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SUBJECT: Forwards responses to draft SER Open Items 355-383,discussed at 830921 meeting w/Auxiliary Sys Branch in Bethesda,MD.

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at [illegible] meeting w/Auxiliary Sys Branch in Bethesda, MD.

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	NRR/DE/CEB 11	1	1	NRR/DE/EHEB	1	1
	NRR/DE/EQB 13	2	2	NRR/DE/GB 28	2	2
	NRR/DE/MEB 18	1	1	NRR/DE/MTEB 17	1	1
	NRR/DE/SAB 24	1	1	NRR/DE/SGEB 25	1	1
	NRR/DHFS/HFEB40	1	1	NRR/DHFS/LQB 32	1	1
	NRR/DHFS/PSRB	1	1	NRR/DL/SSPB	1	0
	NRR/DSI/AEB 26	1	1	NRR/DSI/ASB	1	1
	NRR/DSI/CPB 10	1	1	NRR/DSI/CSB 09	1	1
	NRR/DSI/ICSB 16	1	1	NRR/DSI/METB 12	1	1
	NRR/DSI/PSB 19	1	1	NRR/DSI/RAB 22	1	1
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CP&L

Carolina Power & Light Company

SERIAL: LAP-83-472

October 11, 1983

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

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NOS. 1 AND 2
DOCKET NOS. 50-400 AND 50-401
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Dear Mr. Denton:

Carolina Power & Light Company hereby transmits one original and forty copies of additional information requested by the NRC as part of the safety review of the Shearon Harris Nuclear Power Plant. The enclosed responses relate to the Auxiliary System Branch questions discussed at the September 21, 1983 meeting is Bethesda. These questions are designated as Carolina Power & Light Company Open Item Nos. 355 through 383.

We will be providing responses to other requests for additional information shortly.

Yours very truly,

M. A. McDuffie

M. A. McDuffie
Senior Vice President
Nuclear Generation

FXT/ccc (8156FXT)

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. Travis Payne (KUDZU)
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Shearon Harris Nuclear Power Plant
Draft SER Open Item 355
ASB Question 3.5.1.1

Show that the essential services chilled water system (ESCWS) and waste processing building cooling water system (WPBCWS) are protected against internally generated missiles, or, in the case of the WPBCWS, show that it does not have to be protected.

Response

The following potential sources of internally generated missiles are considered for the Essential Services Chilled Water System:

1) High pressure systems

2) Rotating machinery

3) Gravitational missiles

4) Secondary missiles (those generated from the impact of primary missiles).

As described in FSAR Section 3.5.1.1 and 3.5.1.2, missile generation from the above sources is either not credible or does not affect safety-related equipment in the plant areas where the ESCWS is located.

The Waste Processing Building Cooling Water System is not a nuclear safety-related or Seismic Category I system and is not required for plant safe shutdown, therefore, it does not have to be protected from internally generated missles. There is no adverse safety or radiological impact from the failure of the Non-Nuclear Safety Waste Processing Building Cooling Water System.

FSAR Section 9.2.8.3 will be revised in a future amendment to include the above information on the ESCWS.

(8153SAL)

Shearon Harris Nuclear Power Plant
Draft SER Open Item 356/367
ASB Question 3.6.1 & 9.2.9

Show that the functioning of safety-related systems are not adversely affected by a break in the non-essential services chilled water system.

Show that a break in a NESCWS pipeline as a result of an SSE or other seismic event will not result in reduced functioning of any safety-related equipment to an unacceptable safety level.

Response

As stated in FSAR Section 9.2.9.3, the functioning of safety-related systems are not adversely affected by a postulated failure of the moderate energy non-essential services chilled water system. Refer to FSAR Section 3.6.1 for details on the analysis of dynamic and environmental effects of pipe ruptures, and to FSAR Appendix 3.6A for details on the flooding analysis.

Where it is not feasible or practical to isolate the Seismic Category I piping, the adjacent non-seismic Category I piping was seismically designed in accordance with C.2 and C.4 of Regulatory Guide 1.29. Refer to FSAR Section 3.7.3.13 on the interaction of other piping with Seismic Category I piping.

FSAR Section 3.6.1 will be revised in a future amendment to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 357
ASB Question 5.2.5

Show how flow measurements for identified leakage to the reactor coolant drain tank (RCDT) and pressurizer relief tank (PRT) are capable of detecting leakage with the sensitivity required by the Technical Specifications.

Response

Question 5.2.5 seeks to determine "how flow measurements for identified leakage to the reactor coolant drain tank (RCDT) and the pressurizer relief tank are capable of detecting an increased leakage rate of 10 gpm or less".

These determinations are made by performing an inventory balance on the PRT and the RCDT over a finite period of time. The inventory balance is performed by measuring the increase in level in the PRT and the RCDT. The time interval over which the level increase is measured may be less than or equal to the 72-hour surveillance frequency for identified leakage.

