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DENTON, H. R. Office of Nuclear Reactor Regulation, Director (post 851125

SUBJECT: Forwards marked-up Amend 37 to FSAR. Justification for revs listed chapters also encl. Revs do not alter conclusions in SER through SSER3 (NUREG-1038). Amend will be formally reissued in public form.

See Revs

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Carolina Power & Light Company

SERIAL: NLS-86-381

OCT 3 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NO. 1 - DOCKET NO. 50-400
FSAR AMENDMENT 37

Dear Mr. Denton:

Carolina Power & Light Company (CP&L) hereby submits a hand-marked copy of Amendment 37 to the Shearon Harris Nuclear Power Plant (SHNPP) Final Safety Analysis Report (FSAR). This amendment includes revision to Chapters 2, 3, 5, 6, 7, 9, 11, 12, 13, 14, and 15. The attached table provides justification for these revisions which constitute the known remaining changes which need to be made prior to issuance of an operating license. CP&L has reviewed the changes in this amendment against the SHNPP Safety Evaluation Report (SER) (NUREG-1038) through Supplement 3, and it is our position that these changes do not alter the conclusions of the SER.

Each page bears the amendment number, and changes are indicated by vertical bars in the margin. This amendment is hand marked on current FSAR pages due to the press of time prior to licensing. It will be formally reissued in a published form shortly; therefore, an effective page list and instructions for entering the revised pages are not included with this submittal.

If you have any questions, please contact me.

Yours very truly,

A. B. Cutter - Vice President
Nuclear Engineering & Licensing

ABC/JDK/bmc (5021JDK)

Attachment

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Dr. J. Nelson Grace (NRC-RII)

B001
3|60

A. B. Cutter, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Ruby R. Morgan
Notary (Seal)

RUBY R. MORGAN NOTARY PUBLIC WAKE COUNTY, N.C.

My commission expires: 11/27/89

TABLE 6.2.4-1
CONTAINMENT ISOLATION SYSTEM DATA
SHNPP FSAR

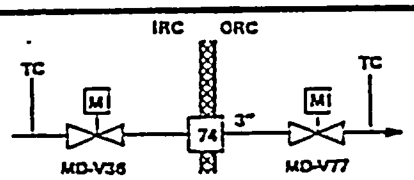
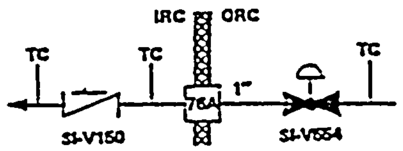
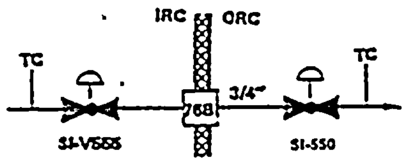
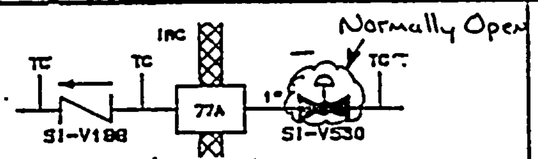
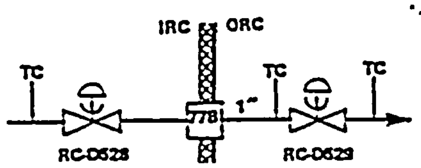
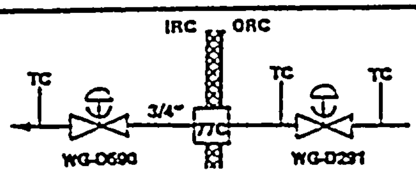
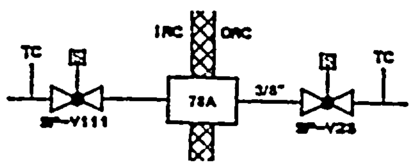
PENETRATION DETAIL	FD NUMBER	GENERAL DESIGN CRITERION	FLUID	HIGH ENERGY LINE	BYPASS LEAKAGE PATH	SYSTEM TITLE	VALVE NUMBER	POWER TRAIN	LENGTH of PIPE, FT	VALVE TYPE	ACTUATOR	PRIMARY ACTUATION MODE	SECONDARY ACTUATION MODE	ISOLATION SIGNAL	ACCIDENT SIGNAL	RESPONSE TIME	VALVE POSITION: NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	ENGINEERED SAFETY FEATURE	CONTAINMENT ISOLATION VALVE	TYPE C TEST	NOTE	
	185	56	W	NO	YES	CONTAINMENT SUMP PUMP DISCHARGE	V36 V77	A B	1 5	GA GA	MO MO	A A	RM RM	1 1	- -	<60 <60	O O	O O	C C	AI AI	C C	NO NO	YES YES	YES YES		~~29~~
75						SPARE																				
	809	56	W	YES	YES	ACCUMULATOR FILL FROM RWST	V150 V554	- B	3 2	CK GL	SA AO	- A	- RM	- 1	- -	- 10	C C	C C	C C	- C	C C	NO NO	YES YES	YES YES	8	~~29~~
	809	56	W	YES	YES	ACCUMULATOR TO RWST	V555 V550	A B	2 1	GL GL	AO AO	A A	RM RM	1 1	- -	10 10	C C	C C	C C	C C	C C	NO NO	YES YES	YES YES	9	
	809	56	G	YES	YES	NITROGEN TO ACCUMULATORS	V188 V530	- B	3 1	CK GL	SA AO	- A	- RM	- 1	- -	- 10	CY CY	C C	C C	- C	C C	NO NO	YES YES	YES YES	10	~~29~~ 37
	801	56	G	NO	YES	PRESSURIZER RELIEF TANK CONNECTION	D528 D529	A B	4 3	DA DA	AO AO	A A	RM RM	1 1	- -	10 10	O O	C C	C C	C C	C C	NO NO	YES YES	YES YES		~~29~~
	813	56	G	NO	YES	RCDT H_2 SUPPLY	D590 D291	A B	3 3	DA DA	AO AO	A A	RM RM	1 1	- -	10 10	O O	C C	C C	C C	C C	NO NO	YES YES	YES YES		~~29~~
	052	55	W	YES	YES	REACTOR COOLANT SAMPLE	V111 V23	B A	2 3	GL GL	SO SO	A A	RM RM	1 1	- -	<60 <60	O O	C C	C C	C C	C C	NO NO	YES YES	YES YES	19	~~29~~

TABLE 6.2.4-1
CONTAINMENT ISOLATION SYSTEM DATA
SHNPP FSAR

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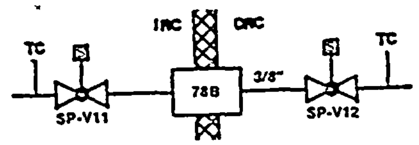
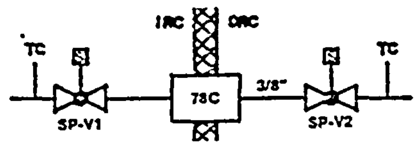
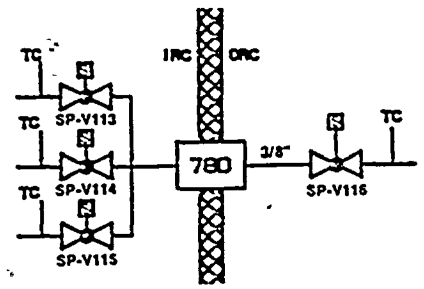
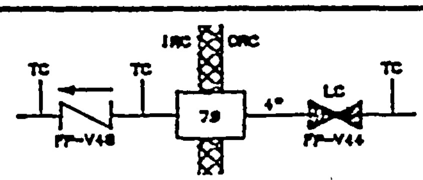
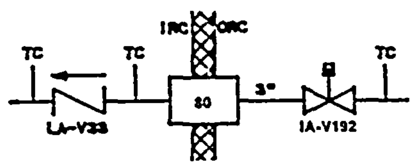
PENETRATION DATA						VALVE DATA											VALVE POSITION									
PENETRATION DETAIL	FD NUMBER	GENERAL DESIGN CRITERION	FLUID	HIGH ENERGY LINE	BYPASS LEAKAGE PATH	SYSTEM TITLE	VALVE NUMBER	POWER TRAIN	LENGTH of PIPE, FT	VALVE TYPE	ACTUATOR	PRIMARY ACTUATION MODE	SECONDARY ACTUATION MODE	ISOLATION SIGNAL	ACCIDENT SIGNAL	RESPONSE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	ENGINEERED SAFETY FEATURE	CONTAINMENT ISOLATION VALVE	TYPE C TEST	NOTE	
	052	55	W	YES	YES	PRESSURIZER LIQUID SAMPLE	V11 V12	B A	4 3	GL GL	SO SO	A A	RM RM	1 1	- -	<60 <60	O O	C C	C C	C C	C C	NO NO	YES YES	YES YES		~~29~~
	052	55	S	YES	YES	PRESSURIZER STEAM SAMPLE	V1 V2	B A	2 3	GL GL	SO SO	A A	RM RM	1 1	- -	<60 <60	O O	C C	C C	C C	C C	NO NO	YES YES	YES YES		~~29~~
	052	55	W	YES	YES	ACCUMULATOR SAMPLE	V113 V114 V115 V116	B B B A	9 3 3 3	GL GL GL GL	SO SO SO SO	A A A A	RM RM RM RM	1 1 1 1	- - - -	<60 <60 <60 <60	O O O O	C C C C	C C C C	C C C C	C C C C	NO NO NO NO	YES YES YES YES	YES YES YES YES		~~29~~
	388	56	W	NO	YES	FIRE WATER STANDPIPE SUPPLY	V48 V44	- -	3 1	CK GA	SA M	- M	- -	- -	- -	- -	C LC	O O	C LC	- C	C LC	NO NO	YES YES	YES YES		~~29~~
	301	56	A	NO	YES	INSTRUMENT AIR SUPPLY	V33 V192	- A	3 2	CK GL	SA AO	- A	- RM	- 1	- -	- <60	O O	O O	C C	C C	C. C	NO NO	YES YES	YES YES		~~29~~
81						SPARE																				
82						SPARE																				
83 SEE PAGE 6.2.4-31c						~~SPARE~~																				37
84						SPARE																				

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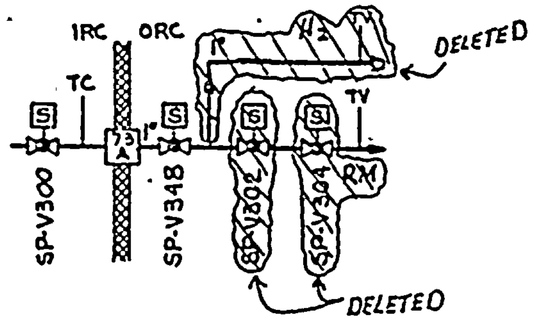
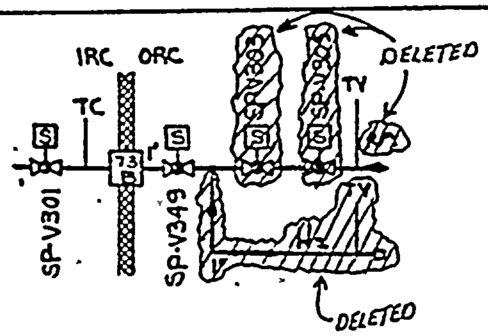
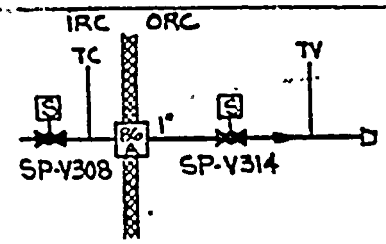
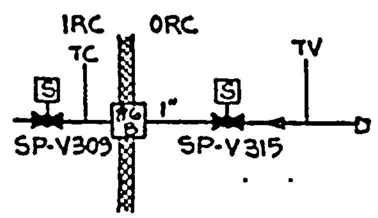
TABLE 6.2.4-1
CONTAINMENT ISOLATION SYSTEM DATA
SHNPP FSAR

PENETRATION DATA						VALVE DATA											VALVE POSITION								
PENETRATION DETAIL	FD NUMBER	GENERAL DESIGN CRITERION	FLUID	HIGH ENERGY LINE	BYPASS LEAKAGE PATH	SYSTEM TITLE	VALVE NUMBER	POWER TRAIN	LENGTH of PIPE, FT	VALVE TYPE	ACTUATOR	PRIMARY ACTUATION MODE	SECONDARY ACTUATION MODE	ISOLATION SIGNAL	ACCIDENT SIGNAL	RESPONSE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	ENGINEERED SAFETY FEATURE	CONTAINMENT ISOLATION VALVE	TYPE C TEST	NOTE
	105	56	A	NO		~~RADIATION MONITOR AND~~ HYDROGEN ANALYZER	V300	A	2	GL	SO	A	RM	1	–	<60	C ~~O~~	C ~~O~~	C	C	C	YES	YES	YES	19
							~~V302~~	~~B~~	~~10~~	~~GL~~	~~SO~~	~~A~~	~~RM~~	~~3~~	~~–~~	~~<60~~	~~O~~	~~O~~	~~C~~	~~O~~	~~O~~	~~NO~~	~~NO~~	~~NO~~	
							~~V304~~	~~A~~	~~10~~	~~GL~~	~~SO~~	~~A~~	~~RM~~	~~3~~	~~–~~	~~<60~~	~~O~~	~~O~~	~~C~~	~~O~~	~~O~~	~~NO~~	~~NO~~	~~NO~~	
							V348	A	1	GL	SO	A	RM	1	–	<60	C ~~O~~	C ~~O~~	C	C	C	YES	YES	YES	
	105	56	A	NO		~~RADIATION MONITOR AND~~ HYDROGEN ANALYZER	V301	A	2	GL	SO	A	RM	1	–	<60	C ~~O~~	C ~~O~~	C	C	C	YES	YES	YES	19
							~~V303~~	~~B~~	~~10~~	~~GL~~	~~SO~~	~~A~~	~~RM~~	~~3~~	~~–~~	~~<60~~	~~O~~	~~O~~	~~C~~	~~O~~	~~C~~	~~NO~~	~~NO~~	~~NO~~	
							~~V305~~	~~A~~	~~10~~	~~GL~~	~~SO~~	~~A~~	~~RM~~	~~3~~	~~–~~	~~<60~~	~~O~~	~~O~~	~~C~~	~~O~~	~~O~~	~~NO~~	~~NO~~	~~NO~~	
							V349	A	1	GL	SO	A	RM	1	–	<60	C ~~O~~	C ~~O~~	C	C	C	YES	YES	YES	
	105	56	A	NO		HYDROGEN ANALYZER	V308	B	1	GL	SO	A	RM	1	–	<60	C	C	C	C	C	YES	YES	YES	19
							V314	B	1	GL	SO	A	RM	1	–	<60	C	C	C	C	C	YES	YES	YES	
	105	56	A	NO		HYDROGEN ANALYZER	V309	B	2	GL	SO	A	RM	1	–	<60	C	C	C	C	C	YES	YES	YES	19
							V315	B	1	GL	SO	A	RM	1	–	<60	C	C	C	C	C	YES	YES	YES	

37

37

~~29~~

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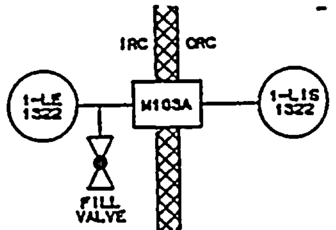
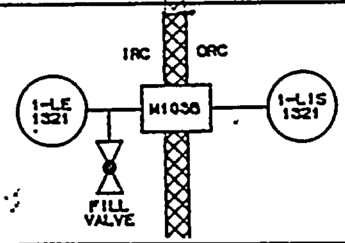
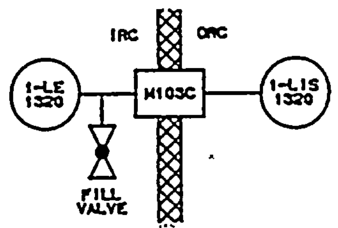
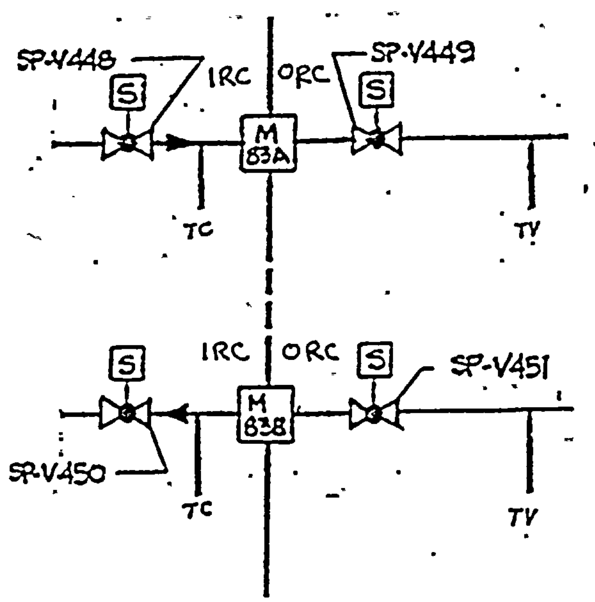
TABLE 6.2.4-1

CONTAINMENT ISOLATION SYSTEM DATA

PENETRATION DATA						VALVE DATA																			
																	VALVE POSITION								
PENETRATION DETAIL	FD NUMBER	GENERAL DESIGN CRITERION	FLUID	HIGH ENERGY LINE	BYPASS LEAKAGE PATH	SYSTEM TITLE	VALVE NUMBER	POWER TRAIN	LENGTH of PIPE, FT	VALVE TYPE	ACTUATOR	PRIMARY ACTUATION MODE	SECONDARY ACTUATION MODE	ISOLATION SIGNAL	ACCIDENT SIGNAL	RESPONSE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	ENGINEERED SAFETY FEATURE	CONTAINMENT ISOLATION VALVE	TYPE C TEST	NOTE
	844	54, 55	W	YES	NO	RVLIS	—	B	—	—	—	—	—	—	—	—	—	—	—	—	—	NO	NO	TYPE A TEST	FILL VALVE IS SEALED AFTER USE 16
	844	54, 55	W	YES	NO	RVLIS	—	B	—	—	—	—	—	—	—	—	—	—	—	—	—	NO	NO	TYPE A TEST	FILL VALVE IS SEALED AFTER USE 16
	844	54, 55	W	YES	NO	RVLIS	—	B	—	—	—	—	—	—	—	—	—	—	—	—	—	NO	NO	TYPE A TEST	FILL VALVE IS SEALED AFTER USE 16
	105	56	A	NO		RCPB LEAK DETECTION RADIATION MONITOR	V-448 V-449 V-450 V-451	A B A B		GL GL GL GL	SO SO SO SO	A A A A	RM RM RM RM	I I I I	- - - -	60 60 60 60	O O O O	O O O O	C C C C	C C C C	C C C C	NO NO NO NO	YES YES YES YES	YES YES YES YES	

~~29~~

ADDED

37

Amendment No. ~~29~~ 37

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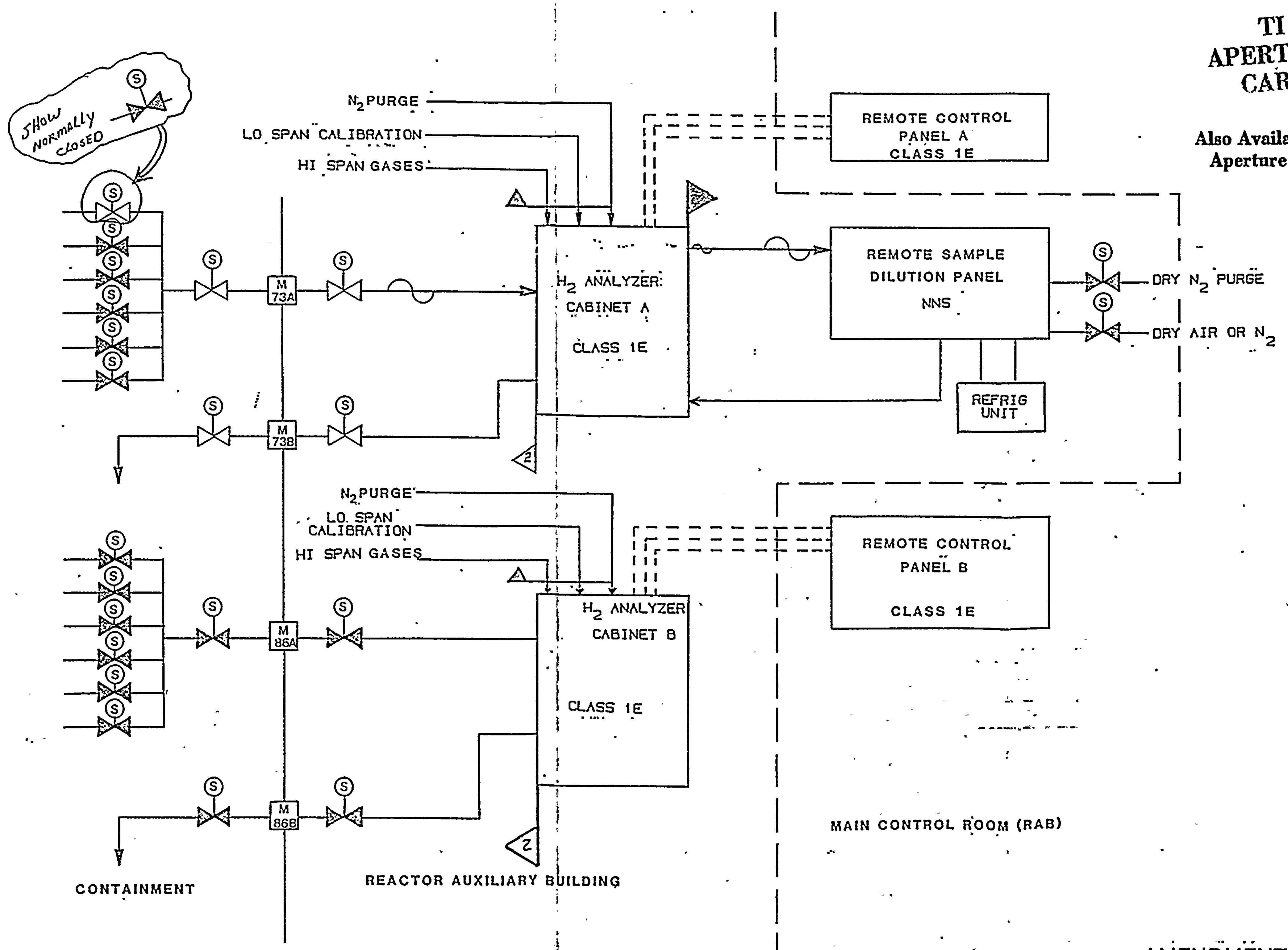

AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

POST ACCIDENT HYDROGEN MONITORING SYSTEM

FIGURE 6.2.5-7

8610140189-05

b) The sample valves, containment penetration and piping up to and including the outermost containment isolation valves for the sample feed header and sample return line are ASME Section III, Safety Class 2, Seismic Category I and are designed to retain their integrity and operability under all conditions following a design basis accident.

c) All materials and equipment required by this system are selected to be compatible with the environmental conditions anticipated during accident operation and are suitable for a lifetime consistent with that of the plant.

d) The system samples containment air, providing the means to measure the containment air hydrogen concentration and to alert the operator in the event that a high hydrogen concentration is detected, in accordance with the requirements of Regulatory Guide 1.7.

e) All containment isolation valves for the Hydrogen Analyzers are normally closed and fail closed on loss of electrical power. Means are provided to reopen valves, when required, after power is restored. In the event of a containment isolation signal, valves 2SP-V301 SA-1 and 2SP-V349SA-1 close and isolate containment penetration 73B. Valves 2SP-V300 SA-1 and 2SP-V348 SA-1 close to isolate penetration 73A. On power failure, all valves fail closed, insuring isolation. 37

~~Containment Penetrations 73A and 73B serve a dual function. During normal operation, these penetrations permit continuous sampling of containment atmosphere for detection of particulate iodine and noble gas airborne radioactivity. During accident conditions, these penetrations permit the withdrawal of containment atmosphere for hydrogen sampling. The radiation monitor is unqualified for post-accident operation. It is therefore isolated from the post-accident sample stream by valves 2SP-V302 SB-1, 2SP-V304 SA-1, 2SP-V303 SB-1, and 2SP-V305 SA-1. With this arrangement, isolation of the post-LOCA unqualified radiation monitors 1LT-3502 is assured.~~ The hydrogen analyzer cabinet, tag number AT-1SP-7438-SA, is qualified for post-accident operation. The sample line coming from and going to penetrations 73A and 73B respectively, contain only safety train A associated valves. Likewise, containment penetrations 86A and 86B use only safety train B associated valves on the hydrogen analyzer sample lines. 37

As a result, if one safety train fails then the required redundancy for post-accident hydrogen sampling is still provided. If the associated valves fail to close when they should close, safety is not compromised since the hydrogen analyzer is qualified for post-accident operation.

f) The Hydrogen Analyzer System consists of two identical units which are completely independent of each other and are powered from independent onsite sources. Therefore, assuming a single failure, process capability is available to monitor the hydrogen concentration in the Containment. See Table 6.2.5-7 which provides a failure modes and effects analysis.

g) The system is designed for remote-manual sampling capability with an intermittent cycle of Hydrogen indication for six (6) different sample

6.2.6 CONTAINMENT LEAKAGE TESTING

The Containment and containment penetrations are designed to permit periodic leakage rate testing in accordance with General Design Criteria (GDC) 52 and 53 and Appendix J to 10CFR50.

Testing requirements for piping penetration isolation barriers and valves have been established by using the intent of GDC 54, as interpreted in Appendix J to 10CFR50. Exceptions taken to Appendix J for Type A, B, or C tests are described and justified in Subsections 6.2.6.1, 6.2.6.2, and 6.2.6.3, respectively.

6.2.6.1 Containment Integrated Leakage Rate Test (Type A Test)

The design leakage rate for the Containment is 0.1 weight percent per day. The actual leakage rate is tested and verified using the methods and requirements of Appendix J to 10CFR50 for Type A tests.

In accordance with Appendix J, a margin for possible deterioration of the Containment integrity during the service intervals between integrated leakage rate test (ILRT) is provided. The measured leak rate (L_{am} at peak test pressure) shall not exceed 0.75 of the maximum allowable value. ~~26~~

The structural integrity test (SIT) is conducted during the same test program as the preoperational peak pressure integrated leakage rate test. The SIT is conducted in conformance with the descriptions contained in Section 3.8.1 and with the exceptions taken to Regulatory Guide 1.18 as specified in Section 1.8. After the SIT peak pressure ~~pressurization~~ requirements and the containment stabilization at required pressurization are completed, the initial peak calculated pressure (P_a = 41.0 psig) ~~and reduced pressure (2 1/2 Pa)~~ ILRT and SIT depressurization phase of the test are conducted. This sequence of testing is chosen to satisfy paragraph II.F of Appendix J to 10CFR50, which specifies that the initial ILRT shall be conducted after the Containment is completed and is ready for operation. ~~26~~ 37 ~~26~~

Subsequent peak calculated pressure tests are conducted as specified in Section 6.2.6.4. ~~26~~

Reduced pressure ILRT's (as described in paragraphs III.A.4 and III.A.5 of Appendix J to 10CFR50) are not performed during pre-operational testing or during periodic ILRT's. Industry experience has shown that extrapolation factors used to correlate the reduced and full pressure tests are not reliable and may be erroneous in some cases. 37 ~~26~~

6.2.6.1.1 Pretest Requirements

Insert "A" from Page 6.2.6-2a

~~A prerequisite to the Containment integrated leakage rate test is the satisfactory completion of a series of local leakage tests. This involves subjecting potential leakage paths through the containment boundary (i.e., containment penetrations) to similar test conditions occurring during the integrated leakage rate test. Conducting local leakage tests allows discovery and elimination of leak paths through the Containment without pressurizing the entire containment structure. These local leakage tests are the Type B and C tests described in Sections 6.2.6.2 and 6.2.6.3.~~

The primary prerequisite for conducting an ILRT is a general inspection of the accessible interior and exterior surfaces of the containment structures and components to uncover any evidence of structural deterioration which may affect either the structural integrity or leaktightness of the Containment. If there is evidence of structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, nondestructive examinations, and tests as specified in the applicable code specified in 10CFR50.55a.

6.2.6.1.2 Valve Positioning for the ILRT

~~Following the completion of the Type C tests,~~ The containment isolation valves are positioned ~~to their normal operational position and subsequently repositioned~~ to their post-accident position by the normal method with no accompanying adjustments. Normal, LOCA, and ILRT positions for each isolation valve are shown on Table 6.2.4-1. 37

6.2.6.1.3 System Preparation for Type A Tests

Systems are properly isolated, drained, or vented to reflect their worst potential status following a LOCA to assure that the Type A test results accurately reflect the most restricting LOCA conditions. Systems required to maintain the Unit in a cold shutdown condition are operable in their normal mode and are not vented or drained. However, any of these system penetrations that require Type C local leakage tests as defined in Section 6.2.4 have the results of the local leakage tests added to the result of the Type A test. Systems used during the Type A test for ~~pressurizing the Containment or~~ sensing the leakage are not lined up in the post-accident positions. Any leakage from the isolation valves in these systems is determined by local methods and the results are added to the Type A test. Systems that operate in post-accident conditions filled with fluid as defined in Section 6.2.4 need not be vented or drained for the Type A test. Systems which form closed Seismic Category I systems inside Containment (as defined by GDC 57) are not vented to the containment atmosphere. 37

Leakage testing of instrumentation lines that penetrate Containment is done in conjunction with the Type A test. These lines will be open to the containment atmosphere. Liner plate weld leak chase channels will not be vented during the Type A test.

All systems which are provided with isolation capabilities to satisfy GDC 55 or 56 are either normally open to the containment atmosphere or are vented to the containment atmosphere during the Type A tests. Table 6.2.4-1 contains the applicable GDC or other defined criteria for the isolation valve arrangements provided.

