March 15, 1985

Docket No. 50-261

Distribution	Docket file
~~NRC PDR~~	LPDR
ORB#1 RDG	Gray file (4)
HThompson	CParrish
GRequa	OELD
SECY	LHarmon
EJordan	PMcKee
TBarnhart (4)	WJones
DBrinkman	ACRS (10)
OPA, CMiles	RDiggs
JPartlow	FRMcCoy

Mr. E. E. Utley, Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated March 21, 1984, as supplemented by November 8, 1984.

The amendment revises the Technical Specifications in order to conform with the revised reporting requirements of 10 CFR 50, Sections 50.72 and 50.73.

In addition, we have corrected certain clerical errors made by the staff in Technical Specification issued in Amendment 85 and 87. Page 4.1-6 (Amendment 85) was inadvertently numbered 4.1-5 consequently the correct page (4.1-5 of Amendment 83) was inadvertently deleted. Therefore, we have enclosed the correct pages 4.1-5 and 4.1-6. Amendment 85 page 3.5-11 item 7 setting limits were approved as "The alarm is set with a method described in the ODCM", but Amendment 87 inadvertently reverted back to a previous limit while deleting a footnote. The enclosed and corrected page 3.5-11 incorporates approved changes from both amendments. We apologize for any inconvenience this may have caused you.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/s/SAVarga

Glode Requa, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 89 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL*	ORB#1:DL	BC-ORB#1:DL*	OELD*	AD:OR:DL	D:DL
CParrish	GRequa/ts	SVarga	03	GLainas	FMiraglia, for
02/09/85	02/14/85	02/13/85	~~02~~ 14/85	03/14/85	03/ /85

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P PDR

Docket No. 50-261

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Sincerely,

Glode Requa, Project Manager
Operating Reactors Branch #1
Division of Licensing

3/14/85

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2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD:OR:DL	D:DL
CParrish	GRequa	SVarga		GLainas	FMiraglia, for
02/ /85	02/11/85	02/ /85	02/ /85	02/ /85	02/ /85

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

March 15, 1985

Docket No. 50-261

Mr. E. E. Utley, Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated March 21, 1984, as supplemented by November 8, 1984.

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Sincerely,

Glode Requa

Glode Requa, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 89 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

Mr. E. E. Utley
Carolina Power and Light Company

H. B. Robinson Steam Electric
Plant 2

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Regional Radiation Representative
EPA Regiona IV
345 Courtland Street, N.E.
Atlanta, GA 30308

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29535

State Clearinghouse
Division of Policy Development
116 West Jones Street
Raleigh, North Carolina 27603

Attorney General
Department of Justice
Justice Building
Raleigh, North Carolina 27602

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
Route 5, Box 413
Hartsville, South Carolina 29550

James P. O'Reilly
Regional Administrator - Region II
U.S. Nuclear Regulatory Commission
Sutie 2900
101 Marietta Street
Atlanta, GA 30303

Mr. R. Morgan
General Manager
H. B. Robinson Steam Electric Plant
Post Office Box 790
Hartsville, South Carolina 29550

Mr. Dayne H. Brown, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
P.O. Box 12200
Raliegh, North Carolina 27605

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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-23

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

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated March 21, 1984, as supplemented by November 8, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-23 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 , 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 its issuance except for the following sections of the Radiological Effluent Technical Specifications which are effective prior to start-up from the Cycle 11 refueling outage: 3.5-2, 3.5-3, 3.9-1, 3.9-2, 3.9-3, 3.9-6, 3.9-7, 3.16-2, 3.16-4, 3.16-5, 3.17-1, 3.17-2, 3.17-3, 3.17-5, 6.9-10 and 6.9-11.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 15, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 89 FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

A. Upon receipt, revise the Appendix A Technical Specifications by adding the pages noted below:

Remove Pages	Insert Pages
3.1-4	3.1-4
3.1-13	3.1-13
3.3-5	3.3-5
3.3-5a	3.3-5a
3.4-2	3.4-2
3.5-11	3.5-11
3.5-18	3.5-18
3.5-19	3.5-19
3.7-2	3.7-2
3.14-1	3.14-1
3.14-2	3.14-2
3.14-3	3.14-3
3.14-4	3.14-4
3.14-5	3.14-5
------	3.14-5a
3.15-1	3.15-1
3.15-2	3.15-2
4.1-5	4.1-5
4.1-6	4.1-6
4.2-4	4.2-4
4.2-5	4.2-5
4.2-6	4.2-6
4.2-7	4.2-7
4.2-9	4.2-9
4.9-1	4.9-1
6.5-5	6.5-5
6.5-7	6.5-7
6.5-7a	6.5-7a
6.5-8	6.5-8
6.5-9	6.5-9
6.5-10	6.5-10
6.6-1	6.6-1
6.7-1	6.7-1
6.9-9	6.9-9
6.10-1	6.10-1

B. Prior to start-up from Cycle 11 Core Refueling Outage revise the Appendix A Technical Specifications as follows.

Remove	Insert
3.5-2	3.5-2
3.5-3	3.5-3
3.9-1	3.9-1
3.9-2	3.9-2
3.9-3	3.9-3
3.9-6	3.9-6
3.9-7	3.9-7
3.16-2	3.16-2
3.16-4	3.16-4
3.16-5	3.16-5
3.17-1	3.17-1
3.17-2	3.17-2
3.17-3	3.17-3
3.17-5	3.17-5
6.9-10	6.9-10
6.9-11	6.9-11

EFFECTIVE AS OF DATE OF ISSUANCE

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2, and are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:

   1. Cooldown and depressurize the RCS or

   2. Heatup the RCS to above 350°F.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

3.1.4 Maximum Reactor Coolant Activity

The total specific activity in μCi/gram of the reactor coolant shall not exceed 1.0 μCi/gram dose equivalent I-131 and $100/\overline{E}$ μCi/gram under all modes of operation. ($\overline{E}$ is the average of beta and gamma energy (MEV) per disintegration of the specific activity.)

Whenever the reactor is critical or the average reactor coolant temperature is greater than 500°F, with the specific activity of the primary coolant > 1.0 μCi/gram dose equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity > 1.0 μCi/gram dose equivalent I-131 exceeding 500 hours in any consecutive six month period, prepare and submit a Special Report to the Commission within 30 days indicating the number of hours above this limit.

