September 10, 1984

Docket No. 50-261

DISTRIBUTION		
Docket File	EJordan	WJones
NRC PDR	LHarmon	CMiles
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GRequa	WVJohnson	
DEisenhut	DBrinkman	
OELD	RBallard	
JNGrace	TBarnhart 4	
ACRS 10	SECY	

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. 83 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 October 24 as clarified by letters dated December 12, 1983 and March 28, 1984.

The amendment would change the Technical Specifications to incorporate Section 4.05 of the Westinghouse Standard Technical Specifications. In addition, miscellaneous other changes were made to correct errors, remove redundancy and provide consistency within the Technical Specifications.

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/GRequa

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

Enclosures:
1. Amendment No. 83 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

*See previous white for concurrence

ORB#1:DL	ORB#1:DL	*METB	*METB	*AD/MCET	*SSPB	*C-ORB#1:DL
CParrish	GRequa	GJohnson	BDLiaw	WVJohnson	DBrinkman	SVarga
9/7/84	9/7/84	8/ /84	8/ /84	8/ /84	8/ /84	8/ /84

OELD	AD:OR:DL
R Bachman	GLainas
9/7/84	9/ /84

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note typo's in dipo
WVJ

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8/14/84	8/17/84	8/ /84	8/21/84	8/31/84	8/20/84	9/5/84

OELD	AD:OR:DL
	GLainas
8/ /84	8/ /84

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

H. B. Robinson Steam Electric
Plant 2

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

State Clearinghouse
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Raleigh, North Carolina 27603

Attorney General
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Raleigh, North Carolina 27602

U.S. Nuclear Regulatory Commission
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. 83
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 October 24, 1983 and December 12, 1983, as supplemented March 28, 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. 83 , 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 #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 10, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 83 FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.5-9	3.5-9
4.1-1 thru 4.1.10a	4.1-1 thru 4.1-14
4.2.1 thru 4.2-25	4.2-1 thru 4.2-9
4.5.2 thru 4.5-6	4.5-2 thru 4.5-4

TABLE 3.5-2 (Cont'd)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

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
16.	Steam Flow/Feedwater Flow Mismatch	1	0	Maintain Hot Shutdown
17.	Low Steam Generator Water Level	1	0	Maintain Hot Shutdown

* 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

4.0 SURVEILLANCE REQUIREMENTS

Specified intervals may be adjusted plus or minus 25% to accomodate normal test schedules. Performance of any surveillance test outlined in these specifications is not required when the system or component is out of service as permitted by the Limiting Conditions for Operation. Prior to returning the system to service, the specified calibration and testing surveillance shall be performed.

4.0.1 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification

4.1.1 Calibration, testing, and checking of instrumentation channels shall be performed as specified in Table 4.1-1.

4.1.2 Sampling tests shall be conducted as specified in Table 4.1-2.

4.1.3 Equipment tests shall be conducted as specified in Table 4.1-3.

Basis

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequencies set forth are deemed adequate for reactor and steam system instrumentation.

Calibration

Calibration is performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels daily calibration against a thermal power calculation will account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, the minimum calibration frequencies set forth are considered acceptable.

Testing(1)

Minimum testing frequency is based on evaluation of unsafe failure rate data and reliability analysis. This is based on operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal. The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hr per channel.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = \frac{W - Q\left(\frac{W}{N-M+2}\right)}{W} = 1 - \frac{N!}{(N-M+2)!\ (M-1)!}(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999968, assuming a channel failure rate, λ, equal to 2.5×10^{-6} hr^{-1} and a test interval, W, equal to 720 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one month is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

<u>Reference</u>

(1) FSAR Section 7.2.2

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)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	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.	Radiation Monitoring System*	D	R	M	
20.	Boric Acid Makeup Flow Channel	N.A.	R	N.A.	

*Process systems radiation monitors and area radiation monitors.