For example, the RCDT is normally maintained at 38 percent of level span by virtue of the automatic drain tank level control system which pumps inleakage to the tank to the boron recycle system. By closing the drain tank level control valve LCV-1003 from the valve control switch, the operator can determine the rate of increase in drain tank level. An inleakage rate of 10 gpm will incrase the indicated level at an approximate rate of 3 percent per minute resulting in a high RCDT level alarm at 75 percent span in ~ 12 minutes.

The pressurizer relief tank level, on the other hand, is manually controlled. An inleakage rate of 10 gpm into the PRT during the 72-hour surveillance interval would increase level from the minimum normal level (~ 64 percent of span) to the high alarm setpoint (83 percent of span) in approximately 3 hours. It is estimated that a monitoring period of about 1 hour would be necessary to detect a quantifiable increase in PRT level at this rate of inleakage.

(8153SAL)

Shearon Harris Nuclear Power Plant
Draft SER Open Item 358
ASB Question 5.4.11

Show that missile fragments resulting from the failure of a rupture disc in the pressurizer relief tank will not adversely affect safety-related equipment.

Response

Missile fragments resulting from the rupture of the Pressurizer Relief Tank rupture disc will not adversely affect any safety-related systems, structures, or components. FSAR Section 5.4.11 will be modified to include the above information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 359
ASB Question 9.1.1

Provide information as to the maximum enrichment of the new fuel to be stored in the new fuel storage facility and a detailed description of the new fuel racks to enable us to make an independent evaluation of K_{eff} in the new fuel storage pools under the most adverse conditions. Describe how control will be maintained over the location of the fuel in the new fuel pools, if this is necessary.

Response

The PWR spent fuel racks will be used to store new fuel at the Shearon Harris plant. The maximum U-235 enrichment is 3.9 for the Westinghouse 17x17 fuel. The rack parameters are shown in Table 1. This table will be included in Section 9.1.1 of the FSAR in a future amendment.

Table 1

SHEARON HARRIS SPENT FUEL RACK DIMENSIONS

Fuel Type: W 17x17

RACK TYPE	POISON
C-C SPACING	10.500
CELL I.D.	8.750
POISON CAVITY	0.090
POISON WIDTH	7.500
CELL GAP (NOMINAL)	1.330
POISON THICKNESS	0.075
WALL THICKNESS	0.075
WRAPPER THICKNESS	0.035
POISON (GM-B10/SQ.CM)	0.020

All Dimensions in Inches

Shearon Harris Nuclear Power Plant
Draft SER Open Item 360
ASB Question 9.1.2

In order to permit us to evaluate K_{eff} for spent fuel pools, provide further information regarding all fuel to be stored, fuel distribution within the pool(s), a detailed description of the storage racks, a description of the calculational methods used in the determination of K_{eff} by CP&L toegther with a discussion of the calculation and mechanical uncertainties considered in the calculation. Describe how control will be maintained over the location of the fuel in the spent fuel pools.

Response

The maximum U-235 enrichment is 3.9 for the Westinghouse 17x17 fuel. The rack parameters are shown in Table 1. This table will be included in FSAR Section 9.1.2 in a future amendment.

Additional information addressing criticality analyses for the Shearon Harris spent fuel racks will be submitted under separate cover due to its proprietary nature.

TABLE 1

SHEARON HARRIS SPENT FUEL RACK DIMENSIONS

Fuel Type: W 17x17 and GE 8x8

RACK TYPE	POISON	BWR
C-C SPACING	10.500	6.250
CELL I.D.	8.750	6.050
POISON CAVITY	0.090	0.060
POISON WIDTH	7.500	5.100
CELL GAP (NOMINAL)	1.330	---
POISON THICKNESS	0.075	0.045
WALL THICKNESS	0.075	0.075
WRAPPER THICKNESS	0.035	0.035
POISON (GM-B10/SQ.CM)	0.020	0.0103

All Dimensions in Inches

Shearon Harris Nuclear Power Plant
Draft SER Open Item 361(1)
ASB Question 9.1.3(1)

Commit to the installation of both fuel pool cooling pumps and heat exchanger trains in Unit 2 before placing the Unit 2 fuel pools cooling system in operation.

Response

There are two cooling pumps and heat exchanger trains for Unit 1 and two cooling pumps and heat exchanger trains for Unit 2. There will be no heat load placed in Unit 2 fuel pool during the construction of Unit 2.

The fuel pool cooling pumps and heat exchanger trains will be installed before the Unit 2 fuel pools cooling system is placed into operation.

FSAR Section 1.2.3 will be revised in a future amendment to the FSAR to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 361(2)
ASB Question 9.1.3(2)

Show that a Seismic Category I water source can be placed into operation in the event of failure of the RWST as a source of makeup, in sufficient time to prevent boiling or uncovering of spent fuel in the Unit 1 spent fuel pool while Unit 2 is under construction.

Response

The emergency service water system will be the Seismic Category I water source that will be available to supply emergency make-up water to the fuel pools cooling system. The ESW will be an emergency backup source of water for.makeup that can be connected to the fuel pool cooling system in sufficient time to prevent boiling or uncovering of spent fuel. It will take approximately five hours to reach boiling in the pools with maximum load and no heat removal. As discussed in FSAR Section 9.1.3.2 and shown on Table 9.1.3-2, time to reach 150°F is one hour and thirty-three minutes with maximum load and both cooling loops operating.