The electrical penetration pressurization system, supplied by dry pressurized nitrogen, serves to exclude moisture-laden air from each containment electrical penetration. During the Type A test, the nitrogen pressure in each electrical penetration will be locked in by shutting each penetration's nitrogen supply valve. Nitrogen supply to the penetration pressurization system will be isolated and the system headers vented to the outside atmosphere.

During a type A test, the steam generator secondary side is to be vented outside the containment atmosphere. The systems connected to the secondary side of the steam generator are identified in Table 6.2.4-1. | ~~20~~

Insert "A"
to Page 6.2.6-1

The performance of Local Leak Rate Tests (LLRT) is not a prerequisite to the ILRT. If a Containment Boundar (isolation valve, airlock seal, etc.) is repaired prior to the ILRT and during the same outage as the ILRT, then the difference between the measured local leak rates before and after the repair are used to adjust the subsequent ILRT measured Type A Leakage Rate to determine the "As-Found" Leakage Rate. The calculated difference is based upon minimum pathway leakage for the affected containment barriers. Minimum pathway leakage is the smaller leakage rate of in-series barriers tested individually, one-half the leakage rate of in-series barriers tested simultaneously by pressurizing between them, and the combined leakage rate for barriers tested in parallel.

The service water lines to the emergency containment air coolers are neither vented or drained, as these lines are designed to GDC 57. The coolers may be required to cool the containment atmosphere during the Type A test.

Pressurized gas and water systems are isolated downstream of the outside isolation valve for the system and vented outside of the Containment. This is done to preclude inleakage into the Containment and to expose the outside isolation valve to an atmospheric back pressure to obtain accurate leakage characteristics.

The reactor coolant drain tank, pressurizer relief tank, and the accumulator tanks are vented to the containment atmosphere. This is done to protect the tanks from the external pressure of the test and to preclude leakage to or from the tanks to help assure the accuracy of the test results.

The following systems are considered closed systems inside containment that need not be vented and drained for a Type A test:

~~a) Main Steam System~~

a ~~b~~) Main Feedwater System

b ~~c~~) Auxiliary Feedwater System

c ~~d~~) Steam Generator Blowdown System 37

d ~~e~~) Safety Related Portion of SW. System to and from emergency fan coolers AH-1 through AH-4

e ~~f~~) Portion of component Cooling Water System (to and from Reactor Coolant Drain Tank HX and Excess Letdown HX)

f ~~g~~) Portion of the Steam Generator Sampling System Inside Containment Out to the Containment Isolation Valve ~~36~~

The system design meets the following requirements of SRP 6.2.4.II.0 for a closed system inside containment:

a) The system does not communicate with either the reactor coolant system or the containment atmosphere.

b) The system is protected against missiles and pipe whip.

c) The system is designated seismic category I.

d) The system is classified Safety Class 2.

e) The system is designed to withstand temperature at least equal to the containment design temperature.

f) The system is designed to withstand the external pressure from the containment structural acceptance test,

g) The system is designed to withstand the loss-of-coolant-accident transient and environment.

6.2.6.1.4 ILRT Test Method

The air used to pressurize the Containment is conditioned for temperature and water vapor to prevent moisture condensation in the Containment at the test pressure. The air used to pressurize the Containment is essentially oil-free to prevent coating of the containment wall with oil or interfering with the test instrumentation.

Sensing devices are located at different locations in the Containment to measure average temperature and humidity. Location of the temperature and humidity sensors are made with consideration to their respective patterns in the Containment. These patterns are employed in determination of the mean representative temperature and humidity for the absolute method of leakage rate testing. These data are periodically monitored during the test and analyzed as they are taken so that the leakage rate and its statistical significance is known as the test progresses.

The leakage rate test period extends to 24 hours of sustained internal pressure. If it is demonstrated to the satisfaction of the NRC that the leakage rate can be accurately determined during a shorter test period, the agreed upon shorter period may be used.

At the conclusion of the leakage rate test, the accuracy of the Type A test is verified by either of the supplemental test methods described in ANSI/ANS 56.8-1981, Appendix C. The supplemental test injects into or bleeds from the Containment an accurately measured amount of air. The supplemental test method selected is conducted for a sufficient duration to establish accurately the change in leakage rate between Type A test and the supplemental test. The difference between the supplemental test data and the Type A test data shall agree within 0.25 L_a. ~~or 0.25 L_x~~. | 37

Except as noted below, the following aspects of Type A testing follow 10CFR50, Appendix J guidelines are adhered to:

a) Pretest requirements including a general inspection

b) Conduct of tests

c) Acceptance criterion

d) Periodic retest schedule

e) Inspection and reporting of test

If during the performance of a Type A test the leakage rate exceeds the criterion of 0.75 L_a ~~or 0.75 L_x~~, corrective action will be required. If excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with the satisfactory completion of the test, these leakage paths will be isolated and the Type A test continued until completion. A local leakage test will be performed before and after the repair of each isolated leakage path. The calculated integrated leak rate will be obtained by adding the post-repair local leak rate to the measured integrated leak rate, and it will be required to fall within the acceptable limits of 0.75 L_a. ~~or 0.75 L_x~~. | 37

If any periodic Type A test fails to meet ~~either~~ 0.75 L_a ~~or 0.75 L_x~~, the test schedule for subsequent Type A tests will be reviewed and approved by the Nuclear Regulatory Commission. If two consecutive Type A tests fail to meet ~~either~~ 0.75 L_a ~~or 0.75 L_x~~, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a ~~or 0.75 L_x~~. At that time, the normal test schedule allowed by 10CFR50, Appendix J shall be in effect. 37 37

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For larger test volumes, a pressure decay method may be utilized to determine the leakage rate.

The total leakage rate for Type B and C tests will be less than 0.6 L_a. The individual testing performed on valves requiring a Type C test is described in Technical Specifications.

In accordance with 10CFR50 Appendix J III.C.1, valves may be tested in the non-accident pressure direction when it can be determined that the results from the tests for the pressure applied in the non-accident direction will provide equivalent or more conservative results. 37

The criteria for determining the direction in which the test pressure is applied to the isolation valves is as follows:

a) Check, ball, plug, and non-wedge disc gate valves are tested in the accident pressure direction.

b) Wedge disc gate, butterfly, and diaphragm valves are tested in either direction since seat leakage is the same in either direction.

c) Globe valves may be tested in the non-accident pressure direction if the test pressure would tend to unseat the valve and the accident pressure would tend to seat the valve. ~~Where globe valves (unbalance plug with flow over the plug) are installed such that the flow and accident pressure are in the same direction, the valve is tested in the non-accident pressure direction. In this case the flow and the accident pressure will tend to seat the valve, while the non-accident pressure will tend to unseat the valve (i.e., force under the plug and acting against the actuator spring force). Where globe valves (with flow from below the seat) are installed such that the flow and accident pressure are not in the same direction, the valve is tested in the non-accident pressure direction. In this case the accident pressure will tend to seat the valve, while the flow and non-accident pressure will tend to unseat the valve. In both of these cases the test results will provide equivalent or more conservative results.~~ 37

6.2.6.4 Scheduling and Reporting of Periodic Tests

Types A, B, and C tests will be conducted at the intervals specified in Technical Specifications. These intervals are in accordance with Appendix J to 10CFR50, with the exception of the testing of the air locks as described in Section 6.2.6.2.

The packing leakage for any valve tested in the non-accident direction shall be included in the reported leak rate for that valve if the packing provides a leakage path from the containment atmosphere to the outside environment (i.e. packing is part of containment isolation boundary).

Periodic leak testing of the containment isolation valves need not be done during a refueling outage but may be scheduled at any time during an operating cycle. However, the test interval for any valve shall not exceed two years.

h) To provide system materials which are compatible with fluid chemistry and applied codes and standards. System component design data parameters are given in Table 6.5.2-1.

6.5.2.2 System Design

The system flow diagram is shown on Figure 6.2.2-1. System component design data parameters are given in Table 6.5.2-1.

A discussion of the spray header design including a description of the number of nozzles per header, nozzle spacing, and nozzle is contained in Section 6.2.2.

~~27~~ System operation is automatically initiated by a HI-3 signal. The signal starts the two spray pumps and the motor operated spray isolation valves. Within 55 seconds, water will reach the nozzles and start spraying (see Section 6.2.2). The motor operated NaOH isolation valves will be opened automatically by the HI-3 signal.

~~27~~ ~~27~~ After the opening of the NaOH Isolation valve, the kinetic energy in the eductor will create a negative pressure to draw the Sodium Hydroxide solution (NaOH) from the containment spray additive tank NaOH solution will be injected into the Containment Spray System (CSS) lines just up stream of the CS pump suction at a rate sufficient to provide the required range of pH 8.5-11 for the containment spray. Turbulence in the fluid passing through the pump is sufficient to assure complete and uniform mixing of the fluid. ~~The operator will manually close the NaOH Isolation valves when the NaOH is spent as indicated by low NaOH tank level.~~ The NaOH isolation valves will automatically close when the containment spray additive tank is empty. Additional NaOH can be added to the tank or through an emergency NaOH addition line outside the Tank Building. If necessary, the operator may reopen these NaOH isolation valves at any later time. The containment spray pumps initially take suction from the refueling water storage tank (RWST). The minimum operating capacity of the RWST (see Section 6.2.2) is more than adequate to supply enough water for the injection mode of operation. ~~27~~ When low-low level tank water level is reached in the RWST, pump suction is transferred to containment recirculating sump automatically by opening the recirculation line valves and closing the valves at the outlet of the RWST. 37

The Containment Spray System can provide one year of operation if required.

The layout of the containment spray system headers and nozzle orientation (see Section 6.2.2) provides a minimum spray coverage of 92.6 percent of the containment free volume and 95 percent of the surface area of the operating floor (Elevation 286 ft.) with only one spray train in operation. This includes the volume beneath the grating in the operating floor. The specified grating has 80 percent free area. The drop size spectrum is discussed in Section 6.2.2.

TABLE 7.3.1-5 (Continued)

ESF ACTUATION SYSTEMS-SAFETY INJECTION SIGNAL(S)

EQUIPMENT IDENTIFICATION	SERVICE	ACTUATION CHANNEL	ACTION	REFERENCE FIGURE NUMBER	SCHEMATIC/LOGIC* NUMBER	
AV-D86SA-1	RAB Normal Ventilation Branch Isol. Dampers	A	Close	9.4.3-2	CAR-2166-B-430SH31.35D	27
AV-D87SB-1	RAB Normal Ventilation Branch Isol. Dampers	B	Close	9.4.3-2	CAR-2166-B-430SH31.330	
AH-16(1A-SA)	Electric Equipment Protection Room Ventilation Supply Fan	A	Start Note 1	9.4.5-1	CAR-2166-B-430SH31.30A	
AH-16(1B-SB)	Electric Equipment Protection Room Ventilation Supply Fan	B	Start Note 1	9.4.5-1	CAR-2166-B-430SH31.30A	
E-10 (1A-SA)	Electric Equipment Protection Room Ventilation Exhaust Fan	A	Stop Note 1	9.4.5-1	CAR-2166-B-430SH31.29	27
E-10 (1B-SB)	Electric Equipment Protection Room Ventilation Exhaust Fan	B	Stop Note 1	9.4.5-1	CAR-2166-B-430SH31.29	
E-17 (1X-NNS)	RAB Normal Exhaust Fan	A&B	Stop	9.4.3-2	CAR-2166-B-430SH31.35P	27
E-18 (1X-NNS)	RAB Normal Exhaust Fan	A&B	Stop	9.4.3-2	CAR-2166-B-430SH31.35Q	
E-19 (1X-NNS)	RAB Normal Exhaust Fan	A&B	Stop	9.4.3-2	CAR-2166-B-430SH31.35R	
E-20 (1X-NNS)	RAB Normal Exhaust Fan	A&B	Stop	9.4.3-2	CAR-2166-B-430SH31.35S	
E-6 (1A-SA)	RAB Emergency Exhaust Fan	A	Start Note 1	9.4.3-2	CAR-2166-B-430SH31.33	
E-6 (1B-SB)	RAB Emergency Exhaust Fan	B	Start Note 1	9.4.3-2	CAR-2166-B-430SH31.33A	
S-3 (1A-NNS)	RAB Normal Supply Fan	A&B	Stop	9.4.3-2	CAR-2166-B-430SH31.37J	27
S-3 (1B-NNS)	RAB Normal Supply Fan	A&B	Stop	9.4.3-2	CAR-2166-B-430SH31.37J	
1A-SA	Emergency Load Sequencer Panel A	A	Start	N/A	CAR-2166-G-509501C	
1B-SB	Emergency Load Sequencer Panel B	B	Start	N/A	CAR-2166-G-509502	
E-61 (1A-SA)	Diesel Generator Bldg. Exhaust Fan	A	Start Note 1	9.4.5-2	CAR-2166-B-430	
E-61 (1C-SB)	Diesel Generator Bldg. Exhaust Fan	B	Start Note 1	9.4.5-2	CAR-2166-B-430	
~~2SP-V405SA-1~~	~~Cont. Atmos. Rad. Monitor Isol.~~	~~A~~	~~Close~~	X	X	~~27~~ 37
~~2SP-V304SA-1~~	~~Cont. Atmos. Sys. Cont. Isol.~~	~~A~~	~~Close~~			
~~2SP-V302SB-1~~	~~Cont. Atmos. Sys. Cont. Isol.~~	~~B~~	~~Close~~			
~~2SP-V303SB-1~~	~~Cont. Atmos. Sys. Cont. Isol.~~	~~B~~	~~Close~~			
~~2SP-V305SA-1~~	~~Cont. Atmos. Sys. Cont. Isol.~~	~~A~~	~~Close~~			

TABLE 7.3.1-7 (Continued)

ESF ACTUATION SYSTEMS-CONTAINMENT ISOLATION PHASE-A (T)

EQUIPMENT IDENTIFICATION	SERVICE	ACTUATION CHANNEL	ACTION	REFERENCE FIGURE NUMBER	SCHEMATIC/LOGIC NUMBER	
AH-37 (1A-NSS)	Non-Nuclear Safety Containment Fan Cooler	A&B	Stop	6.2.2-3	CAR-2166-B-430SH31.64A	
AH-37 (1B-NSS)	Non-Nuclear Safety Containment Fan Cooler	A&B	Stop	6.2.2-3	CAR-2166-B-430SH31.64A	
AH-38 (1A-NSS)	Non-Nuclear Safety Containment Fan Cooler	A&B	Stop	6.2.2-3	CAR-2166-B-430SH31.64A	
AH-38 (1A-NSS)	Non-Nuclear Safety Containment Fan Cooler	A&B	Stop	6.2.2-3	CAR-2166-B-430SH31.64A	
AH-39 (1B-NSS)	Non-Nuclear Safety Containment Fan Cooler	A&B	Stop	6.2.2-3	CAR-2166-B-430SH31.64A	
AH-39 (1B-NSS)	Non-Nuclear Safety Containment Fan Cooler	A&B	Stop	6.2.2-3	CAR-2166-B-430SH31.64A	
2FP-V44SA-1	Fire Protection-Containment Fire Hose Riser Isolation	A	Close	9.5.1-4	-	
2FP-V45SA-1	Fire Protection-Containment Water Sprinkler Isolation	A	Close	9.5.1-4	-	
2CT-V255A-1	Containment Spray Header Recirculation Isolation	A	Close	6.2.2-1	CAR-2166-G-423	
2CT-V49SB-1	Containment Spray Header Recirculation Isolation	B	Close	6.2.2-1	CAR-2166-G-423	
2CT-V8SA-1	Containment Spray Eductor Test Valve Isolation	A	Close	6.2.2-1	CAR-2166-G-423	
2CT-V145SB-1	Containment Spray Eductor Test Valve Isolation	B	Close	6.2.2-1	CAR-2166-G-423	
2SP-V300SA-1	Containment ~~Atmos.~~ H_2 Sampling System A Cont. Isolation	A	Close	-	-	~~27~~ 37
2SP-V308SB-1	Containment ~~Atmos.~~ H_2 Sampling System ~~A~~ B Cont. Isolation	B	Close	-	-	~~27~~ 37
2SP-V309SB-1	Containment ~~Atmos.~~ H_2 Sampling System ~~A~~ B Cont. Isolation	B	Close	-	-	~~27~~ 37
2SP-V348SA-1	Containment H_2 Sampling System A Cont. Isolation	A	Close	-		37
2SP-V349SA-1	Containment H_2 Sampling System A Cont. Isolation	A	Close	-		37

TABLE 7.3.1-7 (Continued)

ESF ACTUATION SYSTEMS-CONTAINMENT ISOLATION PHASE-A (T)

EQUIPMENT IDENTIFICATION	SERVICE	ACTUATION CHANNEL	ACTION	REFERENCE FIGURE NUMBER	SCHEMATIC/LOGIC NUMBER	
2SP-V301SA-1	Cont. H2 Sampling ~~Atmos. Mon.~~ System A Cont. Iso.	A	Close	-	-	~~27~~ 37
2SP-~~V314SA-1~~ V314SB-1	Containment H2 Sampling System B Isolation	~~X~~ B	Close	-	N/A	37
2SP-~~V315SA~~-1 V315SB-1	Containment H2 Sampling System B Isolation	~~X~~ B	Close	-	N/A	37
2AF-V162SAB-1	Hydrazine to AFW Steam Generator 1A	A&B	Close			~~27~~
2AF-V163SAB-1	Ammonia to AFW Steam Generator 1A	A&B	Close			
2AF-V164SAB-1	Hydrazine to AFW Steam Generator 1B	A&B	Close			
2AF-V165SAB-1	Ammonia to AFW Steam Generator 1B	A&B	Close			
2AF-V166SAB-1	Hydrazine to AFW Steam Generator 1C	A&B	Close			
2AF-V167SAB-1	Ammonia to AFW Steam Generator 1C	A&B	Close			
2SP-V408SB-1	PASS Isolation	B	Close			
2SP-V409SA-1	PASS Isolation	A	Close			
2SP-V406SB-1	PASS Isolation	B	Close			
2SP-V407SA-1	PASS Isolation	A	Close			
1A-SA	Containment Spray Pump Interlock Trip	A	Stop Note 1	6.2.2-1	-	
1B-SB	Containment Spray Pump Interlock Trip	B	Stop Note 1	6.2.2-1	-	

Notes:

1. Interlock applicable only when the recirculation valve of its respective safety train is open.

2SP-V448SA-1	Containment Atmos. Mon. System A Cont. Isolation	A	Close	-	37
2SP-V449SB-1		A	Close	-	
2SP-V450SA-1		A	Close	-	
2SP-V451SB-1		A	Close	-	

TABLE 9.1.3-2 (Continued)

FUEL POOL COOLING AND CLEANUP SYSTEM PARAMETERS

Fuel Pool Cooling Pump	
Quantity (per FPCCS)	2
Type	Horizontal Centrifugal
Design flowrate, gpm	~~4800~~ 4560
TDH, ft. H20	~~95~~ 98
Motor horsepower	150
Design pressure, psig	150
Design temperature, F	200
Material	Stainless Steel

37

Spent Fuel Pools	Pool 1	Pool 2	Pool 3
Volume gals.	403,920	403,920	191,480
Boron concentration, ppm	2,000	2,000	2,000
Liner material	Stainless Steel	Stainless Steel	Stainless Steel

New Fuel Pool	
Volume, gals.	147,804
Boron concentration, ppm	2,000
Liner material	Stainless Steel

~~25~~

Fuel Pool Demineralizer Filter	
Quantity (per FPCCS)	1
Type	Flushable
Design pressure, psig	400
Design temperature, F	200
Flow, gpm	325
Maximum differential pressure across filter element at rated flow (clean filter), psi	5
Maximum differential pressure across filter element prior to backflush, psi	60

crane prevents disengagement of a fuel assembly from the gripper during an SSE.

The following safety features are provided for in the fuel transfer system:

a) Transfer car permissive switch - The transfer car controls are located in the Fuel Handling Building; and conditions in the Containment are, therefore, not visible to the operator. The transfer car permissive switch allows a second operator in the Containment to exercise some control over car movement if conditions visible to him warrant such control.

Transfer car operation is possible only when both lifting arms are in the down position as indicated by the limit switches. The permissive switch is a backup for the transfer car lifting arm interlock. Assuming the fuel container is in the upright position in the Containment and the lifting arm interlock circuit fails in the permissive condition, the operator in the Fuel Handling Building still cannot operate the car because of the permissive switch interlock. The interlock, therefore, can withstand a single failure.

b) Lifting arm (transfer car position) - Two redundant interlocks allow lifting arm operation only when the transfer car is at the respective end of its travel and therefore can withstand a single failure.

Of the two redundant interlocks which allow lifting arm operation only when the transfer car is at the end of its travel, one interlock is a position limit switch in the control circuit. The backup interlock is a mechanical latch device on the lifting arm that is opened by the car moving into position.

c) Transfer car (valve open) - An interlock on the transfer tube valve permits transfer car operation only when the transfer tube valve position switch indicates the valve is fully open.

d) Transfer car (lifting arm) - The transfer car lifting arm is primarily designed to protect the equipment from overload and possible damage if an attempt is made to move the car when the fuel upender is in the vertical position. This interlock is redundant and can withstand a single failure. The basic interlock is a position limit switch in the control circuit. The backup interlock is a mechanical latch device that is opened by the weight of the fuel upender when in the horizontal position.

e) Lifting arm (refueling machine) - The refueling canal lifting arm is interlocked with the manipulator crane. Whenever the transfer car is located in the refueling cavity, the lifting arm cannot be operated unless ~~the mast is in the fully retracted position~~ or the manipulator crane is over the core. ~~or the gripper is released and inside the core.~~ 37

f) Lifting arm (fuel handling machine) - The lifting arm is interlocked with the spent fuel bridge crane. The lifting arm cannot be ~~operated~~ lowered unless the spent fuel bridge crane is not over the lifting arm area. 37

the engaged gripper is in the full up position or the disengaged gripper is withdrawn into the mast,

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In the unlikely event of a phase reversal prior to drive operations the crane drives cannot operate by reverse phase relay action. In the event of a phase reversal during hoist motor operation, the reverse phase relay will immediately operate to shut down the hoist drive, set the holding brake, and stop the load.

The crane is designed to maintain its structural integrity and hold its load under the dynamic loading conditions of the SSE. Load drop is precluded due to its redundant supporting system as described in Section 9.1.4.2.2.7 and Table 9.1.4-1.

9.1.4.4 Inspection and Testing Requirements

As part of normal plant operations, the fuel-handling equipment is inspected prior to the refueling operations. During the operational testing, procedures are followed to affirm the correct performance of the fuel handling system interlocks.

The test and inspection requirement for the equipment in the fuel handling system are:

a) Manipulator crane, spent fuel bridge crane, rod cluster control changing fixture, and new fuel elevator.

The minimum acceptable initial test shall include the following: | 37

1) Manipulator Crane and Spent Fuel Bridge Crane shall be load tested at 125 percent of the rated load. | ~~30~~

~~2~~) The equipment shall be assembled and checked for proper functional and running operation.

For condition (4) stresses do not exceed 90 percent of the elastic limit of the material.

c) During the construction phase of the project the use of the crane shall be controlled to assure that the lift capacity and other operating limitations are not exceeded as required in the applicable specifications. Following completion of the construction use and prior to Movement of spent fuel in the fuel handling building ~~final turnover to operations,~~ the crane shall be ~~completely~~ refurbished, tested and inspected in accordance with applicable specifications to assure compliance with performance requirements. ~~28~~ 37

d) Minimum operating temperature of the crane is 50 F.

e) All ferritic material which is used in load bearing structural members are impact-tested to determine fracture toughness of the material. Load bearing structural members are defined as structural members stressed in the process of transferring hook loads (vertical or horizontal) through the crane to the main runway. ASTM A-514 material is not used in any load bearing structural members; other low alloy steel may be used with CP&L's (or it's agent's) written approval.

Either drop weight test per ASTM E-208 or Charpy tests per ASTM A-370 may be used for impact testing. The minimum operating temperature, as obtained by following procedures in Subarticle NC-2300 or ND-2300 of ASME B&PV code, Section III, Div. 1, based on the drop weight test or the Charpy V-notch impact test respectively, are not higher than 50 F.

f) Welding is performed by using welding procedures, welders, welding operators, and tackers qualified in accordance with AWS D1.1. 37

g) Postweld heat treatment of welded assemblies is performed, if necessary, when an assembly is under restraint during welding, when machining is to be performed, or for welded steel greater than 1-1/2 in. in thickness at the welded joint. Welds on all load bearing structural members are Postweld heat-treated in accordance with Subarticle NF-4620 of ASME B&PV Code, 37 Section III, Div. 1, or other requirements as approved by CP&L (or its agent).

h) Where practical, weld joint designs susceptible to laminar tearing are not used. Weld joints susceptible to laminar tearing are ultrasonically tested for soundness of base metal and weld metal of the completed weld joint.

i) Full penetration butt welds on all load-bearing structural members are 100 percent radiographed for soundness of weld metal and base metal where accessible. Full penetration tee welds on all load-bearing structural members and full penetration butt welds on all load-bearing structural members which cannot be radiographed are tested as follows:

1) Magnetic particle or liquid penetrant test of root pass and final weld layer.

2) Ultrasonic test of completed weld joint for soundness of weld metal and base metal.

All fillet welds and partial-penetration welds are visually inspected in accordance with and to the acceptance criteria of AWS D1.1 Paragraph 9.25. Fillet welds and partial-penetration welds joining load-bearing structural

Fire hose is hydrostatically tested in accordance with the recommendations of NFPA 1962, "Fire Hose - Care, Use, Maintenance". Hose stored in outside hose houses will be tested annually. Interior standpipe hose will be tested every three years.

The standpipe system is designed and sized to provide, to the most remote hose station, the flow rate and pressure required for effective hose streams.

Operation of a hose station associated with a particular riser is alarmed locally and alarmed and annunciated at the Main Fire Detection Information Center (MFDIC) in the Plant Communications Room and the Control Room following sensing of water flow in the standpipe riser by system flow switches.

Sectional shutoff valves provided for standpipes serving hose stations in safety related areas are located outside the safety related areas to permit access during a fire.

Portions of the standpipe and hose systems installed in the Containment, Reactor Auxiliary and Fuel Handling Buildings, as shown on (Figures 9.5.1-2 and 9.5.1-4), are designed to be operable, if needed, for manual fire control in areas required for safe plant shutdown following a safe shutdown earthquake (SSE). These portions of the standpipe system were analyzed for SSE loading and seismically supported to assure system pressure integrity. The piping and valves for these standpipes are designed to satisfy ANSI B31.1, "Power Piping."

Normally, the post-SSE standpipe hose station header is supplied from the fire protection water distribution system through seismically qualified check valves. Following an SSE event, water supply for the post-SSE portion of the standpipe system can be obtained by local operator manual positioning of valves to connect the Seismic Category I Emergency Service Water System, located in the Reactor Auxiliary Building, to the post-SSE hose standpipe header. Seismic Category I water supply is provided for the Post SSE Fire Protection Standpipe and Hose System by the emergency service water booster pumps. The ESW booster pumps are normally used following a LOCA to provide high head cooling water to the containment fan coolers. However, in the event of a Post-SSE fire, the ESW pump (A or B) and ESW booster pump would be started. This arrangement provides sufficient TDH to supply the two most remote hose stations with 75 gpm (each) of water at approximately 65 psig as discussed in NFPA. The seismic check valves prevent outflow to other portions of the fire protection water distribution system, which may have failed during the seismic event, and thus avoid loss of hose line protection after the earthquake.

c) <u>Self-Contained Breathing Equipment</u> - Breathing equipment is provided as required for protection against smoke inhalation of personnel required to be in plant areas to control fires or to continue vital plant operations.

Self-contained breathing apparatus, using full face positive pressure masks, approved by National Institute for Occupational Safety and Health (NIOSH), with a minimum capacity of one half hour, are provided for fire brigade and control room personnel.

An additional hour of ~~Two extra~~ air in bottles is ~~are~~ located onsite for each self-contained breathing unit, used by fire brigade and control room personnel, with an onsite six hour 37

During periods of diesel generator standby, the jacket water cooling system is automatically maintained ~~at 150 F~~ between 145F and 170F by means of an electric jacket water keep-warm heater, jacket water keep-warm thermostat, and a motor driven jacket water keep-warm pump (see Figure 9.5.5-1). High and low temperature alarms monitor the jacket water temperature and the "keep warm" pump is tripped automatically upon start of the engine. The jacket water heater is provided with power from a non-safety power distribution panel in the diesel generator building. 37

Engine cooling water system design precludes trapping of air within the engine spaces. Vents are provided in the jacket water cooling system standpipe in order to assure that all spaces are filled with water. Provisions are provided to treat the jacket water by adding or removing chemicals. Corrosion and organic fouling are controlled by ~~utilizing chromate, dichromate~~ using sodium molybdate and/or other suitable chemicals and biocides. The chemicals utilized are compatible with the system materials and each other. The pH of the jacket cooling water is maintained between 8.25 and 9.75 as specified by the manufacturer. 37

The jacket water heater is conservatively sized and will maintain jacket water between 145F and 170F ~~at 150 F when the room ambient temperature drops to a minimum of 60 F. The minimum room temperature of 60 F will be maintained by two electric unit heaters and is based on an outside air temperature of -2 F.~~ A description of the Diesel Generator Building Ventilation System is provided in FSAR Section 9.4.5. 37

The total heat rejection at 110 percent load from the jacket water heat exchanger is 18,078,456 Btu/hr based on 95 F Emergency Service Water maximum inlet temperature. The heat exchanger is designed to a duty of 20,662,000 Btu/hr.