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
21.	Containment Sump Level	N.A.	R	N.A.	
22.	Turbine Trip Setpoint**	N.A.	R	R	
23.	Accumulator Level and Pressure	S	R	N.A.	
24.	Steam Generator Pressure	S	R	M	
25.	Turbine First Stage Pressure	S	R	M	
26.	DELETED				
27.	Logic Channel Testing	N.A.	N.A.	M(1)	(1) During hot shutdown and power operations. When periods of reactor cold shutdown and refueling extend this interval beyond one month, the test shall be performed prior to start-up.
28.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
29.	4 kv Frequency	N.A.	R	R	
30.	Control Rod Drive Trip Breakers	N.A.	N.A.	M	
31.	Overpressure Protection System	N.A.	R	M	

**Stop valve closure or low auto stop oil pressure.

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibration	Test	Remarks
32.	Loss of Power				
	a. 480 Emerg. Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480 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	
38.	Pending - Refer to CP&L Letter Dated February 22, 1982				

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibration	Test	Remarks
39.	Steam/Feedwater Flow Mismatch	N.A.	R	M	
40.	Low Steam Generator Water Level	N.A.	R	M	

** 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 0578 Item 2.1.7.b.

S	-	At least once per 12 hours	Q	-	At least once per 92 days
D	-	At least once per 24 hours	S/U	-	Prior to each reactor startup if not performed in the previous seven (7) days
W	-	At least once per 7 days			
B/W	-	At least once per 14 days	R	-	At least once per 18 months
M	-	At least once per 31 days	N.A.	-	Not applicable

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	- Gross Activity (1)	Minimum 1 Per 72 hrs.	3 days
		- Radiochemical (2)	Monthly	45 days
		- Radiochemical for $\bar{E}$ Determination	1 per 6 mos. (6)(7)	6 months
		- Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days (7)	14 days
		- Isotopic Analysis for Iodine Inluding I-131, I-133 and I-135	a) Once per 4 hours (8) b) One sample (9)	
		- Tritium Activity	Weekly	10 days
		- Cl & O_2	5 day/week	3 days
2.	Reactor Coolant Boron	Boron concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days
4.	Boric Acid Tank	Boron concentration	Twice/week	5 days
5.	Boron Injection Tank	Boron concentration	Weekly (5)	10 days
6.	Spray Additive Tank	NaOH concentration	Monthly	45 days
7.	Accumulator	Boron concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron concentration	Prior to Refueling	NA*
9.	Secondary Coolant	Gross activity	Minimum 1 Per 72 hrs.	3 days
		Isotopic Analysis for Dose Equivalent I-131 Concentration	a) 1 per 31 days (10) b) 1 per 6 months (11)	
10.	Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly (3)	10 days
11.	Steam Generator Samples	Primary to secondary tube leakage	5 days/week	3 days

NOTES TO TABLE 4.1-2

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of µCi/gram

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) When iodine or particulate radioactivity levels excceed 10% of the limit in Specification 3.9.2.1, the sampling frequency shall be increased to a minimum of once each day.

(5) The boron concentration in the boron injection tank shall be checked immediately after any actuation of the safety injection system that might result in dilution of the boron concentration in the boron injection tank.

(6) Sample to be taken after a minimum of 2EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

(7) Samples are to be taken in the power operating condition.

(8) Sample taken at all operating conditions whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/ $\bar{E}$ µCi/gram. These samples are to be taken until the specific activity of the reactor coolant system is restored within its limits.

(9) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one hour period. Samples are required when in the hot shutdown or power operating modes.

(10) Sample whenever that gross activity determination indicates iodine concentrations are greater than 10% of the allowable limit.

(11) Sample whenever the gross activity determination indicates iodine concentrations are below 10% of the allowable limit.

NA* - Not applicable.

