per Specification 3.1.4.1.

SPECIAL TEST EXCEPTION

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirement of Specification 3.4.1.1 that two recirculation loops be in operation may be suspended for up to 24 hours during the performance of startup and PHYSICS TESTS.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With the above specified time limit exceeded, deenergize the scram solenoid valves.

SURVEILLANCE REQUIREMENTS

4.10.4 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during startup and PHYSICS TESTS.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold xenon-free condition and shall show the core to be subcritical by at least R + 0.38% ΔK. The value of R in units of %ΔK is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle. Satisfaction of this limitation can be best demonstrated at the time of fuel loading but the margin must be determined anytime a control rod is incapable of insertion.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

3/4.1.3 CONTROL RODS

The specifications of this section ensure that 1) the minimum SHUTDOWN MARGIN is maintained, 2) the control rod insertion times are consistent with those used in the accident analysis, and 3) the

BASES

CONTROL RODS (Continued)

potential effects of the rod ejection accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the non-fully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent the MPCR from becoming less than 1.07 during the limiting power transient analyzed in Section 14.3 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MPCR remains greater than 1.07. The occurrence of scram times longer than those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within a assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3 and 3.2.1-4.

The calculational procedure used to establish the APLHGR shown on Figure 3.2.1-1 thru 3.2.1-4 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figures 3.2.1-1, and 3.2.1-2; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limitation as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

Bases Table B 3.2.1-1

SIGNIFICANT INPUTS PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR BRUNSWICK-UNIT 1

Plant Parameters;

Core Thermal Power .... 2531 Mwt which corresponds 105% of rated steam flow*

Vessel Steam Output ....10.96×10^6 Lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure..........1055 psia

Recirculation Line
Break Area for Large Breaks
a. Discharge — 2.4 ft^2 (DBA); 1.9 ft^2 (80% DBA)

b. Suction — 4.2 ft^2

Number of Drilled Bundles — 560

Fuel Parameters:

FUEL TYPES	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO**
All	8 x 8	13.4	1.4	1.2

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

*This power level meets the Appendix K requirement of 102%.

**To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

BASES

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity safety limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.45 for 8 x 8 fuel and 2.48 for 8 x 8R fuel. The scram setting and rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and peak flux indicates a TOTAL PEAKING FACTOR greater than 2.45 for 8 x 8 fuel and 2.48 for 8 x 8R fuel. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine by pass. This transient yields the largest Δ MCPR. When added to the Safety Limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.2.3 is obtained. Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4.4 of NEDO-20360[4] and on core parameters shown in Reference 3, response to Items 2 and 9.

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in Attachment 5 of Reference 6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802(5). Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDO-20566(1). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of Specification 3.2.3 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated should the most limiting transient occur at less than rated flow.

The K_f factor values shown in Figure 3.2.3-1 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.2.3-1 are conservative for the General Electric Plant operation because the operating limit MCPR's of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with mimimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape, regardless of magnitude that could place operation at a thermal limit.

3.2.4 LINEAR HEAT GENERATION RATE

The LHGR specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

BASES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, January, 1976.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. Letter from J. A. Jones, Carolina Power and Light Company to B. C. Rusche, NRC transmitting Amendment 31 to the Brunswick Unit 1 Docket No. 50-325, dated November 26, 1975.

4. General Electric BWR Generic Reload Application for 8 x 8 Fuel, NEDO-20360, Revision 1, November 1974.

5. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

6. Letter from J. A. Jones, Carolina Power and Light Company, to B. C. Rusche, NRC dated May 7, 1976.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1, based on the information given in Section 2.2 of the FSAR.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The PRIMARY CONTAINMENT is a steel lined reinforced concrete structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a concrete steel lined pressure vessel in the shape of a torus. The primary containment has a minimum free air volume of (288,000) cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 62 psig.

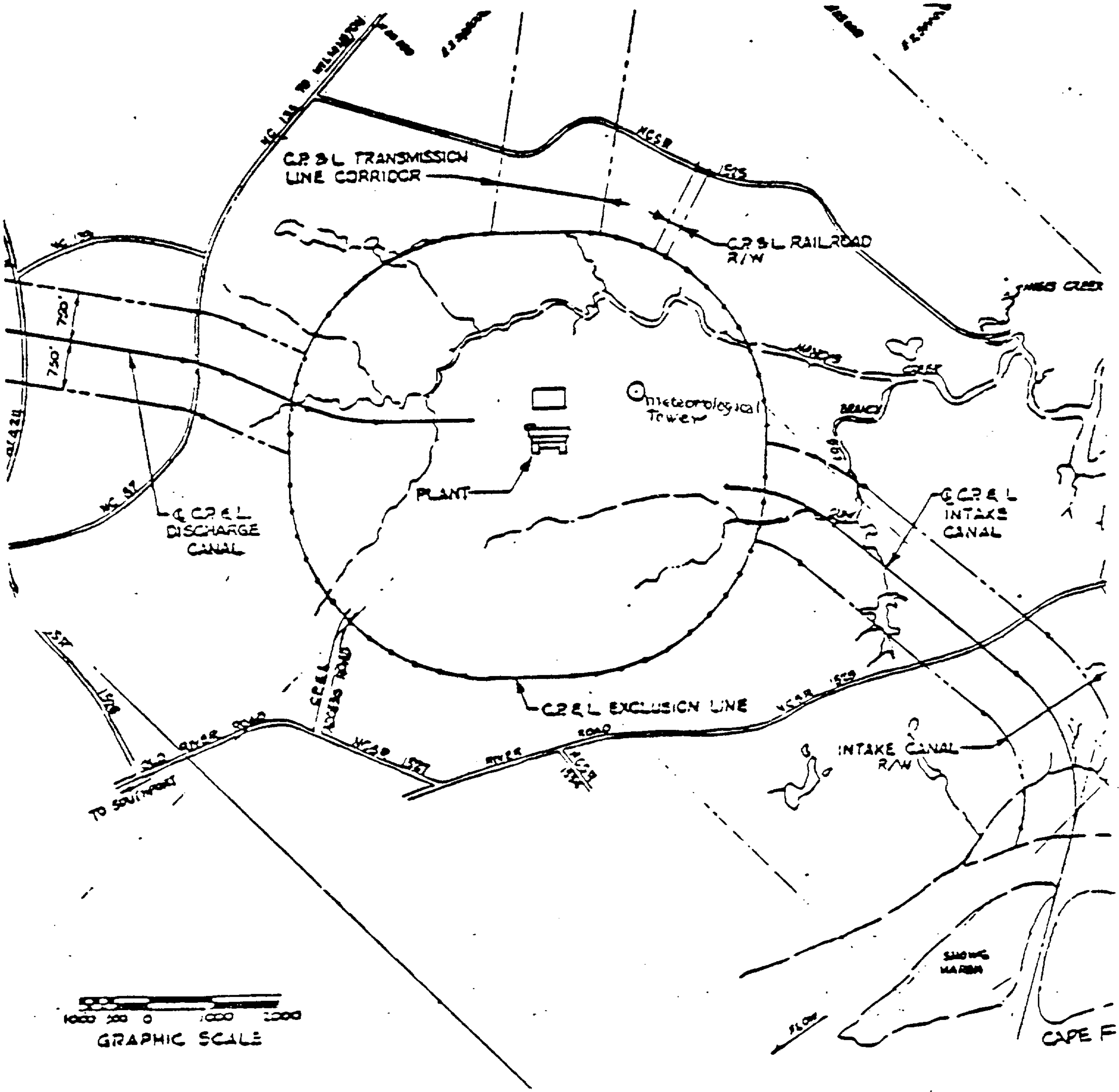
b. Maximum internal temperature: drywell 300°F.
suppression chamber 200°F.

c. Maximum external pressure 2 psig.

5.3 REACTOR CORE

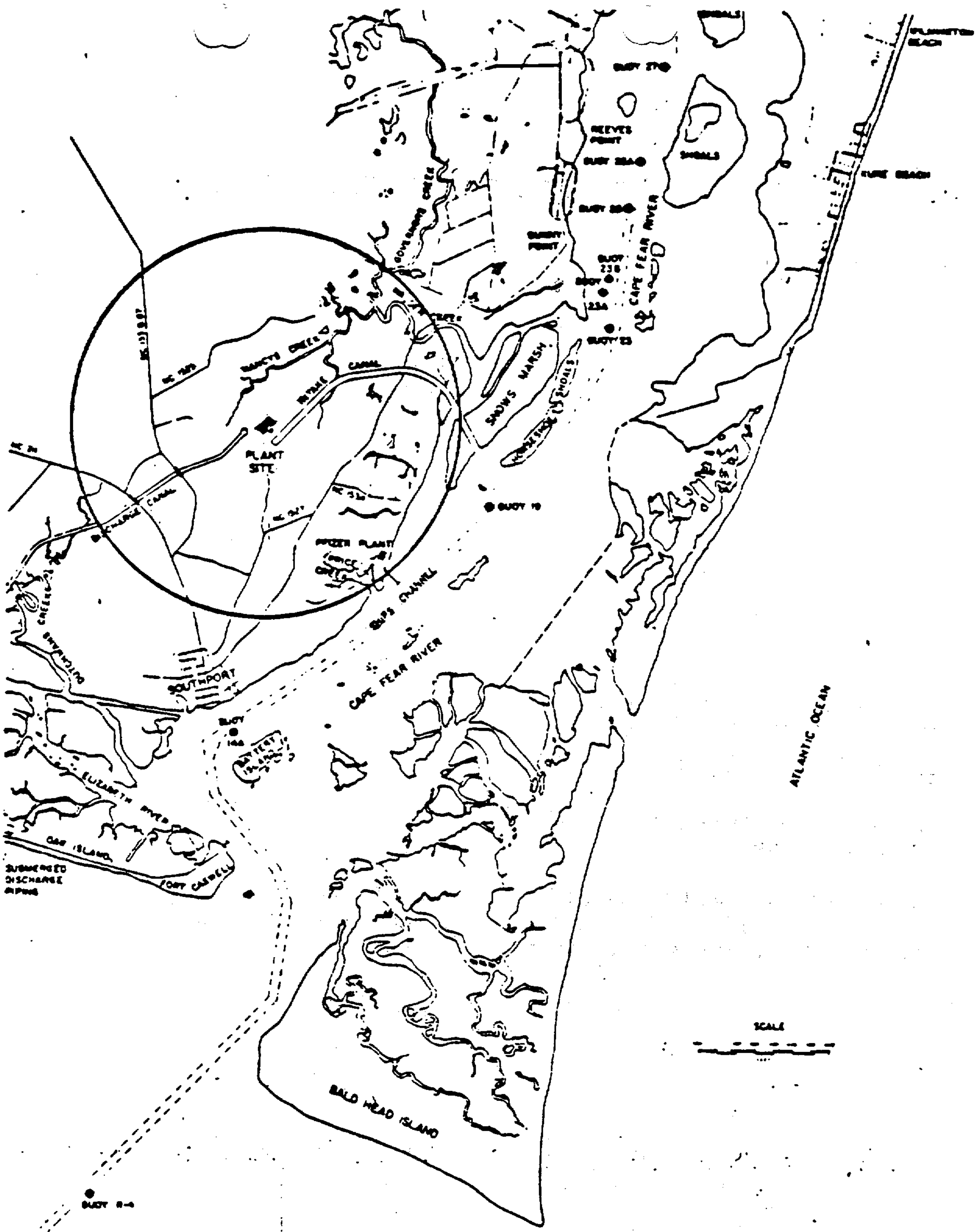
FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 560 fuel assemblies with each fuel assembly containing 63 fuel rods clad with Zircaloy 2. Each fuel rod shall have a nominal active fuel length of 146 inches for 8 x 8 fuel and 150 inches for 8 x 8R fuel and contain a maximum total weight of 3,355 grams of UO_2. The initial core loading



EXCLUSION AREA

Figure 5.1.1-1



LOW POPULATION ZONE

FIGURE 5.1.2-1

5.3 REACTOR CORE

FUEL ASSEMBLIES (Continued)

shall have a maximum enrichment of 2.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 2.85 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 control rod assemblies each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, power surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The nuclear boiler and reactor recirculation system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 1250 psig, and

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 18,670 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1.1-1.