Sufficient emergency backup make-up water can be provided to replace water lost from evaporation and normal leakage, through the use of flexible hoses in conjunction with emergency service water piping; RWST piping and appropriate connections. The RWST shall be isolated from the fuel pool cooling system when using the ESW for emergency makeup. The connection to be used is an emergency connection on both trains 1A-SA and 1B-SB fuel pool cooling system (FHB elevation 236.00 feet) and ESW lines in the Unit 1 and Unit 2 common RAB area elevation 236.00 feet.

FSAR Sections 1.2.3 and 9.1.3 will be revised in a future amendment to include the above information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 362
ASB Question 9.1.4

Verify that loads equal in weight or lighter than a fuel assembly and its handling tool will not be lifted high enough above spent fuel so that the kinetic energy of these loads will exceed that of a fuel assembly and its handling tool.

Response

NUREG-0800, Section 9.1.4 Acceptance Criterion 5 requires that, "The maximum potential kinetic energy capable of being developed by any load handled above the stored fuel, if dropped, is not to exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel storage racks".

Analysis performed by Westinghouse showed that the maximum kinetic energy that can be developed by the BPRA tool is 6677 ft. lbs. while that developed by a fuel assembly and its handling tool is only 4961 ft. lbs.

Analysis of potential fuel damage due to this situation was performed by Westinghouse. This analysis showed that although the kinetic energy for the dropped handling tool is 35 percent greater than the kinetic energy for a combined fuel assembly and tool drop accident, the latter case is more limiting from a fuel rod damage potential. In previous accident analyses it was assumed that the dropped fuel assembly fractures a number of fuel rods in the impacted (stationary) assembly and subsequently falls over and ruptures the remaining rods in the dropped assembly. In the case of a dropped tool accident, it is postulated that the handling tool directly impacts a stationary fuel assembly which can cause fuel rods to be fractured in the impacted assembly. However, no additional fuel rods are fractured due to the tool fallover after impact.

The analytical procedure for assessing fuel damage is to conservatively assume that the total kinetic energy of the dropped assembly is converted to fuel clad impact fracture energy. The energy required to break a fuel rod in compression is estimated to be 90 ft. lbs. If the total kinetic energy for the dropped tool, 6677 ft. lbs., is absorbed by fracturing the fuel rod, a total of 74 fuel rods would be broken.

This value is substantially less than the number of fuel rods that could be potentially fractured by a dropped fuel assembly and subsequent fallover. Based on this analysis, it is concluded that the dropped tool accident is not limiting.

Following this analysis, we analyzed the potential for damage to the fuel racks. Five different locations on the top of a standard PWR poison rack assembly were analyzed for straight drop BPRA impact.

In addition, the effect of dropping the BPRA tool at an angle such that it ended up lengthwise on the top of the rack was analyzed. However, since the energy is applied to a larger number of cells during the inclined drop, the damage to an individual cell is not as great as that of a straight drop.

The different scenarios analyzed indicate that it may be possible for the cell to drop $1/2$-inch to the base or deflect laterally as much as .459-inch. It is possible that the cells located in the drop zone may be damaged enough to obstruct the insertion or removal of fuel. However, in no case does the fuel rack grid structure fail nor is the poison material damaged. Thus, an increase in reactivity between adjacent cells is not considered likely. This is also supported by the fact that the soluble boron in the pool water counteracts any postulated reactivity increase.

Thus, it has been demonstrated that this situation would have no adverse safety impact on the Shearon Harris stored fuel.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 363(2)
ASB Question 9.1.5(2)

Commit to completing Phase I of NUREG-0612 prior to issuance of an operating license.

Response

Phase I of NUREG-0612 is the submittal of information requested by Section 2.1 of Enclosure 3 to NRC's 12/22/80 letter to licensees and OL applicants. Carolina Power & Light Company's response to Section 2.1 was submitted to the NRC on 6/26/81. This submittal did not include the following items:

a. Item 2.1-3(a) - Safe load paths for heavy-load-handling systems.

b. Item 2.1-3(b) - Discussion of measures taken to ensure that load-handling operations remain within safe load paths, including procedures, if any, for deviations from these paths.

c. Item 2.1-3(c) - Complete tabulation of heavy loads and their weights and verification that handling of such loads is governed by written procedures containing, as a minimum, the information identified in NUREG-0612, Section 5.1.1(2).

d. Item 2.1-3(d) - Verification that lifting devices identified in the response to Item 2.1-3(c) comply with the requirements of ANSI N14.6-1978 or ANSI B30.9-1971 as appropriate.

e. Item 2.1-3(e) - Documentation of exceptions taken to the recommendations of NUREG-0612 and ANSI B30.2 for inspection, testing, and maintenance procedures for heavy-load-lifting devices.

f. Item 2.1-3(g) - Identification of deviations from the recommendations of ANSI B30.2 or NUREG-0612 for procedures for training qualification and conduct of crane operators.