With the specific activity of the primary coolant > 1.0 μCi/gram dose equivalent I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1.4-1, be in at least hot shutdown with T_{avg} < 500°F within 6 hours.

With the specific activity of the primary coolant > $100/\overline{E}$ μCi/gram, be in at least hot shutdown with T_{avg} < 500°F within 6 hours.

In any operating mode, with the specific activity of the primary coolant > 1.0 μCi/gram dose equivalent I-131 or > $100/\overline{E}$ μCi/gram, perform the sampling and analysis requirements of Item 1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission within 30 days. This report shall contain the results of the specific activity analyses together with the following information:

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

3.3.1.4 When the reactor is in the cold shutdown condition (except refueling operation when Specification 3.8.1.e applies), both residual heat removal loops must be operable. Except that either the normal or emergency power source to both residual heat removal loops may be inoperable.

a. If one residual heat removal loop becomes inoperable during cold shutdown operation, within 24 hours verify the existence of a method to add make-up water to the reactor coolant system such as charging pumps, safety injection pumps (under adequate operator control to prevent system overpressurization), or primary water (if the reactor coolant system is open for maintenance) as back-up decay heat removal method. Restore the inoperable RHR loop to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the loop to operable status.

b. If both residual heat removal loops become inoperable during cold shutdown operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere prior to the reactor coolant average temperature exceeding 200°F, restore at least one residual heat removal loop to operable status as soon as possible and follow the reporting requirements of Specifications 6.6.1 and 6.6.2.

3.3.2 Containment Cooling and Iodine Removal Systems

3.3.2.1 The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 2505 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. Two containment spray pumps are operable.

c. Four fan cooler units are operable.

d. All essential features, including valves, controls, dampers, and piping associated with the above components are operable.

e. The system which automatically initiates the sodium hydroxide addition to the containment spray simultaneously to the actuation of the containment spray is operable.

3.4.2 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram dose equivalent I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is > 0.10 μCi/gram dose equivalent I-131, be in at least HOT SHUTDOWN within 6 hours and cold shutdown within the following 30 hours.

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

3.4.3 If, during power operations, any of the specifications in 3.4.1, with the exception of 3.4.1.b and 3.4.1.d as it applies to 3.4.1.b above, cannot be met within 24 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. If any of these specifications cannot be met within an additional 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

3.4.4 With one auxiliary feedwater pump and/or essential features inoperable, restore that auxiliary feedwater pump and/or essential features to operable status within 72 hours, or;

a. Submit a Special Report to the Commission within 30 days outlining the cause of the inoperability and the action taken to return the pump and/or essential features to operable status, and;

b. Restore all three auxiliary feedwater pumps and their essential features to operable status within 7 days or be in at least hot shutdown within 6 hours.

3.4.5 With two auxiliary feedwater pumps inoperable, restore at least one inoperable auxiliary feedwater pump to operable status within 24 hours or be in at least hot shutdown within 6 hours.

TABLE 3.5-1 (Continued) (HBR-06)

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6. (Cont'd)	b. 480V Emerg. Bus Undervoltage (Degraded Voltage) Time Delay	Trip Normal Supply Breaker	412 Volts ± 1 Volt 10.0 Second Delay ± 0.5 sec.
7.	Containment Radioactivity High	Ventilation Isolation	The alarm is set with a method described in the ODCM.

* Initiates also containment isolation (Phase A), feedwater line isolation and starting of all containment fans.
** Initiates also containment isolation (Phase B).
*** Derived from equivalent ΔP measurements.

TABLE 3.5-5 (HBR-06)

(THIS TABLE APPLIES WHEN THE RCS IS > 350°F)

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

NO.	INSTRUMENT	1 MINIMUM CHANNELS OPERABLE	2 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 CANNOT BE MET
1	Pressurizer Level	2	See Item 9 Table 3.5-2
2	Auxiliary Feedwater Flow Indication (Primary Indication)		Note 1
	SD AFW Pump	1 per S/G	
	MD AFW Pump	1 per S/G	
3	Reactor Coolant System Subcooling Monitor	1	Note 2
4	PORV Position Indicator (Primary)	1	Note 3
5	PORV Blocking Valve Position Indicator (Primary)	1	Note 3
6	Safety Valve Position Indicator (Primary)	1	Note 3

Note 1: The three AFW lines from the MD AFW pumps and the three AFW lines from the SD AFW pump each contain one primary flow indicator (2 AFW flow paths per steam generator for a total of 6 AFW lines). These primary indicators are backed up by the narrow range steam generator level indications. If one or more of the direct AFW flow indicators becomes inoperable when the RCS is > 350°F, restore the indicator(s) to an operable status within 7 days, or prepare and submit a Special Report to the NRC within the following 14 days detailing the cause(s) of the inoperable indicator(s), the actions being taken to restore the indicator(s) to an operating status, the estimated date for completion of the repairs, and any compensatory action being taken while the indicator(s) is inoperable. The action required when any of the back up indications of AFW flow are inoperable, is described in Table 3.5-2.

(Notes 2 and 3 - See Next Page)

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

Note 2: If both channels of the RCS subcooling monitor become inoperable when the RCS is >350°F, restore at least one channel to an operable status within 7 days, or prepare and submit a Special Report to the NRC within the following 14 days detailing the cause(s) of the inoperable channels, the actions being taken to restore at least one channel to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while both channels are inoperable.

Note 3: The Pzr PORVs and Pzr PORV blocking valves both incorporate limit switches for the direct (primary) means of position indication. The back up method of position indication consists of PRT pressure and a temperature element in a common line downstream of the valves. The Pzr safety relief valves incorporate a vibration monitoring system as the primary method of valve position indication. The back up method of position indication consists of a temperature element downstream of each valve and PRT pressure. If the primary method of position indication for either the Pzr PORVs, Pzr PORV blocking valves, or Pzr safety relief valves becomes inoperable when the RCS is > 350°F, restore the primary method to an operable status within 7 days, or prepare and submit a Special Report to the NRC within the following 14 days detailing the cause of the inoperable primary position indication method, the actions being taken to restore it to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while the primary position indication method is inoperable. If any of the back up methods of position indication for these valves becomes inoperable, it is to be repaired as soon as plant conditions permit.

e. Both batteries and the D.C. distribution systems are operable.