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION#

Condition of Unit 1 - Unit 2 in CONDITION 1, 2 or 3

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	2*
OL**	3	2
Non-Licensed	4	3

Condition of Unit 1 - Unit 2 in CONDITION 4 or 5

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	1*
OL**	2	2
Non-Licensed	3	3

Condition of Unit 1 - No Fuel in Unit 2

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL	1	1*
OL	2	1
Non-Licensed	2	1

* Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

**Assumes each individual is licensed on both plants.

\# Shift crew composition, including an individual qualified in radiation protection procedures, may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for the Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Fire Chief and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.1.1 The PNSC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman:	Operations Maintenance or Technical-Administrative Superintendent Supervisor
Secretary:	Administrative Supervisor
Member:	Maintenance Supervisor
Member:	Engineering Supervisor
Member:	Environmental and Radiation Control Supervisor
Member:	Quality Assurance Supervisor
Member:	Operating Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PNSC activities at any one time.

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. 47
License No. DPR-62

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

   A. The applications for amendments by Carolina Power & Light Company (the licensee) dated February 19, 1979, and March 6, 1979, as supplemented, comply 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 applications, 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 revising paragraph 2.B(7) to read as indicated below:

   2.B(7) - The licensee may proceed with and is required to complete the modifications identified in Paragraph 3.1.1 through 3.1.35 of the NRC's Fire Protection Safety Evaluation Report on the Brunswick facility dated November 22, 1977 and supplements thereto. These modifications shall be completed by the dates identified in the Safety Evaluation Report or Table 3.1 in supplements thereto. In addition, the licensee may proceed with and is required to complete the modifications identified in Section B.2.1 of Supplement 1 to the Fire Protection Safety Evaluation Report, and any future supplements. These modifications shall be completed by the dates identified in Table B.2.1 of the supplement.

The license is further amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) is hereby amended to read as follows:

2.C(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, 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 #3
Division of Operating Reactors

Date of Issuance: April 6, 1979

ATTACHMENTS TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

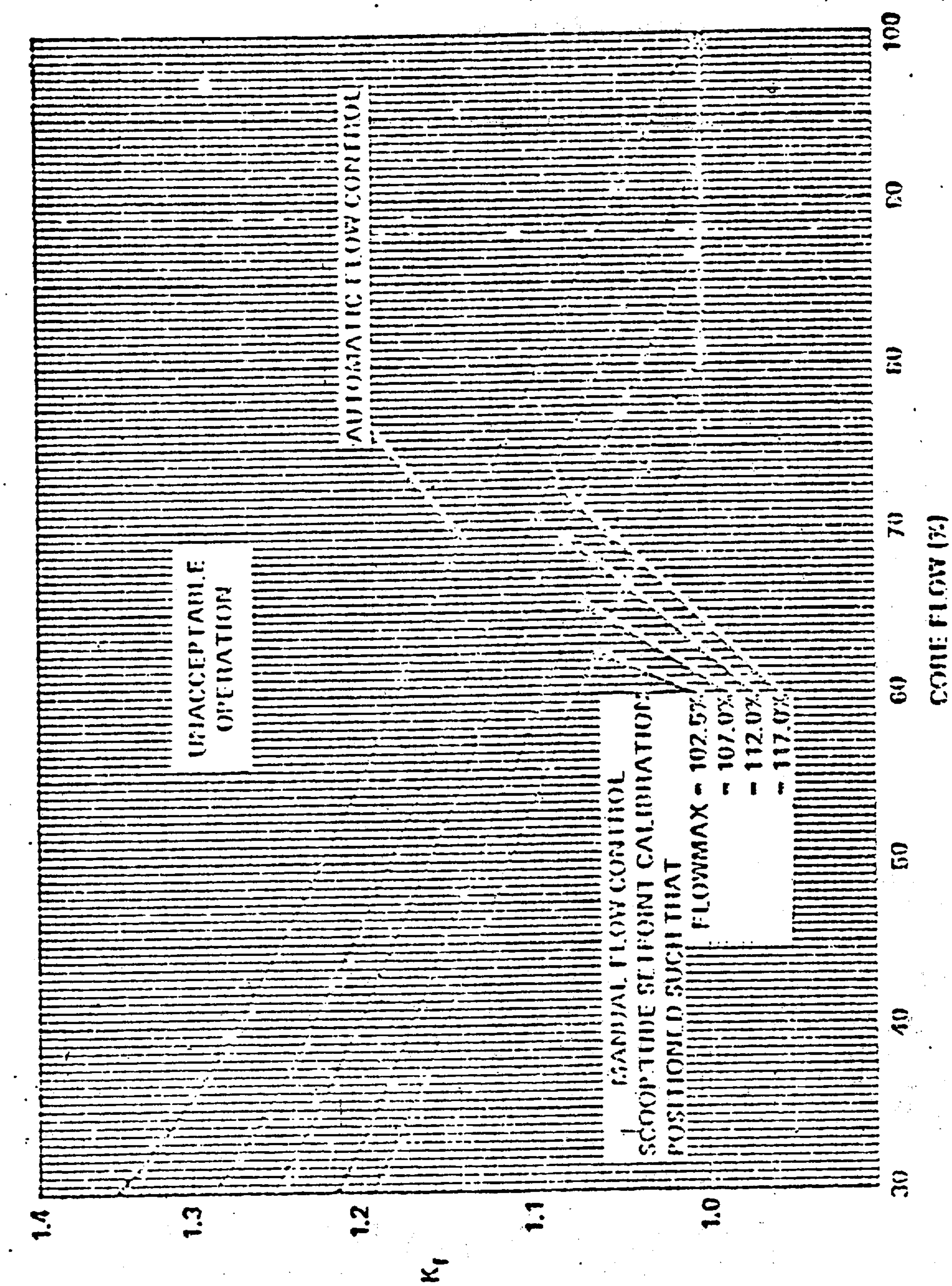
Remove	Insert
viii	viii
3/4 2-10 thru 2-12	3/4 2-10 thru 2-11
3/4 3-48	3/4 3-48
3/4 3-49	3/4 3-49
3/4 3-60	3/4 3-60
3/4 3-61	3/4 3-61
3/4 6-20	3/4 6-20
3/4 7-43	3/4 7-43
3/4 7-47 thru 3/4 7-48	3/4 7-47 thru 7-51
6-5	6-5

Oversides included for convenience.

INDEX

LIMITING CONDITIONS FOR OPERATIONS AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS



K_f FACTOR

FIGURE 3.2.3-1

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 All LINEAR HEAT GENERATION RATES (LHGR's) shall not exceed:

a. For 7 X 7 fuel assemblies, as a function of core height for any fuel rod in an assembly, the maximum allowable LHGR shown in Figure 3.2.4-1.

b. For 8 X 8 and 8 X 8R fuel assemblies, 13.4 kw/ft.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the above limits, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

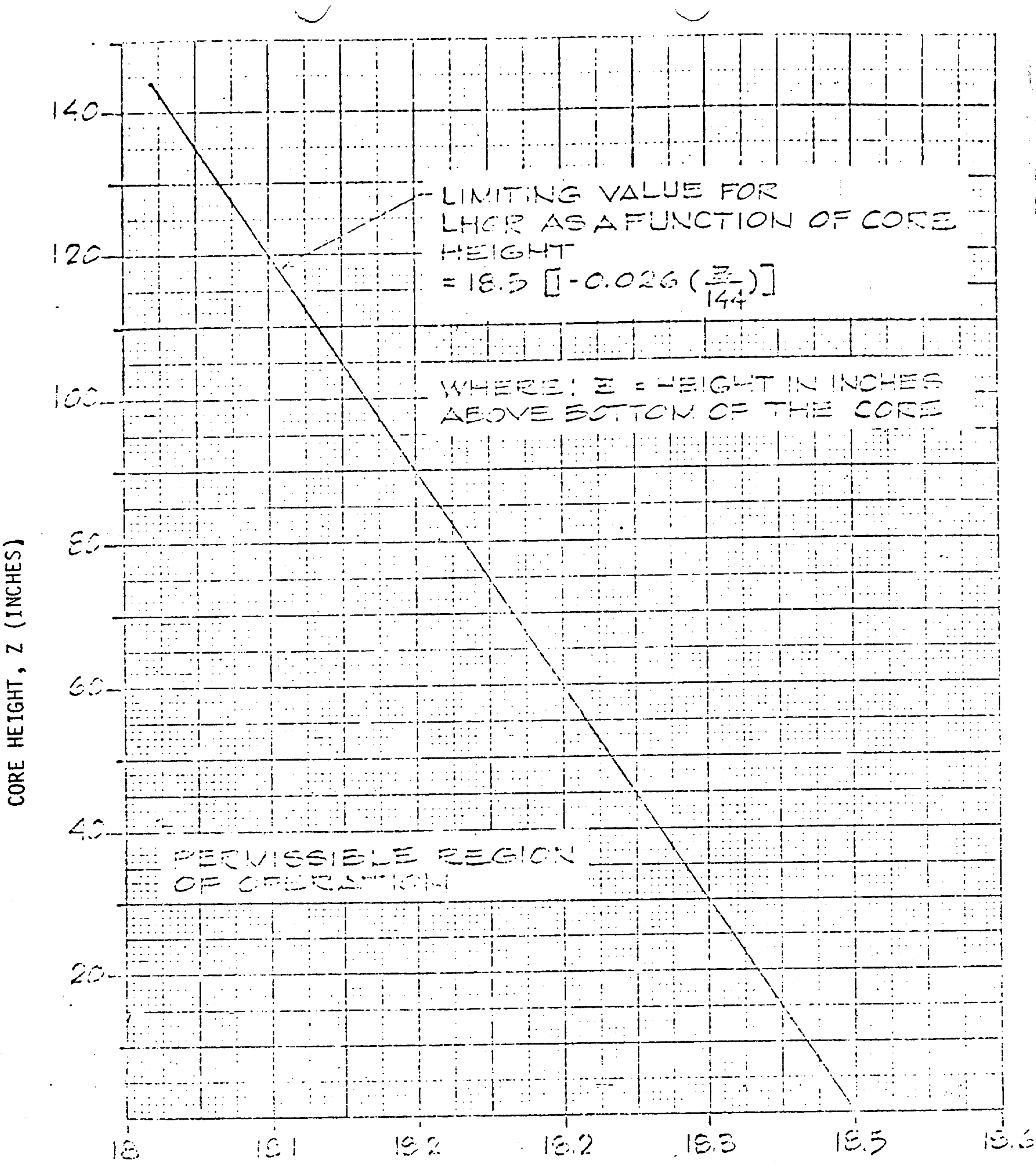
SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the applicable above limit:

a. At least once per 24 hours,

b. When THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.



LHGR (kw/ft)
LIMITING VALUE FOR LHGR
7 x 7 FUEL

FIGURE 3.2.4-1

INSTRUMENTATION

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.2 The remote shutdown monitoring instrumentation channels shown in Table 3.3.5.2-1 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the requirements of Table 3.3.5.2-1, either restore the inoperable channel to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.2 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.2-1.

TABLE 3.3.5.2-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	READOUT LOCATION	MINIMUM CHANNELS OPERABLE
1. Reactor Vessel Pressure (C32-PI-3332 and C32-PT-3332)	RSP*	1
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)	RSP*	1
3. Suppression Chamber Water Level (CAC-LI-3342 and CAC-LT-3342)	RSP*	1
4. Suppression Chamber Water Temperature (CAC-TR-778-7)	RSP*	1
5. Drywell Pressure (CAC-PI-3341 and CAC-PT-3341)	RSP*	1
6. Drywell Temperature (CAC-TR-778-1,3,4)	RSP*	1
7. Drywell Oxygen Concentration (CAC-AT-1259-2)	Local Panel	1
8. Residual Heat Removal Head Spray Flow (E11-FT-3339 and E11-FI-3339)	RSP*	1
9. Residual Heat Removal System Flow (E11-FT-3338, E11-FI-3338 and E11-FY-3338)	RSP*	1
10. Residual Heat Removal Service Water Discharge Differential Pressure (E11-PDT-N002BX and E11-PDI-3344)	RSP*	1

*Remote Shutdown Panel, Reactor Building 20' Elevation

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

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.

TABLE 3.3.5.7-1 (Continued)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE FLAME	HEAT	SMOKE
4. Service Water Building			
Zone 1 4'	0	0	6
Zone 2 20	0	0	5
5. AOG Building			
Zone 1 20'	1	0	0
Zone 2 20'	1	0	0
Zone 3 20'	1	5	1
Zone 4 37' - 49'	1	6	0

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 ΔP 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 ΔP 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.

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.