These items will be prepared prior to fuel load and the appropriate discussion or documentation requested by the NRC will be submitted six months prior to fuel load. The current schedule for this submittal is January 1985. The results of the NRC review will be addressed prior to fuel load.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 363(3)
ASB Question 9.1.5(3)

Commit to completing Phase II of NUREG-0612 prior to the end of the first refueling.

Response

Phase II of NUREG-0612 is the submittal of information requested by Sections 2.2 - 2.4 of Enclosure 3 to NRC's 12/22/80 letter to licensees and OL applicants. Carolina Power & Light Company's response to Sections 2.2 - 2.4 was submitted to the NRC on 9/23/81. This submittal did not include the following items:

a. Item 2.2-3.4.6 - Verification that lifting rigs for the Fuel Handling Building Auxiliary Crane comply with the requirements of ANSI N14.6-1978 or ANSI B30.9-1971 as appropriate.

b. Item 2.2-3.5 - Detailed information on interfacing lift points for the FHB auxiliary crane.

c. Item 2.2-4(b), Item 2.3-4(a), and Item 2.3-4(b) - Identification of deviations from the recommendations of NUREG-0612 for procedures for the removal or bypassing of crane interlocks.

d. Item 2.4-2(b) - Identification of deviations from the recommendations of NUREG-0612 for operations of the jib crane.

These items will be prepared during the first cycle of operation and appropriate documentation submitted to the NRC six months prior to the scheduled end of the first refueling outage. The results of the NRC review will be addressed prior to the end of the first refueling outage.

This response is in conformance with the NRC's schedular request for resolution of this open item.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 363
ASB Question 9.1.5 (4)

Show that a loss of one phase of a 3-phase electrical system or phase reversal will not result in dropping of a heavy load from any crane intended for such use.

NRC Clarification

"For single failure proof cranes, a failure modes and effects analysis of a single failure in the crane electrical system is required."

Response

The auxiliary crane, a single failure proof crane, is fed from a 3-pole circuit breaker located in a motor control center. With this type of scheme, loss of one phase on the power cable to the cranes is not feasible due to the nature of the circuit protective devices. If an overload or short circuit exists on the feed to the cranes, the circuit breaker would open all three phases.

Reversal of two phases is not a credible event at the power source since the power cables are connected directly to a circuit breaker. This event would require two phases to become disconnected and reverse connection. This would be an incredible event. The SHNPP power system design precludes loss of a single phase or reversal of any two phases on the power feeds to the plant crane systems.

Kranco, the manufacturer of the auxiliary crane, stated that in the event of a phase loss before drive operations the crane drives cannot operate. In the event of a phase loss while the hoist is operating, the overspeed switch will disconnect the drive automatically at 140% of drive rated speed to set the holding brake and stop the load.

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.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 365
ASB Question 9.2.2(1)

Show that all nonsafety-related heat loads in the component cooling water (CCW) system are isolated from safety-related loads in the event of suitable emergency initiating signals.

Response

The nonsafety-related heat loads [i.e., sample heat exchangers and gross failed fuel detector (Item Nos. (k) and (l) of FSAR Section 9.2.2.2-1)] are isolated from safety-related heat loads automatically on Safety Injection "S" initiating signal. The air operated valves 3CC-D547SA-1 and 3CC-D548SB-1 (See revised FSAR Fig. 9.2.2-1) and 3CC-L1SA-1 and 3CC-L2SB-1 (FSAR Fig. 9.2.2-4) provided on the inlet lines to sample panel and gross failed fuel detector system respectively shall close on an "S" signal thus isolating CCW supply to nonsafety-related heat loads. Two check valves, in series, are provided on the outlet lines of each of the nonsafety-related heat load.

The non-essential safety-related heat loads (Item Nos. a, b, c, d, e, h, i, and j of FSAR Section 9.2.2.2.1) are remote manually isolated from the control room by closing four motor operated butterfly valves (two on upstream of the CCW pump suction header and two downstream of the CCW heat exchanger header as shown on FSAR Fig. 9.2.2-1). It is not necessary that these valves be closed during the injection phase of a LOCA. Only the following heat sources are considered as essential heat loads which will receive CCW supply:

1) Residual heat removal (RHR) pumps
2) RHR heat exchangers
3) Spent fuel pool heat exchangers (long-term cooling)

FSAR Section 9.2.2 will be revised in a future amendment to include the above information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 366(1)
ASB Question 9.2.8-1

Show that the ESCWS supplies enough water to the safety related components to cool them in the event of an accident coupled with the most adverse single failure.

Response

Figure 9.2.8-2 indicates that the maximum refrigeration load on the ESCWS occurs at Unit 1 during refueling operations. Post-accident refrigeration loads are considerably lower. Refrigeration load is calculated to exceed the ESCWS design capacity by less than 1 percent during refueling; however, there is a 10 percent margin of safety included in the refrigeration loads shown. The design capacity of the ESCWS is, therefore, adequate for the expected maximum refrigeration loads, and there is still greater margin to accommodate accident loads.

