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Docket No. 50-261

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U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 59 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated November 5, 1980. The Technical Specifications changes are supported by the Safety Evaluation Report as transmitted to the Carolina Power & Light Company by letter dated April 18, 1980.

This amendment incorporates the requirements for implementation of the TMI-2 Lessons Learned Category "A" items. They specifically include the areas of emergency power supply requirements, valve position indication, instrumentation for inadequate core cooling, containment isolation, auxiliary feedwater systems, shift technical adviser and the implementation of programs to reduce leakage outside containment and to accurately determine airborne iodine concentrations.

One issue not addressed in this amendment but which was proposed by the licensee in response to NRC letter dated July 2, 1980, is the operability requirements of pilot operated relief valves (PORV) and PORV block valves. We are still reviewing this issue and will take appropriate action at a later date.

We have determined that the amendment does 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 amendment involves 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 or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

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AUG 2 4 1981

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Sincerely,
"ORIGINAL SIGNED BY"

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 59 to DPR-23
2. Notice of Issuance

cc w/enclosures:
See next page

CP 8/19/81

no legal objection to amdt and F.R. notice only

OFFICE	ORB 1	ORB 1	ORB 1	AD:OR	OELD		
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DATE	8/17/81	8/19/81	8/ /81	8/ /81	8/21/81		

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

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Carolina Power and Light Company

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116 West Jones Street
Raleigh, North Carolina 27603

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

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Resident Inspector's Office
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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. 59
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 November 5, 1980, 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 Appendix A, as revised through Amendment No. 59, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **AUG 2 4 1981**

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii	ii
3.1-2	3.1-2
3.1-3	3.1-3
-	3.1-3a
3.4-2	3.4-2
3.4-3	3.4-3
-	3.4-4
-	3.4-5
-	3.4-6
3.5-1	3.5-1
3.5-6	3.5-6
3.5-7	3.5-7
3.5-7a	3.5-7a
3.5-9	3.5-9
3.5-10	3.5-10
3.5-10a	3.5-10a
3.5-11	3.5-11
-	3.5-12
-	3.5-13
-	3.5-14
4.1-6a	4.1-6a
4.6-3	4.6-3
-	4.6-3a
4.8-2	4.8-2
-	4.8-3
-	4.17-1
-	4.18-1
6.2-1	6.2-1
6.3-1	6.3-1

3.1.1.2 Steam Generator

At least two steam generators shall be operable whenever the average primary coolant temperature is above 350^{o}F.

3.1.1.3 Pressurizer (Pzr)

a. At least one Pzr code safety valve shall be operable whenever the Reactor Head is on the vessel.

b. The Pzr including necessary spray and heater control systems shall be operable before the reactor is made critical.

c. Whenever the RCS temperature is above 350^{o}F or the reactor is critical:

 1. All three pressurizer code safety valves shall be operable. Their lift setting shall be maintained between 2485 psig and 2560 psig.

 2. At least 125KW of pressurizer heaters capable of being powered from an emergency power source shall be operable.

d. If the requirements of 3.1.1.3.c.2 are not met and at least 125KW of Pzr heaters capable of being powered from an emergency source cannot be provided within 72 hrs., commence a normal plant shutdown and cooldown to an RCS average temperature of less than or equal to 350^{o}F.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one-half hours. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Specification 3.1.1.1 requires that a sufficient number of reactor coolant pumps be operating to provide core cooling in the event that a loss of flow occurs. The flow provided in each case will keep DNB well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur.

The pressurizer is necessary to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve setpoint.(2) Below 350°F and 450 psig in the Reactor Coolant System (RCS), the Residual Heat Removal System can remove decay heat and thereby control system temperature. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than the capacity of a single valve. One valve therefore provides adequate defense against overpressurization of the RCS for primary coolant temperatures less than 350°F and two valves provide protection for any temperature.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from the complete loss of load[3] without a direct reactor trip or any other control. ASME Section III of the Code allows a maximum variation in the setpoint of 3 percent above the design set pressure.