LIMITING CONDITION FOR OPERATION

3.7.7.2 The following spray and/or sprinkler systems shall be OPERABLE:

a.* Diesel Generator #1 Preaction System - Diesel Generator Building

b.* Diesel Generator #2 Preaction System - Diesel Generator Building

c.* Diesel Generator #3 Preaction System - Diesel Generator Building

d.* Diesel Generator #4 Preaction System - Diesel Generator Building

e.* South Cable Spread Area Sprinkler System - Diesel Generator Building

f.* North Cable Spread Area Sprinkler System - Diesel Generator Building

g. Two Standby Gas Treatment Train 1A Deluge Systems - Reactor Building #2.

h. Two Standby Gas Treatment Train 1B Deluge Systems - Reactor Building #2.

i.* Area Sprinkler System - Reactor Building #2.

j.* Service Water Pump Area Sprinkler System - Service Water Building

k.* Service Water Cable Spread Area Sprinkler System - Service Water Building

l.* Drumming Room Sprinkler System - Radwaste Building

m. Makeup Water Treatment Area Sprinkler System - Makeup Water Treatment Building

APPLICABILITY: Whenever equipment in the areas protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

*Effective July 27, 1979

SURVEILLANCE REQUIREMENTS

4.7.7.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months:

   1. By performing a system functional test which includes simulated automatic actuation of the system, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2. By inspection of the the spray headers to verify their integrity, and

   3. By inspection of each deluge nozzle to verify no blockage.

TABLE 3.7.7.4-1

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Unit No. 2 Reactor Bldg.	-17'	2-RB-19[a]
	-17'	2-RB-20[a]
	-17'	2-RB-24[a]
	-17'	2-RB-25[a]
	-17'	2-RB-26[a]
	20'	2-RB-21[a]
	20'	2-RB-22[a]
	20'	2-RB-23[a]
	20'	2-RB-27[a]
	20'	2-RB-28[a]
	20'	2-RB-29[a]
	50'	2-RB-30[a]
	50'	2-RB-31[a]
	50'	2-RB-32[a]
	50'	2-RB-33[a]
	50'	2-RB-34[a]
	50'	2-RB-35[a]
	67'	2-RB-48A[a]
	80'	2-RB-36[a]
	80'	2-RB-39[a]
	80'	2-RB-41[a]
	80'	2-RB-43[a]
	80'	2-RB-44[a]
	80'	2-RB-45[a]
	98'	2-RB-37[a]
	117'	2-RB-38[a]
	117'	2-RB-40[a]
	117'	2-RB-42[a]
	117'	2-RB-46[a]
	117'	2-RB-47[a]
	117'	2-RB-48[a]
AOG Building	23'	2-AOG-57
	23'	2-AOG-58
	23'	2-AOG-59
	23'	2-AOG-60
	37'	2-AOG-62
	49'	2-AOG-61
Radwaste Building	-3'	RW-49
	-3'	RW-50
	-3'	RW-51
	23'	RW-52
	23'	RW-53
	23'	RW-54
	23'	RW-55
	23'	RW-56

[a] Effective May 31, 1979

TABLE 3.7.7.4-1 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Diesel Generator Building	2'	DGB-1[(b)]
	2'	DGB-2[(b)]
	2'	DGB-3[(b)]
	23'	DGB-4[(b)]
	23'	DGB-5[(b)]
	23'	DGB-6[(b)]
	23'	DGB-7[(b)]
	23'	DGB-8[(b)]
	23'	DGB-9[(b)]
	50'	DGB-10[(b)]
	50'	DGB-11[(b)]
	50'	DGB-12[(b)]
	50'	DGB-13[(b)]
	50'	AFFF HR-2[(c)]
	50'	AFFF HR-3[(c)]
Service Water Building	4'	SW-1[(d)]
	20'	SW-2[(d)]
	20'	SW-3[(d)]
Control Building	23'	1-CB-1[(e)]
	49'	1-CB-2[(e)]
	70'	2-CB-3[(e)]
Diesel Generator Tank Area	NA	AFFF HR-1[(c)]

(b) Effective May 11, 1979

(c) Effective July 27, 1979

(d) Effective April 11, 1979

(e) Effective April 23, 1979

PLANT SYSTEMS

FOAM SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.5 The following foam systems shall be OPERABLE:

a. Diesel Generator Fuel Oil Tank Area Foam System with:

1. The concentrate proportioning and storage subsystem OPERABLE with 240 gallons of concentrate.

2. Each tank room subsystem OPERABLE.

b. Diesel Generator Air Filter Foam System with:

1. The concentrate proportioning and storage subsystem OPERABLE with 40 gallons of concentrate.

2. Each air filter subsystem OPERABLE.

APPLICABILITY: Whenever the diesel generators are required to be OPERABLE.*

ACTION:

a. With one tank room subsystem inoperable, verify the OPERABILITY of the backup foam hose reel within one hour.

b. With one air filter subsystem inoperable, verify the OPERABILITY of two backup foam hose reels within one hour.

c. With any inoperability other than as provided in a and b, above, verify the availability of backup fire suppression equipment for the unprotected area(s) within one hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

*Effective July 27, 1979

SURVEILLANCE REQUIREMENTS

4.7.7.5 Each of the above required foam systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. A visual inspection of the spray headers to verify their integrity.

3. A visual inspection of each nozzle's spray area to verify that the spray pattern is not obstructed.

4. Conducting a performance evaluation of the concentrate.

3/4.7.8 PENETRATION FIRE BARRIERS

LIMITING CONDITIONS FOR OPERATION

3.7.8 All penetration fire barriers protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required penetration fire barriers non-functional, within one hour:

   1. Establish a continuous fire watch on at least one side of the affected penetration, or

   2. Verify the OPERABILITY of the fire detection instruments providing coverage for the fire detection zones on each side of the non-functional barrier(s) by performance of the surveillance requirements of Specifications 4.3.5.7.1 and 4.3.5.7.2, as applicable.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 Each of the above required penetration fire barriers:

a. Shall be verified to be functional by a visual inspection;

   1. At least once per 18 months, and

   2. Prior to declaring a penetration fire barrier functional following repairs or maintenance.

b. That performs a pressure sealing function shall be verified to be functional by performance of a local leakage test prior to declaring a penetration fire barrier functional following repairs or maintenance.

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION#

Condition of Unit 2 - Unit 1 in CONDITION 1, 2 or 3

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	2*
OL**	3	2
Non-Licensed	4	3

Condition of Unit 2 - Unit 1 in CONDITION 4 or 5

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	1*
OL**	2	2
Non-Licensed	3	3

Condition of Unit 2 - No Fuel in Unit 1

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL	1	1*
OL	2	1
Non-Licensed	2	1

* [illegible]es not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

**Assumes each individual is licensed on both plants.

= Shift crew composition, including an individual qualified in radiation protection procedures, may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for the Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Fire Chief and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.1.1 The PNSC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman:	Operations Maintenance or Technical-Administrative Superintendent Supervisor
Secretary:	Administrative Supervisor
Member:	Maintenance Supervisor
Member:	Engineering Supervisor
Member:	Environmental and Radiation Control Supervisor
Member:	Quality Assurance Supervisor
Member:	Operating Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PNSC activities at any one time.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY LICENSE NO. DPR-71 AND AMENDMENT NO. 47 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

A. Brunswick Steam Electric Plant, Unit No. 1, Fuel Cycle No. 2 - Reload Application

1.0 Introduction

By letter dated December 29, 1978 as supplemented January 17, 1979, March 16, 1979, and March 27, 1979, Carolina Power and Light Company (the licensee) requested amendments to Facility Operating License No. DPR-71. The proposed changes relate to the replacement of 176 fuel assemblies constituting refueling of the core for second cycle operation at power levels up to 2436 Mwt (100% power) for Brunswick Steam Electric Plant Unit No. 1 (BSEP 1).

In support of the reload application, the licensee has provided the GE BWR Reload 1 Licensing submittal for BSEP 1 (Reference 1), proposed Technical Specification changes (Reference 2), information on the BSEP 1 Loss of Coolant Accident (LOCA) analysis (References 3 and 4), and responses to NRC requests for additional information on BSEP 1 Physics Startup Tests (Reference 5).

This reload involves loading of General Electric Company Retrofit (8x8R) fuel. The description of the nuclear and mechanical design of the (8x8R) fuel and the (8x8) fuel is contained on GE's licensing topical report for BWR reloads (Reference 6). Reference 6 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing (7x7), (8x8) and (8x8R) fuel.

Values for each plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other various design parameters are provided in Reference 6.

Additional plant and cycle dependent information are provided in the reload application, (Reference 1), which closely follows the outline of Appendix A of Reference 6.

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Reference 8, describes the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 6. The above mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

Our safety evaluation (Reference 8) of the GE generic reload licensing topical report concluded that the nuclear and mechanical design of the (8x8R) fuel, and GE's analytical methods for nuclear, thermal-hydraulic, and transient and accident calculations as applied to mixed cores containing (7x7), (8x8) and (8x8R) fuel are acceptable. Approval of the nuclear and mechanical design of (8x8) fuel was determined based on information in Reference 7 and expressed in the staff's status report (Reference 9) on that document.

Because of our review of a large number of generic considerations related to use of (8x8R) fuel in mixed loadings with (8x8) and (7x7) fuel, and on the basis of the evaluations which have been presented in Reference 8, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 8.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 2 operation of BSEP 1, 52 fresh (8x8R) fuel bundles of type 8DR B 265L and 124 fresh (8x8R) bundles of type 8DR B 283 will be loaded into the core (Reference 1). The remainder of the 560 fuel bundles in the core will be fuel that was used during the previous cycle.

The fresh fuel will be loaded in a core pattern as shown in Figure 1 of Reference 1, which is acceptable.

Based on the data presented in sections 4 and 5 of Reference 1, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 2.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 6, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients (or from undetected fuel loading errors) is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling, and that transition boiling will not occur during steady state operation as the result of the worst possible fuel loading error.

The safety limit MCPR for BSEP 1 is being raised from 1.05 to 1.07 because the distribution of fuel rod power with the (8x8R) fuel bundles is flatter than that of the (8x8) fuel. The reason for the flatter power distribution is the presence of two rather than one water rods in (8x8R) fuel. The issue has been addressed in Reference 8 and 1.07 limit has been found acceptable for BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 6, for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 6. It has been shown in Reference 1 that these conditions are met for BSEP 1, Cycle 2.

In addition to the 1.07 MCPR safety limit discussed above, the reactor fuel must be maintained within the 17.5 KW/ft exposure-dependent Linear Heat Generation Rate (LHGR) safety limit. Maximum LHGR conditions can occur during abnormal operational conditions which affect the fuel locally, e.g., Rod Withdrawal Error and the Fuel Loading Error. In this regard, the staff requires that the calculated maximum transient LHGR for the 8x8 and 8x8R fuel be augmented by a fuel densification power spike allowance. As stated in Reference 11, since implementation of this requirement for BSEP 1 meets the exposure-dependent safety limit for the 8x8 and 8x8R fuel, the staff finds it acceptable that the 8x8 and 8x8R fuel densification power spike penalty be deleted from the BSEP 1 Technical Specifications.

2.2.2. Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the CPR below the intended operating limit MCPR during Cycle 2 operation. The most limiting of these operational transients and the potential fuel loading errors have been analyzed by the licensee to determine which event could induce the largest reduction in the critical power ratio (ΔCPR).

The transients evaluated were the generator load rejection without bypass, feedwater controller failure at maximum demand, loss of a 100 °F feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6, 7 and Figure 2 of Reference 1 were assumed.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Listed below are the limiting ΔCPRs for the various fuel types at the specified cycle exposure. Also shown are the results of the maximum vessel pressure discussed in Section 2.4, and the fuel loading error (Section 2.6).

Transient	Limiting Exposure Time	ΔCPR	MCPR Operating Limit
Load Rejection Without Bypass	(BOC2) TO (EOC2-2000)	.03	*
	(EOC2-2000)TO (EOC2-1000)	.16	1.23
	(EOC2-1000) TO (EOC)	.21	1.28
Loss of 100°F Feedwater Heater	BOC2 TO EOC2	.14	*
Feedwater Controller Failure	(BOC2) TO (EOC2-2000)	.15	1.22
	(EOC2-2000) TO (EOC2-1000)	.09	*
	(EOC2-1000) TO (EOC)	.05	*
Condition			
Rod Withdraw Error	(BOC2) TO (EOC2)	.14	*
Fuel Loading Error	NEW FUEL LOADING ERROR ANALYSIS WITH 0.02 PENALTY	-	*
Overpressurization (MSIV Closure)	Peak Vessel Pressure is 1250 psig		

*Not Limiting

Addition of the most severe ΔCPR to the safety limit (1.07) gives the appropriate operating limit MCPR for each fuel type. This sum will assure that the safety limit MCPR is not violated.