In the event of failure in a single train of the ESCWS during an accident, a redundant 100 percent capacity system would still be available. System redundancy is clearly shown on Figure 9.2.8-1 and the ESCWS design capacity is given in Table 9.2.8-1.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 366(3)
ASB Question 9.2.8-3

Show how the ESCWS has been designed to permit functional and pressure testing.

Response

Pressure testing of the ESCWS can be accomplished by supplying pressure at any convenient valved outlet such as expansion tank, pump inlet or discharge piping etc. shown on Figure 9.2.8-3. Pressure readings can also be taken at those points.

ESCWS are also equipped, at the main chilled water supply source, with orifice type measuring devices which will confirm that the required flow is maintainable. A typical device appears on Figure 9.2.8-3.

In addition each cooling coil circuit is equipped with a flow measuring device which will be used to assure that flows to each coil are adjusted and balanced to the flow rates indicated in the schedules on Figure 9.2.8-3.

FSAR Section 9.2.8-4 will be revised in a future amendment to reflect the above information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 368
ASB Question 9.2.10

Show how radioactive fluids are prevented from entering the service water system in the event of a leak into the Waste Processing Building Cooling Water System.

Response

The only point of probable leakage between the service water system and the WPB cooling water system is the WPB cooling water heat exchanger. At normal operation the service water pressure on the tube side is higher than the WPB cooling water pressure on the shell side. Any rupture or opening on the tubes would result in leakage of the service water system into the WPB cooling water system due to its higher pressure. The service water system will be in operation at all times during operation of the WPB cooling water system. FSAR Section 9.2.10 will be revised in a future amendment to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 138/369
ASB Question 9.3.1

Commit to periodic testing of instrument air quality in accordance with the requirements of ANSI MC 11.101976 (ISA-S7.3).

Response

The design of the instrument air system has been changed such that three micron air filters will be utilized. This commitment was transmitted to the NRC on July 15, 1983.

The testing on the instrument air system will be performed at least once every refueling outage. The testing will be done in accordance with ANSI MC 11.1-1976. Under normal operating conditions the acceptance criteria will be less than 1 ppm w/w or v/v. With regard to particulates in the instrument air system, the three micron particle limit may not be utilized as the acceptance criteria. Since the SHNPP instrument/service air system lines are constructed of carbon steel and the air velocities are relatively low, it is expected that some limits in ANSI MC 11.1-1976 will be difficult to meet, particularly the three micron particle limit at utilization points. Based on startup and operational tests of the instrument air system, CP&L will perform evaluations of deviations from the acceptance criteria of the ANSI standard. These evaluations will include the safety implications, and accordingly recommend appropriate corrective action.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 370(1)
ASB Question 9.4.1(1)

The applicant should clarify the operation sequences for the CRACS after a safety injection actuation signal, chlorine signal or high radiation signal and should justify each sequence on the basis of the operator's safety and ability to continue to maintain the plant in a safe condition.

Response

The operation of the CRACS is as discussed in Sections 9.4.1 and 6.4 except Subsection 6.4.3. After a high radiation signal has automatically isolated the CRACS, the operator will monitor the CRACS air intake radiation detectors and select the emergency air intake from which to draw the least radioactive make-up air.

FSAR Section 6.4.3 will be revised in a future amendment to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 370(2)
ASB Quetion 9.4.1(2)

Show how the control room area air conditioned system (CRACS) complies with the guidelines of Positions C.7 and C.14 of Regulatory Guide 1.78.

Response

Redundant detectors are provided in the Control Room outside air intakes and in the chlorine storage area in order to initiate automatic isolation of the Control Room air intakes. The chlorine detectors will be able to detect chlorine in the range of 1 to 10 ppm. Although the exact setpoint at which the chlorine detectors will initiate isolation has not yet been established, it is anticipated that the setpoint for the detectors in the Control Room outside air intakes and in the chlorine storage area will be approximately 1 ppm and 2 to 3 ppm, respectively. Further information indicating compliance with Position C.7 of Regulatory Guide 1.78 including infiltration, make-up recirculation and Control Room volume may be found in FSAR Subsections 6.4.1, 6.4.2, and 9.4.1.

Information indicating compliance with Postion C.14 of Regulatory Guide 1.78 (Rev. 0) including isolation systems, filtration equipment, air supply equipment and protective clothing may be found in FSAR Subsections 6.4.1, 6.4.2, and 9.4.1

FSAR Section 2.2.3.3 and 9.4.1.2.2 will be revised in a future amendment to include this information.

(8153SAL)

Shearon Harris Nuclear Power Plant
Draft SER Open Item 370(3)
ASB Question 9.4.1(3)

Show how the CRACS complies with the guidelines of Positions C.4.a and C.4.d of Regulatory Guide 1.95.

Response

In compliance with Position C.4.a of Regulatory Guide 1.95 (Rev. 1) the emergency filtration trains will start and operate upon receipt of a high chlorine concentration signal.

The self-contained breathing equipment available for use onsite is discussed in FSAR Section 9.5.1.2.3(c). This section also describes the backups available, specifically two extra air bottles per breathing unit.