3.7.2 During power operation the following components may be inoperable:

a. Provided both diesel generators are operable, power operation may continue with the start-up transformer out of service for 24 hours without reporting to the NRC.

b. Power operation may continue with the start-up transformer out of service beyond 24 hours provided both diesel generators are operable and the reporting requirements of Specification 6.6.1 are followed.

c. Power operation may continue if the start-up transformer and one diesel generator is inoperable provided the reporting requirements of Specifications 6.6.1 and 6.6.2 are followed.

d. Power operation may continue for seven days if one diesel generator is inoperable provided the remaining diesel generator is tested daily to ensure operability and the engineered safety features associated with this diesel generator shall be operable.

3.14 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the operating status of the fire detection instrumentation, fire suppression systems, fire barriers, and to the administrative controls required for a comprehensive fire protection and prevention program. The requirements of these specifications shall apply to an area or areas when equipment in that area or areas is required to be operable as specified by other Limiting Conditions for Operation.

Objectives:

To assure the operability of Fire Protection Systems.

Specification:

3.14.1 Fire Detection and Actuation Instrumentation

3.14.1.1 As a minimum, the fire detection and actuation instrumentation for each fire detection zone shown in Table 3.14.1 shall be operable.

3.14.1.2 With the number of operable fire detection and actuation instruments less than required by Table 3.14.1:

a. For Fire Zones 24, 25A, 25B, 25C and 26 (inside Reactor Containment) initiate an inspection once per shift of the affected zone with particular emphasis on identifying any potential hazards for fire.

b. For all other fire zones, within one (1) hour increase the inspection frequency of the zone with the inoperable instrument(s) to at least once per hour.

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

Basis

Operability of the fire detection and actuation instrumentation ensures that adequate warning capability is available for prompt detection of fires and provides for the actuation of automatic isolation and suppression systems which protect various safety related areas of the plant. The capabilities are required in order to detect, locate, isolate and extinguish fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection and actuation instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

3.14.2 Fire Suppression Water System

3.14.2.1 The Fire Suppression Water System shall be operable with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the yard loop, and

b. An operable flow path capable of taking suction from the Unit 2 intake structure and transferring the water through distribution piping with operable sectionalizing, or isolation valves.

3.14.2.2 With less than the above required equipment operable:

Restore the inoperable equipment to operable status within seven days or prepare and submit a Special Report to the Commission within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

3.14.2.3 With no Fire Suppression Water System operable:

a. Establish a backup fire suppression water system within 24 hours and follow the reporting requirements of Specification 6.6.1, or

b. Proceed to hot shutdown within twelve hours and be in cold shutdown within the next 24 hours.

c. Prepare and submit a Special Report to the Commission within 30 days outlining the plans and procedures to be used to establish operability of the system.

3.14.3 Fire Water Pre-Action System

3.14.3.1 The Fire Water Pre-Action Systems in the first floor Auxiliary Building hallway above the instrument and service air compressor and the Containment Vessel Electrical Penetration Area shall be operable:

a. With no visible water leakage from the spray nozzles,
b. With the air supply to the system operable,
c. With automatic initiation logic operable, and
d. With the system aligned to deliver to the protection area.

3.14.3.2 With the Fire Water Pre-Action Systems in a condition of readiness less than required by the above:

a. For the Containment Vessel Electrical Penetration Area initiate an inspection once per shift with particular emphasis on identifying any potential hazards for fire.

b. For all other areas, within one (1) hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

c. Restore the system to operable status within 14 days or prepare and submit a Special Report to the Commission 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.

3.14.4 Fire Hose Stations

3.14.4.1 Each fire hose station in Table 3.14.2 shall be operable.

3.14.4.2 With a hose station in Table 3.14.2 inoperable:

a. Route an additional equivalent capacity hose to the unprotected area from an operable hose station within one hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. Restore the hose station to operable status within 14 days or prepare and submit a Special Report to the Commission 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.

3.14.5 CO_2 Fire Protection System

3.14.5.1 The CO_2 Fire Protection Systems for 1) the Diesel Generator Rooms and, 2) North and South Cable Vaults shall be operable, each:

a. With a complete bank (19 cylinders for the Diesel Generator Room and 18 cylinders in the North and South Cable Vaults) of fully charged CO_2 cylinders in service,

b. With the system aligned to deliver to the protected areas, and

c. With automatic initiation logic operable. For the Diesel Generators, this includes two dedicated heat detectors per room for CO_2 actuation.

d. A CO_2 cylinder shall be deemed fully charged if it contains not less than 90% of the full charge weight.

3.14.5.2 With any of the CO_2 Fire Protection Systems in a condition of readiness less than required by the above:

a. Within one (1) hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. Restore the affected system to operable status within 14 days or prepare and submit a Special Report to the Commission 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.

c. If a diesel generator CO_2 Fire Protection System is inoperable and the affected diesel generator is running, immediately post a continuous fire watch. A continuous fire watch shall be maintained until the CO_2 fire protection system is restored to operability or until the diesel generator has been shut down.

3.14.6 Halon Fire Protection System

3.14.6.1 The Halon Fire Protection System for the Cable Spread Room Emergency Switchgear Room and the Safeguards Room shall be operable:

a. With a complete bank (10 cylinders, 5 instantaneous and 5 extended discharge) of fully charged Halon cylinders in service.

b. With the systems aligned to deliver to the protected areas.

c. With automatic initiation logic operable.

d. A Halon Cylinder shall be deemed to be fully charged if it contains not less than 90% of its full charge pressure and not less than 95% of its full charge weight.

3.14.6.2 With the Halon Fire Protection System in a condition of readiness less than required by the above:

a. Within one (1) hour establish a continuous fire watch with fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. Restore the system to operable status within 14 days or prepare and submit a Special Report to the Commission 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.

Basis:

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, CO_2, Halon, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for immediate notification to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3.14.7 Fire Barrier Penetration Fire Seals

3.14.7.1 All penetration fire barriers protecting safety related areas shall be operable when equipment in those areas are required to be operable.

3.14.7.2 With the penetration fire barrier inoperable:

a. The operability of the fire detection systems providing coverage for the fire areas on either side of the penetration, as applicable, shall be verified within one hour.

b. If either of the detection systems are inoperable, a continuous fire watch shall be established on at least one side of the affected penetration within one hour.

c. Restore the inoperable fire barrier penetration(s) to operable status within 7 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperable penetration and plans and schedule for restoring the fire barrier penetration(s) to operable status.