We have determined that the operating limit MCPRs listed above are acceptable for Cycle 2 operation at the BSEP 1.

2.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 8. As specified in

Reference 8, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the overpressure design limit (1375 psi) to allow for the failure of at least one valve. Therefore the limiting overpressure event as analyzed by the licensee is acceptable.

2.5 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the Natural Circulation - 105% Rod Line intersection (which is the least stable physically attainable point of operation) are below the 1.0 stability limit.

Because operation in the natural circulation mode is restricted by Technical Specifications, there will be added margin to the stability limit. We find this is acceptable.

2.6 Accident Analysis

2.6.1 ECCS Appendix K Analysis

Input data and results for the ECCS analysis have been given in References 1, 3 and 4. The information presented fulfills the requirements for such analyses outlined in Reference 6.

We have reviewed the analyses and information submitted for the reload and conclude that the BSEP 1 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when (1) it is operated within the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Figures 3.2.1-1, -2, and -3 of Reference 2, and (2) it is operated at a Minimum Critical Power Ratio (MCPR) greater than or equal to 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss of Coolant Accident, as described in Section 2.2.2).

The licensee by letter dated August 3, 1977 requested revision of the Technical Specifications for Brunswick Unit No. 1 and Unit No. 2 to make the following revisions to the low pressure permissive set points:

(1) A change from 325 to 410 psig in the low pressure permissive set point for starting the RHR and CS pumps and for opening the injection valves; and

(2) A change from 325 to 310 psig in the low pressure permissive setpoint for closing the recirculation pump discharge valves.

As described in the Safety Evaluation Report Supporting Amendment No. 38 to Unit 2, the first change (#1 above) is acceptable since it results in earlier availability of ECCS equipment following a postulated LOCA once the change is made.

The second change (#2 above) is also acceptable for implementation because the LOCA analysis assumes a 285 psig to 335 psig pressure permissive set point, and the proposed set point (310 psig) is within this range.

Based on the foregoing, we conclude that this ECCS reevaluation fully meets the requirements of 10 CFR 50.46 and thereby satisfies the conditions of our Order for Modification of License dated March 11, 1977.

2.6.2 Control Rod Drop Accident

For BSEP 1, Cycle 2, the accident reactivity insertion curve (cold) did not satisfy the requirements for the bounding analyses described in Reference 5. Therefore, it was necessary for the licensee to perform plant and cycle specific analyses for the control rod drop accident. Results of the analyses indicate that the peak fuel enthalpy for this event would be less than 280 calories/gram, which is acceptable.

2.6.3 Fuel Loading Error

As discussed in Section 2.2.2, potential fuel loading errors involving misoriented bundles have been explicitly included in the calculation of the operating limit MCPR. Potential errors involving bundles loaded into incorrect positions have also been analyzed by a method which considers the initial MCPR of each bundle in the core, and the resultant MCPR was shown to be greater than 1.07. The GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff (Reference 10).

The analyses which have been performed for potential fuel loading errors for BSEP 1, Cycle 2, are acceptable for assuring that CPRs will not be below the safety limit MCPR of 1.07.

3.0 Physics Startup Testing

The safety analysis for the upcoming cycle is based upon a specifically designed core configuration. We have assumed that, after reloading, the actual core configuration will conform to the designed configuration. A startup test program can provide the assurance that the core conforms to the design. We require that a startup test program be performed and the minimum recommended tests are:

1. Visual inspection of the core using a photographic or videotape record.

2. A check of core power symmetry by checking for mismatches between symmetric detectors.

3. Withdrawal and insertion of each control rod to check for criticality and mobility.

4. Comparison of predicted and measured critical insequence rod pattern for nonvoided conditions.

We find the startup test program, (Reference 5), submitted by the licensee acceptable for Cycle 2 operation.

In the future, as a result of our ongoing generic review of BWR startup test, we anticipate requiring a description of each test sufficient to show how it provides assurance that the core conforms to the design. The description is anticipated to include both the acceptance criteria and the actions to be taken in case the acceptance criteria are not obtained.

In addition to the requirements above, we have requested that a brief written report of the startup tests be submitted to the NRC within 45 days of the completion of the tests.

References

1. "Supplemental Reload Licensing Submittal" for Brunswick Steam Electric Plant Unit No. 1 Reload 1, NEDO-24166, December 1978.

2. Letter, E. E. Utley (CPL) to T. A. Ippolito (NRC), dated January 17, 1979 transmitting Tech Spec changes.

3. Lead Plant LOCA Analysis: James A. FitzPatrick Nuclear Power Plant, July 1977 (NEDO-21662).

4. Plant Specific LOCA Analysis: Brunswick Steam Electric Plant, Unit No. 1, December 1978 (NEDO-24165).

5. Letter, E. E. Utley (CPL) to T. A. Ippolito, (NRC), dated March 16, 1979 transmitting Physics Startup Test Program.

6. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDO-24011-P, May 1977.

7. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.

8. Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-24011-P), April 1978.

9. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

10. Safety Evaluation of New GE Fuel Loading Error Methods, April 1978.

11. Letter, D. G. Eisenhut (NRC) to R. Gridley (G.E.), dated June 9, 1978, transmitting: Safety Evaluation of the General Electric Methods for the Consideration of Power Spiking due to Densification Effects in BWR 8x8 Fuel Design and Performance.

B. Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Suppression Pool - Reactor Building Vacuum Breakers

1.0 Introduction

By letter dated February 19, 1979, the licensee requested revisions to the technical specifications for BSEP Units 1 and 2 to allow implementation of permanent modifications to the BSEP suppression pool reactor building vacuum breaker lines during the 1979 refueling outages. These modifications were described in a letter dated December 29, 1978 and were designed to automate the containment isolation provisions required by General Design Criterion (GDC) 56 of Appendix A to 10 CFR Part 50.

2.0 Discussion

The present configuration of the primary containment (suppression chamber to reactor building) vacuum breakers consists of a single line penetrating the containment connecting to a Tee, each leg of which contains a locked closed manually operated butterfly valve in series with a positive acting normally closed check valve. The present technical specifications allow the butterfly valves to be manually opened in the unlikely event that negative pressures develop in the primary containment.

The proposed modification would install pneumatic actuators on the butterfly valves in the vacuum relief lines. Each butterfly valve will be designed to open automatically upon sensing a pressure differential of 0.5 psig between drywell (-0.5 psig) and reactor building (atmospheric pressure). The valves will be normally closed and will fail closed. Newly installed pressure differential switches (one for each valve) will provide actuation signals. Electrical power supply is from separate power divisions. Air supply for each valve will be from independent sources from a noninterruptible instrument air system. Control switches and valve position indication will be provided in the control room.

3.0 Evaluation

GDC 56 of Appendix A to 10 CFR Part 50 requires that each penetration of the primary containment that connects directly to the containment atmosphere shall be provided with two containment isolation valves, one inside containment and one outside containment, unless it can be demonstrated that the containment isolation provisions are acceptable on some other defined basis. The containment isolation valve outside containment may not be a simple check valve. The configuration proposed by the licensee involves two isolation valves outside containment, with a simple check valve outboard of a normally closed air actuated butterfly actuation valve. To locate one of the isolation valves inside the suppression chamber would be impractical, since the valve could become submerged by the suppression pool, would be exposed to pool swell impact loads following a postulated accident, and would be less accessible for testing and maintenance. The location of the check valve (i.e., vacuum breaker) outboard of the redundant isolation valve (butterfly valve) facilitates the operation of the vacuum breaker function. This arrangement is commonly used in similar containment designs for the same reasons. We conclude that this arrangement constitutes an acceptable "other defined basis" and, therefore, satisfies the requirements of GDC 56.

We have reviewed the proposed changes to the technical specifications for the Brunswick Steam Electric Plant that will implement the automatic operation of the butterfly isolation valves and thereby facilitate the vacuum breaker function. We conclude that the proposed modifications will not degrade the containment isolation provisions required by GDC 56, and will enhance the vacuum breaker function by not requiring operation action. We therefore find the proposed changes to the plant technical specifications acceptable.

C. Brunswick Steam Electric Plant, Units Nos. 1 and 2 Fire Protection Application

See the attached Supplement No. 1 to the Fire Protection Safety Evaluation Report.

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.

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: April 6, 1979

SUPPLEMENT NO. 1

TO THE

FIRE PROTECTION

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT

UNIT NOS. 1 and 2

DOCKET NOS. 50-325 and 50-324

DATE: April 6, 1979

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Introduction

Our SER on fire protection for the Brunswick facility issued November 22, 1977, identified numerous modifications being implemented to upgrade the facility's fire protection program. At that time, the Facility Operating Licenses (FOL NO. DPR-71 for BSEP Unit 1 and FOL No. DPR-62 for BSEP Unit 2) were amended to include a requirement for implementation of these modifications prior to return to operation for Cycle 2 for Unit 1 and Cycle 3 for Unit 2. By letter dated March 6, 1979, the licensee requested an extension for completion of certain modifications. Section A provides our evaluation for this proposed extension.

Our SER also identified certain incomplete items requiring further evaluation. Resolution of these items is described in Section B of this supplemental report.

A. Proposed Extension of Implementation Dates

A.1 Discussion

On November 22, 1977 the Commission issued Amendment No. 11 to FOL No. DPR-71 for Unit No. 1 and Amendment No. 37 to FOL No. DPR-62 for Unit No. 2 of the Brunswick Steam Electric Plant (BSEP). These amendments added a condition to the licenses which required completion of the modifications identified in Paragraphs 3.1.1 through 3.1.35 of the NRC's Fire Protection Safety Evaluation Report (SER) for the facility dated November 22, 1977.

Each license was conditioned to require completion of the modifications during the 1979 refueling outages and prior to return to power operation.

By letter dated January 17, 1979 the licensee informed the NRC that recirculation pump protection required by item 3.1.34 of the Fire Protection SER would not be completed as originally scheduled. In response to NRC letter dated February 21, 1979 the licensee advised by letter dated March 7, 1979, that certain other system modifications would not be completed on a schedule consistent with that required in the SER. As a result, a meeting was held with the licensee on March 9, 1979 to discuss the status of Fire Protection modifications of BSEP. Subsequently, the licensee submitted on March 15, 1979 a status report of fire protection modifications at the BSEP facility, including a completion date for those items delayed and proposed Technical Specifications. This submittal was supplemented with additional information on March 22 and March 29, 1979.

The specific modifications for which the licensee requests the completion date be amended are shown in the following table:

TABLE 3.1

MODIFICATION	SER NUMBER	APPLICABLE TS	REVISED COMPLETION DATE & TS EFFECTIVE DATE
Fire Barriers	3.1.1	3.7.8	7-27-79
Hose Stations - Water	3.1.3	Table 3.7.7 4-1*	
Service Water Building			4-11-79
Control Building			4-23-79
DG Building			5-11-79
Reactor Buildings			5-31-79
Hose Stations - AFFF	3.1.3	Table 3.7.7.4-1*	7-27-79
Ventilation Dampers	3.1.6	3.7.8	
Battery Rooms			6-1-79
DG and Control Building			7-27-79
Fire Doors - Loading Dock	3.1.7	NA	NA*
Drain Systems	3.1.10	3.7.8	7-27-79
Fixed Suppression	3.1.11	3.7.7.2*	7-27-79
Communication Equipment	3.1.17	NA	5-31-79
Sectionalizing Valves	3.1.19	3.7.7.1	7-27-79
Fire Water Piping	3.1.20	3.7.7.1	7-27-79
Portable Foam	3.1.21	3.7.7.5*	7-27-79
Smoke Removal	3.1.22	NA	
Control Building			6-15-79
Control Room			7-27-79
Emergency Breathing	3.1.23	NA	7-15-79
DG Oil Filter	3.1.27	3.7.7.5*	7-27-79
Lube Oil Piping	3.1.30	NA	7.1-79
Fuel Oil Impoundment	3.1.31	NA	7.15-79
Recirculation Pump	3.1.34	NA	NA*

*Proposed

A.2 Evaluation
Modification 3.1.1

The metal wall separating the computer rooms from the control room has been removed and replaced by a 3-hour barrier; however, upgrading of doors and installation of fire dampers in the ventilation system has not been completed.