9.5.5.3 Safety Evaluation

The Diesel Generator Cooling Water System is designed to have adequate capability to carry away the waste heat from diesel generator units under all loading and ambient conditions. The diesel generator is capable of operating fully loaded without secondary cooling for a minimum of one minute. Sufficient water is contained in the engine and standpipe to absorb the heat generated during this period. The normal supply of cooling water for the diesel generator is the normal service water pump. Upon loss of offsite power the emergency service water pump will supply cooling water to the diesel generator after a period of 20-25 seconds.

The Diesel Generator vendor, Transamerica De Laval, ran a continuous 24 hour load test on a diesel engine-generator set similar to Shearon Harris' unit. The test engine ran for 22 hours at 100 percent load, followed by two (2) hours at 110 percent load. Test indicated that less than three (3) gallons of water was lost due to evaporation, boil off, and minor leaks.

The NPSH requirement for the engine jacket water pump corresponds to a minimum standpipe level of 53 1/16 in. Normal water level is indicated by LI-1DG-2466A,BV and is maintained above the low water mark by manual addition of potable water. ~~at 212 3/4 in. and low water level alarm is at 185 1/2 in.~~ Assuming a standpipe water level of 186 1/2 in. at start of seven (7) days of continuous 100 percent load operation, approximately 400 gallons of water is available between elevation 186 1/2 in. and 53 1/16 in. The standpipe, with a capacity of 400 gallons, provides more than adequate water to maintain the required pump NPSH and make-up for seven days of continuous operation. 37

All components of the Diesel Generator Cooling Water System are designed to Seismic Category I requirements. The jacket water heat exchanger and connections to the Emergency Service Water System are also designed to Safety Class 3 requirements. Failure of any non-Seismic Category I structures and components will not affect the safety related performance of the system. The diesel engine-mounted cooling water system piping and components, meet the guidelines as stated by the DEMA standards. The design stresses which includes mechanical, pressure, thermal, and seismic induced loads for the engine-mounted piping have been determined by the Diesel Generator manufacturer (i.e., Transamerica DeLaval Incorporated (TDI)) to be well within the allowable stresses as permitted by ANSI B31.1.

The TDI approved QA/QC program used in conjunction with the manufacture of diesel engines, engine-mounted components, and piping comply with the requirements of Appendix B of 10CFR50.

Each diesel generator has its heat exchanger's tube side connected to the respective emergency service water system train. Therefore, a single failure of a component, or the loss of a cooling source will not reduce the safety related functional performance capabilities of the system. The jacket water standpipe is provided with low level instrumentation for leak detection. In addition each diesel generator room is equipped with a sump and sump pump to collect and dispose of leaking fluids within the Diesel Generator Building. The sumps are provided with safety class 1E level instrumentation with annunciation in the Control Room to alert operators of potential flooding. The sump pumps are automatically actuated on high sump level.

This system is housed in a Seismic Category I Structure (Diesel-Generator Building) that is capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, and missiles. As shown on Figures 1.2.2-86 and 1.2.2-87, each diesel generator is located in a separate room. The protection of safety-related systems from the effects of high and moderate energy piping failures are considered in the design of the diesel generator facility. The facility does not contain any high energy lines but does contain the moderate energy lines of the Emergency Air, Fire Protection, Fuel Oil, Lube Oil, Miscellaneous Drains, Service Water, Potable Water, and Station Air systems. Facility design provides for the effects of failures (cracks) in these moderate energy fluid systems. Flooding from cooling line leaks does not impact other diesel generator areas where the line break has not been postulated. Facility design as shown on Figures 1.2.2-86 and 1.2.2-87, and sump drain design as shown on Figure 9.5.5-2 precludes flooding impact on the unaffected diesel generator area. A further discussion of the postulated piping failures in high and moderate energy fluid systems is located in Section 3.6.

9.5.5.4 Testing and Inspection

The diesel generator jacket system will be tested during the periodic diesel generator tests as described in Section 8.3.1.1, and its standby (keep warm) condition shall be monitored per plant operating procedures. System instrumentation loop checks and setpoint calibration will be performed in accordance with the SHNPP Preventive Maintenance Program or at each refueling outage, whichever is the earliest.

All maintenance on the emergency diesel generator will be followed by a verified line-up and post-maintenance test in accordance with the surveillance requirements of Technical Specifications. The line-up procedure will verify that the keep-warm system is properly aligned. Testing of the diesel generator operation does not require realignment of this keep-warm system.

The cooling water in the closed loop system is periodically analyzed to monitor its condition and treated as required to maintain its quality.

9.5.5.5 Instrumentation Application

The following alarm points with local annunciation are provided in the Diesel Generator Cooling Water System for each diesel generator:

a) jacket water inlet high/low temperature

b) jacket water outlet high/low temperature

c) jacket water high temperature trip

d) standpipe low level

e) jacket water pressure

f) jacket water low pressure trip

Jacket water pressure switch (PS-22C) and jacket water low pressure trip (PS-21C) are separate pressure switches which are connected to a common process tap (refer to Figure 9.5.5-1).

Pressure settings for jacket water pressure switch is 12 psi and decreasing (alarm point), jacket water low pressure trip switch is 10 psi and decreasing.

Operation of any of the above mentioned local alarms is indicated by annunciation on the Diesel Generator Control Panel and also "trip" or "trouble" alarms on the Main Control Board. In addition, pressure and temperature devices are provided for local indication and thermocouples are provided for remote indication of temperature. Temperature settings for jacket water low temperature inlet/outlet alarm switch actuation is 140 F decreasing respectively. ~~This alarm is functional only during the diesel generator operational mode.~~ In addition, temperature settings for the jacket 37
water high temperature inlet/outlet alarm switch actuation is 175 F increasing and 190 F increasing respectively. One thermocouple is placed in the piping between the return header and standpipe (jacket water outlet high/low temperature) and the second thermocouple is placed in the piping between the

described in Section 11.5.2.6.5. The monitor is powered by the Emergency A Bus. A containment isolation actuation signal will isolate this monitor from the Containment.

This monitor provides a radiation alarm when concentrations reach preset limits. The receipt of this alarm will alert the operator to the presence of low level leakage so that additional sampling can be effected in order to locate the leakage source. An interlock ~~will be~~ is provided to terminate continuous purge operation on high radiation. 37

12.3.4.2.8.2 Control Room Normal Outside Air Intake

The control room normal outside air intake plenum has two beta sensitive monitors, one associated with A Bus, and one with B Bus. These monitors are part of the safety related portion of the RMS (Section 11.5.2.3) and use the ambient gas monitors described in Section 11.5.2.6.1.

These monitors provide a high radiation alarm when concentration levels reach preset limits. Upon receipt of the alarm, the monitor closes the normal outside air intake valves associated with a given unit, stops the exhaust fans, closes the exhaust dampers, starts up the emergency filtration fans and opens the required valves and dampers to put the air flow into the recirculatory mode. The receipt of these alarms will also alert the operator to check the radiation levels at both emergency outside air intakes, and to open the intake at which the radiation level is lower (Section 12.3.4.2.8.3).

12.3.4.2.8.3 Control Rooms Emergency Outside Air Intake

There are two emergency outside air intakes. Each intake has two duct-mounted Beta monitors associated with A Bus (Intake 10 and 11A) and B Bus (Intake 10 and 11A). These monitors are part of the safety related portion of the RMS (Section 11.5.2.3), and use the ambient gas monitors described in Section 11.5.2.6.1.

These monitors provide indication to the control room personnel of the radioactivity levels at each emergency air intake, thereby allowing the operator to choose which emergency intake to open (see discussion in Section 7.3.1.5.7 and 12.3.4.2.8.2). These monitors also provide a high radioactivity alarm when concentration levels reach preset limits.

There is one outside air intake monitor for the Technical Support Center (TSC). This monitor is also an in-duct ambient beta monitor. The radiation levels and any alarms are received on the RMS consoles in the Control Room, the WPB Control Room, and the access control point. They are also monitored by the Emergency Response Facility Information System (ERFIS).

13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT ~~27~~

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

13.1.1.1 Organizational Arrangements

Since the first nuclear generating unit belonging to CP&L began commercial operation in March 1971, the amount of nuclear generating capacity on the Company's system derived from nuclear power has increased substantially. Accordingly, the Company's responsibilities in connection with its nuclear facilities have grown. During this period of time, the Company has developed and enhanced its capabilities with respect to the construction, operation, and maintenance of its nuclear facilities. The Company has safely managed H. B. Robinson Unit 2, and Brunswick Units 1 and 2 since they were placed into operation. The Company has also managed the construction of the Brunswick and ~~27~~
Harris facilities. The Company has been, and will continue to be totally committed to safety and quality in the construction and operation of our nuclear facilities.

The Company has reorganized its management structure several times to accommodate and better manage the increased nuclear capacity and additional associated personnel. The most recent major reorganization, announced on September 1, 1983, reflects the strengths developed and lessons learned from the Company's operating experience as well as from the experiences of the rest of the nuclear utility industry. It focuses the authority and responsibility for operation, engineering, and construction under one individual at each of CP&L's three nuclear plant sites. In addition, it ties many of the related offsite nuclear support organizations to the Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBR) plant organizations and places them under one individual, the Senior Vice President - Nuclear Generation. The Vice President, Brunswick Nuclear Project (BNP), who presently reports directly to the Senior Executive Vice President, Power Supply & Engineering and Construction, also benefits from the support services that are under the Senior Vice President - Nuclear Generation (see Figure 13.1.1-1).

The Company's nuclear projects are supported by an extensive organization that provides expertise in a variety of areas. For the most part, the organizations are structured to focus nuclear activities within separate departmental and organizational structures. This philosophy ensures that the Company's other, nonnuclear activities will not divert appropriate management attention from the conduct of its nuclear activities.

The Corporate support for nuclear activities is managed by the Senior Executive Vice President - Power Supply and Engineering & Construction Groups who reports to the President/Chairman/Chief Executive Officer. 37

Reporting to the Senior Executive Vice President - Power Supply and Engineering & Construction are: a) Senior Vice President - Nuclear Generation Group; b) Senior Vice President - Fossil Generation and Power Transmission Group; c) Senior Vice President - Operations Support Group;

d) Vice President - Brunswick Nuclear Project Department; and e) Manager - Corporate Quality Assurance Department (see Figure 13.1.1-2). The responsibilities of each of these groups and departments are described below:

13.1.1.1.1 Nuclear Generation Group

The Nuclear Generation Group is responsible for providing offsite technical and managerial resources to assist and support the operating nuclear plants in areas of nuclear licensing, civil design, instrumentation and controls, computers, mechanical, electrical, nuclear engineering, metallurgical analysis, construction, operations, and industrial security.

37 | Reporting to the Senior Vice President - Nuclear Generation Group are: 1) the Vice President - Harris Nuclear Project Department; 2) the Vice President ~~Manager~~ - Robinson Nuclear Project Department; 3) the Vice President - Nuclear Engineering & Licensing Department; 4) the Manager - Nuclear Plant Construction Section; and 5) the Manager - Nuclear Staff Support Section (see Figure 13.1.1-3).

1) <u>The Harris Nuclear Project Department</u> is responsible for managing the site activities in a manner which will promote the economic, safe, reliable, and effective operations of the plant over its lifetime. The organization, formed on September 1, 1983, represents the Company's concept of providing
37 | more direct on-site management control over ~~all~~ engineering, construction, startup, and operations activities at the plant. Other support functions are provided from other departments in Power Supply and Engineering & Construction.

The Vice President - Harris Nuclear Project is responsible for managing ~~all aspects of~~ engineering, construction, startup, operation, and maintenance of the Harris Nuclear Project. He conducts these activities in a manner which protects the health and safety of the public, is in compliance with the applicable governmental regulations, and is within the policies and guidelines of the Company.

Reporting to the Vice President - Harris Nuclear Project Department are:
37 | a) the General Manager - Harris Plant Operations Section; b) General Manager - Milestone Completion Section; c) General Manager - Harris Plant Engineering Section; d) Manager - Project Administration Section; e) Manager - Planning and Controls Section; and f) Manager - Completion Assurance Section (see Figure 13.1.1-4).

37 | a) The Harris Plant Operations Section is responsible for the startup testing, operation, maintenance, security, environmental and radiological control, and management of the plant in accordance with its construction permit or operating license, Technical Specifications and Plant Operating Manual (see Section 13.1.2).

b) The Milestone Completion Section is responsible for achieving the completion of project milestones in conformance with NRC regulations, code~~,~~s procedures, permits, specifications and drawings; and company policies and commitments. The General Manager - Milestone Completion has direct management responsibility for plant construction and task responsibility for portions of engineering, startup, planning, and scheduling to ensure completion on schedule, within budget, and in strict compliance with commitments to QA and ALARA requirements...

c) The Harris Plant Engineering Section (HPES) is responsible for providing as required, detailed engineering modifications and maintaining design and procurement documents. The Section is supported by Nuclear Engineering and Licensing and/or outside consultants. The Section provides engineering support for the construction and/or operation implementation of these modifications; to the operations organization in areas such as spare parts, Q-list equipment, and equipment qualification; and in the review of plant operating, maintenance and surveillance procedures as requested. A major benefit of this process is that the same technical staff that administered the design of the Harris Plant during its construction will be responsible for the engineering support of plant operations.

d) The Project Administration Section supports the administrative needs of the Harris Nuclear Project Department by providing a centralized source for these services. The Section provides these services either through its own central organization location or through satellite offices located with the various organizations it supports. These activities span a range of responsibilities from coordination of some activities, such as training, employee relations and computer services coordination, to management responsibility for activities such as document control and warehousing. ~~27~~

e) The Planning and Controls Section aids management in ensuring that a consistent, coordinated structure of work activities is achieved which focuses on the objectives and goals of the Department. The Section monitors the resulting structure and reports information to other site management indicating compliance with or variances from the plan. Primary responsibilities of the Section are to identify, develop, and implement programs, systems, methods, and related documents for planning and scheduling, budgeting, cost control, cost assurance, and industrial engineering such that management visibility is maintained to historical accomplishments as well as anticipated variances. Information and forward visibility permits corrective action while managerial alternatives remain open.

f) The Completion Assurance Section is responsible for handling special assignments as directed by the Vice President - Harris Nuclear Project.

2) The Robinson Nuclear Project Department operates and maintains the Company's nuclear generating facility at the H. B. Robinson Plant. ~~Reporting to the Manager - Robinson Nuclear Project Department are: the General Manager - Robinson Plant Section, Manager - Planning & Scheduling Section, Manager - Project Construction Section, Manager - Control & Administration Section, and Manager - Design Engineering Section (see Figure 13.1.1-5). The Robinson Plant Section organization and~~ The responsibilities of the Vice President - Robinson Nuclear Project are similar to those ~~described for the General Manager - Harris Plant Section in Section 13.1.2.2.1~~ of the Vice President - Harris Nuclear Project. 37

3) The Nuclear Engineering & Licensing Department is responsible for the licensing and engineering support of the Company's nuclear generating facilities. Reporting to the Vice President - Nuclear Engineering & Licensing

Department (see Figure 13.1.1-6) are: a) the Manager - Nuclear Licensing Section; and b) the Manager~~s~~ - ~~Engineering Support, Nuclear Plants Sections I and II~~ Nuclear Plant Engineering Section; c) the Manager - Nuclear Engineering Projects Section. ~~and d) Director - Nuclear Engineering Safety Review Unit (see Figure 13.1.1-6).~~

a) The Nuclear Licensing Section acts as the Company's interface with the NRC Office of Nuclear Reactor Regulation and, for multiple plant activities, the Office of Inspection and Enforcement. The section is organized into four units with the following functional responsibilities:

~~27~~ The Project Nuclear Licensing Units are responsible for coordination of Office of Nuclear Reactor Regulation (ONRR) activities affecting the Company's three nuclear projects. This includes the coordination and preparation of responses to ONRR requests, and the preparation of license amendments and licensing documents such as the Harris Final Safety Analysis Report (FSAR). These units are responsible for the maintenance of operating licenses, revisions to the technical

37 specifications, and updating of FSARs. The units participate in industry organizations and utility owners' groups.

The Special Nuclear Programs Unit is responsible for coordination of generic licensing issues. This includes coordination and preparation of responses concerning generic ONRR activities affecting the Company's four nuclear units. It advises Company management on critical licensing issues and ensures that incoming NRC correspondence is routed properly and that responses are prepared to address licensing issues accurately. In addition, Special Nuclear Programs coordinates the Company's regulatory related involvement in industry organizations including AIF, EEI, and EPRI. This Unit also participates in various utility owners' groups and supports other special projects of a technical or regulatory nature as required.

Engineering Support Nuclear Plant Sections'

b) The Nuclear Plant Engineering ~~Support, Nuclear Plants~~ Section~~s~~ is ~~are~~ responsible for providing engineering support for the Company's nuclear plants and for utilizing feedback received from the operating plants so as to prevent identified problems from recurring. The ~~Sections'~~ objectives are to provide engineering and procurement of engineered products on schedule with designs that are economical, safe, efficient, reliable, and compatible with the environment. ~~The Engineering Support, Nuclear Plants Sections are organized into technical units along discipline lines which are headed by Principal Engineers. The Unit Heads are~~

37 ~~responsible to the Section Managers for ensuring the project work which falls into their areas of responsibility is accomplished in such a manner that the Sections' accountabilities are fulfilled.~~ They provide the design engineering necessary to resolve those operating plant problems referred to them ~~their Units~~ and are responsible for utilizing operating plant feedback and for identifying potential problems which might affect the design and engineering of current ~~power plant~~ design ~~construction~~ projects and designed modifications. These Units are staffed with engineers and designers of required experience, education, and capability. Architect/Engineers (A/E) and other consultants may also be retained to assist the Sections in meeting their objectives.

Insert A - from page 13.1.1-5

Reporting to the Manager - Nuclear Plant Engineering Section are: a) the Managers - Engineering Support, Nuclear Plant Sections I and II; and b) Director - Nuclear Engineering Safety Review Unit.

c) The Nuclear Engineering Projects Section is divided into three units: Nuclear Projects Unit I, Nuclear Projects Unit II, and Engineering Administrative Unit. The Section is responsible, through its Nuclear Projects Units, for ensuring that the NELD provides the required design and engineering support for each nuclear project and that the nuclear projects appropriately utilize the resources of NELD. The nature of this support is reflected in defined written agreements with each of the projects and in accordance with other departmental procedures and/or guidelines. The Section establishes the scope, content, and magnitude of projects assigned to A/Es ~~Architect/Engineers~~ and manages the A/E engineering work throughout the final acceptability of the design project. |37

The Engineering Administrative Unit provides the ~~technical~~ support services required by the Sections in the Department. Priorities are set to meet the identified schedules established for the nuclear projects. The Unit serves as the focal point for collecting, processing, and disseminating required information to allow responsible management to monitor schedule and cost progress on ~~all~~ assigned plant modification projects and provides support in engineering schedule preparation, engineering, scheduling services during project implementation, supplement scope development, QA records support, and other engineering administrative support to in-house engineering design sections within the Department. |37 |37

Insert A ... to the ... of current page 13.1.1-4

~~d)~~ The Director - Safety Review, Nuclear Engineering is responsible for reviewing documents generated by the Company's nuclear organization and A/Es identifying problems in engineered safeguards systems and plant safety features; assessing activities and trends in the industry regarding design and operation of safety features; providing feedback to preclude potential nuclear safety problems in ongoing plant designs and design of modifications; and assuring that ALARA concepts for radiation control are considered in engineered designs.

4) The Nuclear Plant Construction Section manages the procurement and contracting activities required to support the completion of construction project assignments. The Section provides both firm-price and reimbursable contracts, onsite procurement and expediting services, and construction equipment and tool management. Onsite procurement staffs have been established at the Harris, Robinson, and Brunswick Nuclear Projects. The Section also provides support services to the other Departments within the Company in the areas of estimating, budgeting, cost control, cost reporting, construction accounting, information management, and construction security.

5) The Nuclear Staff Support Section is primarily responsible for coordinating the implementation and maintenance of operationally oriented programs that require high technical knowledge of methods and procedures and that should be relatively consistent among the plants. The Section is also responsible for preparing reports and documents, performing staff studies, providing operations security support, providing administrative/technical support as required and coordinating the Company's involvement in Institute of Nuclear Power Operations (INPO). These efforts are coordinated with each project. |37

13.1.1.1.2 Fossil Generation and Power Transmission Group

The Senior Vice President - Fossil Generation and Power Transmission Group is responsible for managing the Company's fossil and hydro generating facilities and the Company's transmission line facilities necessary to meet its bulk power requirements.

37 ~~Reporting to the Senior Vice President - Fossil Generation and Power Transmission Group are: the Manager - Fossil Engineering & Construction Department, the Vice President - Fossil Operations Department, the General Manager - System Operations Department, the Vice President - Transmission Department, the Manager - Maintenance Support Section, and the Assistant to Group Executive - Fossil Generation & Power Transmission (see Figure 13.1.1-9).~~

~~21~~ 13.1.1.1.3 Operations Support Group

37 The Senior Vice President - Operations Support Group is responsible for the management of the materials and nuclear fuel needs of the generating and transmission facilities in addition to the training and technical support of those personnel.

37 Reporting to the Senior Vice President - Operations Support Group (see Figure 13.1.1-7) are: ~~1) the Manager - Fuel Department,~~ ~~2~~1) the Manager - Materials Management Department, ~~3~~2) the Vice President - Operations Training and Technical Services Department, and ~~4~~3) the Manager - Nuclear Safety and Environmental Services Department ~~(see Figure 13.1.1-7)~~. Their responsibilities are summarized below:

37 ~~1) The Fuel Department ensures the proper management of nuclear and fossil fuels used for the production of electrical power. Reporting to the Manager - Fuel Department are: (a) the Nuclear Fuel Section, (b) the Fossil Fuel Section, (c) the Administration and Analysis Unit and (d) the Fuel Cost Administration Unit (see Figure 13.1.1-8).~~ The Nuclear Fuel Section is staffed with personnel having both the technical and managerial expertise required to ensure a timely and adequate supply of nuclear fuel, to review fuel and core design, to support nuclear plant outages (including refuelings) and operations, and to provide for spent fuel management. The Nuclear Fuel Section meets with members of the Company's operating nuclear plants on a continuing basis to plan and optimize the fuel operation strategy. Insert B (move to page 13.1.1-7)

37 ~~2~~1) The Materials Management Department is responsible for corporate purchasing, inventory control, warehousing, and salvage of the Company's

37 material needs ~~(see Figure 13.1.1-9)~~.

37 2) The Operations Training and Technical Services Department supports nuclear/fossil training, management of nuclear fuel, the plants' chemistry and health physics programs, and provides radiological environmental support.

Reporting to the Vice President - Operations Training and Technical Services Department are: (a) the Manager - Nuclear Training Section, ~~and (b) the Emergency Preparedness Section (see Figure 13.1.1-10).~~ (see Figure 13.1.1-10) (b) Manager - Nuclear Fuel Section, (c) Manager - Corporate Health Physics and, (d) Manager - Radiation and Chemical Support.

a) The Nuclear Training Section provides support to the Nuclear Project Departments in the areas of Operations, Technical and Craft Training, and the operation of the simulators and other training facilities at the HE&EC and at the respective nuclear projects. The primary purpose of the Nuclear Training Section is to assure that the Company has highly qualified personnel available to maintain and operate its nuclear generating plants in a safe and efficient manner. ~~These responsibilities and services are provided by an organization consisting of eight units which support nuclear projects: the Nuclear and Simulator Training Unit, the Fossil Operator Training Unit, the Craft and Technical Training Unit, the Administrative Unit and the Curriculum Development Unit at the HE&EC; and the Robinson Training Unit, the Brunswick Training Unit, and Harris Training Unit located at the respective nuclear plants.~~

b) Insert B - from page 13.1.1-6

c ~~b~~) The Emergency Preparedness Section is responsible for: directing and coordinating Corporate Emergency Planning to ensure regulatory compliance; assessing the readiness of ~~all~~ CP&L emergency plans and programs; serving as interface with regulatory agencies on emergency preparedness matters; providing emergency preparedness support for CP&L nuclear plants; maintaining training qualifications of plant personnel in emergency response; testing emergency preparedness by preparing and conducting exercises; ensuring the availability and operational readiness of emergency facilities, equipment, and supplies; ~~developing dam failure emergency plans for the hydro plants~~ and providing coordination with federal, state, and local agencies. ~~2~~ 37

Insert C (move to page 13.1.1-9)

The Physical Sciences Unit within the Emergency Preparedness Section is responsible to provide meteorological support to the Company's nuclear and fossil plants and other departments within the Company. The primary purpose of the ~~U~~nit is to provide meteorological data, forecasts and expertise to the Company's nuclear facilities, assisting in the dose projection capabilities of the plants' emergency response personnel by providing micro-meteorological forecasts and dispersion expertise. The ~~U~~nit also provides on a routine basis, early notification to appropriate Company personnel those severe weather events which may affect either the generation of power or customer services. ~~The unit is additionally charged with a responsibility to provide professional consultation services for those meteorological matters to other persons within the Company who need such technical assistance.~~

4) The Nuclear Safety and Environmental Services Department monitors and reviews ~~the~~ plant nuclear safety ~~and health physics programs~~; conducts environmental assessments; performs chemical and materials laboratory services; directs the emergency preparedness programs; and provides staff support in several technical disciplines.

Reporting (see Figure 13.1.1-11) to the Manager - Nuclear Safety and Environmental Services Department are: a) the Manager - Corporate Nuclear Safety Section, b) the Manager-Emergency Preparedness ~~Corporate Health Physics Section, c) the Radiological and Chemical Support Section~~, and c ~~d~~) the Manager Environmental Services Section. ~~(see Figure 13.1.1-11).~~

a) The Corporate Nuclear Safety (CNS) Section combines technical expertise in an integrated off-site/on-site program to monitor and evaluate plant nuclear safety performance. The organization fulfills requirements specified in ANSI N18.7 for Independent Review and NUREG-0737 for the Independent Safety Engineering Group (ISEG). In

c) Insert D - from page 13.1.1-8
d) Insert E - from page 13.1.1-8

addition, CNS provides the base for the operating experience feedback ~~(OEP)~~ program within CP&L as well as input to the establishment of priorities of nuclear safety-related items.

The CNS independent review activity addresses the following:

(1) Procedures and changes meeting 10 CFR 50.59 review criteria,

(2) Licensing actions,

(3) Test or experiments not described in the facility FSAR,

(4) Plant operational occurrences (LERs),

(5) Regulatory violations (IE Reports),

(6) Technical Specification changes

(7) Plant Nuclear Safety Committee (PNSC) meeting minutes, and

(8) Any item deemed appropriate for review relative to safe operations.

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move to page 13.1.1-7

~~b)~~d The Corporate Health Physics Section consists of industrial ~~of~~ personnel with education and/or work experience in fields of ~~radiation~~ hygiene or health physics. The section is also responsible for formulating and recommending corporate level health physics policies and programs, evaluating health physics programs and recommending any needed improvements and modifications in those programs, and providing health physics expertise throughout the Company. The Section provides support to the licensing and corporate nuclear safety activities of the Company, is responsible for the development and distribution of the Corporate ALARA Program, and makes periodic assessments of various ALARA programs developed to comply with the Corporate ALARA Program. This ~~S~~section also conducts a~~n~~ semi annual audit of the QA program.

Insert E
move to page 13.1.1-7

e~~c~~) The Radiological and Chemical Support Section (R&CSS) provides staff support in the areas of health physics, chemistry, and radiological environmental activities and for the effective operation of the environmental, dosimetry, and chemistry laboratories. The R&CSS has responsibilities identified in the Corporate Emergency Plan to provide health physics and environmental support to the nuclear plants in the event of an accident. These responsibilities and services are provided by an organization consisting of three units; ~~headed by two principal specialists and a director:~~ the Health Physics Unit, the Environmental Unit, and the Chemistry Unit.

37 b~~d~~) The Environmental Services Section ~~(ESS)~~ conducts the Company's environmental monitoring assessments and performs analytical chemistry and metallurgical laboratory services at the Harris Energy & Environmental Center (HE&EC) in New Hill, North Carolina. The Analytical Chemistry, Air Quality, Biology, and Metallurgy Laboratories provide an array of services and technical support to generating plants, engineering activities, quality assurance and construction

programs within the Company. ~~One subunit of the Biology Unit is located at BSEP.~~ The Water Environmental Regulations and Permits Unit is responsible for obtaining the National Pollutant Discharge Elimination System (NPDES) permits and together with the Air Environmental Regulation and Permits Unit, any federal, state, and local permits not required by the NRC.

c) Insert C - from page 13.1.1-7

13.1.1.1.4 Brunswick Nuclear Project Department

The Vice President - Brunswick Nuclear Project Department reports to the Senior Executive Vice President - Power Supply & Engineering and Construction. His responsibilities are similar to those of the Vice President - Harris Nuclear Project Department, ~~and he is supported in these responsibilities by the General Manager-Brunswick Plant, the Engineering and Construction Section, the Site Planning and Control Section, and the Brunswick Nuclear Project Outages Section (see Figure 13.1.1-12). These sections are responsible for the operation, maintenance, engineering, construction, and management of the Brunswick Plant.~~ 37 ~~21~~

13.1.1.1.5 Corporate Quality Assurance Department

The Manager of the Corporate Quality Assurance Department reports to the Senior Executive Vice President - Power Supply and Engineering & Construction (see Figure 13.1.1-1) and is responsible for the quality assurance, quality control, and audit functions which were at one time performed separately for engineering and construction, operations, and corporate quality assurance audit activities. In this manner, the Manager - Corporate Quality Assurance oversees the QA/QC activities of the Power Supply and the Engineering & Construction Groups while maintaining independence from any responsibilities within those organizations. Refer to FSAR Section 17.2 for a description of the Quality Assurance Organization.