Basis

The operability of the fire barrier penetration seals ensure that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not operable, verification of fire detection system operability is required to insure that prompt detection capability exists in the vicinity of the penetration barrier. Should an area detection system be inoperable, the fire watch will provide the required protection until the seal is restored to operable status.

3.15 CONTROL ROOM FILTER SYSTEM

Applicability

Applies to the Control Room filter system which is required for the safe operation of the plant. This system incorporates both HEPA filters and a charcoal adsorber bank.

Objective

To provide limiting conditions for operation which ensure the operability of the filter system during plant operation, such that normal operation or accidental plant conditions requiring operation of the system will not result in consequences more severe than those previously analyzed.

Specification

3.15.1 During all modes of operation, except cold shutdown, the Control Room filter system shall be capable of performing its intended function in the required manner, except as described below:

a. If the system is determined to be inoperable, it shall be returned to operable status within seven days or prepare a Special Report which shall be submitted to the Commission within the next 14 days. This report shall outline the cause of the inoperability, the corrective actions taken, and the plans and schedule for restoring the system to an operable status.

3.15.2 If the system is determined to be inoperable while the reactor is in cold shutdown, the system shall be made operable prior to reactor startup.

Basis

Operability of the Control Room filter system ensures that the Control Room will remain habitable during an accidental atmospheric radiation release to the extent that none of the occupants would receive a personnel radiation exposure in excess of 10 percent of the suggested limits in 10CFR100[1]. Because the system's protection is required only during low probability events, the system may be out of service for 7 days for repairs. Following this period, a Special Report detailing the status of the system will be submitted to the Commission. Since reactor startup should not commence without this system in service, the specification prohibits startup with the system inoperable.

(1) FSAR Section 6.4

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M* (3) R* (3)	B/W (2)	(1) Thermal Power calculations during power operations (2) Signal to ΔT; bistable action (permissive, rod stop, trips) (3) Upper and lower chambers for symmetric offset: monthly during power operations. When periods of reactor shutdown extend this interval beyond one month, the calibration shall be performed immediately following return to power.
2.	Nuclear Intermediate Range	S (1)	N.A	S/U (2)	(1) Once/shift when in service (2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A	S/U (2)	(1) Once/shift when in service (2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R	B/W (1) (2)	(1) Overtemperature -ΔT (2) Overpower -ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure	S	R	M	
8.	4 Kv Voltage	N.A	R	M	Reactor Protection circuits only

*By means of the movable in-core detector system

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
9.	Analog Rod Position	S (1,2)	R	M	(1) With step counters (2) Following rod motion in excess of six inches when the computer is out of service
10.	Rod Position Bank Counters	S (1,2)	N.A.	N.A.	(1) Following rod motion in excess of six inches when the computer is out of service (2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	D (1)	R	N.A.	(1) Bubbler tube rodded weekly
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Boron Injection Tank Level	W	R	N.A.	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Containment Pressure	D	R	B/W (1)	(1) Containment isolation valve signal
19.	Deleted by Amendment No.				
20.	Boric Acid Makeup Flow Channel	N.A.	R	N.A.	

(e) Unscheduled inspections shall be conducted in accordance with Specification 4.2.5.1.2 on any steam generator with primary-to-secondary tube leakage (not including leaks originating from tube-to-tube sheet welds) exceeding Specification 3.1.5.3.

All steam generators shall be inspected before returning to power in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineering safeguards, or a main steam line or feedwater line break.

4.2.1.1.5 Acceptance Limits

Definitions:

Imperfection is an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation equal to or greater than 20% of the nominal tube wall thickness.

Defect is an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which a degraded tube must be removed from service by plugging, because the tube may become defective prior to the next scheduled inspection of that tube. The plugging limit is 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the plugging limit for each 12 month period until the next inspection of the inspected steam generator.

4.2.1.2 Corrective Measures

All tubes that leak or are determined to have degradation exceeding the plugging limit shall be plugged prior to return to power.

4.2.1.3 Reports

1. After each inservice examination, the number of tubes plugged in each steam generator shall be reported to the Commission as a Special Report within 14 days after completion of tube plugging.

2. The complete results of the steam generator tube inservice inspection shall be included in the Operating Report for the period in which the inspection was completed.

Reports shall include:

a) Number and extent of tubes inspected
b) Location and percent of wall thickness penetration for each eddy current indication and any leaks.
c) Identification of tubes plugged.

3. All results in Category C-3 of Table 4.2.2 shall be reported to the Commission as a Special Report within 30 days and prior to resumption of plant operation. A written follow-up report, to be submitted within 90 days following completion of the startup test program, shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

BASIS

The inspection program is in compliance with Section XI of the ASME Rules for In-service Inspection of Nuclear Power Plant Components. It should be recognized that examinations in certain areas are desirable but impractical due to the state-of-the-art. The areas indicated for inspection represent those of representative stress levels and therefore will serve to indicate potential problems before significant flaws develop there or in other areas. As more experience is gained in operation of pressurized water reactors, the time schedule and location of inspection may be altered or, should new equipment and/or techniques be developed, consideration may be given to incorporate these into this inspection program.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most primary loop components except the reactor vessel. The reactor vessel presents special problems because of the radiation levels and the requirement for remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps[1] have been incorporated into the design and manufacturing procedures in preparation for nondestructive test techniques which may be available in the future.

The techniques used for in-service inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts during refueling periods.

The primary pressure boundary class 1 components covered by this inspection will include the primary reactor coolant system and branch lines 2" or greater from the reactor coolant system to the second design isolation valve. Credit is taken in the design of this plant for check valves.

Class 2 and Class 3 components were chosen based on Regulatory Guide 1.26 and ANSI N18.2 and N18.2a "Nuclear Safety Criteria for the Design of stationary Pressurized Water Reactor Plants."

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes for evidence of mechanical damage or progressive degradation. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Wastage-type defects will be minimized with proper chemistry treatment of the secondary coolant. If defects or significant degradations should develop in service, this condition is expected to be detected during inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit. Steam generator tube inspections by means of eddy current testing have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications.