To support the acceptability of return to power operation with this modification incomplete, the licensee states that detector capability is maintained for the computer room. In addition, hose stations have been added in the fire towers which will be functional and operable and the modification to provide for remote safe shutdown in the event of a control room fire will be complete. Additionally, portable water extinguishers have been made available. Thus, any fire in this area would not hamper the safe shutdown of either unit.

Fire barriers and hatch cover modifications required for certain office areas have also not been completed. To support the acceptability of return to power operation with those modifications incomplete, the licensee states that:

a) The need for the office area fire barriers will be eliminated since this area will be changed from an office/workroom area to an area containing logic and control cabinets for the analog pressure sensing modifications, thus making it another area of the general control room. The office areas will be removed to another separate fire area being constructed outside the control room area on the roof of the radwaste building. This modification will not be complete by the end of the Unit 1 outage, but protection for these areas has been improved by providing functional and operable hose stations in the fire towers, addition of detectors and portable extinguishers. In addition, the remote safe shutdown modification will be complete, assuring plant safe shutdown in the event of a fire in the areas that affect control room habitability.

b) The addition of a pyrocrete 3-hour barrier around the hatch covers in these areas will also not be completed by the end of the Unit 1 outage. The potential for a fire in the cable spreading room penetrating the present hatch covers has been decreased significantly by coating all cables in the cable spreading room and providing portable extinguishers and functional hose stations to serve the cable spreading room. This provides prompt detection and suppression capability and, in combination with the remote safe shutdown capability, ensures that the plant may achieve safe shutdown.

The licensee has committed to complete the above modifications by July 27, 1979. In the event these modifications are not complete on that date, the licensee has agreed to invoke the provisions of existing Technical Specification 3.7.8 PENETRATION FIRE BARRIERS.

Based on the foregoing, we conclude that sufficient compensatory measures are in place to assure that objectives stated in Section 2.0 of the SER are satisfied in the interim until such time as the modifications are complete. The action required by TS 3.7.8 will provide additional protection in the event the modifications are delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above fire barrier modifications is acceptable.

Modification 3.1.3

Hose stations in the Reactor Buildings, the AOG Building, Diesel Generator Building, Diesel Generator Tank area, Service Water Building, and Control Building will not be completed in all respects. The licensee has committed to complete the hose stations in the Service Water Building by April 11, 1979; in the Control Building by April 23, 1979; in the Diesel Generator Building including the switchgear, fan, DG and basement rooms, by May 11, 1979; and the Reactor Buildings by May 31, 1979: Additonally, AFFF hose stations in the Diesel Generator Building and the Diesel Generator Tank Area will be complete by July 27, 1979.

Hose stations in the AOG Mechanical room will not be completed until such time as the AOG is made operational.

To support the acceptability of return to power operation with these hose stations incomplete, the licensee states that with the exception of the AOG building and the DG tank area these hose stations will be functional and operable by the end of the Unit 1 and Unit 2 outages, providing the desired protection to these areas. The portions of the system that will not be complete by the end of the outages will be electrical in nature, such as alarms and annunciators. This will not degrade the capability of the hose stations to be used to counteract a fire in any of the areas.

With respect to the AOG room, the hose stations in this area will not be installed by the end of the Unit 1 outage. However, since this area is not required to be operable when the AOG system is not operational, a fire in this area would not have an effect on plant safe shutdown. Fire detectors in this area will notify the fire brigade of any fires which might occur.

With respect to the DG tank area, the other modifications to these tank rooms will be completed, assuring that isolation of one tank and its associated equipment can be maintained. This will provide assurance that a fire in any tank room will not prevent achieving safe shutdown of the plant. Hose stations are available in the area of the tank rooms which can be used by the fire brigade to extinguish and control a tank room fire.

In the event these hose stations are not completed on the dates indicated, the licensee has agreed to invoke the provisions of Technical Specification 3.7.7.4 Fire Hose Stations and has proposed a revision to Table 3.7.7.4-1 to include each of the above hose stations. Certain hose stations in the Control Building, the AOG Building, and the DG Building were inadvertently ommitted in the licensee's submittal. We have added these hose stations to Table 3.7.7.4-1.

Based on the foregoing, we conclude that sufficient fire protection is in place to assure that the objectives specified in Section 2.0 of the SER are satisfied in the interim until such time as the modifications are complete. The action required by T.S. 3.7.7.4 will provide additional protection in the event the modifications are delayed beyond the commitment dates. Accordingly, we conclude that the deferral of the above hose station additions is acceptable.

Modification 3.1.6

Ventilation damper modifications in the Battery Rooms will not be completed by the end of Unit 1 outage. The licensee has committed to complete the upgrading of these fire dampers by June 1, 1979. Ventilation damper modifications in the Control Building and the Diesel Generator Building will not be completed by the end of the Unit 1 outage. The licensee has committed to complete the upgrading of these fire dampers by July 27, 1979. Ventilation damper modifications in the AOG Building will not be completed until such time as the AOG is made operational.

To support the acceptability of return to power operation with these fire dampers incomplete, the licensee states that although the upgrading of the fire dampers from 1-1/2 hours to three hours in the supply duct between the cable spreading room and the battery room will not be complete by the end of the Unit 1 outage, coating of cable spreading room cables, addition of fire doors between battery rooms, addition of hose station, portable extinguishers, and addition of fire detectors provides reasonable assurance that fires in the cable spreading room will not likely become large enough to penetrate 1-1/2 hour fire dampers. In addition, although the upgrading of the 1-1/2 hour damper in the exhaust duct which leads to the mechanical equipment room will not be complete, hose stations, remote safe shutdown capability, separation of battery rooms and additional detectors will provide adequate assurance in the interim that a cable spreading room fire will not penetrate this barrier.

With regard to the Diesel Generator Building, even though the intended ventilation dampers will not be installed by the end of the Unit 1 outage, a number of modifications will have been completed that make a fire being carried through this area into adjoining rooms unlikely. The diesel generator exhaust silencers have been installed around the air intakes. In addition, fire hose capability is present in these areas, and fire detectors will provide prompt notification of any fire events. With these modifications, the absence of the ventilation dampers will not prevent the safe shutdown of the plant.

In the Control Building, the ventilation damper between the wash room and the control room will not be installed prior to Unit 1 startup. However, fire detection will be operational, as well as hose stations and remote safe shutdown. Thus, a fire in this area will not hamper safe shutdown of the plant.

The ventilation damper in the AOG Building will not be installed by the end of the Unit 1 outage, but is not required since the system is not used during plant operation. Fire detectors in the area will notify the fire brigade of any fires which might occur.

In the event these ventilation damper modifications are not completed on the dates indicated, the licensee has agreed to invoke the provisions of Technical Specification 3.7.8 Penetration Fire Barriers. Based on the foregoing, we conclude that sufficient compensatory measures are in place to satisfy the objectives of Section 2.0 of the SER in the interim until such time as the modifications are complete. The action required by TS 3.7.8 will provide additional protection in the event the modifications are delayed beyond the commitment dates.

Accordingly, we conclude that the deferral of the above ventilation damper modifications is acceptable.

Modification 3.1.7

By letter dated March 22, 1979, the licensee requested deletion of the commitment to upgrade the doors opening into the loading dock area of the Radwaste Building to 3-hour fire rated doors. The licensee states that the loading dock area is external to the plant area, and thus a fire in the immediate radwaste building area of the loading dock area would not have any effect on other plant areas required for safe shutdown of the plant. In addition, further review of fire protection requirements by the NRC staff and CP&L subsequent to submittal of the Program Review Report established that a fire involving all of the combustibles in the Radwaste Building would result in releases that are only a small fraction of 10 CFR Part 100 limits (Ref. CP&L submittals of October 14, 1977, and December 8, 1977). In addition, installation of a fixed suppression system in the area containing transient combustibles will be completed by July 27, 1979 (Ref. Modification 3.1.11).

Because these doors to the loading dock only separate the radwaste building from outside plant yard areas, upgrading of these doors will not provide any measurable increase in fire protection for the plant. These doors do not separate the radwaste building from any safety-related areas, and therefore the guidelines of Appendix A to BTP 9.5-1 are satisfied without upgrading of this door. Additionally, the analysis of radwaste fires is found acceptable as described in Section B.3.5 and shows that the consequences of such postulated fires are within the limits of 10 CFR Part 100. Based on the foregoing, we conclude that deletion of the commitment to upgrade the doors opening to the loading dock area of the Radwaste Building is acceptable and does not change the conclusions in our SER.

Modification 3.1.10

The backflow devices in the diesel generator drain system will not be complete by the end of the Unit 1 outage.

To support the acceptability of return to power operation with these backflow devices not installed, the licensee states that a fire in any one of the diesel generator rooms will be promptly detected and the fire brigade dispatched to extinguish the fire. Separation of the rooms has been improved by the addition of fire doors and barriers, and any fire fought in a diesel generator area will employ controlled amounts of water since the sprinkler systems will not be operational. In addition, a monthly verification of drain system flow path availability will be conducted until the backflow devices are installed. This, combined with the fact that drain backup is an unlikely occurrence in this area, provides assurance that a fire can be controlled and contained to a single diesel generator area.

The licensee has committed to complete the installation of these backflow devices by July 27, 1979. In the event these devices are not installed by July 27, 1979, the licensee has agreed to invoke the provisions of Technical Specification 3.7.8 Penetration Fire Barriers.

Based on the foregoing, we conclude that sufficient compensatory measures are in place to satisfy the objectives of Section 2.0 of our SER in the interim until such time as the backflow devices are installed. The action required by T.S. 3.7.8 will provide additional protection in the event the installation is delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above fire barrier modification is acceptable.

Modification 3.1.11

Fixed suppression systems for the Reactor Building, Service Water Building, Diesel Generator Building, and the radwaste area will not be operational by the end of the Unit 1 outage. The licensee has committed to complete these fixed suppression systems by July 27, 1979.

To support the acceptability of return to power operation with these fixed suppression systems incomplete, the licensee states that coverage of these areas with backup fire protection measures such as hose station is available.

In addition, the effects of a fire on safe shutdown in reactor building areas have been assessed and modifications performed to assure adequate divisional separation of electrical conduits and trays, thereby assuring that a fire in any one of these areas will not prevent the safe shutdown of the units.

In the Service Water Building significant modifications have been completed to improve the safety of these areas over the original plant installation. The backup fire suppression system in the form of hose racks will be functional and operable, and this in combination with existing fire detection capabilities will ensure rapid and effective control and extinguishment of any fire in this area. Thus, the safe shutdown of the plant will not be hampered by the lack of operability of the fixed suppression systems.

Although the fixed suppression system for the basement of the Diesel Generator Building area will not be operational by the end of the Unit 1 outage, the cables in the basement area have been coated with a fire retardant material. In addition, fire barriers to provide adequate fire protection in areas where redundant division cable trays or conduits are in proximity to each other will have been installed. Portable extinguishers have been added, and hose racks will be functional and operable by the fire brigade. This, in combination with the enclosures and fire doors on the next elevation and improvements in fire detection capability, will provide assurance that a fire in this area will not prevent achieving and maintaining safe shutdown conditions for the Brunswick Plant.

As in the basement area of the building (DG-1), the new sprinkler systems will not be operable in the diesel generator rooms. However, the hose racks will be functional and operable as a backup to the suppression systems. In addition, the upgrading of fire doors and barriers will be functional such that each area will be adequately separated from the other, and single division capability will be realized for safe shutdown of the unit.

Although the fixed suppression system in the radwaste area will not be completed prior to startup of Unit 1, hose racks, detectors and portable extinguishers are available in the area to combat a fire. Additionally, it has been shown that a fire in the radwaste area will not result in a release in excess of 10 CFR Part 100 limits. The adequacy of the analysis of radwaste fires is discussed in Section B.3.5 of this report.

In the event the above suppression systems are not completed by July 27, 1979, the licensee has agreed to invoke the provisions of Technical Specification 3.7.7.2 Spray and/or Sprinkler Systems and has proposed a revision to this specification to include each of the above systems.

Based on the foregoing, we conclude that sufficient fire protection is in place to satisfy the objectives of Section 2.0 of the SER in the interim until such time as the modifications are complete. The action required by T.S. 3.7.7.2 will provide additional protection in the event the modifications are delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above fixed suppression system modifications is acceptable.

Modification 3.1.17

The portable communication units will not be available by the end of the Unit 1 outage.

To support the acceptability of return to power operation with these communication units missing, the licensee states that additional sound powered phones have been supplied for fire brigade use. In addition, communications capability can be realized through the use of portable radios employed by the plant security forces.