Delete

~~Reporting to the Manager - Corporate Quality Assurance Department are: 1) the Harris Plant QA/QC Section, 2) the Operations QA/QC Section, and 3) the QA Services Section (see Figure 13.1.1-13). Their responsibilities are summarized below:~~

~~a) The Harris Plant QA/QC Section has the primary responsibility for the Harris Plant Quality Assurance/Quality Control in the engineering and construction phase. Its purpose is to anticipate and preclude safety-related nonconformances. This section is also responsible for the preparation of the ASME "N" Stamp QA Manual.~~

~~b) The Operations QA/QC Section is responsible for assuring proper application of quality standards, practices, and procedures associated with plant startup, operation, maintenance or modification at CP&L operating plants.~~

~~c) The QA Services Section is responsible for supporting CP&L's nuclear plants in the areas of QA Engineering, vendor qualification/ surveillance and training. This section is also responsible for conducting an independent corporate audit program.~~

13.1.1.2 Qualifications

Carolina Power & Light Company's management and technical support staff positions performing key functions for the SHNPP project are filled by individuals with several years of experience as presented in Table 13.1.1-1.

27
37 Corresponding resumes are provided in Tables 13.1.1-2 through 13.1.1-6. ~~Organizational charts are provided as figures at the end of Section 13.1.~~

The General Manager, Harris Plant Engineering Section is the "Engineer-in-Charge" as specified in ANS 3.1, September 79 Draft.

TABLE 13.1.1-1

EDUCATION AND EXPERIENCE SUMMARIES FOR
KEY PERSONNEL SUPPORTING SHNPP
(as of 1985)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)	
NUCLEAR GENERATION GROUP					
A. Harris Nuclear Project All Caps	Senior Vice President - Nuclear Generation Group	M. A. McDuffie	BS Civil Eng.	37	37
	Vice President - Harris Nuclear Project	R. A. Watson	BS Nuclear Eng. MS Physics	29	~~24~~
1. Harris Plant Engineering Section	General Manager - Harris Plant Engineering Section	E. J. Wagner	BS Mechanical Eng.	33	37
	Manager - Harris Plant Engineering Management Section	M. F. Thompson, Jr.	BS Nuclear Eng. MS Nuclear Eng.	22	
	Manager - Harris Plant Engineering	L. I. Loflin	BS Electrical Eng. BS Nuclear Eng.	22	
	~~Principal Engineer - Mechanical~~	~~L. S. Whatley~~	~~BS Nuclear Eng.~~	~~23~~	
	~~Principal Engineer - Mechanical~~	~~M. Holvock~~	~~BS Civil Eng.~~ ~~MS Civil Eng.~~	~~13~~	37
	~~Principal Engineer - Mechanical~~	~~R. A. Stewart~~	~~BS Mechanical Eng.~~	~~13~~	
	~~Project Engineer - Mechanical~~	~~J. T. Duncan, Jr.~~	~~BS Mechanical Eng.~~	~~13~~	
	~~Project Engineer - Mechanical~~	~~D. P. Knepper~~	~~BS Aerospace Eng.~~ ~~MS Mechanical Eng.~~	~~16~~	

TABLE 13.1.1-1 (Cont'd)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)
	~~Principal Engineer - Electrical~~	~~J. Bullock~~	~~BS Electrical Eng.~~	~~26~~
	~~Principal Engineer - Electrical~~	~~R. W. Prunty, Jr.~~	~~BS Electrical Eng.~~	~~14~~
	~~Resident Engineer - Electrical~~	~~A. Cockerill~~	~~National Certificate Electrical Eng.~~	~~18~~
	~~Project Engineer - Civil~~	~~H. L. Williams~~	~~BA Mathematics BS Civil Eng.~~	~~7~~
2. Milestone Completion Section	~~Project~~ General Manager - Milestone Completion	G. Meyer	BS Nuclear Maritime Eng.	19
	~~Director - Electrical Construction~~	~~R. K. Stephens~~	~~HS Diploma~~	~~38~~
	~~Principal Engineer - Mechanical~~	~~A. G. Fuller~~	~~BS Civil Eng.~~	~~12~~
	~~Resident Engineer - Metallurgy/Welding~~	~~R. Hanford~~	~~BS Metallurgical Eng.~~	~~25~~
	~~Construction Superintendent Night Shift~~	~~A. Rager~~	~~HS Diploma~~	~~17~~
	~~Project Engineer - Mechanical~~	~~E. M. McClean~~	~~BS Mechanical Eng.~~	~~18~~
	~~Resident Engineer - Civil~~	~~J. W. McKay~~	~~BS Civil Eng.~~	~~13~~
	~~Resident Engineer - Mechanical~~	~~E. E. Willet~~	~~BS Mechanical Eng.~~	~~16~~
3. Completion Assurance Section	Project General Manager - Completion Assurance	R. M. Parsons	BS Civil Eng.	21
4. Harris Plant Operations Section	General Manager - Harris Plant Operations Section	J. Willis	BS Electrical Eng.	34

37 ~~24~~

TABLE 13.1.1-1 (Cont'd)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)
5. Harris Project Administration	Manager - Harris Project Administration	W. J. Hindman, Jr.	BS Civil Eng.	20
	~~Principal Specialist - Document Service~~	~~E. E. Johnson~~	~~BS Applied Mathematics~~	~~21~~
	~~Superintendent - Materials and Custodial~~	~~L. P. Capps~~	~~BS Accounting~~	~~12~~
6. Harris Project Planning and Controls	Manager - Planning and Control	T. J. Allen	BS Civil Eng. MBA	18
	Director - Planning and Scheduling	N. L. Blair	BS Engineering Mechanics	22
	Manager - Projects and Accounting	H. W. Rhodes, Jr.	HS Diploma	33
B. NUCLEAR ENGINEERING AND LICENSING DEPARTMENT	Vice President - Nuclear Engineering and Licensing Department	A. B. Cutter	BS Chem. Eng. MS Nuclear Science & Eng.	29
1. Nuclear Licensing Section	Manager - Nuclear Licensing Section	S. R. Zimmerman	BS Eng.	22
	~~Principal - Harris Nuclear Licensing~~	~~D. C. McCarthy~~	~~BS Nuclear Eng.~~	~~9~~
	~~Principal - Special Nuclear Projects~~	~~M. R. Oates~~	~~BS Electrical Eng.~~ ~~MS Electrical Eng.~~	~~25~~
3. Nuclear Plant Engineering Section	Manager - Nuclear Plant Engineering	E. J. Wagner	BS Mechanical	33
a. Engineering Support Nuclear Plants Section I	Manager - ESNPS I	R. L. Sanders	BS Eng. MS Nuclear Eng.	27
2. Nuclear Engineering Projects Section	Manager - Nuclear Engineering Projects	M. G. Zaalouk	BS Electrical Eng. MS Nuclear Eng. PhD Nuclear Eng.	28
	~~Principal Engineer - Mechanical/Electrical~~	~~H. Hines~~	~~BS Chemistry~~ ~~BS Nuclear Eng.~~	~~23~~

~~27~~ 37

Amendment No. ~~27~~ 37

TABLE 13.1.1-1 (Cont'd)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)
	~~Principal Engineer - Civil~~	~~D. Koss~~	~~BS Civil Eng.~~ ~~MS Water Resources Eng.~~	~~12~~
b. Engineering Support Nuclear Plants Section II	Manager - ESNPS II	J. Nevill	BS Civil Eng.	18
C. Nuclear Engineering Safety Review	Director - Nuclear Engineering Safety Review	S. McManus	BS Industrial Eng. BS Nuclear Eng. & Eng. Mathematics	25
C. Nuclear Plant Construction Section	Manager - Nuclear Plant Construction Section	S. N. Hamilton	BS Science	33
D. Nuclear Staff Support Section	Manager - Nuclear Staff Support Section	M. D. Hill	BS Mechanical Eng.	16
II. OPERATIONS SUPPORT GROUP	Senior Vice President - Operations Support Group	~~J. M. Davis, Jr.~~	BS Electrical Eng. ~~BS Mechanical Eng.~~	27 ~~20~~
~~FUEL DEPARTMENT~~	~~Manager - Fuel Department~~	~~W. J. Hurford~~	~~BS Metallurgical Eng.~~ ~~MS Industrial Mgt.~~	~~36~~
Nuclear Fuel Section	Manager - Nuclear Fuel Section	L. H. Martin	BS Nuclear Eng. MBA	20
	~~Principal Engineer - Incore Analysis~~	~~K. Karcher~~	~~BS Physics~~ ~~MS Nuclear Eng.~~	~~10~~
	~~Principal Engineer - Irradiated Fuel Services~~	~~R. K. Kunita~~	~~BS Physics~~ ~~MS Nuclear Science & Eng.~~	~~28~~
	~~Principal Engineer - Fuel Projects~~	~~R. G. Mathews~~	~~BS Electrical Eng.~~	~~26~~

All Caps

Insert F
move to page 13.1.1-15

~~27~~ 37

TABLE 13.1.1-1 (Cont'd)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)
	~~Project Engineer - Fuel Projects~~	~~W. R. Farr~~	~~BS Physics~~ ~~MS Nuclear Eng.~~	~~12~~
~~Administration and Analysis Unit~~	~~Principal Fuel Specialist - Fuel Cost Administration~~	~~S. R. Williams~~	~~BS Home Economics Ed.~~	~~11~~
A. MATERIALS MANAGEMENT DEPARTMENT	Manager - Materials Management Department	R. B. Richey	BS Engineering MS Industrial Eng.	21
B. OPERATIONS TRAINING AND TECHNICAL SERVICES DEPARTMENT	Vice President - Operations Training Department	B. J. Furr	BS Mechanical Eng.	23
1. Insert F - from page 13.1.1-14				
1. Emergency Preparedness Section	Manager - Emergency Preparedness Section	R. B. Black, Jr.	BS Industrial Eng.	15
2. Nuclear Training Section	Manager - Nuclear Training Section	A. C. Tollison, Jr.	BS Chemical Eng.	24
C. NUCLEAR SAFETY AND ENVIRONMENTAL SERVICES DEPARTMENT	Manager - Nuclear Safety and Environmental Services Department	R. B. Starkey, Jr.	BS Physics	21
2. Corporate Nuclear Safety Section	Manager - Corporate Nuclear Safety Section	J. D. E. Jeffries	BS Eng. MS Nuclear Eng. PhD Nuclear Eng.	21
	Director - Onsite Nuclear Safety, SHNPP	H. W. Bowles	BS Physics\Eng.	14
	Director - Nuclear Safety Review	J. G. Hammond	BS Mechanical Eng. MS Industrial Mgt.	18
3. Corporate Health Physics Section	Manager - Corporate Health Physics Section	R. L. Mayton, Jr.	BS Nuclear Eng. MS Nuclear Eng.	23
4. Radiological and Chemical Support Section	Manager - Radiological and Chemical Support Section	B. H. Webster	BS Physics	27

TABLE 13.1.1-1 (Cont'd)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)
	~~Principal Specialist - Chemistry~~	~~M. McDowell~~	~~BS Chemical Eng.~~	~~16~~
	~~Principal Specialist - Environmental~~	~~G. H. Warriner~~	~~BS Chemistry~~	~~15~~
	~~Principal Health Physics Specialist Health Physics Support~~	~~B. A. Meyer~~	~~BS Nuclear Eng.~~	~~9~~
	~~Director - Health Physics Services~~	~~J. A. Padgett~~	~~MSPH Health Physics~~	~~20~~
3. Environmental Services Section	Manager - Environmental Services	G. H. Warriner ~~R. B. Starkey (acting)~~	BS Chemistry	15
	~~Principal Engineer - Environmental Compliance~~	~~T. J. Crawford~~	~~BS Civil Eng. MS Civil Eng.~~	~~21~~
	~~Principal Scientist - Analytical Chemistry~~	~~L. L. Ball~~	~~BA Biology & Chemistry~~	~~33~~
	~~Principal Engineer - Metallurgy~~	~~D. M. Sullivan~~	~~BS Metallurgical Eng.~~	~~21~~
III. CORPORATE QUALITY ASSURANCE DEPARTMENT	Manager - Corporate Quality Assurance Department	H. R. Banks	HS Diploma	37
A. QA/QC Harris Plant Section	Manager - Material Quality ~~QA/QC Harris Plant Section~~	N. J. Chiangi	HS Diploma	34
	Director - QA/QC Harris Plant	G. L. Forehand	HS Diploma	33
	~~Principal QA/QC Specialist Nondestructive Examination~~	~~G. R. Osman~~	~~BS Engineering Physics~~	~~15~~
	~~Principal Engineer - QA Engineering Harris Plant Unit~~	~~H. V. Hatel~~	~~BS Metallurgical Eng. MS Materials Eng. MS Management~~	~~15~~

37 ~~[illegible]~~

TABLE 13.1.1-1 (Cont'd)

SECTION OR ORGANIZATION	TITLE	NAME	EDUCATION	APPLICABLE EXPERIENCE (YEARS)
B. QA/QC Operations Section	Manager - Operations QA/QC	C. H. Moseley, Jr.	BS General Eng. MS Nuclear/Civil Eng.	20
	~~Director - QA/QC Harris Operations~~	~~C. L. McKenzie~~	~~BS Industrial Eng.~~	~~14~~
C. Quality Assurance Services Section	Manager - Quality Assurance Services Section	R. E. Lumsden	BS Marine Eng.	32
	~~Project QA Engineer - QA Engineering~~	~~T. L. James~~	~~BS Mechanical Eng.~~	~~10~~
	~~Principal QA Specialist - Vendor Surveillance~~	~~W. T. Poteat~~	~~BS Electrical Eng.~~	~~23~~
	~~Principal QA Specialist Performance Evaluation~~	~~C. A. Rosenberger~~	~~BS Agricultural Eng.~~	~~32~~
	~~Principal QA Specialist - Training and Administration~~	~~H. J. Love, Jr.~~	~~BS Chemistry~~ ~~BS Electrical Eng.~~	~~29~~

37
~~21~~

TABLE 13.1.1-2 (CONT'D)

27 37
E. E. Utley
Page 2

L. June 1, 1979 - Appointed in charge of the Power Supply & Customer Services Groups - CP&L

M. May 1, 1980 - Appointed in charge of the Power Supply and Engineering & Construction Groups - CP&L

N. Senior Executive Vice President- Power Supply and Engineering and Construction Groups

III. Professional Societies:

A. American Society of Mechanical Engineers

B. North Carolina Society of Engineers

C. Raleigh Engineers Club

D. American Nuclear Society (National)

E. Eastern Carolinas Section of American Nuclear Society

F. Association of Edison Illuminating Companies - Committee on Power Generation

TABLE 13.1.1-3 (CONT'D)

Manager - Nuclear Plant Engineering 37

E. J. Wagner, ~~General Manager - Engineering~~

I. Education:

A. B. S. in Mechanical Engineering from Carnegie-Mellon University - 1953

B. Additional Courses - Case Western Reserve, George Washington University, and Ohio State University

II. Experience:

A. Exxon

1. Summer 1952 - Maintenance Engineer

B. Babcock and Wilcox Company

1. 1953 to 1955 - Test Engineer

C. Naval Nuclear Propulsion Program

1. 1955 to 1970 - Chief, Nuclear Components Branch

D. Westinghouse Electric Corporation

1. 1970 to 1975 - Division Engineering Manager

E. Burns and Roe, Inc.

1. 1975 to 1983

a. Assistant to the Executive Vice President
b. Director of Engineering and Design
c. Deputy Director for Engineering
d. Deputy Director for Technical Evaluation

F. Cincinnati Gas & Electric Company

1. 1983 to 1984 - Assistant Vice President, Nuclear Engineering

G. Carolina Power & Light Company

1. May 1984 to January 1985 - Assistant to Executive Vice President - PSE&C

2. January 1985 to Present - General Manager - Engineering, Harris Nuclear Project Department, Nuclear Generation Group

3. June 1986 - Assigned additional responsibilities of Manager - Nuclear Plant Engineering Section, Nuclear Engineering and Licensing Department. 37

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TABLE 13.1.1-3 (CONT'D)

Haines William Rhodes, Jr., Manager - Project Costs & Accounting

I. Education:

A. Diploma - Orrum High School, Orrum, North Carolina - 1943

B. Attended Evening Classes in Economics at North Carolina State University

II. Experience:

A. Ebasco Services, Inc.

1. 1951 - 1958

a. Served in Capacities of Chief Timekeeper, Assistant Accountant and Cost Engineer on Construction of Weatherspoon Unit 3, Sutton Units 1 and 2, and Cape Fear Units 5 and 6

B. Bechtel Corporation

1. 1958 - 1960

a. Cost Engineer on Construction of Port Everglades Units 1 and 2

C. Ebasco Services, Inc.

1. 1960 - 1970

a. Cost Engineer on Construction of Lee Unit 3, Asheville Unit 1, Roxboro Unit 1 and Robinson Unit 2

D. Carolina Power & Light Company

1. 1970 to Present

a. Employed as Cost Control Specialist in the Construction Section, Power Plant Design & Construction Department, located in the General Office

b. February 15, 1973 - Promoted to Senior Cost Control Specialist in the Construction Section of the Power Plant Engineering and Construction Department, located in the General Office

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TABLE 13.1.1-3 (CONT'D)

Haines William Rhodes, Jr.
Page 2

c. July 21, 1973 - Promoted to Principal Cost Control Specialist in the Construction Section, Power Plant Engineering and Construction Department, located in the General Office

d. September 1, 1973 - Transferred as Principal Cost Control Specialist to the Power Plant Construction Department, located in the General Office

e. February 11, 1974 - Assigned as Principal Cost Control Specialist to Construction Engineering and Estimating Section of the Power Plant Construction Department, located in the General Office

f. February 28, 1976 - Transferred as Principal Cost Control Specialist to Construction Services Section of the Power Plant Construction Department, located in the General Office

g. January 1, 1977 - Transferred as Principal Cost Control Specialist to Cost Control/Cost Reporting Section of the Engineering & Construction Support Services Department, located in the General Office

h. January 15, 1977 - Promoted to Manager, Cost Control and Cost Reporting in the Administrative Section of the Engineering and Construction Support Services Department, located in the General Office

i. January 10, 1981 - Transferred as Director - Cost Control & Cost Reporting to the Power Plant Construction Department in the Harris Site Management Section, located in New Hill, North Carolina

j. January 31, 1981 - Reorganization - Department renamed Nuclear Plant Construction

k. December 25, 1982 - Reclassified to Manager - Project Costs & Accounting in the Harris Site Management Section of the Nuclear Plant Construction Department, located at the Harris Site, New Hill, North Carolina

III. Professional Societies:

Raleigh Engineers Club

DELETE

TABLE 13.1.1-4

Resumes of Management Personnel in the
Operations Support Group

Lynn W. Eury
~~James M. Davis, Jr.~~, Senior Vice President - Operations Support Group

I. Education and Training:

B. S. Degree in Electrical ~~Mechanical~~ Engineering, North Carolina State University, Raleigh, North Carolina - ~~1958~~ 1959

II. Experience:

A. Companies (other than CP&L) and Military Experience

1. April 1960 - October 1960 - U.S. Army

B. Carolina Power & Light Company

1. Summers of 1957 and 1958 - Employed at Carolina Power & Light Company as a Student Summer Worker in the Northern Division Relay Office and Substation Design Office, respectively

2. June 1959 to April 1960 - Employed as a Junior Engineer in the Northern Division Relay Office in Raleigh, North Carolina.

~~April 1960 to October 1960 - U. S. Army~~

3. October 1960 through March 1970 - Employed at CP&L as a Junior Engineer in the Northern Division Relay Office in Raleigh, North Carolina, and progressed to Senior Engineer in the System Operations Section of the Power Supply Department in Raleigh, North Carolina

4. April 1970 - Employed as System Operating Engineer in the System Operations Section of the Generation & System Operations Department located in the General Office in Raleigh, North Carolina

5. January 1972 through December 1976 - Employed as Manager - System Operations Section and Manager - System Operations & Maintenance Section in the Generation & System Operations and Power Supply Departments located in the General Office in Raleigh, North Carolina

6. January 1977 - Employed as Manager - System Operations & Maintenance in the System Operations & Maintenance Department located in the General Office in Raleigh, North Carolina

7. June 1979 - Employed as Vice President - System Planning & Coordination Department in the Corporate Services Group located in the General Office in Raleigh, North Carolina

8. May 1980 through July 1983 - Employed as Vice President/Sr. Vice President - Power Supply Group located in the General Office in Raleigh, North Carolina

9. August 1983 - Title changed to Senior Vice President - Fossil Generation & Power Transmission. Located in the General Office in Raleigh, North Carolina

10. June 1986 - Employed as Senior Vice President - Operations Support. Located in the General Office in Raleigh, North Carolina

TABLE 13.1.1-4 (CONT'D)

Lynn W. Eury
~~James M. Davis, Jr.~~
Page 2

III. Professional Societies:

A. Registered Professional Engineer - North Carolina & South Carolina

B. Institute of Electrical and Electronics Engineers

~~27~~ 37 C. Professional Engineers of North Carolina

D. North Carolina Society of Engineers

E. American Nuclear Society

F. ANS - Eastern Carolinas Section

G. N. C. Chapter of the Health Physics Society

H. N. C. MATHCOUNTS Director

I. Director - N. C. Engineering Foundation, Inc.

TABLE 13.1.1-4 (CONT'D)

Walter J. Hurford, Manager - Fuels Department

I. Education and Training:

A. B. S. Degree in Metallurgical Engineering - Carnegie Institute of Technology, Pittsburgh, Pennsylvania - 1942

B. S. M. Degree in Industrial Management - Massachusetts Institute of Technology - Boston, Massachusetts - 1960

II. Experience:

A. 1949 - 1976 - Manager - Light Water Breeder Reactor Core Activity - Westinghouse Bettis Laboratory (Westinghouse Electric Corporation)

B. 1976 - 1981 - Vice President Corporate Production - Wyoming Mineral Corporation (Westinghouse Electric Corporation)

C. 1981 - 1982 - Manager of Production - Western Zirconium Division (Westinghouse Electric Corporation)

D. January 1983 - Employed at Carolina Power & Light Company as Manager - Technical Services Department in the Power Supply Group located in the General Office, Raleigh, North Carolina

E. September 1983 - Manager - Fuels Department, General Office, Raleigh, North Carolina

III. Professional Societies:

A. American Society for Metals

DELETE

TABLE 13.1.1-4 (CONT'D)

L. H. Martin, Manager - Nuclear Fuel Section

I. <u>Education</u>:

A. B. S. Degree in Nuclear Engineering - North Carolina State University - 1965

B. M. B. A. Degree - University of South Carolina - 1971

II. <u>Experience</u>:

A. 1965 to 1970 - Engineer in Reactor Technology Section - Savannah River Plant

B. April 1972 - July 1973 - Senior Engineer - Carolina Power & Light Company, Bulk Power Supply Department, Fuel Section, General Office, Raleigh, North Carolina

C. July 1973 - August 1974 - Principal Engineer - Surveillance & Accountability (In-Training) Bulk Power Supply Department, Fuel Section - CP&L

D. August 1974 - January 1977 - Principal Engineer - Surveillance & Accountability Bulk Power Supply Department, Fuel Section - CP&L

E. January 1977 - May 1977 - Principal Fuel Analyst, Fuel Department, Fuel Analysis Unit - CP&L

F. May 1977 - Present - Manager - Nuclear Fuel, ~~Fuel Department,~~ Nuclear Fuel Section - CP&L 37

III. <u>Professional Societies</u>:

A. Registered Professional Engineer - North Carolina - 1975

B. Member of American Nuclear Society

TABLE 13.1.1-4 (CONT'D)

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37

B. J. Furr, Vice President - Operations Training and Technical Services Department

I. Education and Training:

A. B. S. Degree in Mechanical Engineering - North Carolina State University - 1962

B. Basic Surveying Course - 1965

C. Basic Radiological Health Course Conducted by the Public Health Service, Winchester, Massachusetts - 1966

D. Reactor Safety and Hazards Evaluation Conducted by the U. S. Public Health Service, Rockville, Maryland - 1968

E. Westinghouse Nuclear Reactor Training Program - 1968

F. Senior Reactor Operator License on H. B. Robinson 27

II. Experience:

A. June 1955 to July 1958 - U. S. Army - Instructor in Aviation Maintenance

B. Summer 1960 - Summer Student Worker - Substation Shops - Carolina Power & Light Company - Raleigh, North Carolina

C. Summer 1961 - Summer Student Worker - Cape Fear Steam Electric Plant - Carolina Power & Light Company - Moncure, North Carolina

D. June 1962 to May 1963 - Engineer - E. I. DuPont de Nemours Company

E. May 1963 - Employed as a Junior Engineer at the W. H. Weatherspoon Plant, Lumberton, North Carolina

F. February 1964 - Employed as a Junior Engineer at the H. B. Robinson Plant, Hartsville, South Carolina

G. July 1964 - Employed as a Mechanical Engineer at the H. B. Robinson Plant, Hartsville, South Carolina

H. January 1966 - Employed as a Mechanical Engineer at the Roxboro Steam Electric Plant, Roxboro, North Carolina

I. February 1966 - Employed as Operating & Results Supervisor at the H. B. Robinson Plant, Hartsville, South Carolina

TABLE 13.1.1-4 (CONT'D)

B. J. Furr
Page 2

J. September 1971 - Employed as a Principal Engineer in the Nuclear Generation Section of the Generation & System Operations Department in the General Office

K. June 1972 - Employed as Plant Superintendent in the Nuclear Generation Section of the Generation & System Operations Department at the H. B. Robinson Plant, Hartsville, South Carolina

L. July 1974 - Employed as Manager - Nuclear Generation Services in the Nuclear Generation Section of the Bulk Power Supply Department in the General Office

M. May 1976 - Employed as Plant Manager II (Temporary) in the Nuclear Generation Section of the Bulk Power Supply Department at Brunswick Steam Electric Plant, Southport, North Carolina

N. December 1976 - Employed as Manager - Nuclear Generation Services in the Nuclear Generation Section of the Bulk Power Supply Department in the General Office

O. January 1977 - Employed as Manager - Generation Department in the Power Supply Group in the General Office

P. October 1979 - Employed as Manager - Nuclear Operations in the Power Supply Group in the General Office

Q. December 1979 - Employed as Vice President - Nuclear Operations in the Power Supply Group in the General Office

R. September 1983 - Employed as Vice President - Operations Training & Technical Services Department in the Operations Support Group in the General Office

S. October 1985 - Employed as Vice President - Operations Training Department in the Operations Support Group in the General Office

T. August 1986 - Employed as Vice President - Operations Training and Technical Services Department.

III. Professional Societies:

A. Member of American Society of Mechanical Engineers

B. Member of American Nuclear Society

TABLE 13.1.1-4 (CONT'D)

A. C. Tollison, Jr.
Page 2

7. 1981 to 1983 - On loan to Institute of Nuclear Power Operations

   a. 1981 - Evaluator, Evaluation Team Manager - Manager - Organization & Administration Department

   b. 1982 to 1983 - Director - Evaluation & Assistance Division

8. September 1983 - Employed as Manager - Nuclear Training, Operations Training & Technical Services Department, Shearon Harris Energy & Environmental Center, New Hill, North Carolina

37 9. October 1985 - Employed as Manager - Nuclear Training, ~~Operations Training Department,~~ Shearon Harris Energy & Environmental Center, New Hill, North Carolina

III. Professional Societies:

A. American Nuclear Society

TABLE 13.1.1-4 (CONT'D)

Robert G. Black, Jr., Manager - Emergency Preparedness Section

I. Education and Training:

A. B. S. Degree in Industrial Engineering - Georgia Institute of Technology - 1969

B. Attended various schools while in the U. S. Navy

C. Registered Professional Engineer - February 1979

II. Experience:

A. June 1969 to June 1973 - U. S. Navy Nuclear Program

B. September 1973 - Senior Engineer - Environmental & Technical Services Section Special Services Department, CP&L, Raleigh, North Carolina

C. January 1976 to June 1976 - Project Engineer - Licensing & Technological Services Section, Special Services Department, CP&L, Raleigh, North Carolina

D. June 1976 to December 1979 - Project Engineer - Nuclear Licensing Unit, Licensing & Siting Section, Technical Services Department, CP&L, Raleigh, North Carolina

E. December 1979 to March 1981 - Project Engineer - Nuclear Licensing Unit, Licensing & Permits Section, Technical Services Department, CP&L, Raleigh, North Carolina

F. March 1981 to August 1983 - Director - Emergency Preparedness - Technical Services Department, General Office, Raleigh, North Carolina

G. August 1983 to May 1985 - Director - Emergency Preparedness Unit, Operations Training & Technical Services Department, General Office, Raleigh, North Carolina

H. May 1985 to October 1985 - Manager - Emergency Preparedness Section, Operations Training & Technical Services Department, General Office, Raleigh, North Carolina

I. October 1985 to Present - Manager - Emergency Preparedness Section, ~~Operations Training Department,~~ General Office, Raleigh, North Carolina 37

TABLE 13.1.1-4 (CONT'D)

R. B. Starkey, Jr.
Page 2

E. July 1972 - September 1973 - Lt. Commander, U. S. Navy - Assistant Director of the Engineering Division - U. S. Navy Submarine School

F. September 1973 - February 1974 - Senior Engineer - Nuclear Generation Section, Bulk Power Supply Department, Carolina Power & Light Company, Raleigh, North Carolina

G. February 1974 - February 1975 - Senior Engineer, Quality Assurance Section, Bulk Power Supply Department, Carolina Power & Light Company, Raleigh, North Carolina

H. February 1975 - April 1975 - Principal Engineer, Quality Assurance Section, Bulk Power Supply Department, Carolina Power & Light Company, Raleigh, North Carolina

I. April 1975 - May 1976 - Quality Assurance Supervisor, Nuclear Generation Section, Bulk Power Supply Department, Carolina Power & Light Company, Brunswick Plant, Southport, North Carolina

J. May 1976 - December 1976 - Superintendent - Technical and Administrative, Nuclear Generation Section, Bulk Power Supply Department, Carolina Power & Light Company, Brunswick Plant, Southport, North Carolina

K. December 1976 - November 1977 - Superintendent - Operation and Maintenance, Nuclear Generation Section, Generation Department, Carolina Power & Light Company, Brunswick Plant, Southport, North Carolina