4.2.2 Materials Irradiation Surveillance Specimens

The reactor vessel surveillance program includes eight specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation testing of specimens. The specimens are located about three inches from the vessel wall at the axial midplace and are spaced radially at 0°, 10°, 20°, 30°, and 40°.[1]

Capsule No. 1 is scheduled to be removed at the first region replacement. The exposure of this capsule leads the vessel maximum exposure by a factor of 2.1. Thus, this capsule provides information for approximately a four-year exposure to the vessel.

Capsule No. 2 is scheduled to be removed at the fourth region replacement. This capsule leads the vessel maximum exposure by a factor of 0.8 and thus will provide data for a four-year exposure to the vessel. This sample also contains weld metal which is not present in Capsule No. 1.

Capsule No. 3 leads the vessel maximum exposure by a factor of 2.2 and is scheduled to be removed after twenty years. Thus, sample No. 3 will provide data for an exposure to the vessel of approximately forty years.

Capsule Nos. 4 and 5 lag the maximum vessel exposure by factors of 0.7 and 0.5, respectively. Thus, Capsule No. 4, which is scheduled to be removed after thirty years, provides data for a vessel exposure of twenty-one years and Capsule No. 5, which is scheduled to be removed at forty years, provides data for a vessel exposure of twenty years.

In addition to the capsules discussed above, there are three spares. Two are located at the same location as Capsule No. 5 and one is located at the same location as Capsule No. 4.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling after each ten year inspection. At the fourth refueling after each ten tear inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods.

References

(1) FSAR, Section 4.4
(2) FSAR, Volume 4, Tab VII, Question VI.C

TABLE 4.2-2 (HBR-06)

STEAM GENERATOR TUBE INSPECTION
H. B. ROBINSON UNIT NO. 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (SG) $S=3(N/n)\%$ where: N is the number of steam generators in the plant = 3	C-1	Acceptable for Continued Service	N/A	N/A	N/A	N/A
	C-2	Plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued service	N/A	N/A
			C-2	Plug tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for continued service
					C-2	Plug tubes exc. plug limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
n is the number of steam generators inspected during an examination	C-3	Inspect all tubes in this SG, plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in each other steam generator not included in the inservice inspection program. Report results to NRC.	All other SGs are C-1	Acceptable for Continued Service	N/A	N/A
			Some SGs are C-2 but no additional SGs are C-3	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			Additional SG is C-3	Inspect all tubes in SG and plug tubes exceeding the plugging limit. Report results to NRC.	N/A	N/A

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, submit a Special Report to the Commission within 30 days.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should

6.5.1.2.4 Modifications that are determined to either constitute an unreviewed safety question, as defined in 10CFR50.59(a)(2), or a change to the Technical Specifications, shall be reviewed by the Plant Nuclear Safety Committee and submitted to the NRC for approval prior to implementation. All such modifications shall be approved by the Corporate Nuclear Safety Section prior to implementation.

6.5.1.2.5 Modifications which constitute changes to the facility as described in the FSAR shall also be reviewed by the Corporate Nuclear Safety Section. This review may be conducted after plant management approval, and implementation may proceed prior to completion of review.

6.5.1.3 Technical Specifications and License Changes

6.5.1.3.1 Each proposed Technical Specification or Operating License change shall be reviewed by the Plant Nuclear Safety Committee and submitted to the NRC for approval.

6.5.1.4 Review of Technical Specification Violations

6.5.1.4.1 All violations of Technical Specifications shall be investigated and a report prepared that evaluates the event and that provides recommendations to prevent recurrence. Such reports shall be reviewed by the PNSC and approved by the Plant General Manager or his designee and submitted to the Manager - Robinson Nuclear Project and to the Manager - Corporate Nuclear Safety.

6.5.1.5 Nuclear Safety Review Qualification

6.5.1.5.1 Individuals shall be designated by the Manager - Robinson Nuclear Project for the safety reviews of Specifications 6.5.1.1.2, 6.5.1.1.3, 6.5.1.2.1, and 6.5.1.2.2. These reviewers shall have a Bachelor of Science in engineering or related field or equivalent and two (2) years related experience.

6.5.1.6.5 A quorum of the PNSC shall consist of the Chairman, and three members, of which two may be alternates.

6.5.1.6.6 The PNSC activities shall include the following:

a. Perform an overview of Specifications 6.5.1.1 and 6.5.1.2 to assure that processes are effectively maintained.

b. Performance of special reviews, investigations, and reports thereon requested by the Manager - Corporate Nuclear Safety.

c. Annual review of the Security Plan and Emergency Plan.

d. Perform reviews of Specifications 6.5.1.1.6, 6.5.1.2.4, 6.5.1.3.1, and 6.5.1.4.1.

e. Perform review of all reportable events.

f. Review of facility operations to detect potential nuclear safety hazards.

g. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrences to the Manager - Robinson Nuclear Project, Manager - Corporate Nuclear Safety and the Manager - Corporate Quality Assurance.

h. Review of changes to the Process Control Program and the Offsite Dose Calculation Manual.

6.5.1.6.7 In the event of disagreement between the recommendations of the Plant Nuclear Safety Committee and the actions contemplated by the General Manager, the course determined by the General Manager to be more conservative will be followed. The Manager - Robinson

Nuclear Project and the Manager - Corporate Nuclear Safety will be notified within 24 hours of the disagreement and subsequent actions.

6.5.1.6.8 The PNSC shall maintain written minutes of each meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications; and copies shall be provided to the Manager - Robinson Nuclear Project and to the Manager - Corporate Nuclear Safety.

6.5.2 Corporate Nuclear Safety Section - Independent Review

The Corporate Nuclear Safety Section of the Corporate Nuclear Safety & Research Department shall provide independent review of significant plant changes, tests, and procedures; verify that reportable events are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer. Specific review subjects are defined in Specification 6.5.2.1.d.

6.5.2.1 The Manager - Corporate Nuclear Safety, under the Vice President - Corporate Nuclear Safety & Research, is charged with the overall responsibility for administering the independent review function as follows:

a. Approves selection of the individuals to conduct safety reviews under Specification 6.5.2.

b. Has access to plant records and operating personnel in performing independent reviews.

c. Prepares and retains written records of reviews.

d. Assures independent reviews are conducted on the following subjects:

(1) Written safety evaluations of changes in the facility as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report, and tests or experiments not described in the Safety Analysis Report that are completed without prior NRC approval under the provisions of 10CFR50.59(a)(1). This review is to verify that such changes, tests, or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as

defined in 10CFR50.59(a)(2). These reviews may be conducted after appropriate management approval, and implementation may proceed prior to completion of the review.