The licensee has committed to supply the portable communication units by May 31, 1979.

We conclude that communication capability is sufficient in the interim until the required portable communication units are provided to support manual fire fighting activities, and to satisfy the objectives of Section 2.0 of the SER. Accordingly, the deferral of this provision for improved communications capability until May 31, 1979 is acceptable.

Modification 3.1.19

Most of the sectionalizing valves which were proposed to be installed will have been provided by the end of the Unit 1 outage.

To support the acceptability of return to power operation with certain sectionalizing valves not installed, the licensee states that those which will not be installed provide isolation between the two turbine building tapoffs, which will have no effect on safe shutdown. Additional sectionalizing valves for secondary feeds to the Diesel Generator, Reactor Building, Radwaste and Service Water buildings will not be complete, but water supply will be available to the buildings through other tapoffs, thus providing assurance that hose stations will be operable in these buildings in the interim.

The licensee has committed to complete the installation of all sectionalizing valves by July 27, 1979. In the event these modifications are not complete on that date, the licensee has agreed to invoke the provisions of existing Technical Specification 3.7.7.1 Fire Suppression Water System.

Based on the foregoing, we conclude that sufficient compensatory measures are in place to satisfy the objectives of Section 2.0 of the SER in the interim until such time as the sectionalizing valves are installed. The action required by T.S. 3.7.7.1 will provide additional protection in the event the modifications are delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above modifications is acceptable.

Modification 3.1.20

The second water supply to the diesel generator building will not be completed by the end of the Unit 1 outage.

To support the acceptability of return to power operation with this water supply incomplete, the licensee states that the additions of hose stations, upgrading of fire barriers, provision of separation of redundant division cables and other modifications provide assurance that in the interim, the capability to safely shut down the plant will not be prevented.

The licensee has committed to complete the installation of this water supply by July 27, 1979. In the event these modifications are not complete on that date, the licensee has agreed to invoke the provisions of existing Technical Specification 3.7.7.1 Fire Suppression Water System.

Based on the foregoing, we conclude that sufficient compensatory measures are in place to satisfy the objectives of Section 2.0 of the SER in the interim until such time as the second water supply to the diesel Generator Building is installed. The action required by T.S. 3.7.7.1 will provide additional protection in the event the modification is delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above modification is acceptable.

Modification 3.1.21

The two portable Aqueous Film Forming Foam (AFFF) concentrate stations will not be delivered prior to the end of the Unit 1 outage.

To support the acceptability of return to power operation with this modification incomplete, the licensee states that the capability of existing and modified systems in the diesel generator and yard areas is sufficient to assure that safe shutdown of the units can be achieved without the need for the portable AFFF systems in the interim.

The licensee has committed to complete the above modification by July 27, 1979. In the event this modification is not complete on that date, the licensee has agreed to invoke the provisions of proposed Technical Specification 3.7.7.5 Foam Systems, Action Statement C.

Based on the foregoing, we conclude that sufficient fire protection is in place to satisfy the objectives of Section 2.0 of the SER in the interim until such time as the modification is complete. The action required by T.S. 3.7.7.5 will provide additional protection in the event the modification is delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above modification is acceptable.

Modification 3.1.22

The Control Building exhaust cowl and ventilation dampers in the mechanical room supply air for smoke exhaust to improve the capability for smoke removal from the control room will not be installed by the startup of Unit 1.

To support the acceptability of return to power operation with these smoke removal modifications incomplete, the licensee states that the availability of the remote safe shutdown system will ensure safe shutdown of the plant in the event the control room becomes uninhabitable as a result of not having this modification completed.

The licensee has committed to complete the control building exhaust cowl by June 15, 1979 and the mechanical room dampers by July 27, 1979.

We conclude that the remote safe shutdown system will provide sufficient protection to satisfy the objectives of Section 2.0 of the SER in the interim until the required smoke removal modifications are completed. Accordingly, the deferral of these modifications until the above dates is acceptable.

Modification 3.1.23

The air compressor to provide 6-hour backup supply of air will not be complete by the end of the Unit 1 outage.

To support the acceptability of return to power operation without this air compressor, the licensee states that there are 30 air packs and 70 spare bottles available, assuring at least a 1-1/2 hour supply of air for fire brigade members and operators. In addition, the existing cascade recharging system will provide additional capacity, although not a full 6-hour supply.

The licensee has committed to complete the installation of this air compressor by July 15, 1979.

We conclude that sufficient emergency breathing air is available in the interim to support fire fighting operations and emergency shutdown actions until the required backup supply is installed. Accordingly, the deferral of this modification until July 15, 1979 is acceptable.

Modification 3.1.27

The AFFF suppression system for the air intake filters will not be completed by the end of the Unit 1 outage.

To support the acceptability of return to power operation with this modification incomplete, the licensee states that the oil retention system will be in place, limiting the possibility to widespread oil dispersion from the filters. In addition, the exhaust silencers have been removed, thus reducing the fire hazard by removing a potential source of fire. As well, hose racks, detection and dampers provide additional assurance that a fire in this area can be controlled and not prevent achieving safe shutdown of the plant.

The licensee has committed to complete the above modification by July 27, 1979. In the event this modification is not complete on that date, the licensee has agreed to invoke the provisions of proposed Technical Sepcification 3.7.7.5 Foam Systems.

Based on the foregoing, we conclude that sufficient fire protection is in place to satisfy the objectives of Section 2.0 of the SER in the interim until such time as the modification is complete. The action required by T.S. 3.7.7.5 will provide additional protection in the event the modification is delayed beyond July 27, 1979. Accordingly, we conclude that the deferral of the above modification is acceptable.

Modification 3.1.30

The original design for the lube oil piping barrier was determined to be unsuitable from a protection and also an operation and maintenance standpoint. The barrier is being redesigned and will not be completed prior to startup of Unit 1.

To support the acceptability of return to power operation with this modification incomplete, the licensee states that a fire in this area (turbine building accessway) affecting or including the fuel oil line would be promptly detected in the interim time until the modification is installed due to the number of people normally in this area. The only safety-related equipment affected by such a fire would be the battery cables which have been rerouted into this area to circumvent the cable spreading room fire. The other division battery will be available and capable of carrying the loads required for normal shutdown of either unit. Thus, the safety of the plant will not be decreased.

The licensee has committed to complete the above modification by July 1, 1979.

Based on the foregoing, we conclude that sufficient compensatory measures are in place to assure that the objectives stated in Section 2.0 of the SER are satisfied in the interim until such time as the lube oil piping barrier is complete. Accordingly, we conclude that deferral of the above modification is acceptable.

3.1.31 Modification

The modification to increase the height of the dike around the diesel fuel oil tank will not be completed prior to Unit 1 startup.

To support the acceptability of return to power operation with this modification incomplete, the licensee states that the fuel oil level will be lowered and maintained administratively at a level equal to 90% of the impoundment capacity. This will preclude the overflow of the impoundment due to water added to fight a postulated fire following rupture of a diesel fuel oil tank. This capability will provide an equivalent measure of safety until the dike is modified.

The licensee has committed to complete this modification by July 15, 1979.

We conclude that if the fuel oil level is maintained at 90% of the impoundment capacity in the interim until the impoundment modification is completed, fire protection for the yard area will satisfy the objectives stated in Section 2.0 of the SER. Accordingly, we conclude that the deferral of the above modification is acceptable.

3.1.34 Modification

In a letter dated January 17, 1979 the licensee stated that the installation of proposed reactor coolant recirculation pump motor fire protection systems would be postponed until such time as power generation with a de-inerted containment is authorized. Operating with a nitrogen inerted containment in the meantime, serves as protection by preventing the initiation of fires.

We will require the completion of the above modification prior to authorizing power generation with a de-inerted containment. On this basis, we concur that sufficient fire protection is available and conclude that the deferral of this modification is acceptable.

B. Resolution of Incomplete Items

B.1 Discussion

Our initial safety evaluation report (SER) pertaining to the reevaluation of fire protection at the Brunswick Steam Electric Plant-Unit 1 & 2 was issued by letter from A. Schwencer to CP&L dated November 22, 1977. In Section 3 of the SER, certain items were identified as incomplete and requiring further information from the licensee and evaluation by the staff. The SER also listed several modifications proposed by the licensee to improve fire protection.

The licensee in his letter(s) dated October 14, November 8, December 8, and December 16, 1977; March 30 and September 29, 1978; and March 15, 1979, submitted additional information in response to staff requests and positions to resolve these incomplete items.

We have reviewed the additional information submitted by the licensee to assure that for the incomplete items the fire protection guidelines identified in Section 2.0 of our SER are satisfied.

Section B.2 of this report summarizes the additional modifications proposed by the licensee and the remaining incomplete items. Section B.3 of this report provides the results of our evaluation of the resolution of the incomplete items.

B.2 Summary of Modifications and Remaining Incomplete Items

B.2.1 Modifications

The licensee has proposed the modifications summarized below. The implementation schedule for these proposed modifications is shown in Table B.2.1. A complete description of each proposed modification is given in the licensee's documentation.

B.2.1.1 Safety System Modifications

Various modifications are being performed on safety systems to provide an alternate shutdown capability independent of the cable spreading room and control room. These modifications include: re-routing of cables; addition of isolation, transfer, and control switches; addition of fuses in control circuits; addition of new instrument loops; re-aligning of certain equipment to different power sources; and relocation of certain instrument racks (B.3.2, B.3.6).

B.2.1.2 Sprinkler Heads

Sprinkler heads are being located at numerous locations where redundant safe shutdown cables are in proximity (B.3.2).

B.2.1.3 Procedural Changes

Certain procedures are being changed to define operator actions to restore power to certain equipment required for safe shutdown for fires affecting certain diesel generators (B.3.2).

Certain procedural changes are being made to administrative controls related to fire brigade training drills and physical examinations, and review of work requests (B.3.4).

A procedure will be developed defining actions to be taken to effect safe shutdown using the alternate shutdown capability (B.3.6).

B.2.1.4 Fire Barriers

Three-hour rated fire barriers are being provided at certain locations to protect redundant safe shutdown cables that are in proximity (B.3.2). Test results will be provided to demonstrate the adequacy of the "Pyrocrete" type fire barriers being installed in the various areas on a schedule consistent with Table B.2.1.

At locations where sprinkler heads are being added (2.1.2), Kao-wool type insulation or Marinite fire barriers will also be installed (B.3.2).

B.2.2 Incomplete Item-Protection for Redundant Cables Not in Close Proximity

The adequacy of protection provided for redundant safe shutdown cabling that is separated by greater than five feet horizontal distance is still under review by the staff. The results of this evaluation will be discussed in a later supplement.

B.3 Evaluation

The following provides our evaluation of the incomplete items. Numbers in parentheses following each heading refer to the sections of our previously issued SER which address these incomplete items.

B.3.1 Firestop Qualification (3.2.1)

Prior to the SER issuance, the licensee had not provided data to demonstrate the adequacy of the cellular concrete and mineral wool with Flamemastic type firestops which are installed at the Brunswick facility. Subsequent to the SER, the licensee has performed tests on these firestop designs to demonstrate their ability to sustain an ASTM E-119 exposure fire in excess of 3 hours duration. We have reviewed the test configurations, test procedure, and test results and find that these firestop designs will satisfactorily withstand a 3-hour fire severity without passage of flames to the unexposed side. We conclude that these firestop designs conform to the provisions of Appendix A to BSEP 9.5-1 and are, therefore, acceptable.

B.3.2 Effects on Safe Shutdown Where Redundant Cables are in Proximity (3.2.2)

Our SER noted that the plant contained several areas where a fire potentially could affect cables from redundant safety divisions due to: a common exposure fire; cables providing a pathway between redundant divisions; crossovers of redundant safety division cabling; or the large fire loading due to cable concentration; and that the licensee's fire hazards analysis did not evaluate the effects on safe shutdown for fires in these areas that damage cables from redundant safety divisions. The licensee has performed a cable study evaluating the effects on safe shutdown capability for fires that damage cabling in various areas. This study identified the following problem areas:

(a) locations where a fire may affect the alternate shutdown capability where this capability was being relied on for safe shutdown due to damage to normal shutdown cabling;

(b) locations where a fire may affect redundant safe shutdown equipment;

(c) locations where a fault due to a fire may prevent operation of safe shutdown equipment, yet the cabling is not required to be functional to achieve safe shutdown.

(d) locations where a fire potentially may affect two diesel generator units and the remaining two diesels, although in the same safety division, may not be able to supply required safe shutdown equipment.