The requirement that 125 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency power source provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The reactor cannot be made critical without water in all three steam generators, since the low-low steam generator water level trip prevents this mode of operation. Two operable steam generators are therefore adequate.

References

(1) FSAR Section 14.1.12
(2) FSAR Table 4.1-3
(3) FSAR Section 14.1.10

3.4.2 The specific activity of the secondary coolant system shall be $\leq$ 0.10 uCi/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 uCi/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 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^{o}F using normal procedures.

3.4.4 In the event that the number of channels of the Auxiliary Feedwater Initiation circuits falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirements shown in Column 3 of Table 3.4-1. The Auxiliary Feedwater System Automatic Initiation Setting Limits are shown in Table 3.4-2. If the setpoint is less conservative than the value shown in the Allowable Values column of Table 3.4-2, declare the channel inoperable and operation shall be limited according to the requirement shown in Column 3 of Table 3.4-1.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

The twelve main steam safety valves have a total combined rated capability of 10,068,845 lbs/hr. The total full power steam flow is 10,068,845 lbs/hr.; therefore, twelve (12) main steam safety valves will be able to relieve the total steam flow if necessary.[1] Following a loss of load, which represents the worst transient, steam flows are below the total capacity of the 12 safety valves. Therefore, over-pressurization of the secondary system is not possible.

In the unlikely event of complete loss of turbine-generator and offsite electrical power to the plant, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps operated from the diesel generators and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant.[2] The minimum amount of water in the condensate storage tank is the amount needed for at least two-hours operation at hot standby conditions. If the outage is more than two hours, deep well or Lake Robinson water may be used.

An unlimited supply is available from the lake via either leg of the plant Service Water System for an indefinite time period.

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10CFR, Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

References

(1) FSAR Section 10.3
(2) FSAR Section 14.2.5

TABLE 3.4-1

AUXILIARY FEEDWATER FLOW AUTOMATIC INITIATION*

NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1.	Stm. Gen. Water Level-low-low			
	a. Start Motor-Driven Pumps	2/Steam Generator	1/Steam Generator	Maintain Hot Shutdown
	b. Start Turbine-Driven Pump	2/Steam Generator	1/Steam Generator	Maintain Hot Shutdown
2.	Undervoltage-4KV Busses 1 & 4 Start Turbine-Driven Pump (15 Second Time Delay Pickup)	2 Per Bus	0	Note 1
3.	S.I. Start Motor-Driven Pumps	See Table 3.5-3, Item No. 1		
4.	Station Blackout Start Motor-Driven Pumps (40 Second Time Delay Prior to Starting MD AFW Pumps on Blackout Sequence)	2 Per Bus	0	Note 2
5.	Trip of Main Feedwater Pumps Start Motor-Driven Pumps	1/Pump	0	Note 2

*: This table is applicable whenever the RCS is >200^{o}F except Item 2. Item 2 is applicable only when the R is at normal operating temperature and the reactor is critical.

Note 1: 4KV Busses 1, 2, and 4 each have two undervoltage relays. One relay on each of the three busses provides an input to the reactor trip logic. Both relays on Busses 1 and 4 provide inputs to the SD AFW pump start logic. If the undervoltage relay on Busses 1 or 4 that provides the input to the reactor trip logic fails, follow the requirements of Table 3.5-2 Item 14 in addition to the following. If either 4KV undervoltage relay on Busses 1 or 4 fails, within 4 hours insert the equivalent of an undervoltage signal from the affected relay in the SD AFW pump start circuit and repair the affected relay within 7 days. If the affected relay is not repaired in the 7 days, then commence a normal plant shutdown to hot standby.

Note 2: Restore the inoperable channel to operable status within 48 hours. If the inoperable channel is not restored to an operable status within 48 hours, then commence a normal plant shutdown and cooldown to $\leq 200^{o}$F.