(2) Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or an unreviewed safety question pursuant to 10CFR50.59(c). Matters of this kind shall be referred to the Corporate Nuclear Safety Section by the Plant General Manager or by other functional organizational units within Carolina Power & Light Company prior to implementation.

(3) Proposed changes to the Technical Specifications or this operating license, prior to implementation.

(4) All reportable events.

(5) Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section, deems appropriate for consideration of which is referred to the Manager - Corporate Nuclear Safety Section, by the on-site operating organization or by other functional organizational units within Carolina Power & Light Company.

(6) Reports and minutes of the PNSC.

6.5.2.2 Results of Corporate Nuclear Safety reviews, including recommendations and concerns, shall be documented.

a. Copies of documented reviews shall be retained in the CNS files.

b. Recommendations and concerns shall be submitted to the plant General Manager and Manager - Robinson Nuclear Project within 14 days of determination.

c. A summation of Corporate Nuclear Safety recommendations and concerns shall be submitted to the Chairman/President; Executive Vice President - Power Supply and Engineering & Construction; Senior Vice President - Nuclear Generation; Manager - Robinson Nuclear Project; Vice President - Corporate Nuclear Safety & Research; Plant General Manager; and others, as appropriate on at least a bimonthly frequency.

d. The Corporate Nuclear Safety Review program shall be conducted in accordance with written, approved procedures.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for events requiring immediate notification:

a. The NRC shall be notified pursuant to the requirements of 10CFR50.72.

b. Each reportable event shall be reviewed in accordance with Specification 6.5.1.6.6 and submitted to the Manager - Corporate Nuclear Safety Section, and the Manager - Robinson Nuclear Project.

6.6.2 The following actions shall be taken for reportable events requiring a Licensee Event Report:

a. A report shall be submitted to the NRC pursuant to the requirements of 10CFR50.73.

b. Each reportable event shall be reviewed in accordance with Specification 6.5.1.6.6 and submitted to the Manager - Corporate Nuclear Safety Section, and the Manager - Robinson Nuclear Project.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10CFR50.72 shall be complied with.

b. The provisions of 10CFR50.36(c)(1)(i) shall be complied with.

c. The Safety Limit violation shall be reported to the NRC Region II within one hour and to the Manager - Robinson Nuclear Project and the Manager - Corporate Nuclear Safety Section within 24 hours.

d. A Safety Limit Report shall be prepared. The report shall be reviewed in accordance with Specification 6.5.1.6.6. This report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems, or structures; and (3) corrective action taken to prevent recurrence.

e. The Safety Limit Violation Report shall be submitted to the NRC Region II, Manager - Robinson Nuclear Project, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

6.9.2 Deleted

6.9.3 Special Reports

6.9.3.1 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

	Area	Reference	Submittal Date
a.	Containment Leak Rate Testing	4.4	Upon completion of each test
b.	Containment Sample Tendon Surveillance	4.4	Upon completion of the inspection at 25 years of operation
c.	Post-operational Containment Structural Test	4.4	Upon completion of the test at 20 years of operation
d.	Fire Protection System	3.14	As specified by limiting condition for operation
e.	Overpressure Protection System Operation	3.1.2.1.e	Within 30 days of operation
f.	Auxiliary Feedwater Pumps	3.4	Within 30 days after becoming inoperable

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records of facility operation covering time interval at each power level.

b. Records of principal maintenance activities, inspections, repair and replacement of principal items of equipment, related to nuclear safety.

c. Reportable Event Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak test and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

EFFECTIVE PRIOR TO START UP FROM CYCLE 11 REFUELING OUTAGE

3.5.2 Radioactive Liquid Effluent Instrumentation

Applicability

Applies to the radioactive liquid effluent instrumentation system.

Objective

To define the operating requirements for the radioactive liquid effluent instrumentation system.

Specification

3.5.2.1 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5-6 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.1.1 are not exceeded. The alarm/trip setpoints shall be determined in accordance with the ODCM.

3.5.2.2 With a radioactive liquid monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluent monitored by the affected channel, change the setpoint so it is acceptably conservative, or declare the channel not operable.

3.5.2.3 With less than the minimum number of radioactive liquid effluent monitoring instrumentation operable, take the action shown in Table 3.5-6.

3.5.2.4 The provisions of Specification 3.0 are not applicable.

3.5.3 Radioactive Gaseous Effluent Instrumentation

Applicability

Applies to the radioactive gaseous effluent instrumentation system.

Objective

To define the operating requirements for the radioactive gaseous effluent instrumentation system.

Specification

3.5.3.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.5-7 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.3.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

3.5.3.2 With a radioactive effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents, change the setpoint so it is acceptably conservative, or declare the channel not operable.

3.5.3.3 With less than the minimum number of radioactive effluent monitoring instrumentation channels operable take the action shown in Table 3.5-7.

3.5.3.4 The provisions of Specification 3.0 are not applicable.

3.9 RADIOACTIVE EFFLUENTS

3.9.1 Compliance With 10 CFR Part 20 - Radioactive Materials in Liquid Effluents

Applicability

Applies to radioactive material in liquid effluents released from the site to unrestricted areas.

Objective

To define the concentration limits of 10CFR20 for radioactive material in liquid effluents released to unrestricted areas.

Specification

3.9.1.1 The concentration of radioactive material in liquid effluents released at any time from the site to unrestricted areas (see Figure 1.1-1) shall be limited to the concentrations specified in 10CFR20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} μ Ci/ml total activity.

3.9.1.2 With the concentration of radioactive material in liquid effluents released from the site to unrestricted areas exceeding the above limits, without delay restore the concentration to within the above limits. In addition, notification must be made to the Commission in accordance with Specification 6.6.

3.9.1.3 In the event that the immediate action required by 3.9.1.2 above cannot be satisfied, the facility shall be placed in hot shutdown within 12 hours and in cold shutdown within the next 30 hours, and entry into the power operating condition shall not be made unless Specification 3.9.1.1 is met.

3.9.1.4 The provisions of Specification 3.0 are not applicable.

3.9.2 Compliance With 10 CFR Part 50 - Radioactive Materials in Liquid Effluents

Applicability

Applies to radioactive materials in liquid effluents released from the site to unrestricted areas.