L. November 1977 - November 1979 - Plant Manager - H. B. Robinson Plant, Nuclear Generation Section, Generation Department, Hartsville, South Carolina (CP&L)

M. November 1979 - September 1983 - General Manager - Robinson, Nuclear Operations Department, Carolina Power & Light Company, Hartsville, South Carolina

37 N. September 1983 - April 1984 - (left CP&L) Executive Director - Nuclear Operations at the Marble Hill Nuclear Plant (under construction), Public Service Indiana

O. April 1984 - October 1985 - Manager - Environmental Services, Environmental Services Section, Operations Support Group, Carolina Power & Light Company, Harris E&E Center, New Hill, North Carolina

P. October 1985 - Present - Manager - Nuclear Safety & Environmental Services Department, Operations Support Group, Carolina Power & Light Company, Raleigh, North Carolina

TABLE 13.1.1-4 (CONT'D)

J. D. E. Jeffries
Page 2

27

J. April 1978 - Employed as Principal Engineer, Nuclear Safety, CNS&QAA Section, System Planning & Coordination Department, Carolina Power & Light Company, Raleigh, North Carolina

K. June 1979 - August 1981 - On loan to the Electric Power Research Institute as Project Manager, Nuclear Division, Palo Alto, California

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37

L. August 1981 - Employed as Manager - Corporate Nuclear Safety Section, ~~Corporate Nuclear Safety & Research Department,~~ Carolina Power & Light Company, Raleigh, North Carolina

III. Professional Societies:

A. American Nuclear Society

B. Society of the Sigma Xi

C. Health Physics Society - North Carolina Section

27

D. Registered Professional Engineer in North Carolina and Pennsylvania

TABLE 13.1.1-4 (CONT'D)

H. W. Bowles
27 Page 2

H. September 1981 - November 1982 - Employed as Project Engineer, Corporate Nuclear Safety Section, Corporate Nuclear Safety & Research Department, Carolina Power & Light Company, Raleigh, North Carolina

37 I. November 1982 - Present - Employed as Director - Onsite Nuclear Safety (SHNPP), Corporate Nuclear Safety Section, ~~Corporate Nuclear Safety & Research Department,~~ Carolina Power & Light Company, Raleigh, North Carolina

III. <u>Professional Societies</u>:

A. American Nuclear Society

B. Professional Engineer of North Carolina

TABLE 13.1.1-4 (CONT'D)

B. H. Webster
Page 2

L. February 1982 - Employed as Manager - Radiological & Chemical Support Section in the Technical Services Department. (Located at the Harris Energy & Environmental Center) - CP&L

M. January - 1981 - Employed as Manager - Environmental & Radiation Control in the Technical Services Department. (Located at the Harris Energy & Environmental Center) - CP&L

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III. Professional Societies:

A. North Carolina Chapter - Health Physics Society

B. American Nuclear Society of East Carolina Section

C. Power Reactor Health Physics Group

N. October 1985 - Employed as Manager - Radiological and Chemical Support.

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TABLE 13.1.1-5

Resume of Vice President
Brunswick Nuclear Project

P. W. Howe, Vice President - Brunswick Nuclear Project

I. Education:

A. Bachelor of Science Degree in Chemistry from The Citadel, Charleston, South Carolina in 1951

B. Certificate - Engineering Management - UCLA - 1963

C. Member of U.S.A.E.C. Atomic Safety & Licensing Board from 1962-1966

II. Experience:

A. September 1951 to February 1956 - Laboratory Supervisor - E. I. du Pont de Nemours & Company, Inc., Savannah River Plant, Aiken, South Carolina

B. February 1956 to August 1956 - Senior Nuclear Engineer - The Martin Company, Nuclear Division, Baltimore, Maryland

C. August 1956 to August 1957 - Superintendent - Olin Mathieson Chemical Company, Nuclear Fuels Division, New Haven, Connecticut

D. August 1957 to June 1966 - Department Head - Lawrence Radiation Laboratory, University of California, Berkely, California

E. September 1967 to March 1971 - Chief, Site Environmental and Radiation Safety Group - Division of Reactor Licensing, U. S. Atomic Energy Commission, Washington, DC

F. March 1971 to November 1971 - Manager - Environmental & Technical Services Section of the Generation & System Operations Department, Carolina Power & Light Company

G. November 1971 to February 1974 - Manager- Environmental & Technical Services Section, Special Services Department - CP&L

H. February 1974 to February 1975 - Manager - Licensing & Technological Services Section, Special Services Department - CP&L

I. February 1975 - Manager - Special Services Department, Engineering, Construction & Operation Group - CP&L

J. June 1976 - Manager - Technical Services Department, Engineering, Construction & Operation Group - CP&L

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TABLE 13.1.1-5 (CONT'D)

P. W. Howe
Page 2

K. December 1976 - Vice President - Technical Services Department, Engineering & Construction Group - CP&L

L. October 1982 - Vice President - Brunswick Nuclear Project - CP&L

III. Professional Societies:

A. American Nuclear Society

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TABLE 13.1.1-6 (CONT'D)

H. R. Banks
Page 2

I. June 1960 - June 1962 - Chief Engineman, EOOW, Nuclear Power Training Unit, (SIW) Nuclear Submarine Prototype - Idaho Falls, Idaho

J. June 1962 - October 1964 - USS Andrew Jackson, SSBN 619 - EOOW, Leading Machinery Division Chief, Supervisor in charge of operation of the nuclear power plant

K. October 1964 - January 1965 - Naval Officer's Candidate School

L. January 1965 - August 1968 - Nuclear Ship Superintendent - San Francisco Bay Naval Shipyard

M. August 1968 - July 1970 - Resident Project Engineer - H. B. Robinson Plant - Unit No. 2 - CP&L, Power Supply Department, Hartsville, South Carolina

N. July 1970 - August 1971 - Resident Project Engineer - Brunswick Plant - Units 1 & 2 - CP&L, Power Plant Design & Construction Department, Southport, North Carolina

O. August 1971 - February 1972 - Manager - Quality Assurance, Power Plant Design & Construction Department, CP&L, Raleigh, North Carolina

P. February 1972 - July 1973 - Manager - Quality Assurance Audit, Special Services Department - CP&L, Raleigh, North Carolina

Q. July 1973 - August 1975 - Manager - Quality Assurance & Training Audit, Special Services Department - CP&L, Raleigh, North Carolina

R. August 1975 - March 1976 - Manager - Corporate Quality Assurance Audit, Special Services Department - CP&L, Raleigh, North Carolina

S. March 1976 - October 1979 - Manager - Nuclear Generation, Bulk Power Supply Department, Nuclear Generation Section - CP&L, Raleigh, North Carolina

37 T. October 1979 - February 1981 - General Manager - Shearon Harris Nuclear Power Plant - CP&L, Raleigh, North Carolina

U. February 1981 - Present - Manager - Corporate Quality Assurance - CP&L, Raleigh, North Carolina

III. <u>Professional Societies</u>:

A. ASME Standards Committee - Main Committee NQA, Subcommittee Operations N45-2.12 & N45.2.23
B. EEI QA Committee
C. American Society of Nondestructive Testing
D. American Society of Mechanical Engineers
E. American Nuclear Society
F. North Carolina Society of Engineers.

TABLE 13.1.1-6 (CONT'D) | ~~27~~ 37

N. J. Chiangi, Manager - ~~QA/QC Harris Plant~~ Material Quality

I. Education:

A. Graduate of Norwich Free Academy, Norwich, Connecticut

B. Special Schools: Nuclear Submarine Systems, Navyships 250-1500-1, Mil. Std. 271 D-271A, Navyships 250-693-1 693-3 (structural), Health Physics Monitoring, Management Schools - Electric Boat Company, Electronics School - U. S. Navy, Welding School - EBC, Radiography School, Magnetic Particle Testing School - EBC, Liquid Penetrant Test School - EBC, Ultra Sonic Testing Classes - EBC, Eastman Kodak School for Automatic Film Processing Equipment, Job Cost Estimating - EBC. Qualified: AEC Licensed Radiographer and Radiographer Supervisor

II. Experience:

A. 1947 - 1952 - U. S. Navy, Sonar Man - Radar Man. Special Training, Electronics School, Sonar School, Radio School

B. 1952 - 1967 - Electric Boat Company, Groton, Connecticut

1. 1952 - 1954 - Welding-Field Work-Piping-Structural

2. 1954 - 1967 - Lead-Supervisor - Radiography Department. Responsible for all Nuclear Radiography Structural-Piping-Castings, Polaris Missile Program, Radiographer, Film Readers. Setup, wrote, and reviewed Radiography Test Procedures for Casting-Piping-Structural Radiography. Instructed Piping and Mechanical design personnel, instructed Radiography Classes for New Hires, reviewed and interviewed personnel for hire. Attended Management-Quality Control meetings.

C. 1967 - 1973 - Ebasco Services, Inc., New York, New York

1. 1967 - 1970 - Quality Compliance-Quality Control Supervisor for Ebasco at H. B. Robinson NPS Unit No. 1. Responsible for implementation of the site for the H. B. Robinson project. This included supervising Ebasco site Quality Compliance Representatives in the performance of their inspection duties in the following areas: welding, civil, electrical, nondestructive testing, receiving, storage, and testing. Responsible for the review of site purchase orders for quality requirements and documentation to assure its adequacy. Responsible for maintaining Quality Assurance documentation.

TABLE 13.1.1-6 (CONT'D)

N. J. Chiangi
Page 2

2. 1970 - 1972 - Site Quality Compliance Supervisor for Ebasco at St. Lucie No. 1 Nuclear Power Plant, with responsibility for implementing the site phase of the Ebasco Quality Program as modified for St. Lucie. Responsible for auditing field construction activities as required by the Quality Program, auditing the performance of construction quality control tasks through the Site Quality Compliance Staff, meeting with AEC representatives in performance of their site audits, and maintaining quality compliance files as described in the Ebasco Quality Program for representation to the client at the completion of the project.

3. 1972 - 1973 - Senior Quality Compliance Engineer for Ebasco at Chin-Shan Unit Nos. 1 and 2. Had overall responsibility for Ebasco Quality Compliance Program on site. Duties at Chin-Shan site included the following: instructed personnel in inspection of welding, mechanical, civil, and electrical functions. Responsible for interpretation of all codes and specifications having to do with this project where compliance or control was required. Instructed and trained Taipower Personnel in Quality Compliance and Quality Control functions. Developed quality control and compliance programs for Taipower. Responsible for a vendor inspection. Interpreted all radiographs on site. Responsible for maintaining radiographs and quality assurance documentation.

D. October 1, 1973 - Carolina Power & Light Company, Raleigh, North Carolina - Employed as Quality Assurance Manager - Construction, Quality Assurance Section of the Power Plant Construction Department, located in the General Office

E. November, 1976 - Manager, Engineering and Construction QA Section, Technical Services Department - CP&L

F. March 1983 - Manager, QA/QC Harris Plant Section of the Corporate Quality Assurance Department - Harris Site, New Hill, North Carolina

37

III. <u>Professional Societies:</u>

A. Member - ASNT - ASME

B. Qualified ANST - Level III - 2/4/77 Radiographic - Magnetic Particle - Liquid Penetrate

C. Professional Engineer - State of California - January 1977

G. June 1986 - Manager - Material Quality Section of the Corporate Quality Assurance Department - New Hill, North Carolina

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TABLE 13.1.1-6 (CONT'D)

Charles Lewis McKenzie, Jr., Principal Quality Assurance Engineer - Acting Director - QA/QC Harris (Operations)

I. Education and Training:

A. B. S. Industrial Engineering, University of Florida - 1971

B. Supplemental at Charleston Naval Shipyard

1. Nuclear Cleanliness; Reactor Fundamentals, 11/71 - 1/72

2. Nuclear Quality Control - 9/72 - 12/72

3. S5W Reactor Plant - 1/73 - 5/73

C. Basic Ultrasonic Testing Course presented by Magnaflux Corporation - 11/5 - 9/73

D. Orientation of Newly Appointed Supervisors presented by CP&L, General Office - 11/18 - 20/75

E. Basic Principles of Supervisory Management presented by CP&L. General Office - 6/8 - 11/76

II. Experience:

A. U. S. Post Office

1. April 1966 to December 1967

a. Postal Clerk

B. Pratt & Whitney Aircraft-Turbine Engine Division

1. June 1968 to March 1969

a. Engineering Aid Co-Op Student

C. Bates & Daily Construction Company

1. June 1969 to September 1970

a. Laborer (summer employment while attending college)

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TABLE 13.1.1-6 (CONT'D)

Charles Lewis Mckenzie, Jr.
Page 2

D. Charleston Naval Shipyard

1. August 1971 to August 1973

   a. Nuclear Auditor for Naval S5W Overhaul and Refueling Program

E. Carolina Power & Light Company

1. September 1973 - Employed as a Junior Quality Surveillance Specialist in the Quality Assurance Section of the Power Plant Engineering Department, located in the General Office, Raleigh, North Carolina

   a. October 1974 - Promoted to Vendor Surveillance Specialist in the Quality Assurance Section of the Power Plant Engineering Department, located in the General Office, Raleigh, North Carolina

   b. April 1976 - Promoted to Vendor Surveillance Specialist III in the Quality Assurance Section of the Power Plant Engineering Department, located in the General Office, Raleigh, North Carolina

   c. November 1976 - Transferred as a Vendor Surveillance Specialist III in the Engineering & Construction Quality Assurance Section of the Technical Services Department, located in the General Office, Raleigh, North Carolina

   d. June 1978 - Promoted to Senior Vendor Surveillance Specialist in the Engineering & Construction Quality Assurance Section of the Technical Services Department, located in the General Office, Raleigh, North Carolina

   e. August 1980 - Reclassified to Senior Quality Assurance Engineer in the Engineering & Construction Quality Assurance Section of the Technical Services Department, located in the General Office, Raleigh, North Carolina

   f. March 1981 - Transferred as a Senior Quality Assurance Engineer in the Engineering & Construction Quality Assurance/Quality Control Section of the Corporate Quality Assurance Department, located in the General Office, Raleigh, North Carolina

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| 24

TABLE 13.1.1-6 (CONT'D)

Charles Lewis Mckenzie, Jr.
Page 3

g. August 1981 - Promoted to Project Quality Assurance/Quality Control Engineer in the Engineering & Construction Quality Assurance/Quality Control Section of the Corporate Quality Assurance Department, located in the General Office, Raleigh, North Carolina

h. February 1983 - SECTION TITLE CHANGE - Project QA Engineer in the QA Engineering Unit of the Quality Assurance Services Section of the Corporate Quality Assurance Department, located in the General Office, Raleigh, North Carolina

i. March 1983 - Promoted to Principal Quality Assurance Engineer in the QA Engineering Unit of the Quality Assurance Services Section of the Corporate Quality Assurance Department, located in the General Office, Raleigh, North Carolina

j. May 1985 - Transferred as a Principal QA Engineer [Acting Director - QA/QC Harris (Operations)] in the QA/QC Harris (Operations) Unit of the Operations QA/QC Section of the Corporate Quality Assurance Department, located in the General Office, Raleigh, North Carolina

III. Professional Societies:

A. American Society for Quality Control

B. Registered Professional Engineer in the State of North Carolina - February 1976

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13.1.2 OPERATING ORGANIZATION*

13.1.2.1 Introduction

The SHNPP organization is based on the considerable experience that CP&L has operating its three nuclear units, Robinson Unit No. 2 and Brunswick Units 1 and 2. Carolina Power & Light Company will comply with ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," as indicated in Section 1.8, in the operation and administration of the Shearon Harris Nuclear Power Plant. The succession of responsibility in the event of absences, incapacitation of personnel, or other emergencies are outlined by the organization chart (Fig. 13.1.2-1). The staff loading schedule is shown in Table 13.1.2-1.

13.1.2.2 Personnel Functions, Responsibilities, and Authorities

13.1.2.2.1 Plant General Manager - Harris Plant Operations Section 37 ~~27~~

The Plant General Manager is responsible for all phases of plant management, including administration, operation, maintenance, and technical support. He manages and controls the organization through personal contact with the Assistant Plant General Manager and ~~seven~~ unit heads and through written reports, meetings, conferences, and in-plant inspections. He is responsible for adherence to ~~all~~ the requirements of the operating license, technical specifications, Corporate Quality Assurance Program, and Corporate Health Physics and Nuclear Safety policies. He is responsible for reviewing incoming and outgoing correspondence with the NRC Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement concerning the Harris Plant; the establishment and approval of qualification requirements for all Harris Plant Operations staff positions; the personal review of the qualifications of specific personnel for managerial and supervisory positions in the Harris Plant Operations Section; and the review of and concurrence in the plant radiation protection, radiological security, quality assurance, fire protection, training, operations, and maintenance programs. He is supported in these responsibilities by the Assistant Plant General Manager, Director - Plant Programs and Procedures, Manager - Maintenance, Manager - Environmental and Radiation Control, Manager - Operations, Manager - Technical Support, the Manager - Startup and Test, and Director - Regulatory Compliance. He has the authority to issue procedures, standing orders, and special orders. In the absence of the Plant General Manager, the Assistant Plant General Manager assumes his authority and responsibilities. The Plant General Manager reports directly to the Vice President - Harris Nuclear Project Department.

13.1.2.2.2 Assistant Plant General Manager

The Assistant Plant General Manager has the responsibility and accountability for the safe, reliable, and efficient daily operation of the Harris Plant. He has direct control over the operations, maintenance, environmental/chemistry/radiation control, and technical support functions. He is responsible for adherence to ~~all~~ requirements of the Operating License, Technical Specifications, the Corporate Quality Assurance Program, and Corporate Health

*Further information is contained in the TMI appendix.

Physics and Nuclear Safety policies. He is responsible for the personal review of the training and qualification requirements of the following managers who report directly to him: Manager - Operations, Manager -
~~27~~ Maintenance, Manager - Environmental and Radiation Control, and Manager - Technical Support. ~~In the absence of the Plant General Manager, the Assistant~~
37 ~~Plant General Manager assumes that position's authorities and responsibilities.~~ The Assistant Plant General Manager reports directly to the Plant General Manager.

13.1.2.2.3 Plant Programs and Procedures Unit

The Plant Programs and Procedures Unit provides support functions such as security, procedure control, and emergency preparedness.

37 ~~27~~ The Director~~y~~-Plant Programs and Procedures provides direct support to the Plant General Manager in the areas of security, emergency preparedness, procedure development and control, personnel administration and plant administrative coordination; directs plant security planning and activities; directs emergency preparedness planning and activities at the plant staff level; supervises the preparation, review, approval and distribution of plant procedures and directives. He is assisted in these duties by ~~an~~ a
37 ~~27~~ ~~Administrative Supervisor,~~ Security Supervisor, a Senior Specialist-Plans and Programs, and a Senior Specialist - Emergency Preparedness. The Director, Plant Programs and Procedures reports to the Plant General Manager - Harris Plant.

The ~~Administrative Supervisor supervises~~ Senior Specialist-Plans and Programs provides the administrative functions of the plant including incoming correspondence screening and action assignment;
~~27~~ action item/response development and follow-up; outgoing correspondence preparation, screening and coordination; ~~supervision~~ and coordination of plant
37 procedure preparation, review, and approval. ~~and distribution functions and supervision of personnel administration functions at the plant level.~~

~~27~~ The Security Supervisor develops, implements, and maintains a security program which ensures that the security of the plant is maintained in accordance with NRC requirements. He maintains a close working relationship with local law enforcement agencies to ensure compliance with NRC regulations. He provides input to the Training Unit so that employees requiring access to the plant are properly trained and badged. He ensures that equipment and guards are available and in a state of readiness. The Senior Specialist - Security is assisted by Technical Aides and a contract security guard force. The Security
37 ~~27~~ Supervisor reports to the Director~~y~~-Plant Programs and Procedures.

The Senior Specialist - Emergency Preparedness is responsible for the continuing refinement of the plant Emergency Preparedness Program which ensures that a "state of readiness" is maintained at the plant to cope with any classification of emergency. He incorporates the provisions of the plant Emergency Plan in the program and revises the program and related procedures as changes are made in the plant Emergency Plan. He coordinates the training of Technical Support Center participants and the annual Emergency Drill. The Senior Specialist - Emergency Preparedness reports to the Director~~y~~-Plant Programs and Procedures.

13.1.2.2.3.1 Environmental & Radiation Control Unit

The Manager - Environmental & Radiation Control (E&RC) is responsible for the plant radiation safety and control (health physics) programs, the plant chemical control programs, and the environmental programs. These programs are designed to ensure that environmental and radiation control is maintained in a manner which will protect the plant, employees, visitors, general public, and the surrounding community. He has the authority to issue special orders. His primary responsibility is organizing, planning, and controlling E&RC resources to provide the required support while ensuring compliance with plant Technical Specifications, the ALARA concept, and all applicable state and federal regulations and permit requirements.

Some of his major responsibilities include: (1) ensuring that programs and related procedures are developed and administered to meet plant needs and regulatory requirements; (2) maintaining an awareness of current and pending regulations in the areas of radiation control, chemistry, and environmental matters concerning plant operations; and (3) providing adequate documentation pertaining to individual radiation exposures, radioactive effluents, chemical control of plant systems and environmental surveillance and ensuring that these records are maintained in an up-to-date, retrievable manner. He is assisted in these functions by an Environmental & Chemistry Supervisor, a Radiation Control Supervisor, a Project Specialist - Environmental and Chemistry, a Project Specialist - Radiation Control, and a staff of radiation control and environmental and chemistry specialists, foremen, and, E+RC technicians. The Manager - ~~Environmental & Radiation Control~~ reports to the Assistant Plant General Manager. The Manager ~~- Environmental & Radiation~~ - E+RC ~~Control~~ has direct access to the Plant General Manager for matters relating to radiological health and safety of employees and the public. 21 37 21

The Environmental & Chemistry Supervisor plans, organizes, and directs chemistry control and environmental surveillance programs, maintains laboratory procedures, test results and records, and adheres to the requirements of the operating license and technical specifications. He accomplishes these responsibilities through foremen and technicians. The Environmental and Chemistry Supervisor reports to the Manager - ~~Environmental & Radiation Control~~ E+RC Unit. 37

The Radiation Control Supervisor is responsible for the plant Radiation Control (Health Physics) Program and for ensuring that ~~all~~ plant activities are conducted in a manner which will protect the plant, employees, visitors, general public, and the surrounding community. His primary responsibility is organizing, planning, and controlling Radiation Control Subunit resources to provide the required support while ensuring compliance with plant Technical Specifications and all applicable state and federal regulations and permit requirements. He accomplishes this through foremen and radiation control technicians. The Radiation Control Supervisor reports to the Manager - E+RC ~~Environmental & Radiation Control~~ Unit. 37 37

The Project Specialist - Environmental & Chemistry provides technical advice and recommendations for program enhancement to the Manager - E&RC, and ensures that the Environmental and Chemistry Programs support efficient, reliable

plant operations. He is the Environmental Chemistry technical expert for the Manager - E&RC. He is supported by a staff of specialists and technicians and
37 reports to the Manager - ~~Environmental & Radiation Control~~ E+RC Unit.

The Project Specialist - Radiation Control provides technical advice and recommendations for program enhancement and ALARA program considerations to the Manager - E&RC, and ensures that the Radiation Control Programs support efficient and reliable plant operations. He is the Radiation Control technical expert for the Manager - E&RC. He is supported by a staff of specialists, technicians, and clerks and reports to the Manager - Environmental and Radiation Control Unit.

13.1.2.2.3.2 Maintenance Unit

37 The Maintenance Unit performs ~~all~~ corrective and preventive maintenance on plant systems and equipment. The Manager - Maintenance is responsible for corrective and preventive maintenance for the equipment of the unit and in the support facilities. This includes ensuring that the equipment and associated instrumentation and controls, mechanical, and electrical systems in the unit and support facilities are maintained at optimum dependability and operating efficiency. He is responsible for the coordination of these functions and for approval of Special Orders, working procedures and standards. He is assisted by the Mechanical Maintenance Supervisor, Electrical Maintenance Supervisor, Project Engineer - Maintenance, Project Specialist - Maintenance, Project Engineer - Computer, and a staff of engineers and specialists, foremen, mechanics, electricians, painters/pipe coverers, planner/analysts, and technicians. The Manager - Maintenance reports to the Assistant Plant General Manager - Operations.

The Maintenance Supervisor - Electrical ensures that equipment, instrumentation, controls, and electrical systems are maintained at optimum dependability, safety, and operating efficiency to comply with plant technical specifications, QA, Security, Radiation Control and plant procedures, and regulatory requirements. He accomplishes this by planning, directing, and controlling a trained staff, inspecting maintenance work, providing effective maintenance procedures and standards, and developing improvements in the Preventive and Corrective Maintenance Program. He is assisted in these functions by a staff of foremen, technicians, and electricians. The Maintenance Supervisor - Electrical reports to the Manager - Maintenance Unit.

The Maintenance Supervisor - Mechanical ensures that mechanical systems are maintained at optimum dependability, safety, and operating efficiency to comply with plant technical specifications, QA, Security, Radiation Control,
37 and plant procedures, and regulatory requirements. He is responsible for all required painting and pipe covering activities necessary to maintain neat, properly insulated plant systems. He accomplishes this by planning, directing, and controlling a trained staff, inspecting maintenance work, providing effective maintenance procedures and standards, and developing improvements in the Preventive and Corrective Maintenance Programs. He is assisted by a staff of foremen, mechanics, and painter/pipe coverers. The Maintenance Supervisor - Mechanical reports to the Manager - Maintenance ~~Subunit~~ Unit.

The Project Engineer - Maintenance provides technical support to plant electrical and mechanical maintenance and assists the Manager - Maintenance in assuring that plant instrumentation, control, electrical systems and mechanical systems are maintained at optimum dependability, safety, and operating efficiency, and remaining in compliance with ~~all~~ technical 37
specifications and regulatory requirements. He is responsible for administration of the Maintenance Management System to accomplish the planning and scheduling of maintenance, ensuring parts availability, and establishing clearances necessary for preplanned work; he is assisted by a staff of engineers, specialists, technicians, and planner/analysts. The Project Engineer - Maintenance reports to the Manager - Maintenance Unit.

The Project Engineer - Computer provides process computer system maintenance support and technical expertise to ensure that ~~all~~ plant process computer 37
systems are fully operational for the safe, reliable, and efficient operation of the plant. He is assisted by a staff of specialists and technicians. The Project Engineer - Computer reports to the Manager - Maintenance Unit.

13.1.2.2.3.3 Operations Unit

The Manager - Operations ensures that the safe and efficient operation of the unit and required support facilities. He is responsible for primary and secondary system performance and the timely completion of the scheduled periodic tests, and for adherence to the requirements of the operating license and technical specifications. He is also responsible for coordinating and overseeing the duties of the Operating Supervisors assigned to the plant, the 37
Radwaste Supervisor, and the Operations Support Supervisor. He is responsible for orderly and safe operations, turnovers, and compliance with operating instructions. He shall hold a ~~Senior Operator's~~ SRO's License. He has the 37
authority to issue Special Orders. He is supported in these responsibilities by a staff of the Operating Supervisor, Radwaste Supervisor, Operations Support Supervisor, engineers/specialists, Shift Technical Advisors, Shift Foremen, and Operators. The Manager - Operations Unit reports to the Assistant Plant General Manager.

The Operating Supervisor supervises plant operations. He is responsible for adherence to the requirements of the Operating License and Technical Specifications. The Operating Supervisor is responsible for scheduling and ~~27~~
reviewing surveillance tests, reviewing operating data, logs and records, shift reports of equipment malfunctions or unusual system behavior, and initiating corrective action. The Operating Supervisor reports to the Manager - Operations. The Operating Supervisor shall hold an SRO License.

The Harris Plant Operations Section will have shift operating crews assigned to provide 24-hour coverage of plant activities. Each shift operating crew will be manned in accordance with Technical Specification Table 6.2-1.

Each Shift Operating Crew in the Harris Plant Section shall meet the following requirements:

a) When the unit has fuel in the reactor core, there shall be a Shift Foreman with an SRO license on site at all times.

b) When the unit has fuel in the core, there shall be a licensed operator in the control room at all times.

c) When the reactor is operating, there shall also be a licensed SRO in the control room at all times.

37 d) When the reactor is operating ~~being operated~~, there shall be an additional licensed operator in the control room to provide relief for the control room operator and to perform duties outside the control room that need to be performed by a licensed operator.

e) When the reactor contains fuel, there shall be an auxiliary operator in addition to the individuals required in (a) through (d) above. An additional auxiliary operator is required for the control room when the reactor is ~~being~~
37 ~~operated.~~ operating.

f) For all core alterations, there shall be a licensed SRO or SRO limited to Fuel Handling to directly supervise the core alteration. This SRO shall not be assigned any other concurrent operational duties.

g) The Shift Foreman shall be assigned only the minimal administration duties required to operate his shift.

An extensive training program has been established to ensure that each onsite crew collectively has the requisite technical qualifications in reactor physics and control, nuclear fuel, thermal hydraulics, transient analysis, instrumentation and control, mechanical and structural engineering, radiation control and health physics, electric power, chemistry, and plant operation and maintenance.

The Shift Foremen ensure the safe, dependable, and efficient operation of the plant during their assigned shift. ~~and are the designated individuals in charge~~
37 ~~of the plant on that shift unless specifically relieved by the Operations Supervisor or his superior.~~ They are responsible for adherence to the operating procedures, the operating license, and technical specifications. It is the responsibility and authority of the Shift Foreman to maintain the broadest perspective of operational conditions affecting the safety of the
37 plant and to keep this as the highest priority ~~at all times~~ when on Control Room duty. The Shift Foreman shall hold a Senior Operator's license. The Shift Foreman, until properly relieved, remains in the Control Room at all times during an accident to direct the activities of Control Room Operators. He may be relieved only by qualified persons holding SRO licenses. During routine operations when the Shift Foreman is temporarily absent from the Control Room, a Senior Control Operator will be designated to assume the Control Room command function. He is supported by and supervises Senior Control Operators, Control Operators, and Auxiliary Operators. The Shift Foreman reports to the Operating Supervisor.