Numerous modifications have been proposed by the licensee to correct the above deficiencies as summarized in the following:

(a) To preserve the alternate shutdown capability for control room or cable spreading room fires; various cables are being re-routed around these areas; isolation switches are being installed to isolate control circuitry in the cable spreading room and control room for required safe shutdown equipment; control switches are being installed to operate safe shutdown equipment remote from the control room; and for certain instrumentation, new instrument loops are being installed.

(b) To preclude a fire from causing loss of safe shutdown capability, at numerous locations sprinkler heads, as well as Kaowool or Marinite type barriers, are being installed where redundant safe shutdown cables are in proximity. At other locations, 3-hour fire barriers are being installed, instrument racks are being relocated, and cables are being re-routed.

(c) To remove faults that may affect the function of certain safe shutdown equipment, fuses are being installed in these circuits.

(d) Certain equipment is being re-aligned to different power sources, and manual transfer switches are being provided to allow powering certain valves from an alternate diesel to its normal supply.

(e) Certain procedural changes are being made to identify required operator actions to maintain operability of safe shutdown equipment.

Details on specific modificatons are contained in licensee submittals of November 8, 1977 and March 15, 1979 (Attachment 2).

We have reviewed the assumptions used in this cable study, the methodology of evaluating separation of safe shutdown cabling, and details on proposed modifications. The study evaluated the effects on safe shutdown where cabling from redundant divisions has less than five foot horizontal separation or if they cross each other. This may not be a sufficient criterion for minimum separation without cable damage to redundant divisions due to a fire. Heat buildup due to a large concentration of cabling, or heat concentrated in the plume from a fire, may damage cables at a higher elevation, even if redundant divisions are horizontally separated by greater than five feet. We are continuing to review the adequacy of protection for redundant cables that are horizontally separated by greater than five feet, and will address resolution of this in a later supplement to the SER.

We agree with all other assumptions and the methodology used in the cable study, and with the proposed modifications where it was determined that a fire may affect redundant safe shutdown cabling.

We have also reviewed the details of design changes to safety systems to assure that safety systems are not degraded by such modifications.

We find that where redundant cables are routed within five feet of each other horizontally, the effects on safe shutdown have been adequately analyzed, and the protection proposed for safe shutdown cables in proximity satisfies the objectives of Section 2.0 of our SER and is, therefore, acceptable. We will address the adequacy of protection for redundant cables that are separated by greater than five feet horizontally in a later supplement.

B.3.3 <u>Effects on Safe Shutdown For a Fire in the Cable Tunnels (3.2.3)</u>

The Cable Tunnels contain certain cables that are denoted as safety-related cables, however, the effects of their loss on safe shutdown capability had not been analyzed. The licensee has analyzed the effect of the loss of these cables due to a fire and found that they would have no adverse effects on safe shutdown capability. Numerous detectors are located in the tunnels, firestops are provided by coating a 10-foot length of the cable trays every 50 feet with a flame retardant coating, and manual hose stations are provided with access to the tunnels to afford manual suppression capability. We find that fire protection for the cable tunnels satisfies the objectives of Section 2.0 of our SER and is, therefore, acceptable.

B.3.4 <u>Administrative Controls (3.2.4)</u>

At the time our SER was issued, the licensee had provided a description of his administrative controls for fire protection which demonstrated that staff guidelines were met in all areas with the exception of the following:

(a) Fire brigade requirements did not require physical examinations to verify capability to perform strenuous activities;

(b) Refresher training was not provided for fire brigade members to repeat the classroom instruction.

(c) Work requests were not reviewed by an individual qualified in fire protection to assure proper fire protection provisions, and;

(d) A procedure had not been established for including off-site fire fighting organizations in fire brigade drills at least once per year.

To resolve these concerns, the licensee has subsequently proposed the following:

(a) Including fire brigade members in the respiratory protective program which requires a comprehensive medical examination, including medical history, X-rays, pulmonary function tests, and psychological consultation;

(b) Providing quarterly training sessions for brigade members to refresh training previously provided and to advance knowledge in various fire fighting aspects;

(c) Requiring that shift foremen, who are required to review and approve all work requests, be adequately trained to recognize fire hazards and fire protection impact of the work requests; these shift foremen are brigade members and serve as brigade chiefs; and

(d) Off-site fire companies will be invited to participate in an unannounced fire drill at least once per year.

We find that these proposed procedural changes resolve staff concerns identified in Section 6.0 of our SER. Subject to implementation of the above described changes, we find that administrative controls for fire protection at this facility conform to staff guidelines and satisfy the objectives of Section 2.0 of our SER and are, therefore, acceptable.

B.3.5 Radwaste Fires (3.2.5)

At the time our SER was issued, the licensee had not analyzed the effects of fires that consumed combustible materials containing radioactive material. The licensee has subsequently performed an analysis of fires in radwaste areas using the following conservative assumptions:

(a) Seven hundred twenty cubic feet of contaminated Class A material are stored prior to compaction and drumming for shipment offsite although normal procedures would be to drum this waste before such quantities are accumulated;

(b) The waste is assumed to have an activity of 0.3 millicuries/gram;

(c) All materials are assumed to be burned; and

(d) All radioactive material is assumed to be released as a ground level release.

The results of this analysis show that such conservatively postulated releases are a small fraction of 10 CFR Part 100 limits.

The staff has reviewed the licensee's analysis including assumptions and methodology used and the radionuclide concentrations used in the licensee's analysis and agree that the results are conservative for the postulated fire.

The staff performed a further analysis using the 5% meteorological data for accidents described in the Brunswick Safety Evaluation Report (Pgs. 2-8). This analysis showed a larger postulated offsite release than that calculated by the licensee, although the releases are still a small fraction of 10 CFR Part 100 limits.

We, therefore, conclude that the consequences of postulated fires in radwaste materials are acceptably small and that the fire protection in radwaste areas satisfies objectives of Section 2.0 of our SER and is, therefore, acceptable. As an added margin of safety to preclude offsite releases due to fires, despite the results of the above analyses, the licensee has proposed to install an automatic sprinkler system in the waste drumming area.

B.3.6 Alternate Shutdown Capability

As noted in Section 4.10 of our SER, the licensee will be providing an alternate shutdown capability independent of cabling and equipment in the cable spreading room and control room. This capability will include the following:

(a) Control capability at various remote shutdown stations;

(b) Equipment, control circuits, electrical distribution system, and emergency power supplies for the following systems: Service Water Systems (SWS), Reactor Core Isolation Cooling System (RCIC), Residual Heat Removal System (RHR) and Automatic Depressurization System (ADS).

Also included are the diesel generators and critical components, certain ventilation and room cooling systems, and critical instrumentation including reactor vessel pressure, reactor vessel water level, RHR flow, RHR heat exchanger differential pressure, RHR head spray flow, drywell pressure, and torus water temperature and level.

These systems are the same as those that would be used to perform safe shutdown on a loss of normal A.C. power. The adequacy of these systems to perform safe shutdown had previously been analyzed in the FSAR and found adequate.

(c) Procedures to perform the remote shutdown;

(d) spare fuses to replace fuses, if required, after isolating faults in the cable spreading room by operating isolation or transfer switches at the remote shutdown stations; and

(e) Communications equipment for communications between remote shutdown stations.

To assure that adequate personnel are available to perform a remote safe shutdown independent of fire fighting demands, technical specification changes are being issued with this report requiring that a minimum of five (5) operators be available onsite at all times for one unit operation and nine (9) operators for two unit operation. These operators are excluded from satisfying the technical specification requirement to maintain a five (5) man fire brigade onsite at all times.

We have reviewed the proposed systems and functions for the alternate shutdown capability and the available manpower and find these adequate to assure that safe shutdown can be achieved.

To provide this alternate shutdown capability, the licensee is performing various modifications as identified in Section 3.2 of this supplemental report. We have reviewed these modifications and find that:

(a) Changes to safety systems including addition of fuses, rerouting of cables, and installation of isolation switches, transfer switches, and control switches will not degrade these systems. These new devices are being designed, procured, installed, and tested to at least the same criteria as similar components in the existing safety systems.

(b) Opens, hot-shorts, or faults to cabling in the cable spreading room will not preclude operation of required safe shutdown equipment.

(c) Adequate fire barriers are provided to separate the alternate shutdown operability from cable spreading room or control room fires.

(d) Adequate measures have been taken to preclude inadvertent or unauthorized operation of isolation or transfer switches that would remove control capability from the control room. These measures include use of keylocked switches, strict administrative control keys, annunciation lights in the control room to indicate position of transfer switch (i.e., loss of lights indicates "local" control position), and routine operator checks of RTG Board indications.

(e) Adequate tests are being performed to verify the operability of new or modified control circuits and devices.

We find that the proposed alternate shutdown capability will include the required personnel, procedures, equipment, controls, and separation from cable spreading room and control room equipment to safely bring the reactor to cold shutdown conditions for fires that may occur in the cable spreading room or control room. We, therefore, conclude that, subject to implementation of the above described modifications, the alternate shutdown capability satisfies the objectives of Section 2.0 of our SER and is accordingly acceptable.

B.4 Conclusions

The licensee has performed a fire hazards analysis and has proposed certain modifications to improve the fire protection program as described in our SER of November 22, 1977. Additional modifications have been proposed by the licensee to resolve those issues identified as incomplete items in that SER. These additional proposed modifications are summarized in Section 2 of this report.

We find that the licensee's proposed modifications described herein are acceptable both with respect to the improvements in the fire protection program that they provide and with respect to continued safe operation of the facility.

In the report of the Special Review Group on the Browns Ferry Fire (NUREG-0050) dated February 1976, consideration of the safety of operation of all operating nuclear power plants pending the completion of our detailed fire protection evaluation was presented. The following quotations from the report summarize the basis for the Special Review Group's conclusion that the operation of the facility need not be restricted for public safety:

> "Fires occur rather frequently; however, fires involving equipment unavailability comparable to the Browns Ferry fire are quite infrequent (see Section 3.3 of [NUREG-0050]). The Review Group believes that steps already taken since March 1975 (see Section 3.3.2) have reduced this frequency significantly."

> "Based on its review of the events transpiring before, during and after the Browns Ferry fire, the Review Group concludes that the probability of disruptive fires of the magnitude of the Browns Ferry event is small, and that there is no need to restrict operation of nuclear power plants for public safety. However, it is clear that much can and should be done to reduce even further the likelihood of disabling fires and to improve assurance of rapid extinguishment of fires that occur. Consideration should be given also to features that would increase further the ability of nuclear facilities to withstand large fires without loss of important functions should such fires occur."

It is our conclusion that the operation of the facility, pending the implementation of all facility modifications, does not present an undue risk to the health and safety of the public based on our concurrence with the Browns Ferry Special Review Group's conclusions identified above, as well as the significant improvements in fire protection already made at the facility since the Browns Ferry fire. These include establishment of administrative controls over combustible materials and use of ignition sources, training and staffing of a fire brigade, and issuance of technical specifications to provide limiting conditions for operation and surveillance requirements for fire protection systems.

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.

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.

B.5.0 CONSULTANT'S REPORT

Brookhaven National Laboratory under contract to the NRC has provided the services of fire protection consultants who participated in the evaluation of the fire protection program. They have also participated in the preparation and review of the safety evaluation report issued on November 22, 1977. Their report, "Fire Protection in Operating Nuclear Power Stations, Brunswick Units 1 & 2", dated December 1977, discusses many items which have been addressed in this report. The consultants' recommendations which we have not totally adopted are discussed in Appendix "I". Our basis for not adopting these recommendations is given therein.

Additionally, the consultant participated in the evaluation of licensee information submitted March 15, 1979 to resolve incomplete items 3.2.1 and 3.2.2 from the SER. Appendix "II" provides the consultant's report on the results of his evaluation. All concerns raised by the consultant have been incorporated into this SER Supplement.

APPENDIX I

DISCUSSION OF CONSULTANT'S REPORT ON SER

Under Contract to Nuclear Regulatory Commission, Brookhaven National Laboratory has provided the services of fire protection consultants who participated in the evaluation of the licensee's fire protection program and in the preparation of the Safety Evaluation Report (SER) issued on November 22, 1977. Their report, "Fire Protection in Operating Nuclear Power Station Brunswick Units 1 & 2," dated December 1977 discusses several matters which have been addressed in the SER. The consultant's report contains recommendations which have for the most part, been implemented during our evaluation. The consultant's recommendations which we have not adopted, along with basis therefore, are identified herein.