Objective

To define the calculated dose limits of 10CFR50 for radioactive materials in liquid effluents released to unrestricted areas.

Specification

3.9.2.1 The dose commitment at all times to a member of the public from radioactive materials in liquid effluents released to unrestricted areas (See Figure 1.1-1) shall be limited:

a. During any calendar quarter to $\leq$1.5 mrem to the total body and to $\leq$5 mrem to any organ, and
b. During any calendar year to $\leq$3 mrem to the total body and to $\leq$10 mrem to any organ.

3.9.2.2 With the calculated dose commitment from the release of radioactive materials in liquid effluents exceeding any of the limits prescribed by Specification 3.9.2.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2.

3.9.3 Compliance With 10 CFR Part 20 - Radioactive Materials in Gaseous Effluents

Applicability

Applies to radioactive materials in gaseous effluents released from the site to unrestricted areas.

Objective

To define the dose rate limits for radioactive materials in gaseous effluents released to unrestricted areas.

Specification

3.9.3.1 The dose rate due to radioactive materials in gaseous effluents released from the site boundary (see Figure 1.1-1) shall be limited to the following:

a. For radionoble gases: $\leq$500 mrem/yr to the total body, $\leq$3000 mrem/yr to the skin, and

b. For I-131, I-133, and tritium, and for all radioactive materials in particulate form, inhalation pathway only, with half lives greater than 8 days: $\leq$1500 mrem/yr to any organ.

3.9.3.2 With the dose rate(s) exceeding the above limits, without delay decrease the release rate to within the above limits. In addition, a notification must be made to the Commission in accordance with Specification 6.6.

3.9.3.3 In the event that the immediate action required by 3.9.3.2 above cannot be satisfied, the facility shall be placed in hot shutdown

3.9.6 Compliance With 40 CFR Part 190 - Radioactive Effluents From Uranium Fuel Cycle Sources

Applicability

Applies to radioactive effluents from uranium fuel cycle sources.

Objective

To define the dose limits of 40CFR190 for radioactive effluents from uranium fuel cycle sources.

Specifications

3.9.6.1 The dose commitment to any member of the public, due to releases of licensed materials and radiation, from uranium fuel cycle sources shall be limited to $\leq$25 mrem to the total body or any organ except the thyroid, which shall be limited to $\leq$ 75 mrem over 12 consecutive months. This specification is applicable to Robinson Unit 2 only for the area within a five mile radius around the Robinson Plant.

3.9.6.2 With the calculated doses from the release of the radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.9.2.1.a, 3.9.2.1.b, 3.9.4.1.a, 3.9.4.1.b, 3.9.5.1.a, or 3.9.5.1.b, calculations should be made including direct radiation contributions from the reactor unit and from outside storage tanks to determine whether the above limits of Specification 3.9.6.1 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3.2.d, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the same request is complete.

3.9.6.3 The provisions of Specification 3.0 are not applicable.

Basis

Compliance With 10 CFR Part 20 - Radioactive Materials in Liquid Effluents

This specification is provided to ensure that the concentration of radioactive materials in liquid effluents released from the site to unrestricted areas will be less than the concentrations specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides the additional assurance that the concentrations of radioactive materials in bodies of water outside the site will result in exposures within the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radionuclide and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in

Objective

To define the operating requirements for the liquid holdup tanks.

Specification

3.16.2.1 The quantity of radioactive material contained in each of the following tanks shall at all times be limited to $\leq$10 curies, excluding tritium and dissolved or entrained noble gases.

a. A monitor tank
b. B monitor tank
c. C Waste Condensate tank
d. D Waste Condensate tank
e. E Waste Condensate tank
f. Any Outside temporary tank*

3.16.2.2 With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and the event should be described in the Semiannual Radioactive Effluent Release Report, Specification 6.9.1.d.

3.16.2.3 If Specification 3.16.2.2 is not completed within 48 hours a notification must be made to the Commission in accordance with Specification 6.6.

3.16.3 Gaseous Radwaste and Ventilation Exhaust Treatment Systems

Applicability

Applies to the gaseous radwaste and ventilation exhaust treatment systems.

* A temporary tank is defined as any tank having a capacity of $\geq$ 100 gallons used for the receipt or transfer of radioactive liquids.

Objective

To define operating requirements for the Waste Gas Decay Tanks.

Specification

3.16.4.1 The oxygen concentration in the four Waste Gas Decay Tanks should be limited to $\leq$ 4% by volume when the hydrogen concentration in the same tank exceeds 4% by volume. The hydrogen concentration in the four Waste Gas Decay Tanks should be limited to $\leq$ 4% by volume when the oxygen concentration in the same tank exceeds 4% by volume.

3.16.4.1.a When the concentration of oxygen in a Waste Gas Decay Tank is $>$ 4% but $\leq$ 6% by volume and the hydrogen concentration in the same tank is $>$ 4% by volume, or the concentration of hydrogen in a Waste Gas Decay Tank is $>$ 4% but $\leq$ 6% by volume and the oxygen concentration in the same tank is $>$ 4% by volume, restore one or both to $\leq$ 4% by volume within 48 hrs.

3.16.4.1.b When the concentration of oxygen in a Waste Gas Decay Tank is $>$ 6% by volume and the hydrogen concentration in the same tank is $>$ 4% by volume, or the concentration of hydrogen in a Waste Gas Decay Tank is $>$ 6% by volume and the oxygen concentration in the same tank is $>$ 4% by volume, immediately suspend all additions of waste gas to the affected tank and immediately commence efforts to lower the concentration of one or both to $\leq$ 4% by volume.

3.16.4.2 If the requirements of paragraph 3.16.4.1.a cannot be met within the 48 hour limit, submit a Special Report to the NRC within the following 14 days which outlines the cause of the occurrence, corrective actions taken to date, corrective actions which will be taken, and any compensatory actions being taken to minimize the potential hazard.

3.16.4.3 If the actions taken to comply with paragraph 3.16.4.1.b do not reduce the concentration of hydrogen and/or oxygen in the affected tank to $\leq$ 6% by volume within 24 hours, a notification must be made to the Commission in accordance with Specification 6.6. Once the concentration of hydrogen and/or oxygen in the affected tank is $\leq$ 6% by volume, paragraphs 3.16.4.1.a and 3.16.4.2 apply.