The Shift Foreman is the designated individual in charge of the plant on back shifts unless specifically relieved of the responsibility by either the Operating Supervisor, Manager - Operations, Assistant Plant General Manager, or the Plant General Manager. They are responsible for all personnel assigned on the back shifts including operators, mechanics, electricians, RC technicians, and I&C technicians.

a) Licensed Operators - The licensed operators are responsible for performing shift operations in accordance with the procedures, instructions, set points, limitations, and precautions contained in the Plant Operating Manual and the Technical Specifications. They exercise continuous monitoring of plant conditions and system parameters. They manipulate the controls and equipment to start up, change output, and shut down the plant as required by operating schedules and load demands. They initiate the immediate actions necessary to maintain the plant in a safe shutdown condition during abnormal and emergency situations. They maintain required records of plant data, shift events, and performance checks. They initiate plant corrective maintenance to report and document equipment problems. Licensed Senior Control Operators (SROs) have the responsibility and authority to assume the control room command function during the temporary absence of the Shift Foreman. The licensed operators report to the Shift Foreman.

b) Non-Licensed Operators - The non-licensed auxiliary operators are responsible to the Shift Foreman for assisting in the performance of assignments associated with shift operations or refueling. The non-licensed operators' duties are normally associated with the operation of auxiliary systems and equipment outside the control room. Non-licensed radwaste operators perform shift operations of the Waste Processing Systems. Non-routine operations are performed under the direction of a licensed control operator or Shift Foreman. Radwaste Operators report to the Radwaste Shift Foreman.

c) Radwaste Supervisor - The Radwaste Supervisor supervises the shift operations of the Waste Processing System. This includes the working procedures for the maintenance and implementation of the waste process equipment, and the operation of the equipment necessary to generate ~~all~~ the process water utilized within plant systems. The Radwaste Supervisor is responsible for ensuring safe and efficient handling and storage of plant-generated contaminated wastes until final disposition. He is assisted by the Radwaste Shift Foremen, Radwaste Control Operators, Project Specialist - Radwaste, Engineers, and Radwaste Auxiliary Operators. The Radwaste Operations Supervisor reports to the Manager - Operations ~~Subunit~~. | 37

d) Shift Foremen Radwaste - The Shift Foremen - Radwaste ensure the safe, dependable, and efficient operation of the Waste Processing System. It is the responsibility and authority of the Shift Foremen Radwaste to direct the activities of the Radwaste Operators to ensure efficient handling, processing, storage, and shipment of plant generated contaminated wastes. They are supported by and supervise Radwaste Control Operators and Radwaste Auxiliary operators. The Shift Foremen-Radwaste functionally report to the Radwaste Supervisor but are under the direction of the Shift Foreman to ensure that radwaste operations support is compatible with overall plant operations.

e) Operations Support Supervisor - The Operations Support Supervisor provides technical and engineering support to the plant operating personnel. He is responsible for the implementation and efficient operation of the Shift Technical Advisor (STA) program at the plant as well as for providing direct technical support in the areas of: (1) Plant Operations, and (2) Fire | 37

Operations Engineers

37 Protection as necessary to support safe, efficient, reliable operations. He is assisted by Shift Technical Advisors, and a Fire Protection ~~Specialist~~. The Operations Support Supervisor reports to the Manager - Operations.

Senior Specialist

f) Senior Specialist - Fire Protection - The Senior Specialist - Fire Protection is responsible for fire detection equipment, fire protection equipment, and general safe working conditions for employees. He is responsible for keeping current on "Fire Protection Guidelines for nuclear power plants," Regulatory Guide 1.120, and Branch Technical Position APCSB 9.5-1 and 9.5-1 Appendix A, and informing plant management of changes affecting the plant. He will evaluate damage to plant fire protection equipment under warranty and make recommendations as to course of action. He will coordinate plant inspections for insurance purposes. He is assisted by a Specialist and Fire Protection Technical Aides. The Senior Specialist - Fire

~~27~~ Protection reports to the Operations Support Supervisor.

g) Shift Technical Advisor - The Shift Technical Advisor provides accident assessment and technical advice concerning plant safety to shift operations personnel. He performs 10 CFR 21 evaluations for the shift operations personnel. He accomplishes this by performing engineering evaluations of plant operations, maintaining and broadening his knowledge of normal and off-normal operations, and diagnosing off-normal events. The Shift Technical Advisors report to the Operations Support Supervisor.

13.1.2.2.3.4 Technical Support Unit

The Technical Support Unit provides engineering support for the entire plant staff. Their support involves investigations of day-to-day equipment and system operation. Based on their investigations, they recommend modification tasks to keep the plant in compliance with new regulations or to improve efficiency of operation.

The Manager - Technical Support also manages the Power Ascension Test Program.

The Manager - Technical Support Unit develops and tests maintenance modifications and provides technical support for plant outages, plant operation, and maintenance and manages the plant Inservice Inspection (ISI)

37 and performance programs. The Manager - Technical Support has the authority to issue procedures, Standing Orders, and Special Orders. He is supported by the Engineering Supervisors and a Principal Engineer. The Manager - Technical Support Unit reports to the Assistant Plant General Manager.

The Engineering Supervisors and a Principal Engineer are responsible for providing technical direction and coordination for plant engineering studies. They develop and implement the inservice inspection program and plant performance programs as well as procedures, instructions, and guidelines for plant engineering functions. They are supported in these tasks by a staff of engineers, specialists, engineering technicians, and draftsmen. The Principal Engineer and the Engineering Supervisors report to the Manager - Technical Support.

13.1.2.2.4 Startup and Test Unit

The Manager - Startup and Test is responsible for successfully implementing and accomplishing, on schedule, the Harris Nuclear Project preoperational ~~and~~

37 ~~startup test~~ program in accordance with the Startup Manual. The Manager -

Startup and Test Unit reports to the General Manager - Harris Plant Operations Section.

The Manager - Startup and Test is responsible for the following:

a) Supervises the activities of the Startup Organization through the Startup Supervisors.

b) Prepares and updates the startup schedule.

c) Assigns overall test responsibility to the Startup Supervisors.

d) Reviews and approves requests for vendor assistance as recommended by the Startup Organization.

e) Reviews and approves/recommends approval of test procedures, test procedure modifications, and test data in accordance with the Startup Manual instructions.

f) Reviews and recommends approval of startup requests for construction and engineering modifications or changes required during the test program.

g) Issues periodic progress reports and work schedules for the Startup Organization.

h) Issues special reports concerning startup activities as he deems necessary.

i) Reviews progress of startup activities with contractors, vendors, and Company management.

j) Maintains liaison with the plant management, keeping them informed of the test program status, and coordinates with them the activities of plant personnel assigned to startup activities in conjunction with their training program.

k) Represents the Startup Organization on interdepartmental and interorganizational committees associated with the test program.

l) Maintains liaison with contractors and vendors to coordinate their activities relating to the test program.

m) Is responsible for the preparation and maintenance of the Startup Manual.

n) Accepts release for tests from Harris Plant Construction Section.

He is supported in the accomplishment of these tasks by a staff of Startup Supervisors, ~~E~~engineers, specialists, technicians, and clerks. The Manager - Startup and Test reports to the General Manager - Harris Plant. 37

The Startup Supervisors are responsible for checking out and starting up on schedule the systems assigned in their areas in accordance with the Startup Manual and regulatory requirements. Each supervisor is assigned engineers and technicians and reports to the Manager - Startup and Test Unit.

13.1.2.2.5 Regulatory Compliance

37 The Regulatory Compliance Unit provides staff functions to the entire plant for regulatory compliance activities and routine reporting of ~~all~~ noncompliance items. The Unit is responsible for the continual updating of the FSAR and Technical Specifications, and it serves as the on-site contact for the NRC.

The Director - Regulatory Compliance coordinates activities at the plant to ensure that commitments, responses, records, and reports are prepared, submitted, and maintained in accordance with regulatory requirements. He also maintains a tracking system for the resolution of all plant safety and environmental concerns. He serves as the on-site contact with NRC and provides the expertise necessary to support plant activities in accordance with the operating license and technical specifications. He is assisted by a staff of technicians and specialists. The Director - Regulatory Compliance reports to the General Manager - Harris Plant Operations Section.

ASSIGNMENT OF ON-SITE SHIFT OPERATIONS

The Operating Supervisor is responsible for ~~all~~ operating activities at the plant. 37 The shift complement consists of one Shift Foreman (SRO), one Senior Control Operator (SRO), two Control Operators (RO), four Auxiliary Operators, and at least one Radiation Control Technician qualified in radiation protection measures. Each shift will also have personnel fulfilling roles in Fire Protection and Radwaste Control (normally five). The Harris Plant Operations Section will have shift operating crews assigned to provide 24-hour coverage of plant activities. Each shift operating crew will be manned in accordance with Technical Specification Table 6.2-1. ~~24~~ Additional support, for example the I&C Technicians, Mechanics, Chemistry Technicians, and Plant Storekeepers, will be available on a normal two shift basis, but this schedule will be subject to change as plant conditions require. On-call personnel will be available ~~at all times~~ to support emergencies. Reactor and Performance 37 Engineers will also be available as required, although they will normally work a regular schedule.

During fuel movement operations or core alterations there will be one Senior Reactor Operator in Reactor Containment and an operator in the Fuel Handling Building. This Senior Reactor Operator will direct and supervise the operation and will report to the Shift Foreman.

The following chart contains the minimum shift assignments of the Operation Unit:

MINIMUM SHIFT CREW COMPOSITION
Operating Mode

LICENSE CATEGORY	APPLICABLE OPERATIONAL MODES 1, 2, 3, 4	5 & 6
SRO	2	1*
RO	2	1
Non-Licensed	2	1

Shift crew composition, including a Radiation Control technician qualified in radiation protection procedures, may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. In the unlikely event an unexpected absence occurs that would involve the health physics technician on duty, it is possible this position would be covered by the individual qualified in radiation protection procedures for short periods of time, e.g., a few hours.

Operational Modes listed above are defined in the Technical Specifications. It is expected that the number of personnel as outlined in Table 13.1.2-1 will be used to support the operation of the plant. In the event that additional health physics personnel are required, it is projected that contract health physics services will be used. The number of contract health physics personnel required and their ANSI qualifications will be situationally dependent.

* Does not include the licensed Senior Reactor Operator or Senior Reactor Operator limited to Fuel Handling, supervising core alterations.

TABLE 13.1.2-1 (Cont'd)

PROJECTED SHNPP STAFF LOADING

TITLE	NO. OF POSITIONS	
STARTUP & TEST		
Manager - Startup & Test	1	27
Startup Supervisor	5	
Startup Engineer	29	
Engineering Technician 1	8	
Senior Clerks	4	
ENVIRONMENTAL & RADIATION CONTROL		
Manager - Environmental & Radiation Control	1	
Supervisor - Environmental & Chemistry	1	
Environmental & Chemistry Foreman	2	
Environmental & Chemistry Technician	13	27
Project Specialist - Environmental & Chemistry	1	
Senior Specialist - Environmental & Chemistry	2	27
~~RADIATION CONTROL~~		37
Supervisor - Radiation Control	1	
Project Specialist - Radiation Control	1	
Senior Specialist - Radiation Control	2	27
Senior Specialist - ALARA	1	
Traveling Radiation Control Foreman	1	
Radiation Control Foreman	3	
Radiation Control Technician	23	27
Senior Clerk	1	
Traveling Radiation Control Technician	7	27
TECHNICAL SUPPORT		
Manager - Technical Support	1	
Engineer - Supervisor	2	
Principal Engineer	1	
Project Engineer	6	
Engineer	19	
Senior Specialist	9	
Co-Op Engineer	3	
Co-Op Technician	1	
Engineering Technician 1	11	
Senior Draftsman	2	

13.1.3 QUALIFICATION REQUIREMENTS FOR PLANT PERSONNEL*

13.1.3.1 Minimum Qualifications

Minimum qualifications for plant personnel are listed in CP&L's position on Regulatory Guide 1.8 in Section 1.8.

13.1.3.2 Qualification of Plant Personnel

Resumes for key plant personnel ~~positions presently filled~~ are provided in Tables 13.1.3-1 through 13.1.3-35. 37 ~~27~~

*Further information is contained in the TMI Appendix.

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TABLE 13.1.3-1

James L. Willis
General Manager - Harris Plant Operations Section

I. Education and Training

A. B. S. Degree in Electrical Engineering - 1955 - U. S. Naval Academy - Annapolis, MD

B. Navy Nuclear Power School - 1958

II. Experience

A. June 1951 - June 1979 - U. S. Navy

B. August 1979 - September 1980 - Project Manager, System Development Corporation Santa Monica, CA

C. September 1980 - September 1981 - Manager, Nuclear Training - Southern California Edison Company

D. October 1981 - employed as Manager - Plant Operations in the Nuclear Operations Department, Harris Plant Section. Located in the General Office.

E. April 1982 - General Manager - Harris Plant in the Nuclear Operations Department located at the Harris Plant, New Hill, NC

F. September 1983 - General Manager - Harris Plant in Harris Nuclear Project Department located at the Harris Plant, New Hill, NC

III. Professional Societies

A. American Nuclear Society

B. N. C. Society of Engineers

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TABLE 13.1.3-5

Lloyd R. Hancock
Administrative Supervisor

I. Education and Training

A. Associate in Applied Science Degree - Mechanical Technology from Davidson County Community College - 1971

B. Diploma in Mechanical Engineering from the International Correspondence School (ICS), Scranton, Pennsylvania - 1974

II. Experience

A. September 1970 through June 1971 - Draftsman, Croft, Inc.

B. June 1971 employed as an Engineering Aide I in the Power Plant Engineering Department in the General Office

C. June 1972 employed as a Technician II in the Power Plant Engineering Department in the General Office

D. October 1973 employed as a Technician I in the Power Plant Engineering Department in the General Office

E. June 1974 employed as a Junior Engineer in the Power Plant Engineering Department located in the General Office and at the Brunswick Plant, Southport, North Carolina

F. October 1975 employed as an Engineer in the Power Plant Engineering Department located in the General Office and at the Brunswick Plant, Southport, North Carolina

G. February 1977 employed as an Engineer in the Generation Services Section of the Generation Department in the General Office

H. January 1979 employed as a Senior Engineer in the Generation Services Section of the Generation Department in the General Office

I. May 1979 employed as a Senior Engineer in the Generation Services - O&M Section of the Generation Department in the General Office

J. June 1979 employed as a Senior Engineer in the Nuclear Generation Section of the Generation Department in the General Office

K. November 1981 employed as an Administrative Supervisor in the Nuclear Operations Department located at the Harris Plant, New Hill, North Carolina

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TABLE 13.1.3-5 (continued)

Lloyd R. Hancock
Administrative Supervisor

L. September 1983 employed as an Administrative Supervisor in the Harris Nuclear Project Department, Harris Plant, New Hill, NC

III. Professional Societies

A. Member of American Society of Mechanical Engineers

TABLE 13.1.3-7 27

Dean L. Tibbitts
Project Specialist - Regulatory Compliance

I. Education and Training 27

A. Webb High School, Reedsburg, Wisconsin - 1969

B. University of Missouri-Rolla, Rolla, Missouri - 1975 - B.S. and M.S. in Nuclear Engineering

C. University of Maryland, College Park, Maryland - 1978

II. Experience Prior to Joining CP&L

A. June 1975 to March 1980 - U. S. Nuclear Regulatory Commission - Project Engineer - Washington, DC

B. March 1980 to February 1981 - NUTECH - Senior Consultant, Bethesda, Maryland

C. February 1981 - March 1983 - Phoenix Power Services - Principal - Washington, DC

III Experience with CP&L

A. June 1983 - Employed as a Senior Specialist - Regulatory Compliance in the Nuclear Operations Department at the Shearon Harris Nuclear Power Plant located in New Hill, North Carolina

B. September 1983 - Reorganization - Department renamed Harris Nuclear Project Department 27 | 37

C. August 1985 - Promoted to Project Specialist - Regulatory Compliance in the Harris Plant Section of the Harris Nuclear Project Department

IV. Professional Affiliations and Achievements 27

None

TABLE 13.1.3-23 | ~~27~~ 37

Danny G. Batten
Shift Foreman

I. Education and Training

A. Bladenboro High School, Bladenboro, North Carolina - 1965

B. U. S. Navy

1. Electricians's Mate "A" School - 4 months
2. Nuclear Power School - 6 months
3. Nuclear Prototype - 6 months

II. Experience Prior to Joining CP&L

A. April 1967 to January 1970 - U.S.S. Truxtun DLGN-35. Qualified Electrical Operator, Auxiliary Electrician and Reactor Plant Shutdown Watch. Maintained electrical equipment.

B. February 1970 to May 1971 - Monob YAG-61. In charge of maintaining electrical system on board this research vessel.

III. Experience with CP&L

A. July 1971 employed as Auxiliary Operator "A" in the Generation and System Operations Department at the H. B. Robinson Plant, Hartsville, South Carolina

B. November 1972 employed as Control Operator in the Bulk Power Supply Department at the H. B. Robinson Plant, Hartsville, South Carolina

C. August 1977 employed as Senior Control Operator in the Generation Department at the H. B. Robinson Plant, Hartsville, South Carolina

D. June 1981 employed as Shift Foreman - Nuclear in the Nuclear Operations Department at the H. B. Robinson Plant, Hartsville, South Carolina

E. May 1982 employed as Shift Foreman - Nuclear in the Nuclear Operations Department at the Shearon Harris Nuclear Power Plant, New Hill, North Carolina. (Temporarily assigned to the H. B. Robinson Plant, Hartsville, South Carolina.)

F. July 1982 employed as a Shift Foreman in the Nuclear Operations Department at the Shearon Harris Nuclear Power Plant, New Hill, North Carolina

G. September 1983 employed as a Shift Foreman in the Harris Nuclear Project Department, New Hill, NC

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TABLE 13.1.3-25

John W. Digby
Shift Foreman

I. Education and Training

A. Miami Edison Senior High - Miami, Florida - 1960

B. George T. Baker Aviation - Miami, Florida - 1964 to 1966 - No Degree - Avaiation Mechanics

C. Purdue - West Lafayett, Indiana - 1980 to 1983 - No Degree - STA Program

D. Electrical Power Production Technical - Sheppard Air Force Base, Watchta Fall, Texas - October 1961 to June 1961 - Electrical Power Production

E. Electric Power Production Missile School - Sheppard Air Force, Watchta Fall, Texas - June 1961 to November 1961 - Electrical Power Production

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II. Experience Prior to Joining CP&L

A. June 1961 to March 1964 - EPPT - United States Air Force

B. June 1965 to September 1966 - Truck Driver - Lou-Mack Transfer - Miami, Florida

C. September 1966 to June 1978 - Watch Engineer (SRO) - Florida Power and Light Company - Miami, Florida

D. June 1978 to June 1980 - Reactor Control Operator I - Washington Public Power Supply System - Richland, Washington

E. August 1980 to January 1984 - Shift Supervisor - Public Service Indiana - New Washington, Indiana

III. Experience with CP&L

A. February 1984 - Employed as Shift Foreman - Nuclear in the Harris Nuclear Project Department, New Hill, North Carolina

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TABLE 13.1.3-32

L. P. Capps
Superintendent - Materials and Custodial

I. Education

A. Pembroke State University - Pembroke, North Carolina - 1973 B. S. in Accounting

II. Experience

A. 1973 - Junior Accountant; Plant Accounting Section, Treasury and Accounting Department, Carolina Power & Light Company, Raleigh, North Carolina

B. 1975 - Accountant, Plant Accounting Section, Treasury and Accounting Department, Carolina Power & Light Company, Raleigh, North Carolina

C. 1975 - Accountant, Construction Engineering and Accounting Section, Power Plant Construction Department, Carolina Power & Light Company, Raleigh, North Carolina

D. 1976 - Senior Accountant, Nuclear Construction Section, Power Plant Construction Department, Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, New Hill, North Carolina.

E. 1978 - Supervisor - Project Accounting, Harris Site Management Section, Nuclear Plant Construction Department, Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, New Hill, North Carolina

F. 1984 - Superintendent - Materials and Custodial Project in the Administration Section, Harris Nuclear Project Department, Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, New Hill, North Carolina.

III. Professional Societies

None

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TABLE 13.1.3-33

E. E. Johnson
Principal Specialist - Document Services

I. Education

A. North Carolina State University - Raleigh, North Carolina - 1964 - B. S. in Applied Mathematics

II. Experience Prior to Joining CP&L

A. 1964 - Engineer in Atomic Power Division, Newport News Shipbuilding & Dry Dock Company

B. 1971 - Engineering Supervisor in Atomic Power Division, Newport News Shipbuilding & Dry Dock Company.

C. 1973 - Structural Design Specialist, Gilbert Associates, Inc.

D. 1975 - Project Control Engineer - Gilbert Associates, Inc.

E. 1977 - Records Management Consultant of VEPCO's North Anna Nuclear Power Plant for Gilbert Associates, Inc.

F. 1979 - Supervisor, Administration and Records Management, at VEPCO's North Anna Nuclear Power Plant

G. 1981 - Information Management Consultant at SCE&G's V. C. Summer Nuclear Power Plant for Gilbert Associates, Inc.

III. Experience with CP&L

A. 1983 - Project Specialist - Nuclear Operations Administration, Carolina Power & Light Company

B. 1984 - Principal Specialist - Document Services in the Project Administration Section of the Harris Nuclear Project Department, Carolina Power & Light Company

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TABLE 13.1.3-33 (continued)

E. E. Johnson
Principal Specialist - Document Services

III. Professional Societies

A. Member of Institute of Certified Records Managers

B. Member of Association of Information and Image Management (formerly National Micrographics Association)

C. Member of Association of Records Managers and Administrators

D. Member of Nuclear Records Management and Administrators

(1) Member on Micrographics Committee
(2) Member on Technical Support Center/Emergency Offsite Facility Records Committee

DELETE

DELETE

TABLE 13.1.3-34

Ronald E. Gurganus
Project Engineer - Project Analysis, Harris Nuclear Project Department

I. Education

A. North Carolina State University - Raleigh, North Carolina - 1970 B.S. in Forest Management.

B. Duke University - Durham, North Carolina - 1980 - MBA in Business Administration

C. International Correspondence School - Scranton, Pennsylvania - 1982 Diploma in Electrical Engineering, Power Option

D. 3286th United States Army Reserve School - Raleigh, North Carolina - 1982 - First Sergeant School - Diploma

II. Experience Prior to Joining CP&L

A. 1970 - Assistant to Vice President of General Construction, Stackhouse, Inc..

III. Experience with CP&L

A. 1973 - Transmission Line Right-Of-Way Forester, CP&L System Operations Department

B. 1978 - Senior Specialist, CP&L System Operations Department

C. 1983 - Senior Engineer - Project Analysis, CP&L Harris Site Management

D. 1984 - Project Engineer - Project Analysis, Harris Project Administration, CP&L, Harris Nuclear Project Department

IV. Professional Societies

None

DELETE

DELETE

TABLE 13.1.3-35

R. E. Butler
Project Engineer - Site Industrial Engineering

I. Education

A. Maryland University - College Park, Maryland - 1959 B. S. in Chemical Engineering/Math

B. Maryland University - College Park, Maryland - 1967 B. S. in Industrial Engineering

II. Experience Prior to Joining CP&L

A. 1964 - Distribution Supervisor - UPS - Landover, Maryland

B. 1966 - Personnel Supervisor - UPS - Landover, Maryland

C. 1967 - I.E. Manager - UPS - New Jersey District

D. 1974 - Industrial Engineering Manager Assigned to UPS National Staff - New York, New York

E. 1976 - District Operations/I.E. Manager Carolina District - UPS - Raleigh, North Carolina

III. Experience with CP&L

A. 1980 - Industrial Engineering Supervisor - Harris Site Construction

B. 1983 - Project Engineer - Site Industrial Engineering, Shearon Harris Nuclear Power Plant

IV. Professional Societies

A. Member American Institute of Industrial Engineers

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~~24~~

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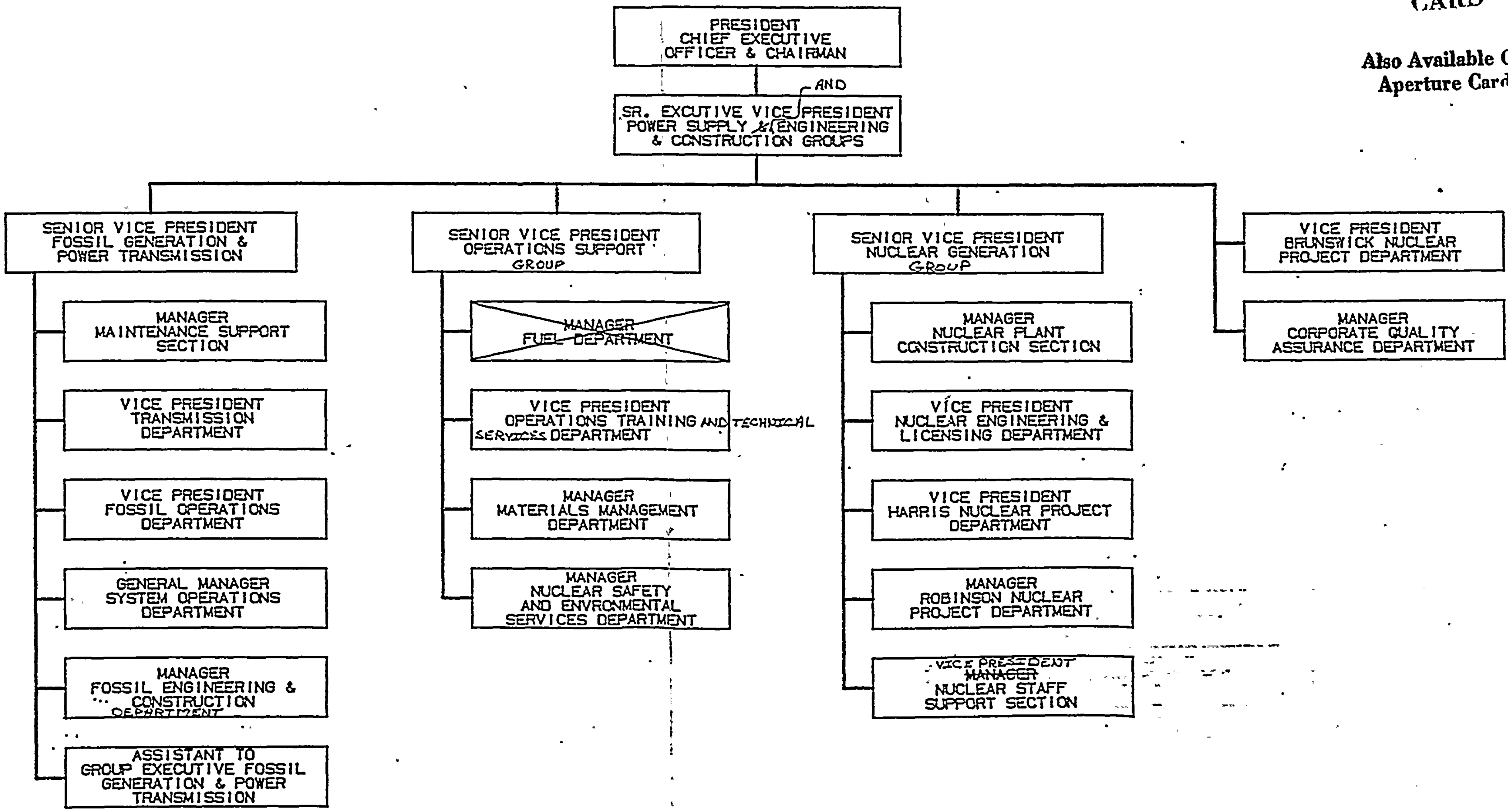
Also Available On Aperture Card



AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

OPERATIONS AND ENGINEERING & CONSTRUCTION ORGANIZATION

FIGURE 13.1.1-1

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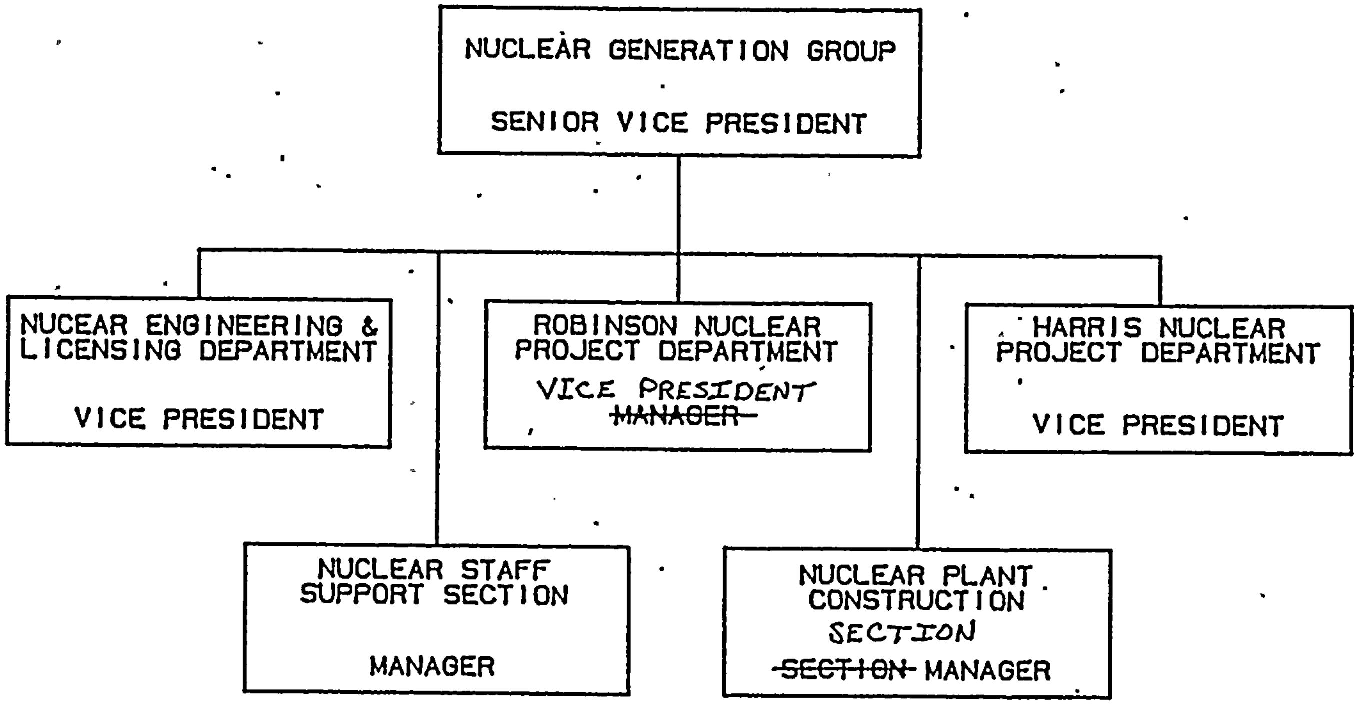
[illegible]
[illegible]