Figure 9.4.1-1 will be revised to show that the control room HVAC design meets the regulatory requirements to include the purge make-up system which is designed with isolation valves in series.

(8153SAL)

Shearon Harris Nuclear Power Plant
Draft SER Open Item 370(4)
ASB Question 9.4.1(4)

Show that the CRACS can maintain suitable control room conditions under the most adverse conditions (including failures of heating and cooling systems such that it provides extra heating in summer or extra cooling in the winter) in order to show compliance with General Design Criterion 4.

NRC Clarification

Discuss the possibility of a control system failure which would provide inadequate heating or cooling of the control room.

Response

The CRACS is designed to ensure that any postulated single failure will not adversely affect the capability of the system to satisfy its design objectives as stated in FSAR Section 9.4.1.

The ventilation system has sufficient redundancy to preclude inadequate heating or cooling as described in Section 7.3.1.5.7 and as shown on Figure 7.3.1-17 and Table 9.4.1-4.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 370(5)
ASB Question 9.4.1(5)

Demonstrate that the control room will not be adversely affected by a pipe crack or break in an adjacent area which pressurizes that area, especially if the broken line is used to convey radioactive fluids. Include the most adverse single failure.

Response

The high energy piping systems outside containment are listed and described in Section 3.6A.2. Those listed systems are the CVCS, SGBS, FWS and the AFS. A discussion of the analysis is provided for those systems in Section 3.6A.2 that demonstrate that the control room will not be adversely affected by a pipe crack or break in an adjacent area. Figure 3.6A-2 shows the only high energy piping adjacent to the control room with indication of break locations and jet impingement envelopes.

The steam lines near the control room (elevation 305 feet) are located in the turbine building at elevation 286.00 feet and separated by the Category I wall of the RAB. The control room, including outside air intakes for control room ventilation, will not be adversely affected by any high energy pipe breaks. The main steam line may be slightly radioactive but as discussed in Sections 15.1.5, 6.4.4, and 9.4.1 radiological impact would be limited to below acceptable levels by ventilation systems and wall shielding. Systems are designed to include the effects of the most adverse single failure.

FSAR Figures 1.2.2-35 and 1.2.2-68 show the control and indicate the location of the main steam lines.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 371
ASB Question 9.4.5

Provide assurance that all air intakes to the diesel generator room (whether indirect or direct) will comply with recommendations A.2 and C.1 of NUREG/CR-0660.

NRC Clarification from the September 21, 1983 meeting

Commit to visually inspect the air filters for the diesel generator building HVAC system at least one per month.

Response

The cited recommendations from NUREG/CR-0660 address features of the design of the diesel generator building HVAC system and dust protection for electrical equipment. The recommendations do not address surveillance intervals for the air filters on this HVAC system.

In an earlier response to the NRC, (refer to the response to Open Item 103 submitted to the NRC on June 29, 1983), CP&L committed to develop and implement a detailed preventive maintenance program for the diesel generator. The inspections of the diesel generator building air intake filters (AH-99 and AH-100, refer to FSAR Figure 9.4.5-2) will be included in that program. The inspection will be at monthly intervals (refer to Technical Specifications for definition of monthly surveillance interval).

Shearon Harris Nuclear Power Plant
Draft SER Open Items 372 And 373
ASB Question 10.4.9(1),(2)

(1) Provide assurance that at least 475 gpm will be provided to the steam generators by the AFW system after a loss-of-feedwater (LOFW) event.

(2) Clarify the discrepancy between Table 10.4.9-1 of the FSAR which shows the capacity of the AFW pumps to be 450 gpm while the narrative states the pump capacities to be 475 gpm.