Consultant's Comment:

1. Damage Limits

"SER Item 8.0(2) concludes that fire detection and suppression will minimize the effects of fire on safety-related systems. The consultant does not concur in this conclusion. There are usually several protective approaches that can be utilized for a given fire hazard, with each approach offering certain advantages and disadvantages in terms of limiting the fire extent, damage due to the fire suppression agents employed, reliability, and cost effectiveness. In most cases, it is technically possible to reduce the damage potential to a very low level, but the cost penalties often become severe. The fire protection systems that are being provided and recommended are to assure safe shutdown capability and will not necessarily minimize fire damage to all safety-related systems."

Staff Response:

The fire protection systems that are being provided and recommended are to assure safe shutdown capability and will not necessarily minimize fire damage to all safety-related systems in the strict sense of the meaning. The term "minimize" is used by the staff in the context of GDC 3 as interpreted in Appendix A to BTP 9.5-1. The staff agrees that the effects of fires have not been literally "minimized."

Consultant's Comment:

2. Control Valves

"SER Item 4.3.1(3) indicates that the position of fire protection system valves will be controlled by locks or seals with periodic inspections. Locking or sealing programs depend upon ongoing administrative controls that are subject to human failure. Locks can also prevent prompt water shut off if piping ruptures. It is recommended that electrical supervision be required on all control valves for fire protection systems protecting areas containing or exposing safety-related equipment. Reference letter dated July 13, 1977 to Mr. R. L. Ferguson from Mr. R. E. Hall."

Staff Response

The guidelines of Appendix A to BTP 9.5-1 allow electrical supervision locking, or sealing with tamper proof seals with periodic inspection as a means of assuring that valves in fire protection water systems are in the correct position. Valves on other systems in the plant are presently under administrative control. The plant Technical Specifications require a monthly check of all valves in the flow path to fire suppression systems. A review by the staff of Licensee Event Reports on all plants using such periodic checks indicates that valves being in the incorrect position has not been a significant contributor to valve related failures. Additionally, standing water as a result of failure of suppression system piping will not damage safety-related equipment due to curbs, drains, mounting of equipment above floor level, grating, and doorways. To date, the staff has not found any data that indicate that electrical valve supervision will significantly improve the availability of fire suppression systems in nuclear power plants.

Consultant's Comment:

3. Charcoal Filters

"SER Item 4.4.2 indicates that some charcoal filters are protected by automatic water deluge systems. Other charcoal filters are to be provided with fire detectors. The consultant recommends that further guidance be developed as to when and what type of protective systems are required for various charcoal filters."

Staff Response

The need for fire protection on charcoal filters is dependent on a number of factors: the probability of a fire, the potential quantity of radioactive material released and other factors which determine the doses resulting from such releases. These are all clearly within the jurisdication and expertise of the NRC staff and the need for fire protection will be determined by an analysis of such.

Most of the charcoal filters found in the plant are in ventilation systems and contain insignificant amounts of activity, and consequently do not present a problem. These filters also do not have the inherent capability to become an ignition source because of the low heat generation from radioactive decay due to the significant amount of contained radioactive material. However, these filters are considered to represent a possible exposure to safety related equipment and are evaluated on that basis for fire protection. Most of such have been found to be adequately separated from safety equipment and are encased in sheet steel structures and therefore, need only manual hose stations for protection.

Currently, Regulatory Guide 1.52 provides guidance on those charcoal filters which are susceptible to excessive heating due to decay of contained radioactive material. Accordingly, charcoal filters at Brunswick in the Standby Gas Treatment System are provided with automatically actuated deluge systems. Other filter systems do not require suppression systems due to insufficient decay heat to cause ignition.

In addition, the off-gas charcoal filters are of concern due to the quantity of contained radioactive material and the inherent possibility for ignition. A number of off-gas explosions and at least one charcoal fire have occurred none of which has proven to cause unacceptable releases. The problem is currently under review by the NRC staff during which the need for fire protection will be determined.

Consultant's Comment:

4. Smoke Removal

"SER Item 4.4.1 indicates that portable fans and ducts will be accepted as the means for removing smoke from many plant areas. Fires in electrical insulation can generate copious amounts of dense smoke which hamper fire control efforts by rendering the atmosphere toxic and reducing visibility in the area. Properly used, self-contained breathing apparatus can minimize the problem of toxic atmosphere, but little can be done to improve visibility except to remove the smoke from the building. Massive changes will be required in most areas of this plant if effective permanent smoke removal systems are required, the design of which would also have to include consideration of radioactivity releases. While portable fans and ducts may be effective for smoke control in many instances, there is some concern that they will not be sufficient for a major fire in some specific areas of this plant. It is recommended that this item be held open until better guidelines are developed for the evaluation of smoke generation potential and smoke removal system design."

Staff Response

Additional information and improved equipment would provide some benefit in the design and construction of fixed ventilation systems to be used for smoke removal in future plants. However, a massive plant redesign of current plant ventilation systems is not warranted because portable smoke removal equipment can be used in those plant areas with inadequate fixed smoke removal systems. Portable smoke removal units have been used in fire service for a sufficient length of time so that the limits on their use is well understood.

In plants where smoke removal is dependent on such equipment, smoke removal is not generally initiated until the room atmosphere is cooled sufficiently by fixed sprinkler operation or manual hose fogging to permit entry by fire fighting personnel. Ventilation prior to this time serves no purpose but to add oxygen to active fire sites. The current fire service portable smoke removal units have a sufficiently high temperature capability to remove smoke when the hot gases are cooled enough for fire brigade entry. The manual fire fighting consultants have made their evaluations of the fire fighting capabilities of a number of plants and have recommended use of the portable smoke exhaust systems. We require the licensees and applicants to develop prefire plans which include the proper use of ventilation equipment in each plant area of concern. This is addressed in our Administrative Controls review.

Consequently, there is adequate information at this time to continue to evaluate plant smoke removal capability. The use of fire suppression equipment, fire barriers and other fire protection measures is evaluated based on the need for immediate access into an area and the limitations imposed by the currently available portable smoke removal units. These concerns are evaluated on an area basis at each plant with due consideration of the advice of the manual fire fighting consultants.

Consultant's Comment:

5. Condenser Off-Gas Removal System

"The SER does not consider the discharge piping from the condenser off-gas removal system which is buried beneath the control building. It is recommended that an analysis of the potential consequences of an explosion in this system be required of the licensee."

Staff Response

Subsequent to our SER, the licensee provided the results of an evaluation of the off-gas and the potential for an explosion in this system to affect safe shutdown, as requested in the above consultant's comment. Based on our review of this information we find that:

(1) The piping, which is only 8 inches in diameter, passes approximately 30 feet below the control building;

(2) The control building is built on a pad of approximately 4 feet of reinforced concrete;

(3) Water seals will blow-out in an explosion to relieve pressure and dissipate energy;

(4) The off-gas system is designed to sustain the effects of an explosion and its resultant shock waves without damage;

(5) Several explosions and a fire have been experienced in this system, with no adverse effects on safety systems; and

(6) CP&L is planning to install a hydrogen-oxygen recombiner to preclude the potential for explosions.

We find that adequate measures are taken to preclude a hydrogen off-gas explosion from affecting safe shutdown equipment. After installation of the hydrogen-oxygen recombiner, the potential for hydrogen off-gas explosions will be greatly reduced.

Consultant's Comment:

6. Fire Detection System

"SER Item 42 indicates increased testing will be required for unsupervised wiring in the fire detection systems. Prompt fire detection is crucial for effective manual fire control in many areas containing safety-related systems and a detection system failure could go undetected between test periods. It is recommended that electrical supervision of these circuits be required."

Staff Position

The staff considered the need for requiring electrical supervision of this part of the fire detection system. The wiring from the fire detection device to the local control panel is electrically supervised. The probability of a failure in the electrical wiring from the local alarm panel to the control room is low in comparison to the remainder of the electrical components of the fire detection system (i.e., electrical panels and detection devices), which are already supervised. Periodic testing with a frequency of every 31 days will confirm the reliability of these circuits. This testing frequency is consistent with the testing frequency of safety related instrumentation. Therefore, we find that adequate measures are taken to detect detection alarm circuit failures and maintain availability of these systems.

Consultant's Comment:

7. Fire Pump Controls

"SER Item 4.3.1(2) does not address a concern of the consultant about the single pressure sensing line extending from the fire water system to the two fire pump controllers. If this line were to become plugged, or if the portion of water system to which the line is connected were valved off under pressure, neither fire pump would start automatically. It is recommended that a separate pressure sensing line be required for each fire pump."

Staff Response

In response to the consultant's concern, CP&L has proposed to install a redundant pressure sensing line so that isolation of one pump will not cause loss of starting signal to the remaining operable pump upon low pressure in the yard loop. We find that this resolves the consultant's concern.

Consultant's Comment:

8. Fire Pump Fuel Line

"SER Item 4.3.1(2) incidates that the licensee has proposed to install an automatic sensing device and shutoff valve in the fuel line of the diesel fire pump. The purpose of this installation is to prevent a broken fuel line from feeding a fire near the fire pump. However, if the sensing mechanism or valve should malfunction, it would shutoff the fuel supply to the diesel fire pump engine. The probability of such a malfunction appears to be greater than that of a broken fuel line, and this type of system is not required in any existing fire protection standards such as NFPA 20. It is recommended that this proposal be rejected."

Staff Response

The staff has not rejected the licensee's proposal because both fire pumps and their controllers are located in the water treatment building, and could be subject to damage by a fire in that structure. To preclude such an event, the licensee has proposed to provide automatic sprinklers and barriers to prevent flame impingement between the pumps and between the pumps and their controllers, and three-hour fire barriers between the building and the fuel tank. A flow switch and cutoff valve will be provided to detect a rupture in the supply line and shut off fuel flow to the diesel driven fire pump.

One of the primary means of reducing the effects of a combustible liquid fire is to reduce the quantity of fuel available. That is the purpose of the modifications proposed by the licensee. The staff believes the benefit to be gained by such a feature overrides the possibility of failure. A second pump is available in the event of such failures. A second pump may not be available if the cutoff valve is not installed and the fuel line were to rupture.

Consultant's Comment:

9. Fire Pump Controller

"The SER does not mention the 4160 volt controller for the electric fire pump. Controllers of this size are not approved for fire pump service by a recognized testing laboratory such as Underwriters Laboratories Inc. or Factory Mutual. However, NFPA 20 requires that they have certain features incorporated in their design. It is recommended that the licensee docket and justify any differences between the controller and the NFPA standard."

Staff Response

Subsequent to our SER, CP&L provided information identifying differences between their fire pump controller design and the guidelines of NFPA 20. This information indicated that all requirements of NFPA 20 are met with the exception that instead of a circuit breaker to serve as a disconnecting device, a thermal relay is provided in series with magnetic overcurrent trip relays to provide short circuit protection. The staff has evaluated this protection for the fire pumps and concludes that adequate alternative measures are provided to the NFPA 20 guidelines.

Consultant's Comment:

10. Seismic Damage

"The SER does not consider the effect of seismic damage on primary and backup fire protection systems, although Branch Technical Position 9.5-1 addresses this item for new plants. It is recommended that the potential that a seismic event could cause both a fire and damage to the protective features provided to cope with the fire be further evaluated. This should include fires started in nonseismically qualified systems or areas that spread to safety-related systems because protective systems are damaged. This item requires a policy decision by the NRC staff to remove it as a generic concern."

Staff Response

The guidelines of Appendix A to BTP 9.5-1 do not require fire protection systems at operating plants to be seismically designed. In developing the guidelines of Appendix A to BTP 9.5-1, the staff performed a study of the likelihood of a fire being caused by a seismic event concurrent with failure of fire suppression water systems as a result of a seismic event. The staff found that the contribution to overall risk from potential seismically induced fires is low and would not be significantly affected whether the fire protection system is designed to Category I requirements or not. Seismic qualification of the fire protection system was not a part of the evaluation of the Brunswick fire protection program.

7590-01

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. 23 and 47 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the license and Technical Specifications for operation of the Brunswick Steam Electric Plant, Units 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendment for BSEP Unit No. 1 changes the Technical Specifications to establish revised safety and operating limits for operation in fuel Cycle No. 2.

The amendments for BSEP Unit Nos. 1 and 2 change the Technical Specifications to allow implementation of permanent modifications to the suppression pool-reactor building vacuum breaker lines. In addition these amendments change the operating licenses for both units to allow revised implementation dates for certain modifications intended to improve the level of fire protection.

The applications for amendments comply 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.

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