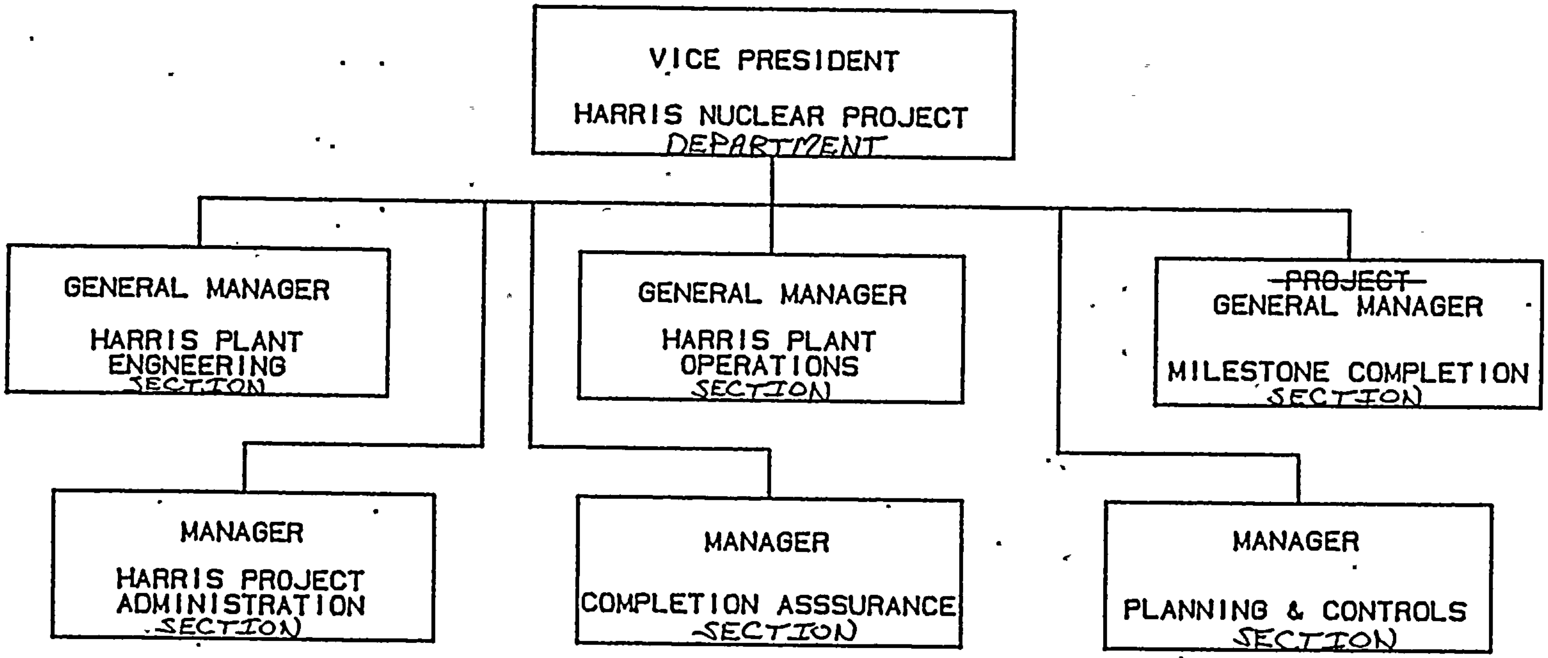
AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

NUCLEAR GENERATION GROUP ORGANIZATION

FIGURE 13.1.1-3

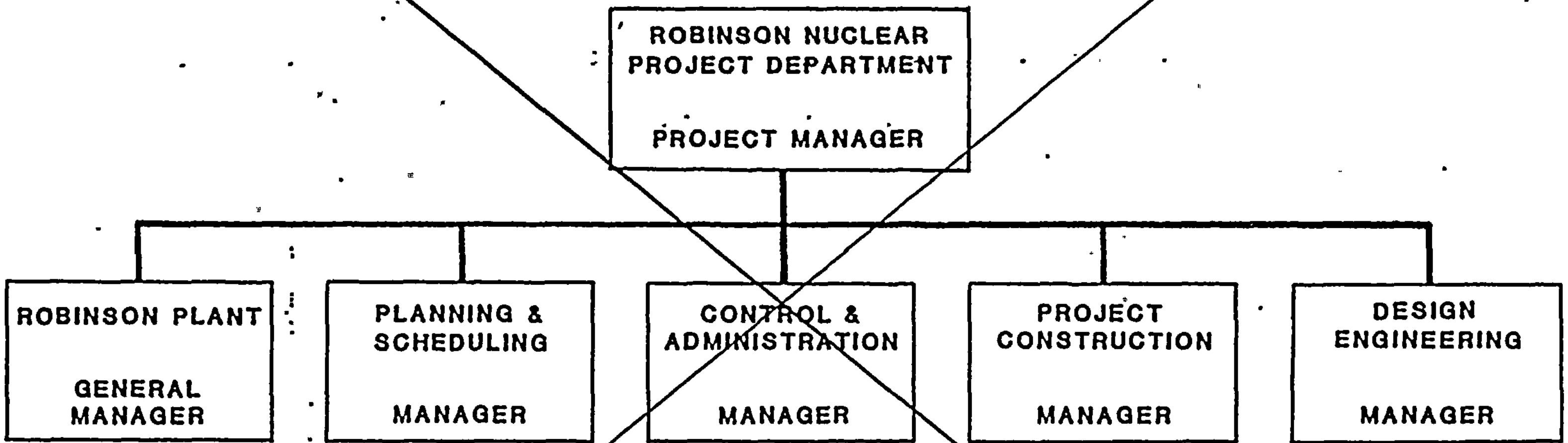
SHEARON HARRIS
NUCLEAR POWER PLANT
Carolina
Power & Light Company
FINAL SAFETY ANALYSIS REPORT



HARRIS NUCLEAR PROJECT ORGANIZATION

FIGURE 13.1.1-4

AMENDMENT NO. ~~27~~ 37



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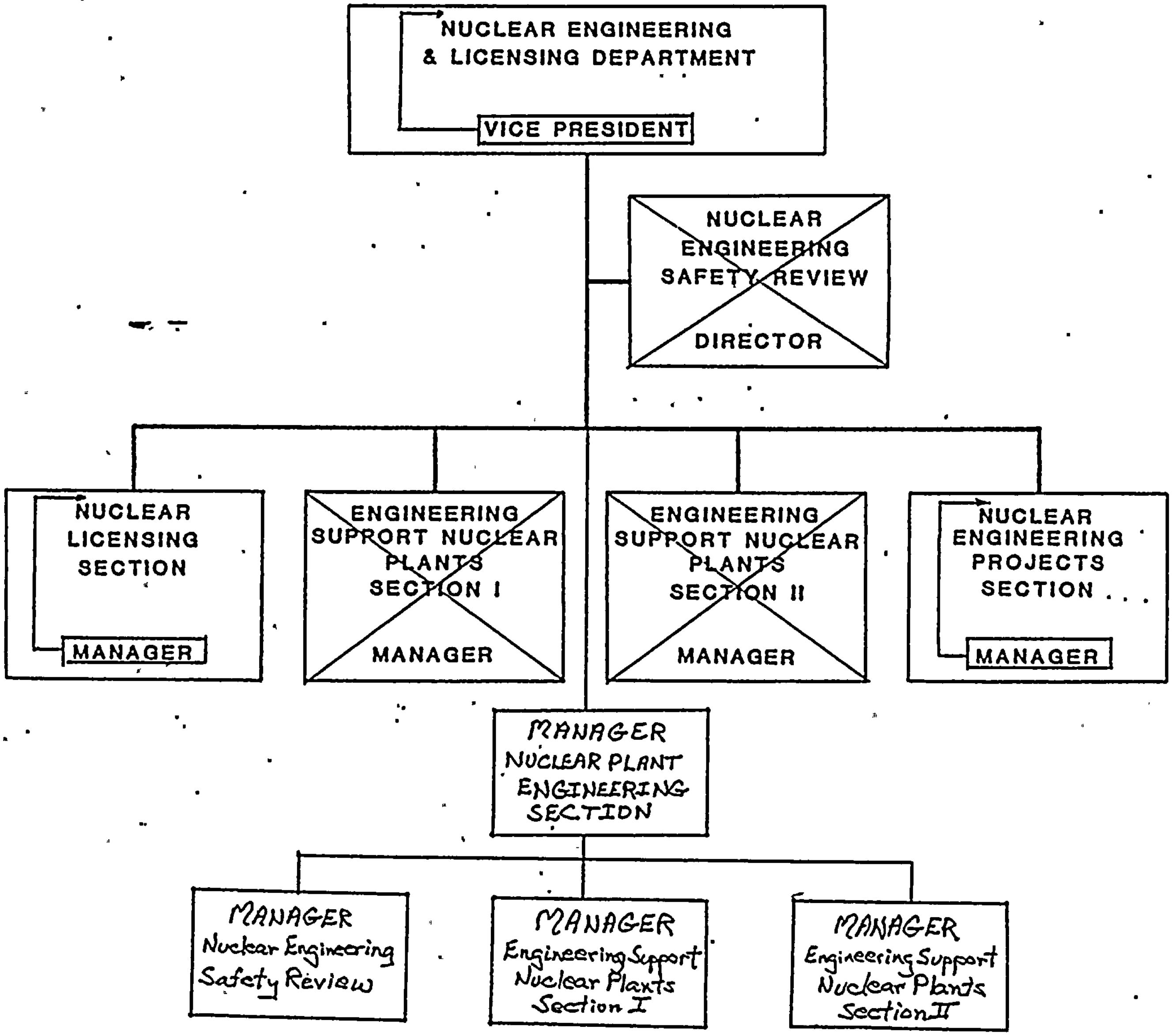
Amendment No. ~~23~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

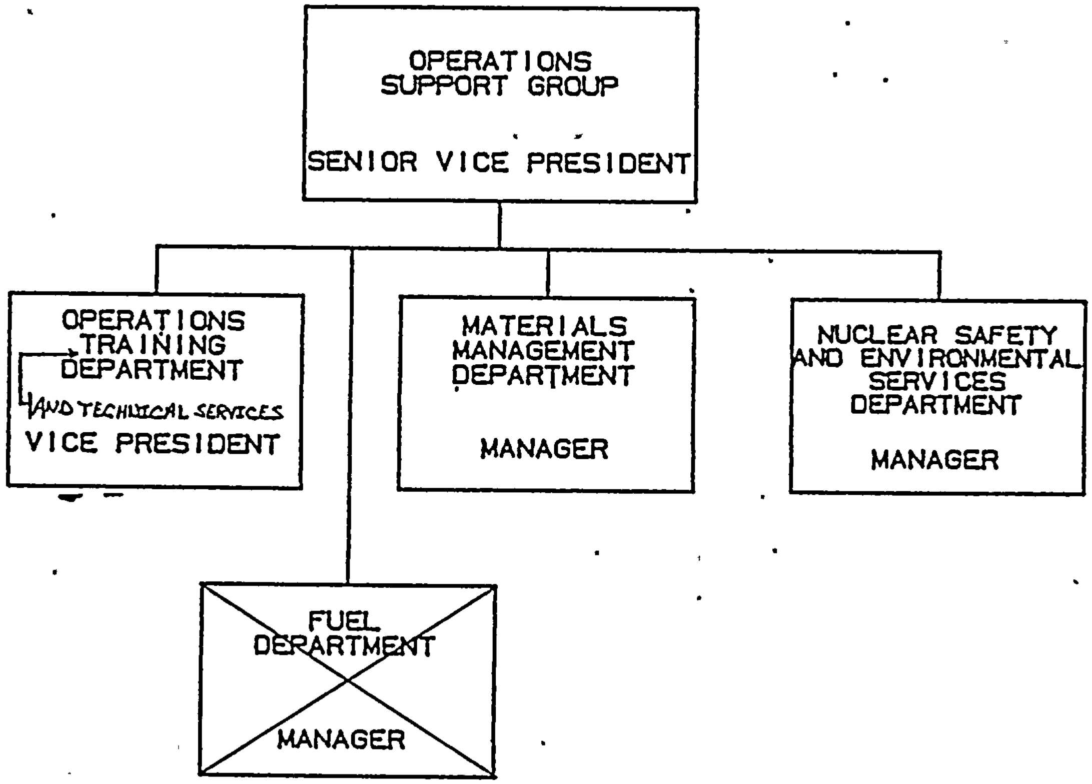
ROBINSON NUCLEAR PROJECT ORGANIZATION

FIGURE 13.1.1-5



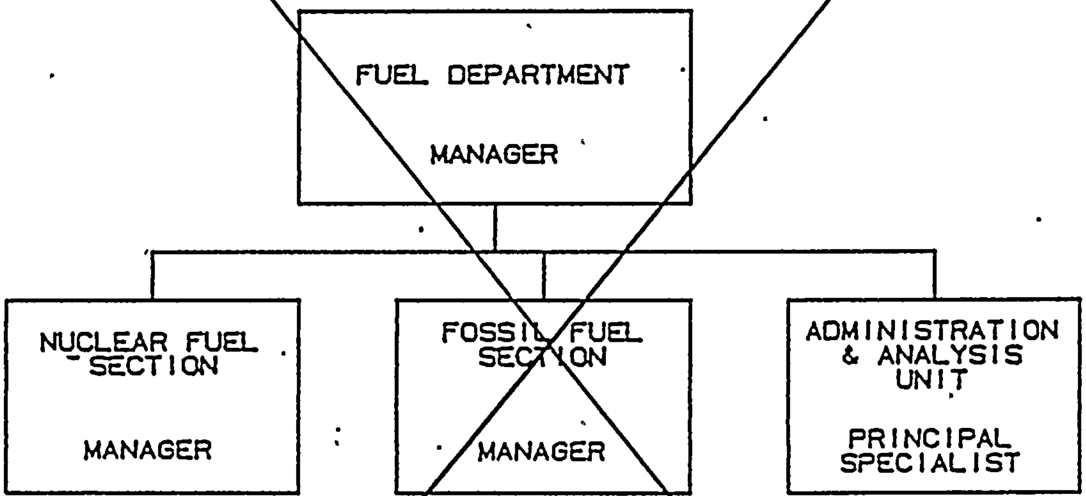
Amendment No. ~~13~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	NUCLEAR ENGINEERING AND LICENSING DEPARTMENT ORGANIZATION	FIGURE 13.1.1-6



AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	OPERATIONS SUPPORT GROUP ORGANIZATION	FIGURE 13.1.1-7



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AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	FUEL DEPARTMENT ORGANIZATION	FIGURE 13.1.1-8

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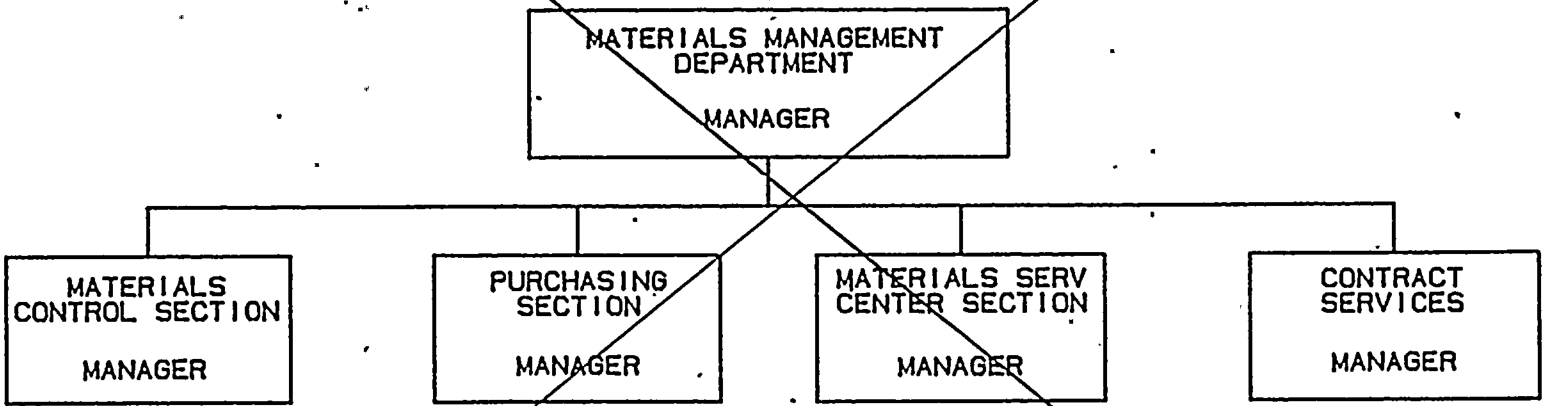
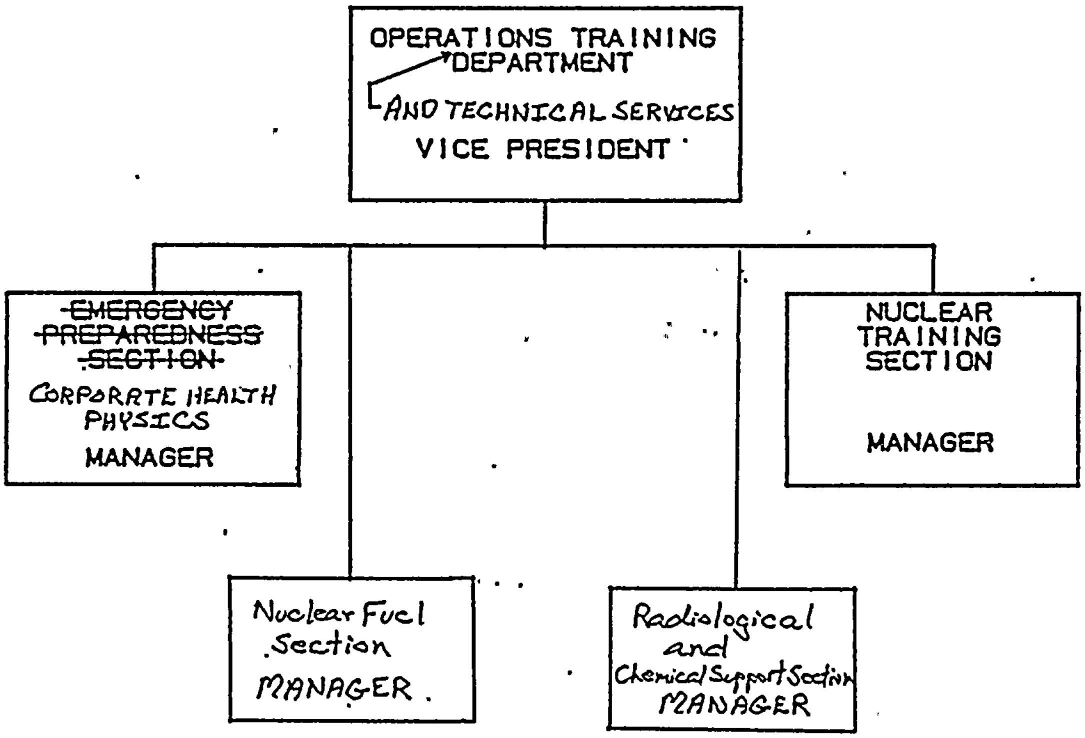


AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

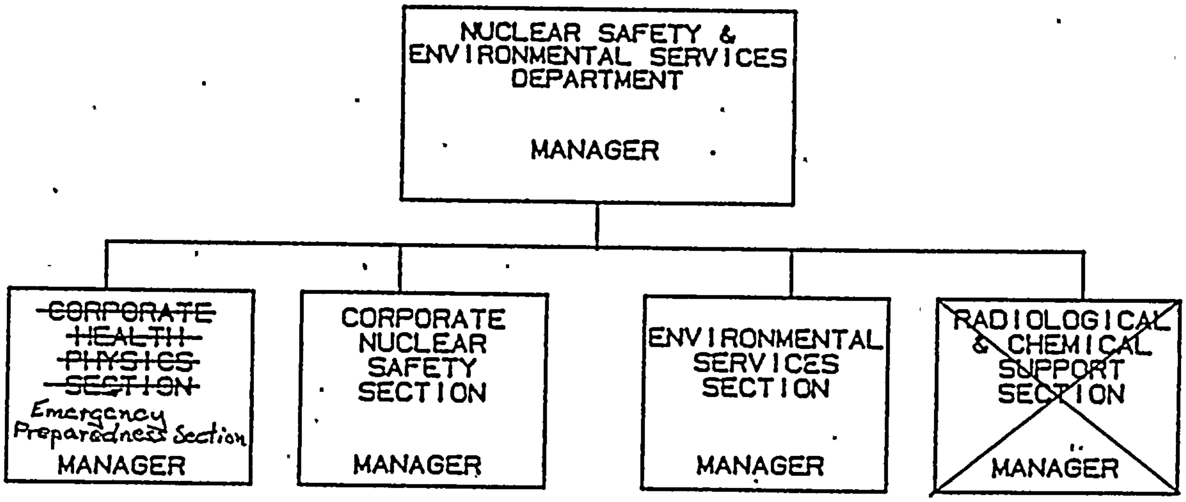
MATERIALS MANAGEMENT DEPARTMENT ORGANIZATION

FIGURE 13.1.1-9



AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	OPERATIONS TRAINING AND TECHNICAL SERVICES DEPARTMENT ORGANIZATION	FIGURE 13.1.1-10



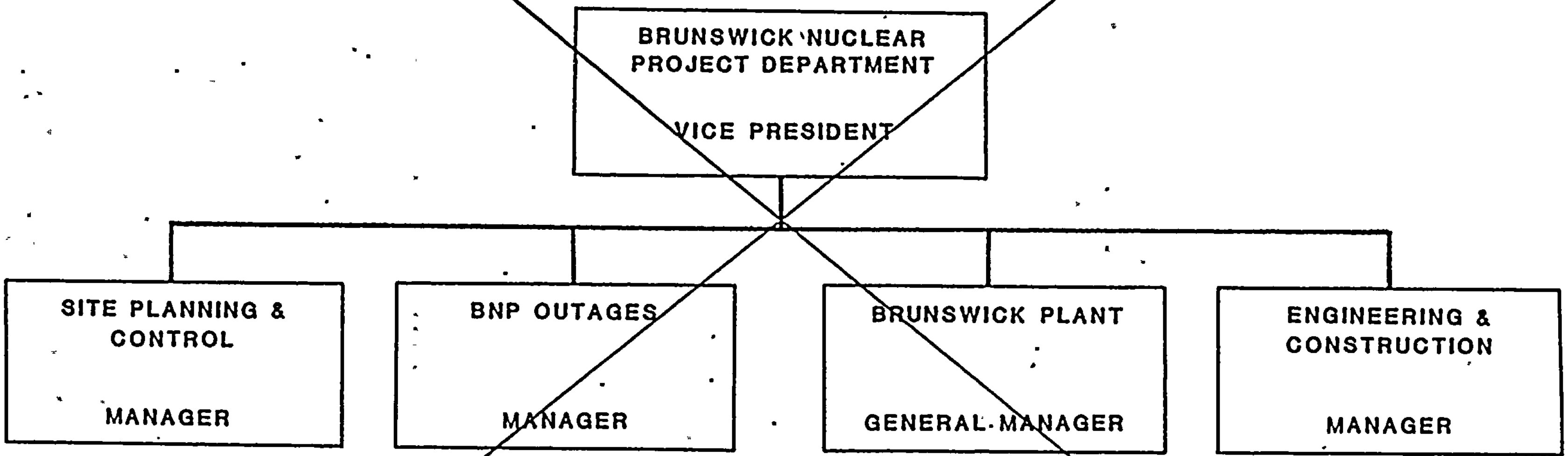
AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	NUCLEAR SAFETY & ENVIRONMENTAL SERVICES DEPARTMENT	FIGURE 13.1.1-11

SHEARON HARRIS
NUCLEAR POWER PLANT
Carolina
Power & Light Company
FINAL SAFETY ANALYSIS REPORT

BRUNSWICK NUCLEAR PROJECT ORGANIZATION

FIGURE
13.1.1-12



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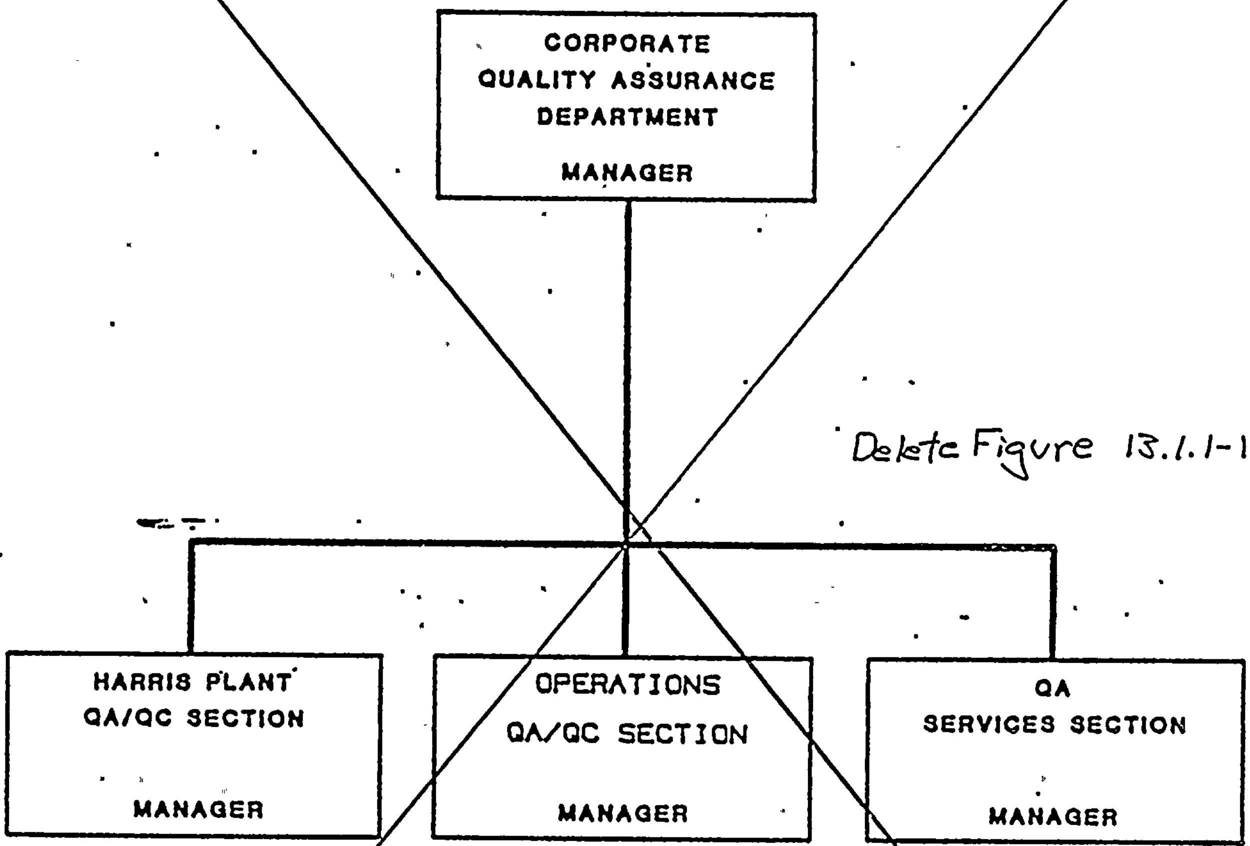