TABLE 3.4-2

AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION SETPOINTS

FUNCTIONAL UNIT	SETTING LIMIT
AUXILIARY FEEDWATER	
a. Steam Generator Water Level-low-low	$\geq$ 14% of narrow range instrument span each steam generator
b. Undervoltage - 4KV Busses 1 & 4	$\geq$ 70% of 4KV Busses 1 & 4 Normal Voltage
c. S.I.	See Table 3.5-3, Item No. 1 and Table 3.5-1
c. Station Blackout	See Table 3.5-1, Item No. 6

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to plant instrumentation systems.

Objective

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

3.5.1 The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-5.

3.5.3 In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 3 of Tables 3.5-2 through 3.5-4 and Column 2 of Table 3.5-5.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features.(1)

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; (b) defeating the $\Delta T/T_{avg}$ protection CHANNEL SET that is being fed from the NIS channel, and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Instrumentation to Access Plant Conditions During and Following an Accident

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

Reference

(1) FSAR Section 7.5
(2) FSAR Section 14.3
(3) FSAR Section 14.2.5
(4) CP&L Letter to the Directorate of Licensing dated October 23, 1973.

TABLE 3.5-1

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1.	High Containment Pressure (HI Level)	Safety Injection*	$\leq$ 5 psig
2.	High Containment Pressure (HI-HI Level)	a. Containment Spray** b. Steam Line Isolation	$\leq$ 25 psig
3.	Pressurizer Low Pressure	Safety Injection*	$\geq$ 1700 psig
4.	High Differential Pressure Between any Steam Line and the Steam Line Header	Safety Injection*	$\leq$ 150 psi
5.	High Steam Flow in 2/3 Steam Lines***	a. Safety Injection* b. Steam Line Isolation	$\leq$ 40% (at zero load) of full steam flow $\leq$ 40% (at 20% load) of full steam flow $\leq$ 110% (at full load) of full steam flow
	Coincident with Low T_{avg} or Low Steam Line Pressure		$\geq$ 541°F T_{avg} $\geq$ 600 psig steam line pressure
6.	Loss of Power		
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage) Time Delay	Trip Normal Supply Breaker	328 Volts $\pm$ 1 Volt .75 $\pm$ .25 sec.

TABLE 3.5-1 (Continued)

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 $\pm$ 1 Volt 10.0 Second Delay $\pm$ 0.5 sec.
7.	Containment Radioactivity High	Ventilation Isolation	$\leq$ 2 X Reading at the Time the Alarm is Set with Known Plant Conditions

* 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 3.5-2 (Cont'd)

NO.	FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
12.	Lo Lo Steam Generator Water Level	2	1	Maintain Hot Shutdown
13.	Underfrequency 4 KV System	2	1	Maintain Hot Shutdown
14.	Undervoltage on 4 KV System	2	1	Maintain Hot Shutdown
15.	Control Rod Misalignment Monitor****			
	a) Rod Position Deviation	1	0	Log individual rod positions once/hour, and after a load change >10% or after >30 inches of control rod motion
	b) Quadrant Power Tilt Monitor (upper and lower ex-core neutron detectors)	1	0	Log individual upper and lower ion chamber currents once/hour and after a load change >10% or after >30 inches of control rod motion

* For zero power physics testing it is permissible to take one channel out of service.

** When two of four power channels are greater than 10% full power, hot shutdown is not required.

*** When one of two intermediate range channels is greater than 10^{-10} amps, hot shutdown is not required.

**** If both rod misalignment monitors (a and b) are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93 percent of rated power in addition to the increased surveillance noted.