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown	NA*
2.	Control Rod	Partial movement of all full length rods	Every 2 weeks during reactor critical operations	20 days
3.	Pressurizer Safety Valves	Set point	Each refueling shutdown	NA
4.	Main Steam Safety Valves	Set point	Each refueling shutdown	NA
5.	Containment Isolation Trip	Functioning	Each refueling shutdown	NA
6.	Refueling System Interlocks	Functioning	Prior to each refueling shutdown	NA
7.	Service Water System	Functioning	Each refueling shutdown	NA
8.	Fire Protection Pump and Power Supply	Functioning	Monthly	45 days
9.	Primary System Leakage	Evaluate	Daily when reactor coolant system is above cold shutdown condition	NA
10.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days
11.	Critical Headers of Auxiliary Coolant System	100 Psig Hydrostatic Test	Every five years	6 years
12.	Turbine Steam Stop, Control, Reheat Stop, and Interceptor Valves	Closure	Monthly during power operation and prior to startup	45 days

TABLE 4.1-3 (Continued)
FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency	Maximum Time Between Tests
13.	Turbine Inspection	Visual, Magnaflux and Die Penetrant	Every five years	6 years
14.	Fans and associated charcoal and Absolute Filters for Residual Heat Removal Compartments (HVE-5a and 5b)	Fans functioning. Laboratory tests on charcoal must show $\geq$ 99% iodine removal. In-place test must show $\geq$ 99% removal of polydispersed DOP particles by the HEPA filters and Freon by the charcoal filters.	Once per operating cycle.	NA
15.	Isolation Seal Water System	Functioning	Each refueling shutdown	NA
16.	Overpressure Protection System	Functioning	Each refueling shutdown	NA
17.	Primary Coolant System check valves	Functioning	1. Periodic leakage testing[a][b] on each [c] valve listed in Table 3.1-1 shall be accomplished prior to entering reactor operation condition (1) after every time the plant is placed in the cold shutdown condition for refueling, (2) after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, (3) after maintenance, repair or replacement work is performed.	

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum test differential pressure shall not be less than 150 psid.

(c) More than one valve may be tested in parallel. The combined leakage shall not exceed 5.0 gpm. Redundant valves in each line shall not be tested in series.

TABLE 4.1-3 (Continued)

Check	Frequency	Maximum Time Between Test
	2. Whenever integrity of a pressure isolation valve listed in Table 3.1-1 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.	

4.2 PRIMARY SYSTEM SURVEILLANCE

Applicability

Applies to in-service structural surveillance of the reactor vessel and primary system boundary.

Objective

To assure the continued integrity of the primary system boundary.

Specification

4.2.1.1 Inservice Inspection of Steam Generator Tubes

4.2.1.1.1 Tube Inspection

Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.

4.2.1.1.2 Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the following basis:

(a) One steam generator shall be inspected during inservice inspection in accordance with the following requirements:

1. The inservice inspection may be limited to one steam generator on a rotating sequence basis. This examination shall include at least 9% of the tubes if the results of the first or a prior inspection indicate that all three generators are performing in a comparable manner.

2. When other steam generators are required to be examined by Table 4.2-2 and if the condition of the tubes in one or more generators is found to be more severe than in the other steam generators, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator or generators with the more severe condition.

(b) The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements specified in Table 4.2-2. The results of each sampling examination of a steam generator shall be classified into the following three categories:

<u>Category C-1</u>: less than 5% of the total number of tubes examined are degraded but none are defective.

<u>Category C-2</u>: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective <u>or</u> one tube to not more than 1% of the sample is defective.

<u>Category C-3</u>: More than 10% of the total number of tubes examined are degraded, but none are defective <u>or</u> more than 1% of the sample is defective.

In the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by the prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the nominal tube wall thickness.

(c) Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

(d) The tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded, plus additional tubes are required to satisfy the minimum sample size specified in Table 4.2-2. If any selected tube does not permit passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection. This information shall be included in the report required by Specification 4.2.1.3.2.

(e) During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.

(f) During subsequent inservice inspections, the tube inspection may be limited to certain areas of the tube sheet array and those sections of the tube lengths where imperfections were detected during previous inservice inspections.

4.2.1.1.3 Examination Method and Requirements

Steam generator tubes shall be examined in accordance with the method prescribed in Appendix IV, "Eddy Current Examination of Non-Ferromagnetic Steam Generator Heat Exchanger Tubes," as contained in ASME Boiler and Pressure Vessel Code - Section XI- "Inservice Inspection of Nuclear Power Plant Components."