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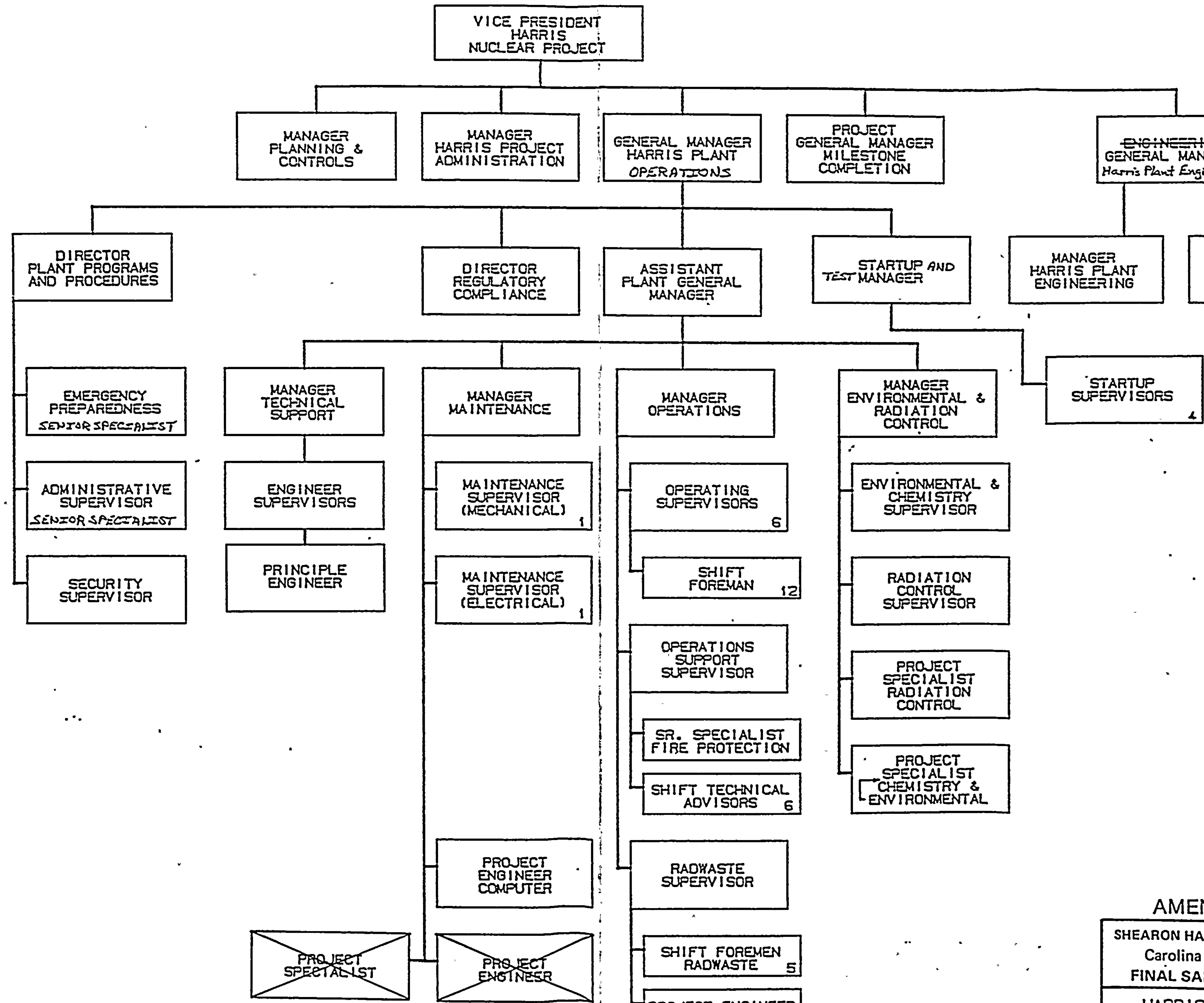
AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	CORPORATE QUALITY ASSURANCE DEPARTMENT ORGANIZATION	FIGURE 13.1.1-13

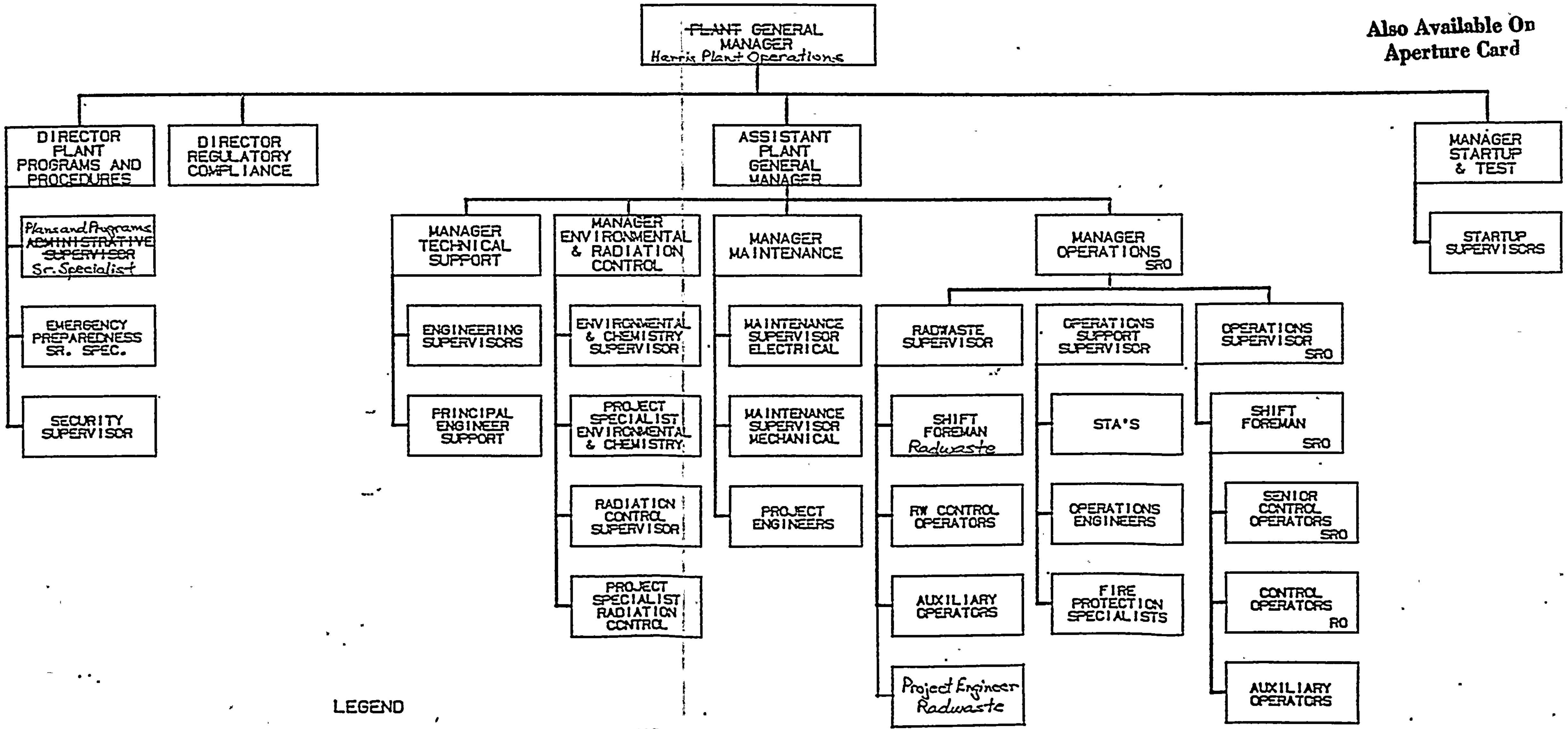


AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

HARRIS NUCLEAR PROJECT ORGANIZATION

FIGURE 13.1.2-1



LEGEND

——— ADMINISTRATIVE ORGANIZATION

SRO - SENIOR REACTOR OPERATORS LICENSE

RO - REACTOR OPERATORS LICENSE

AMENDMENT NO. ~~27~~ 37

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

FACILITY ORGANIZATION
FIGURE 13.1.2-2

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[illegible]

13.2.2 REPLACEMENT AND RETRAINING

A training program will be utilized to maintain the proficiency of the plant operating organization after the initial plant start-up. This training program will include, as described below, requalification training for licensed personnel, and replacement training for replacement personnel.

13.2.2.1 Licensed Operator Requalification Training

Following the initial licensing of cold license candidates, a requalification training program will be initiated to evaluate, document ~~maintain~~ and demonstrate the continued proficiency and competence of all licensed personnel. This requalification training program ~~will be conducted on an annual basis and~~ will include pre-planned lectures, on-the-job training, and regular and continuing operator evaluation. The SHNPP simulator will be used to fulfill appropriate portions of this retraining program. 37

The program ... is based on a two year cycle in accordance with 10CFR55. Requalification training will be preplaned and conducted on an annual basis meeting all 10CFR55 requirements over a two year period.

13.2.2.1.1 Lectures

A minimum of six pre-planned lectures will be presented during each requalification cycle. These lectures will be scheduled throughout the two year cycle taking into account heavy vacation periods and infrequent operations such as refueling periods and forced outages. Lectures may be deferred due to unanticipated shutdowns. However, these lectures shall be conducted as soon as practicable thereafter. Content of the lectures shall take into consideration the categories as listed in 10CFR Part 55, Appendix A, heat transfer, fluid flow, thermodynamics, mitigation of accidents involving a degraded core, operating experiences from similar plants and the results of the annual examination. Training aids such as films, video tapes, and slides may be used and some self-study may be required in conjunction with the lectures. An instructor will present or attend as an auditor at least 50 percent of the lecture series. 37

All licensed individuals will be required to attend every pre-planned lecture except those specifically exempted. Exemptions will be allowed only for individuals scoring greater than 80 percent in the corresponding area on the previous examination.

13.2.2.1.2 On-the-job Training

The on-the-job training portion of the requalification program will consist of the following:

a) Control Manipulation -Licensed reactor operators shall manipulate and senior reactor operators shall manipulate or direct or evaluate the activities of those manipulating the station controls through a minimum of ten reactivity changes during each annual cycle. These manipulations may consist of any of the following, providing that asterisked items are performed annually and all other items are performed on a two year cycle:

*1) Start-up to the point of adding heat

2) Orderly shutdown

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of auxiliary feedwater flow due to loss of off-site power, loss of main feedwater flow, safety injection signal, or low-low level in one or more steam generators.

6) Visually observe system piping and components during system operation for abnormal vibration or piping response to system and component operations. Perform instrumented tests as required by the System Dynamic Test and Analysis Test Procedures.

~~7) During Low Power Testing, prior to attaining 25% power perform a 48-hour endurance test of each motor-driven auxiliary feedwater pump using a flow path to the steam generators and recirculation flow to Condensate Storage Tank. On completion, each pump will remain idle approximately eight hours to permit pump cooldown, and then it will be restarted and operated using the same flow path for one hour.~~ 37 ~~23~~

d) Acceptance Criteria

1) Automatic initiation of feedwater flow from the Auxiliary Feedwater System shall occur within one minute of the automatic auxiliary feedwater actuation signal listed in FSAR Section 10.4.9.2.4 (a) and (b).

2) The turbine-driven auxiliary feedwater pump and turbine performance shall meet or exceed vendor performance data supplied in SHNPP Technical Manual 16-P043-3065.

3) The turbine-driven auxiliary feedwater pump reliability shall be demonstrated by performance of five consecutive cold starts.

4) The motor driven auxiliary feedwater pumps pressure control valves maintain pump discharge pressure above 1000 psig.

5) The motor-driven auxiliary feedwater pumps flow to the steam generators shall meet or exceed the rate shown on the vendor pump curves. This rate shall be equal to or greater than 400 gpm per pump as indicated by installed flow elements. ~~23~~

Note: Steam Generator capacity in gallons/inches determined during initial steam generator filling.

6) System dynamic testing completed per SHNPP Dynamic Test and Analysis, FSAR Section 3.9.2.

7) The motor driven auxiliary feedwater pump shall not exceed the limitations for vibration, bearing, and bearing oil outlet temperatures as specified in SHNPP Technical Manual 16-P043-3065. In addition, the RAB Ventilation and Equipment Cooling Systems shall maintain environmental conditions of the Auxiliary Feedwater piping area within the design requirements of FSAR Sections 9.4.3 and 9.4.5.

d) Acceptance Criteria

1) The ESFAS components actuate into the states shown on FSAR Tables 7.3.1-5 through 7.3.1-11 as appropriate for A train operation, B train operation, and both train operation. During single train operation, the opposite train 6.9 kV buses remain de-energized.

2) The emergency diesel generators start and sequence loads, including capability to carry manual loads of FSAR Table 8.3.1-2 when offsite power is not available.

3) Upon resetting the initiating ESFAS signals, the safety related components actuated above remain in their ~~actuated state/position.~~ emergency mode. 37

14.2.12.1.60 Process Computer Test Summary

a) Test Objective

To demonstrate that the ERFIS computer system functions as per vendor's technical manual to provide monitoring, alarming, displaying, reporting, and archiving capabilities to the Control Room Operator, the Technical Support Center, and the Emergency Operations Facility.

b) Prerequisites

1) The general prerequisites are met.

2) Hardware is complete and verified by the vendor test plan and procedures.

3) Software is installed and proven by satisfactory completion of the performance sections of the Vendor Test Plan and Procedures.

~~24~~

c) Test Method

1) Verify each process input is operable by simulating the input or application of a known input.

2) Verify that the alarm and conversion of each type process input provides valid information by simulating various input of conditions or by monitoring various levels of known inputs.

3) Verify the ERFIS system's capability to give proper computational results by simulating the inputs or using static test cases and comparing the result against independently computed values.

4) Verify the system display capability in the Control Room, the Technical Support Center (TSC) and in the Emergency Operations Facility (EOF).

5) Verify the redundancy capability of the ERFIS system by inducing system faults and observing system performance.

3) With continuous sampling flow established, verify the operation of the temperature control system.

4) Test the operation of the Hotwell Sample Pumps.

5) Operate the Steam Generator Blowdown sample isolation valves and verify closure times specified by FSAR Section 6.2.4.

d) Acceptance Criteria

1) Sample points have been verified per FSAR Table 9.3.2-1 and 9.3.2-2.

2) All controls and alarms function in accordance with latest design documents.

3) ~~Automatic~~ Temperature control system has demonstrated the capability to maintain sample temperatures at 77 ± 5 F. 37

4) Steam Generator Blowdown samples are cooled to less than 120 F.

5) Hotwell Sample Pumps operate in accordance with vendor instruction manual (16-P175).

6) Steam Generator Blowdown sample isolation valves have demonstrated closure times specified by FSAR Section 6.2.4.

14.2.12.1.79 Loss of Instrument Air Test Summary

a) Test Objectives

1) To demonstrate that a reduction and loss of instrument air pressure causes fail-safe operation of pneumatically-operated valves and dampers both safety and nonsafety related located in the reactor building, auxiliary building and fuel handling building.

b) Prerequisites

1) The general prerequisites are met.

2) Specific prerequisites will be delineated in the system preoperational test procedure.

c) Test Method

1) Where safe to personnel and equipment, a slow reduction in pressure and a loss of pressure test will be performed. Testing will be done in small segments/individually and response noted for both safety and nonsafety-related valves and dampers. The loss of pressure test will be conducted by isolating segments/individual items and venting the air from the isolated segment.

d) Acceptance Criteria

1) Proper fail-safe operation of valves and dampers subject to a reduction and loss of instrument air is verified.

3) Repeat steps 1 and 2 for low pressure heaters #3 and 4.

d) Acceptance Criteria

1) With the bypassing of either heater group, the resultant transient will be less severe than the predicted response of FSAR Section 15.1.1.

2) The most severe loss of feedwater heater transient will be identified based upon the resultant data from the series of tests.

14.2.12.2.30 Main Steam Isolation Valve Test Summary

a) Test Objective

1) This test will demonstrate the capability of the main steam isolation valve (MSIV) "Test Feature" to operate as designed under maximum steam flow conditions.

b) Prerequisites

1) Plant is at approximately 100% power.

2) The general prerequisties are met.

c) Test Methods

1). The MSIV control switch is put in the "Test" mode. (Each valve is tested separately)

2) Monitor plant response and valve stem travel.

d) Acceptance Criteria

1) MSIV stem travels from 100% open to 90% open and back to 100% open. ~~automatically.~~ 37

2) The plant's control systems operate to maintain steady state conditions.

14.2.12.2.31 Steam Generator Test for Condensation Induced Water Hammer

a) Test Objective

1) To demonstrate the capability of transferring feedwater flow from the auxiliary feedwater nozzles to the main feedwater nozzles. ~~34~~

b) Prerequisites

1) Plant conditions are established as required by test instructions for 15 percent power level. ~~34~~

c) Test Method

1) Operate the plant at approximately 15 percent of full power by feeding the steam generators through the auxiliary feedwater nozzles.

34 2) Transfer the feedwater flow to the main feedwater nozzles by opening the main feedwater isolation valves per general plant operating procedures.

3) Observe and record the transient that follows.

d) Acceptance Criteria

Either low amplitude or no condensate-induced water hammer is observed in the region of the main feedwater nozzle/preheater section of steam generator.

and motor driven

14.2.12.2.32 Steam Turbine-Driven/Auxiliary Feedwater Pumps Endurance Test 37

a) Test Objective

1) To demonstrate the capability of the steam turbine-driven and motor driven auxiliary feedwater pumps to continuously feed two or more steam generators for a 48-hour period without exceeding the design conditions of FSAR Section 9.4.3 and 9.4.5. 37

b) Prerequisites

1) The general prerequisites have been met.

2) The plant is operating at a power level of 25 percent or less ~~for the 48-hour endurance run~~ with a main feedwater pump in operation. 37

Insert "A" From Page 14.2.12-100a

c) Test Method

~~1) With the steam turbine-driven auxiliary feedwater pump supplying feedwater to two or more steam generators, increase reactor power until maximum flow from the pump is obtained or reactor power is 25 percent.~~

~~2) Maintain maximum flow condition or 25 percent reactor power for 48 hours.~~

~~3) On completion of 48-hour endurance run, shift feedwater supply to the motor-driven auxiliary feedwater and secure the turbine-driven auxiliary feedwater pump per normal plant procedure. Additional feedwater needs during this time will be provided by the main feedwater pumps.~~

~~4) After obtaining ambient pump conditions, start the turbine-driven auxiliary feedwater pump and operate for 1 hour.~~

~~5) During the 48-hour run of the pump, the pump cubicle humidity will be measured at specified intervals and cubicle temperature shall be recorded at a minimum frequency of hourly.~~

d) Acceptance Criteria

1) The turbine-driven auxiliary feedwater pump and turbine, vibration, bearing, and bearing lube oil temperature shall not exceed ~~manufacturer's specification per SHNPP Technical Manuals 16-P043-3065.~~ 200F metal temperature, 160F return oil temperature and 2.0 mils vibration.

2) The motor-driven auxiliary feedwater pump motors temperatures shall not exceed 160F for bearing temperatures, 60F rise for motor winding temperatures, and vibration of 2.0 mils. 37

Insert "A" to Page 14.2.12-100

1) Operate the steam turbine-driven and/or the motor-driven auxiliary feedwater pumps to supply feedwater to two or more steam generators. The initial total flow from the pumps, with all three pumps in operation, will be approximately 500 to 600 gpm. If temperatures and vibration at this flow are acceptable, the test will continue at this flow rate. If temperatures or vibration are not acceptable, the flow rate will be increased until it is acceptable. The test for each pump will continue for 48 hours from the time of initial pump start. During this time, pump operating parameters, (vibrations, bearing temperature, discharge pressure, etc.) and pump cubicle environmental conditions (temperature on an hourly frequency, humidity at specified intervals) will be recorded.

2) Makeup to the Condensate Storage Tank will be via an additional condensate transfer line downstream of the Condensate Polishing System in order to maintain chemistry.

3) Upon completion of the 48-hour endurance run, maintain feedwater supply through the Main Feedwater Pump and secure all three auxiliary feedwater pumps.

4) After obtaining ambient pump conditions for the turbine-driven pump, restart the turbine-driven pump and operate for one hour.

5) Upon cooldown of the motor-driven pumps (approximately eight hours after completion of the 48-hour run), restart pumps and operate for one hour.

3~~2~~) The Reactor Auxiliary Building Ventilation and Equipment Cooling Systems shall maintain environmental conditions of the auxiliary feedwater pump ~~piping~~ areas within the design requirements of FSAR Sections 9.4.3 and 9.4.5. 37

14.2.12.2.33 Resistance Temperature Detector (RTD) Bypass Flow Verification Test Summary

a) Test Objectives

1) To determine that the flowrate in each RTD bypass loop is sufficient to achieve the design RTD bypass transport time.

b) Prerequisites

1) The reactor is in the hot shutdown condition with all reactor coolant pumps running.

2) RTD bypass loop flow instrumentation is calibrated and in service.

3) The general prerequisites are met.

c) Test Method

1) The flow required to achieve the design reactor coolant transport time is determined by measuring and recording the lengths of installed piping in each hot and cold leg bypass loop and then calculating the flow necessary to achieve design transport time.

2) Total bypass flow rate for each loop is measured and recorded, and then actual bypass transport time is calculated.

d) Acceptance Criteria

1) The flow rate in each RTD bypass loop yields design transport time as per the Westinghouse NSSS Startup Manual.

14.2.12.2.34 Secondary Sampling System Test Summary

a) Test Objective

1) To verify the proper operation and performance of the Secondary Sampling System.

2) To verify the operation of the following sample points: S7, S8, S9, S10, S11. ~~27~~

b) Prerequisites

1) To the extent practical systems to be sampled are at operating pressure and temperature. ~~27~~

2) Cooling water is available to sample panels for sample coolers and chiller condensers.

c) Test Method

1) By valving "in and-out" sample lines, verify samples to sample panels are correctly identified.

2) With continuous sampling flow established, verify the operation of the ~~automatic~~ temperature control system. 37

d) Acceptance Criteria

1) Sample points have been verified per FSAR Table 9.3.2-2.

2) ~~Automatic~~ Temperature control system has demonstrated the capability to maintain sample temperatures at 77 ± 5 F. 37

TABLE 15.6.5-6

RADIOLOGICAL CONSEQUENCES OF A POSTULATED LOSS-OF-COOLANT ACCIDENT

	Doses (rem)			
	Thyroid	Whole Body	Skin	~~25~~
0-2 Hour Dose at the Exclusion Area Boundary*	1.5×10^{2}	2.6×10^{0}		
0-30 Day Dose at the Low Population Zone*	1.7×10^{2}	1.6×10^{0}		
0-30 Day Dose to the Control Room Personnel	5.1×10^{0}	$6.\cancel{8}0 \times 10^{-1}$	1.4×10^{1}	~~25~~ 37

* These doses include contributions from ECCS outside containment (refer to Section 15.6.5.4.3.c for further information).

15.7.4.3.2 Postulated Fuel Handling Accident Inside Containment

The possibility of a fuel handling accident inside Containment during refueling is relatively small due to the many physical, administrative, and safety restrictions imposed on refueling operations.

a containment ventilation

redundant area radiation monitors.

During fuel handling operations, the Containment is kept in an isolable condition, with all ventilation penetrations to the outside atmosphere either closed or capable of being closed on ~~an~~ isolation signal initiated by ~~a duct mounted radiation monitor~~. At least one of the two interlock doors on the personnel locks is kept closed. In addition, there are ~~area and~~ airborne radiation monitors in the Containment ~~which can isolate ventilation penetrations~~ and portable monitors with audible alarms located in the fuel handling area during refueling. Should a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors sound an audible alarm, the Containment is isolated and the personnel are evacuated. The containment pre-entry purge lines are automatically closed, thus minimizing the escape of any 37 ~~29~~

upon a containment ventilation isolation signal radioactivity. The consequences of dropping a fuel assembly in the Containment are less severe than the consequences of dropping the assembly in the Fuel Handling Building, since the Containment provides a considerably greater holdup time than the Fuel Handling Building, allowing for radioactive decay of the released fission products.

For analytical purposes, consideration is given to one accident; a drop of a fuel assembly into the refueling cavity by the manipulator crane inside Containment. Assumptions and parameters used in evaluating the fuel handling accident inside Containment are shown in Table 15.7.4-3.

The radiological consequences of a fuel handling accident inside Containment were conservatively evaluated by assuming that containment releases ~~via the containment pre-entry purge line~~ occur ~~during~~ for the first ~~27~~ 20 second period. It was assumed that further releases were ~~Any subsequent release will be~~ made in a controlled manner through the Reactor Auxiliary Building Filtration System charcoal adsorbers. The assumption of a controlled release was made due to the availability of these charcoal adsorbers and based on calculations showing that containment isolation can be achieved prior to ~~releases exceeding 10CFR100 guidelines.~~ radionuclides reaching the first containment isolation valve. 37 ~~28~~

The activity released inside Containment as a result of a fuel handling accident will be detected by ~~the containment pre-entry purge~~ redundant area radiation monitors when the monitor setpoint is reduced to 150 mR/hr for refueling operations. The response time for ~~this~~ these monitors is expected to be less than 5 ~~12~~ seconds. Following activity detection, the monitors will initiate the closure of the containment ventilation ~~pre-entry~~ isolation valves. The valves will require a maximum 15 seconds to close. The Containment will be isolated in ~~27~~ 20 seconds after detection of the accidental release of radioactivity. 37 ~~28~~

The time required for airborne activity to reach the containment isolation valve is based on 1) travel time from the surface of the reactor pool to the nearest intake header and 2) travel time through the duct. The nearest intake header from the pool is located at a distance of 14.8 feet. The average airflow velocity within a distance of less than 3 feet and the velocity at 3 feet from the intake header was estimated using equations from Reference 15.7.4-1. These equations are as follows:

$$V = \frac{Q}{10X^2 + A} \qquad (1)$$

$$V_{av} = \frac{Q}{X\sqrt{10A}} \tan^{-1} \frac{X\sqrt{10A}}{A} \qquad (2)$$

where,

X = distance outward along axis, ft. (Note: Equation is accurate only when X is less than 1 1/2 D)

V = centerline velocity at distance X from hood, ft/min.

V_{av} = average velocity within a distance X, ft/min.

Q = air flow, cfm

A = area of hood opening, ft^2

D = diameter of round hoods or side of essentially square hoods.

The air velocity beyond 3 feet, though expected to be smaller, is assumed to remain the same as that at 3 feet from the intake header. The size of the intake header is 24" x 24". The average air velocity up to and including 3 feet was estimated at 179.6 ft/min and the average air velocity beyond 3 feet was estimated to be for a given intake rate of 2500 cfm through the header. Therefore the activity would take 27.6 seconds to reach the intake header from the surface of the pool. The travel time in the 26 in. x 30 in. duct (106 ft. in length) would be 3.4 seconds. Therefore the total time required before the activity would reach the isolation valve is 31 seconds.

The following conservative assumptions are based on Regulatory Guide 1.25 and inherent plant design parameters used to calculate the activity releases and offsite doses for the postulated fuel handling accident inside Containment.

a) The accident is assumed to occur 48 hours following reactor shutdown for refueling.

b) All rods in one fuel assembly are ruptured.

c) The assembly damaged is assumed the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. A radial peaking factor of 1.65 is used.

d) All of the gap activities in the damaged rods is released and consist of the 10 percent of the total noble gases other than krypton-85, 30 percent of the krypton-85, and 10 percent of the total radioactive iodine in the rods at the time of the accident.

e) The iodine gap inventory is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

f) The refueling cavity water decontamination factors for the inorganic and organic iodine are 133 and 1 respectively, giving an overall effective decontamination factor of 100.

g) The retention of noble gases in the refueling cavity water is negligible.

h) The accident occurs during refueling with the Containment Purge System operating.

i) Containment isolation occurs 20 seconds after detection of the accident with resulting filtration. 37

j) A filter efficiency of 90 percent is used for halogens. 37

The doses from a fuel handling accident occurring inside Containment have been calculated, and have been found to be below the guidelines of 10CFR100. The results of this analysis are presented in Table 15.7.4-4.

TABLE 15.7.4-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Parameter	Design Basis Assumptions	Realistic Assumptions
Source Data:		
Power level, MWt	2900	2775
Radial peaking factor	1.65	1.55
Burnup	3 full-power years	3 full-power years
Decay time, hr	48	48
Number of failed assembly	1	1/17
Fraction of fission product gases contained in the gap region of the fuel rods, percent		
Kr-85	30	30
Other Noble Gases	10	10
Iodine	10	10
Activity Release Data		
Fraction of gap activity released to pool	100	100
Minimum water depth above damaged rods, ft.	23	23
Pool decontamination factor for noble gases	1	1
Pool decontamination factor for iodine		
Inorganic	133	---
Organic	1	---
Overall	100	500
Iodine chemical form released to fuel building		
Inorganic iodine percent	75	75
Organic iodine, percent	25	25
Containment Isolation Time Following ~~detection of~~ the Accident (Sec)	~~27~~ 20	~~27~~ 20
Containment Pre-Entry Purge ~~(cfm)~~ acfm	37,000	37,000
Containment Volume (cu ft.)	2.37×10^6	2.37×10^6

37 ~~29~~

TABLE 15.7.4-3 (Cont'd)

Parameter	Design Basis Assumptions	Realistic Assumptions	
Filter Efficiency			
Iodine, inorganic percent	~~95~~ 90	99	37
Iodine, organic percent	~~95~~ 90	99	~~29~~
Noble gas percent	0	0	
Activity released to atmosphere, (Ci)			

Isotope	Before Isolation	Total*	
I-131	2.8 ~~3.8~~	5.6 ~~3.1~~ $\times 10^{1}$	~~29~~
I-133	3.3 ~~4.5~~ $\times 10^{-1}$	6.6 ~~3.6~~	37
Xe-131m	2.4 ~~3.2~~	4.6 $\times 10^{2}$	
Xe-133	5.4 ~~7.3~~ $\times 10^{2}$	1.0 $\times 10^{5}$	
Xe-135	1.3 ~~1.8~~ $\times 10^{-1}$	2.5 $\times 10^{1}$	
Kr-85	1.4 ~~1.9~~ $\times 10^{1}$	2.6 $\times 10^{3}$	

Parameter	Design Basis Assumptions	Realistic Assumptions
Dispersion Data		
Atmospheric dispersion factors	5 percentile level χ/Qs, (Table 2.3.4-5)	50 percentile level χ/Qs, (Table 2.3.4-5)
Dose Calculation Model	Dose model as discussed in Appendix 15.0A	

* Total is the combination of doses before isolation and after through controlled purge.

TABLE 15.7.4-4

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Result	Design Basis Assumptions Before Isolation	Total*	
Exclusion Area Boundary Dose (0 to 2 hr.) (rem)			
Thyroid	~~1.3~~ 9.5×10^{-1}	1.9 ~~1.0~~ $\times 10^{1}$	37 ~~29~~
Whole body	3.4×10^{-3}	4.7×10^{-1}	
LPZ Outer boundary Dose (duration) (rem)			
Thyroid	2.2 ~~3.0~~ $\times 10^{-1}$	4.3 ~~2.3~~	37 ~~29~~
Whole body	7.6×10^{-4}	1.1×10^{-1}	

* Total is the combination of doses before isolation and after isolation through controlled purge.