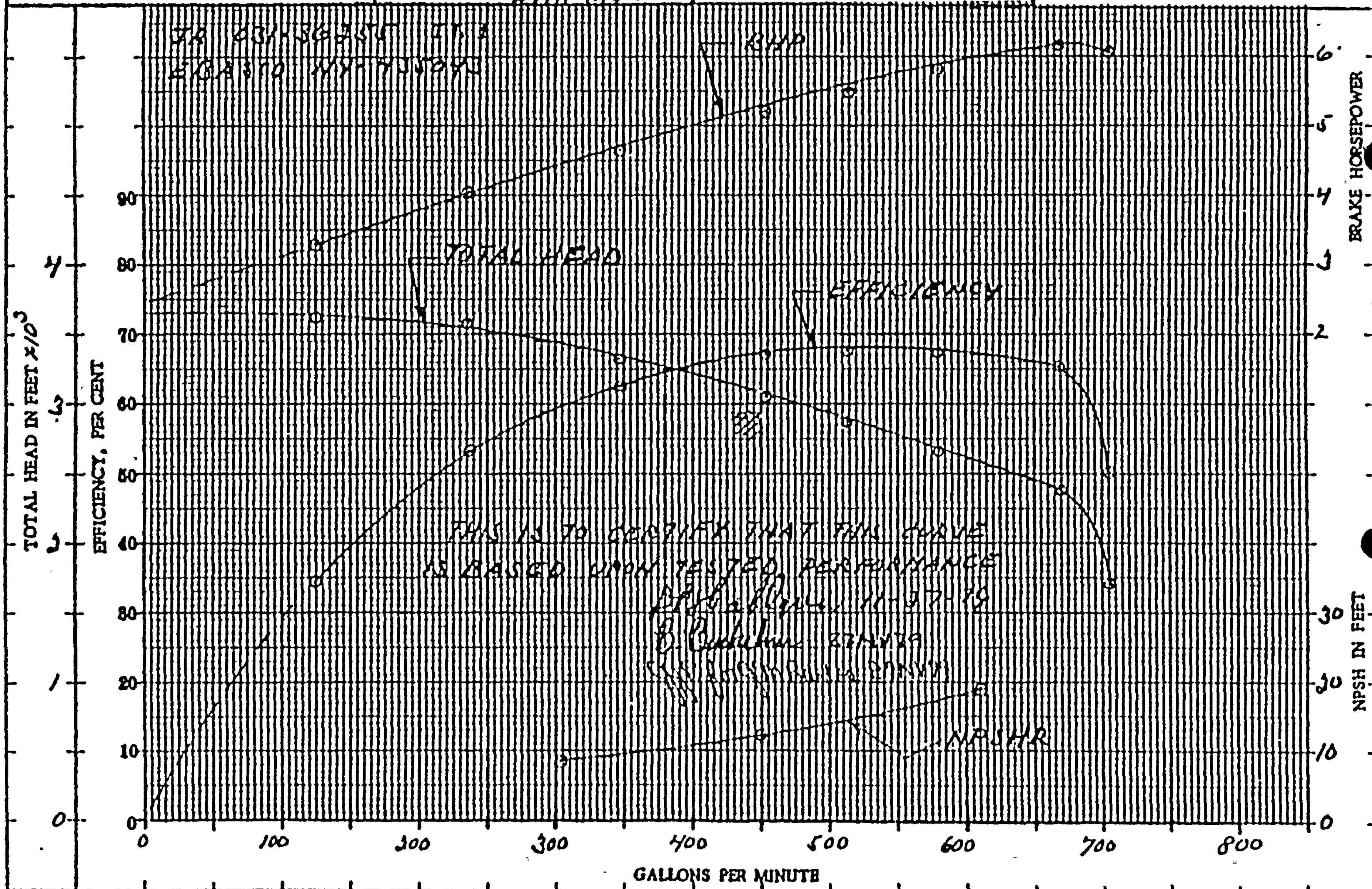
Response

As shown on the attached AFW motor driven pump curve No. 1212, Rev. 0 the pump design condition is 450 gpm but the capacity to deliver 475 gpm is well within the pump capacity.

FSAR Table 10.4.9-1 lists the correct AFW motor driven pump design capacity at 450 gpm. However, as shown on the attached pump curve the pump can deliver the 475 gpm against the pressure relief valve setpoint pressure in the steam generator(s).

(8153SAL)

CUSTOMER EBASCO/CP&L PROPOSAL NO. ITEM SPECIAL NOTES: SPARE ROTOR 79P90990 TESTED WITH PUMP SN 027958	DESIGN CONDITIONS GPM 450 EFF 69.5% T. H. (FT.) 2940 BHP 481 SG. 1.0 RPM 3550 DRIVER HP 500 HP MOTOR WITH 1.15 SF	Ingersoll-Rand DRAWN BY D.P.W DATE 11-27-79	CURVE N-1212 REV. 0 PUMP 3HMTA-9 Curves are approximate. Pump is guaranteed for one set of conditions. Capacity, head and efficiency guarantees are based on shop test and when handling clear, cold, fresh water at a temperature of not over 85 degrees and not over 15' suction lift.



CURVE N-1212, REV. 0

PAGE 1 OF 5

Shearon Harris Nuclear Power Plant
Draft SER Open Item 375
ASB Question 10.4.9(4)

Evaluate the possibility of water hammer because of trapped air at high points.

Response

Water hammer in the auxiliary feedwater system (AFWS) is minimized by designing the system to remain full of water as discussed in Section 10.4.9.3. The suction piping to the AFW pumps and part of the discharge piping are always maintained under a positive head of water due to the higher elevation of the CST. The discharge piping from the steam generator nozzle to the first check valve is pressurized to steam generator pressure. Feedwater system pressure, CST static head pressure and leakage across pumps and valving will maintain a water solid system on the AFWS pump discharge side. Vents are provided at all high points. Frequent system testing and inspections as described in Section 16.2 will ensure that AFWS is full of water. Vent lines are opened prior to start-up to vent any trapped air.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 376
ASB Question 10.4.9(5)

Provide procedures for providing water to the AFW pumps for an emergency startup in the event the CST is unavailable.

Response

The Condensate Storage Tank (CST) is described in FSAR Section 10.4.9. This facility is provided with tornado missile protection as described in FSAR Table 3.5.1-2. No single active failure can preclude the availability of the CST. The CST is relied upon as a source of water to the auxiliary feedwater system during accidents; the minimum capacity of the CST is 240,000 gallons. The alternative to the CST when it is depleted is the emergency service water system. Switchover to this system is performed manually from the main control board or the auxiliary control board. Operating procedures for the AFW will identify the point at which switchover to the ESW should occur. Instrumentation available to monitor CST level are identified in FSAR Section 10.4.9.5.