4.2.1.1.4 Inspection Intervals

(a) Inservice inspections shall not be more than 24 calendar months apart, except that reduced or tightened inspection intervals shall be governed as specified in 4.2.1.4(c) and (d).

(b) The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 4.2.1.1.4(a), (c) or (d), as applicable, are not exceeded.

(c) If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals between inspections.

(d) If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 4.2-2 at 40 month intervals fall in category C-3, the inspection frequency shall be reduced to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.2.1.1.4(c) and the interval can be extended to a 40 month period.

(e) Unscheduled inspections shall be conducted in accordance with Specification 4.2.1.1.2 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 engineered 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 in accordance with Specification 6.9.2.a(3).

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 prompt notification of Specification 6.9.2.a prior to resumption of plant operation. The written follow-up 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 greater than 1" 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."

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][2]

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.

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 promptly reported to the Commission pursuant to Specification 6.9.2.a 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.

4.2.3 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 4.2-1

DELETED

TABLE 4.2-2
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 (S.G.) S=3(N/n)% where: N is the number of steam generators in the plant =3 n is the number of steam generators inspected during an examination	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
	C-3	Inspect all tubes in this S.G, 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 within 24 hours in accordance with Technical Specification 6.9.2.a(3).	All other S.G.s are C-1	Acceptable for Continued Service	N/A	N/A
			Some S.G.s C-2 but no additional S.G are C-3	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in S.G. and plug tubes exceeding plugging limit. Report to NRC within 24 hours in accordance with Technical Specification 6.9.2.a(3).	N/A	N/A

Containment Spray System

4.5.1.3 System tests shall be performed at each refueling interval. The test shall be performed with the isolation valves in the spray supply lines at the containment and spray additive tank blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

4.5.1.4 The spray nozzles shall be checked for proper functioning at least every five years.

4.5.1.5 The tests discussed in 4.5.1.3 and 4.5.1.4 will be considered satisfactory if visual observations indicate all components have operated satsifactorily.

Containment Fan Coolers

4.5.1.6 Each fan cooler unit shall be tested at intervals not to exceed one month to verify proper operation of all essential features including valves, dampers and piping.

4.5.2 Component Verification

4.5.2.1 When the reactor coolant pressure is in excess of 1,000 psi, it shall be verified at least once per 12 hours (from the RTGB indicators/controls) that the following valves are in their proper position with control power to the valve operators removed.

Valve Number	Valve Position
1- MOV 862 A&B	Open
2- MOV 863 A&B	Closed
3- MOV 864 A&B	Open
4- MOV 866 A&B	Closed

4.5.2.2 At monthly intervals during power operations each valve (manual, power operated, or automatic) in the safety injection (low and high pressure) and containment spray system flow paths that is not locked, sealed or otherwise secured in position shall be verified as correctly positioned.

Basis

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1][2][4]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked each shift and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupt the logic matrix output to the master relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the logic matrices to the master relay is complete is accomplished by use of an ohmmeter to check continuity. In addition, the active components (pumps and signal valves) are to be tested quarterly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The quarterly test interval is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), and that more frequent testing would result in increased wear over a long period of time.

Quarterly testing of valves is consistent with the requirements of ASME Section XI.

Quarterly testing of the safety injection pumps, residual heat removal pumps, containment spray pumps and the boron injection tank isolation valves is not required when in the cold shutdown condition. These components are not required for plant safety when the reactor is in cold shutdown and testing during this condition will result in unnecessary wear on the equipment.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

References

(1) FSAR Section 6.2
(2) FSAR Section 6.4
(3) FSAR Section 6.1
(4) CP&L report and supplemental letters of September 29, November 5, December 8, 1971, and March 20, 1972.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 83 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

By letters dated October 24 as clarified by letters dated December 12, 1983, and March 28, 1984, Carolina Power and Light Company (the licensee) submitted a request for a license amendment concerning Technical Specification surveillance requirements. The December 12, 1983 submittal reinstated two sections of the Technical Specification that were deleted by error by the original submittal; Section 4.5.2.1 (originally 2.5.2.7) and 4.5.2.2 (originally 4.5.2.8). The March 28, 1984 submittal corrected a typographical error and corrected an error in Section 4.2.1.1.4 Bases to conform with ASME Section XI. The purpose of the original (October 24, 1983) submittal was to make the Technical Specification conform to the ASME Section XI. Therefore, the additional submittals did change the intent of the original submittal.