3.16.5 Waste Gas Decay Tanks (Radioactive Materials)

Applicability

Applies to the four Waste Gas Decay Tanks.

Objective

To define the operating requirements for the Waste Gas Decay Tanks.

Specification

3.16.5.1 The quantity of radioactivity contained in each Waste Gas Decay Tank shall at all times be limited to $\leq$ 6.0E5 curies noble gases (considered as Xe-133).

3.16.5.2 With the quantity of radioactive materials in any Waste Gas Decay Tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

3.16.5.3 If Specification 3.16.5.2 is not completed within 48 hours, a prompt notification must be made to the Commission in accordance with Specification 6.6.

3.17 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

3.17.1 Monitoring Program

Applicability

Applies to the radiological environmental monitoring program.

Objective

To define the requirements for implementation of the radiological environmental monitoring program.

Specification

3.17.1.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.17-1.

3.17.1.2 With the radiological environmental monitoring program not being conducted as specified in Table 3.17-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.e, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

3.17.1.3 With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.17-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a member of the public is less

* The methodology and parameters used to estimate the potential annual dose to a member of the public shall be indicated in this report.

than the calendar year limits of Specifications 3.9.2.1, 3.9.4.1, and 3.9.5.1. When more than one of the radionuclides in Table 3.17-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ . \ . \ . \geq 1.0$$

When radionuclides other than those in Table 3.17-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to a member of the public is equal to or greater than the calendar year limits of Specifications 3.9.2.1, 3.9.4.1, and 3.9.5.1. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3.17.1.4 With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.17-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.d, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

* The methodology and parameters used to estimate the potential annual dose to a member of the public shall be indicated in this report.

3.17.1.5 The provisions of Specification 3.0 are not applicable.

3.17.1.6 Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

3.17.2 Land Use Census

Applicability

Applies to the land use census.

Objective

To define the requirements for the conduct of the land use census.

Specification

3.17.2.1 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

3.17.2.2 With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.10.4.1, identify the new location(s) in the next Semiannual Radioactive Effluent Release report, pursuant to Specification 6.9.1.d.6.

Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.e.

3.17.3.3 The provisions of Specification 3.0 are not applicable.

3.17.3.4 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.e.

Basis

Monitoring Program

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initally specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLD). The LLDs required by Table 3.17-3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined

6.9.3.2 Special Radiological Effluent Reports

The special radiological effluent reports discussed below shall be the subject of written reports to the Regional Administrator of the NRC Regional Office within thirty days of the occurrence of the event.

a. Exceeding any of the limits prescribed by Specification 3.9.2.1, 3.9.4.1, and/or 3.9.5.1. This report shall include the following information:

   1. The cause for exceeding the limit(s)

   2. The corrective action(s) to be taken to reduce the releases of radioactive materials in the affected effluents (i.e. liquid, radionoble gas, and/or radioiodines, particulates, etc.) within the Specification and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

   3. If any of the limits of Specification 3.9.2.1 were exceeded, the report must include a statement that no drinking water source exists that could be affected or include the results of radiological impact on finished drinking water supplied with regards to the requirements of 40CFR141 Safe Drinking Water Act.

b. Exceeding any of the limits prescribed by Specification 3.16.1.1 and/or 3.16.3.1. This report shall include the following information:

   1. Identification of equipment or subsystem that rendered the affected radwaste treatment system not operable.

2. The corrective action(s) taken to restore the affected radwaste treatment system to an operable status.

3. A summary description of the action(s) taken to prevent a similar recurrence.

c. Exceeding the reporting level for environmental sample media as specified in Specifications 3.17.1.3. This report shall include the following information:

1. An evaluation of any environmental factor, release condition or other aspect which may have caused the reporting level to be exceeded.

2. A description of action(s) taken or planned to reduce the levels of licensed materials in the affected environmental media to below the reporting level.

d. Exceeding the limits prescribed by Specification 3.9.6.1. This report shall be made in lieu of any other report and shall include the following information:

1. The corrective action(s) to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits prescribed by Specification 3.9.6.1.

2. An analysis which estimates the dose commitment to a member of the general public from uranium fuel cycle source including all effluent pathways and direct radiation, for a 12 month period that includes releases covered by this report.

3. If the release conditions resulting in violation of 40CFR190 have not already been corrected, include a request for a variance in accordance with the provisions of 40CFR190 and include the specified information of 40CFR190.11(b).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

Introduction

Generic Letter (GL) 83-43, dated December 19, 1983, informed utilities of changes to Title 10 of the Code of Federal Regulations (10 CFR) with respect to notification requirements for operating nuclear power reactors and requested utilities to revise Technical Specifications to incorporate these changes. These changes involved revision to Section 50.72 of 10 CFR for immediate notification requirements and addition of a new Section 50.73 of 10 CFR for a revised Licensee Event Report System. Both of these changes became effective January 1, 1984. GL 83-43 provided a model Technical Specification showing the revisions which should be made in the "Administrative Control" and "Definitions" sections of Technical Specifications to incorporate these regulation changes.

Carolina Power and Light Company, the licensee for H. B. Robinson Unit No. 2, provided a response to GL 83-43 and forwarded proposed Technical Specification changes in order to incorporate the provisions of GL 83-43 by letters dated March 21, 1984, and November 8, 1984. The proposed Technical Specification changes of the November 8, 1984 letter superseded the proposed changes of the March 21, 1984 letter.

The licensee response to GL 83-43 and the proposed Technical Specification changes have been reviewed.

Evaluations and Conclusions

The licensee's response to GL 83-43 and the proposed Technical Specification changes forwarded by the November 8, 1984 letter meet the intent of GL 83-43 and are acceptable. The proposed Technical Specification changes forwarded by letter dated November 8, 1984 are approved as proposed.

The licensee's second submittal dated November 8, 1984 was largely due to Amendments 83, 84, and 85 issued subsequent to their March 21, 1984 submittal. The subsequent amendments affected pages of the reporting requirements as described in the licensee's November 8, 1984 forwarding letter. Minor changes of a clarification nature were also made as a result of the NRC review process. Therefore, no substantive changes were made by the licensee's November 8, 1984 resubmittal.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 15, 1985

Principal Contributors:

F. R. McCoy, RII