R.P. = Rated Power

TABLE 3.5-3

INSTRUMENTATION OPERATING CONDITIONS FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1.	SAFETY INJECTION			
	a. Manual	1	0	Cold Shutdown
	b. High Containment Pressure (Hi Level)	2	1	Cold Shutdown
	c. High Differential Pressure Between Any Steam Line And The Steam Line Header	2	1	Cold Shutdown***
	d. Pressurizer Low Pressure	2	1	Cold Shutdown***
	e. High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} or Low Steam Pressure	1/Steam Line 2 T_{avg} Signals 2 Pressure Signals	***** 1 1	Cold Shutdown****

TABLE 3.5-3 (Continued)

INSTRUMENTATION OPERATING CONDITIONS FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
2.	CONTAINMENT SPRAY			
	a. Manual*	2	0**	Cold Shutdown
	b. High Containment Pressure* (Hi-Hi Level)	2/set	1/set	Cold Shutdown
3.	LOSS OF POWER			
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage)	2/bus (a)	1/bus(b)	Maintain Hot Shutdown
	b. 480V Emerg. Bus Undervoltage (Degraded Voltage)	2/bus	1/bus	Maintain Hot Shutdown(c)

* Also initiates a Phase B containment isolation.
** Must actuate two switches simultaneously.
*** When primary pressure is less than 2000 psig, channels may be blocked.
**** When primary temperature is less than 547°F, channels may be blocked.
***** In this case the 2/3 high steam flow is already in the trip mode.
(a) During testing and maintenance of one channel, may be reduced to 1/bus.
(b) During testing and maintenence of one channel, may be reduced to 0/bus.
(c) The reactor may remain critical below the power operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps.

Amendment No. 59

TABLE 3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1.	CONTAINMENT ISOLATION			
	a. Phase A			
	i. Safety Injection	See Item No. 1 of Table 3.5-3		Cold Shutdown
	ii. Manual	1	0	Hot Shutdown
	b. Phase B	See Item No. 2 of Table 3.5-3		
	c. Ventilation Isolation			
	i. High Containment Activity	1	0	Containment shall not be purged and personnel shall not enter the containment
	ii. Phase A	See Item No. 1.a of Table 3.5-4		

TABLE 3.5-4 (Continued)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
2.	STEAM LINE ISOLATION			
	a. High Steam Flow in 2/3 steam lines coincident with Low T_{avg} or Low Steam Pressure	See Item No. 1 of Table 3.5-3		Cold Shutdown
	b. High Containment Pressure	See Item No. 1 of Table 3.5-3		Cold Shutdown
	c. Manual	1/Line	0	Hot Shutdown
3.	FEEDWATER LINE ISOLATION			
	a. Safety Injection	See Item No. 1 of Table 3.5-3		Cold Shutdown

TABLE 3.5-5

(THIS TABLE APPLIES WHEN THE RCS IS $> 350^{\circ}$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 $> 350^{\circ}$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 & 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.

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
32.	Loss of Power				
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480V Emerg. Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
33.	Auxiliary Feedwater Flow**** Indication	M	N.A.	R	
34.	Reactor Coolant System** Subcooling Monitor	M	R	N.A.	
35.	PORV Position Indicator***	N.A.	N.A.	R	
36.	PORV Blocking Valve*** Position Indicator	N.A.	N.A.	R	
37.	Safety Relief Valve Position*** Indicator	N.A.	N.A.	R	

** Instrumentation for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.3.b.
*** Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 Item 2.1.3.a.
**** Auxiliary Feedwater Flow Indication to Steam Generator NUREG Item 2.1.7.b.

S - Each Shift
D - Daily
W - Weekly
B/W - Every two weeks
A/R - After each refueling startup
M - Monthly
Q - Quarterly
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown
N.A. - Not applicable

4.6.3.4 The batteries shall be subjected to a load test once every five years. The battery voltage as a function of time shall be monitored to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

4.6.4 Pressurizer Heaters' Emergency Power Supply

The emergency power supply for the pressurizer heaters shall be demonstrated operable each refueling shutdown by transferring power from the normal to the emergency power supply and energizing the heaters.