Based on the above features, SHNPP concludes that adequate provisions have been made to maintain a water supply to the Auxiliary Feedwater System.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 377
ASB Question 10.4.9(6)

Provide procedures for valve alignment checks after testing in addition to those done after operating and maintenance.

Response

The surveillance procedures for all operational surveillance tests will specify the valve lineups to restore the system to operation or standby status, as appropriate. The valve lineup will require double independent verification of the lineup. This practice is also followed to restore the system to service after maintenance. This question is also covered in our response to Open Item 225(3), Sub Item 5.2-6 submitted to the NRC staff in a letter dated July 15, 1983.

(8042NECccc)

Shearon Harris Nuclear Power Plant
Dfaft SER Open Item 379
ASB Question 10.4.9(8)

Provide assurance that there are redundant alarms to indicate low level in the CST; the operator must have at least 20 minutes following the alarm to switch to a new supply.

Response

Two (2) redundant safety grade level transmitters (LT-1CE-9010A-SA, LT-1CE-9010B-SB) are provided on the CST for tank inventory monitoring. There are two separate tank taps and tubing for transmitter monitoring. Both level transmitters provide a signal via train A and train B to separate safety grade level indicators. The redundant level indicators provide indication to operators at the MCB and ACP of CST water inventory. Also each level transmitter provides a signal to redundant CPU's. Each CPU, on a separate train has alarm and indication of tank water inventory.

Also, the level transmitter for train B provides a signal to six (6) level switches which in turn provide the following control functions:

1. one High level - Controls tank make-up, no alarm
2. one Low level - Controls tank make-up, no alarm
3. one High-High - Tank overflow, alarm
4. one Low-Min - Indicates minimum level to meet tech spec. quantity of water to meet accident require-ments, alarm
5. one Low-Low - Indicates approach to minimum water level for accident, alarm
6. one Empty - Indicates approach to water depletion, and allows operator twenty (20) minutes to switch to alternate water supply, alarm.

The level switches provide four (4) alarms and indication on the MCB and ACP. System will fail in the alarm mode.

FSAR Sections 9.2.6.5 and 10.4.9.2.2 will be revised in a future amendment to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 380
ASB Question 10.4.9(9)

The applicant should provide a means for protecting the AFW pumps from the possible failure of the single valve in the single supply line from the CST.

Response

As shown on Figures 7.3.1-9 and 7.3.1-10, the AFW pumps have an alarm for low pump suction, and a pump trip and alarm on low-low pump suction. Also included are low pump discharge pressure alarms.

FSAR Sections 7.3.1.3.3 and 10.4.9.5 will be revised in a future amendment to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 381
ASB Question 10.4.9(10)

The applicant should verify that a break in the line deliverng steam to the turbine of the AFW turbine-driven pump will still permit the AFW system to deliver the necessary coolant flow to the undamaged steam generator(s) when coupled with the most adverse single failure.

Response

The AFWS design as described in Section 10.4.9 ensures that there is no initiating failure and assumed single failure that will render all three AFW pumps and associated systems as unavailable in providing the necessary coolant flow to the appropriate steam generator(s).

The turbine driven AFW pump steam supply line downstream of the normally closed steam supply valves is classified as a moderate energy line under the 2% rule described in SRP 3.6.2, BTP MEB 3-1. The motor driven AFW pumps and its associated systems will be the system used for AFW during startup, hot standby or shutdown. The turbine driven pump and its associated piping up to the normally closed supply valves are not used during normal operation, startup, hot standby or normal shutdown. Also, as discussed in Section 10.4.9.3 the steam supply piping to the AF pump turbine is sloped toward the turbine in order to avoid collection of condensate and thereby prevent damage to the piping system due to water slugging effects.

FSAR Section 10.4.9.3 will be revised in a future amendment to include this information.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 382
ASB Question 9.4.1

Show that the moveable barrier between the cask pool and the northern new fuel pool will not damage spent fuel or safety-related equipment as a result of a seismic event.

Response

The removeable barrier is designed to withstand DBE seismic loads in accordance with Positions C.2 and C.4, Regulatory Guide 1.29.

FSAR Section 9.1.4 will be revised in a future amendment to include this statement.

(8153SAL)

Shearon Harris Nuclear Power Plant
Draft SER Open Item 383
ASB Question 3.5.1

State whether or not missiles generated by the failure of fan components in HVAC equipment were considered in the analysis of internally-generated missiles.

Response

Specifications for in-line, axial and centrifugal fans for use at SHNPP explicitly require that material gage and fan housing design be shown to be sufficient to withstand equipment-generated missile penetration at the maximum operating condition to which it can be field-adjusted. Missiles generated by failure of fan components, therefore, need not be further evaluated for their effects. FSAR Section 3.5.1.1.2 will be revised in a future amendment to reflect this information.