The licensees proposed changes include:

1. Changes to delete detailed inservice inspection requirement from the current Technical Specifications and replace these with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code in accordance with 10 CFR 50.55a. This will maintain consistency and conformance with the Code.

2. Miscellaneous additional changes of administrative nature to make the Technical Specifications consistent with other sections, the FSAR or to correct errors.

Evaluation

ASME XI Surveillance Requirements

Section XI of the ASME Boiler and Pressure Vessel Code is referenced in 10 CFR 50.55a as the document that should govern examination and testing of code classed systems. The incorporation of the Section XI into the H. B. Robinson Technical Specifications would remove any inconsistency between the current H. B. Robinson No. 2 examination and testing programs and the ASME Section XI requirements. This would ensure that appropriate requirements are applied.

The staff has reviewed these changes and finds that:

The changes ensure that inservice examination and testing of safety related piping and components will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a.

This change includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice examination and testing activities.

Therefore, the staff finds these changes acceptable.

Turbine Trip Set Point

The Turbine Trip Set Point, listed as Item 22 in Table 4.1-1, Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels, was defined as a stop valve closure or low EH fluid pressure. The table has been revised to clarify the set point as being a stop valve closure or low auto-stop oil pressure.

This is a clarification in nomenclature and therefore acceptable.

Emergency Plant Portable Survey Instruments

Emergency Plant Portable Survey Instruments, listed as Item 26 in Table 4.1-1, Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels, has been deleted. This requirement is not governed by an LCO nor is it a protection instrument channel as defined in TS 1.4. The requirement for maintaining emergency monitoring equipment is defined in Section 6.3.1 of the Emergency Plan and plant procedure RST-003, "Emergency Kit Inventory."

Since the requirement for maintaining emergency monitoring equipment is defined in the Emergency Plant and plant procedure, and since it is not governed by an LCO, we find the change will remove redundancy while maintaining the requirement and therefore find the change acceptable.

Steam Flow/Feedwater Flow Mismatch

This change provides an additional limitation not currently included in the TS but are discussed in Section 7.2.1.1.1 of the FSAR, i.e., steam flow/feedwater flow mismatch in coincidence with low steam generator water level. This requirement was added to Tables 3.5-2 and 4.1-1 as items 16, 17, 39, and 40 contained on pages 3.5-9 and 4.1-6a. This is an additional limitation not previously included in the technical specifications.

This requirement is contained in Section 3 of the Standard Technical Specifications NUREG-0452, therefore, the addition is acceptable.

Test Frequency for Startup

The licensee's request, which resulted in Amendment 65, inadvertantly removed a phrase "if not performed in the previous seven (7) days" from the surveillance frequency notation of Table 4.1-1.

This change reinstated an inadvertently removed phrase and therefore is acceptable.

Control Room Filters

The laboratory testing criteria stated in Item 14 of Table 4.1-3, Frequencies for Equipment Tests, conflicts with that in Section 4.15.1.d for the charcoal in the Control Room's heating, ventilation, and exhaust system (HVE-19). When the Technical Specifications were changed (Amendment No. 45) to reflect an upgrade in HVE-19's testing requirements due to the uprating of the unit's power level, two new sections were added to the Technical Specifications. These sections, 3.15 and 4.15, stated that the charcoal would be subjected to a methyl iodide removal test ($\geq$ 90% efficient) in accordance with the ANSI/ASME N509-1976 standard. This was approved by the staff in a Safety Evaluation dated December 5, 1979. The testing requirements for HVE-19 is therefore being removed from Table 4.1-3 since it is now discussed in detail in Section 4.15. This change removes conflicting requirements and therefore is approved.

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 occupation 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: September 10, 1984

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