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safety features equipment will function automatically in the event of a loss of all normal 480 V AC station service power.(1)

The test to ensure proper operation of engineered safety features upon loss of AC power is initiated by tripping the breakers supplying normal power to the 480 volt buses and initiating a safety injection signal. This test demonstrates the proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, operation of the diesel generators, and sequential starting of essential equipment. The test of the diesel protective bypass circuits is performed to verify their operability.

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and any faults are alarm indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

On-site emergency power is available from two emergency diesel-generator sets. Each engine-generator set consists of a Fairbanks-Morse Model 38TD8-1/8 engine coupled to a Fairbanks-Morse 3125 kva, 0.8 power factor, 900 RPM, 3 phase, 60 cycle, 480 volt generator. The units have a continuous rating of 2500 kW with a 2-hour overload capability of 2750 kW in any 24-hour period.

4.8.5 The surveillance requirements for auxiliary feedwater system automatic initiation shall be as stated in Table 4.8-1.

Basis

The monthly testing of the auxiliary feedwater pumps by recirculation will verify their operability. The capacity of any one of three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.[(1), (2), (3)]

Proper functioning of the steam turbine admission valve and the starting of the feedwater pumps will demonstrate the integrity of the steam driven pump. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps. Testing of the steam turbine auxiliary feedwater pump is not required during periods of cold shutdown when steam is not available. In this condition the pump is not required for plant safety.

References

(1) FSAR Section 10.4

(2) FSAR Section 14.1.11

(3) FSAR Section 14.2.5

TABLE 4.8-1

AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
AUXILIARY FEEDWATER			
a. Steam Generator Water Level--Low-Low	N.A.	See table 4.1-1, Item 11	
b. Undervoltage - 4kv busses 1 and 4	N.A.	R	R
c. S. I.	(all Safety Injection surveillance requirements)		
d. Station Blackout - E1 and E2 busses	N.A.	N.A.	R
e. Trip of Main Feedwater Pumps	N.A.	N.A.	R

4.17 Systems Integrity

4.17.1 A program shall be implemented to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

a. Provisions for preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

4.18 Iodine Monitoring

4.18.1 The capability will exist to determine the airborne iodine concentration in vital areas under accident conditions. This capability shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.2 ORGANIZATION

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

a. The shift complement shall consist of at least one shift foreman holding a Senior Reactor Operator's License, two control operators each holding a Reactor Operator's License, one additional shift member, and one shift technical advisor.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed ANSI N18.1-1971 with regard to the minimum qualifications for comparable positions.

6.3.2 The Environmental and Radiation Control Supervisor shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975. The Guide says that he shall have a bachelor's degree or equivalent in a science or engineering subject. Equivalent in this case is defined as follows:

(a) 4 years of formal schooling in science or engineering

(b) 4 years of applied radiation protection experience at a nuclear facility,

(c) 4 years of operational or technical experience/training in nuclear power, or

(d) Any combination of the above totaling 4 years.

This requirement is in addition to the requirement for five years of professional experience in applied radiation protection.

6.3.3 The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 59 to Facility Operating License No. DPR-23 issued to Carolina Power and Light Company (the licensee), which revised Technical Specifications for operation of the H. B. Robinson Steam Electric Plant, Unit No. 2, (the facility) located in Darlington County, South Carolina. The amendment is effective as of the date of issuance.

This amendment incorporates the requirements for implementation of the TMI-2 Lessons Learned Category "A" items. They specifically include the areas of emergency power supply requirements, valve position indication, instrumentation for inadequate core cooling, containment isolation, auxiliary feedwater system, shift technical adviser and the implementation of programs to reduce leakage outside containment and to accurately determine airborne iodine concentrations.

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

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

For further details with respect to this action, see (1) the application for amendment dated November 5, 1980, (2) Amendment No. 59 to License No. DPR-23, and (3) the Commission's letter dated August 24, 1981. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina 29550. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 24th day of August 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing