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Waste Awareness and Reduction Network
NC WARN

Sept. 20, 1999

Russell Powell, Chief
FOIA-LPDR Branch
Division of Freedom of Information and Publication Services Office of Administration
U.S. Nuclear Regulatory Commission Washington, DC 20036

FOIA/PA REQUEST
Case No: 99-367
Date Rec'd: 9-21-99
Action Off: Pugh
Related Case:

BY FAX: (301) 415-5130

Dear Mr. Powell:

On behalf of North Carolina Waste Awareness and Reduction Network (NC-WARN), and pursuant to the Freedom of Information Act, 5 U.S.C. 552(b), et. seq., I hereby request that you make available copies of all documents in the U.S. Nuclear Regulatory Commission's possession which describe or discuss:

1) Actual or projected heat loads and/or heat production from spent fuel assemblies and/or any other source, in spent fuel pools at the Shearon Harris reactor site, including any calculations and/or data of any kind (including sources of such data and all references used) used to calculate or estimate such loads, explicitly including any documents or information whatsoever relating to any of the estimated heat loads and/or the "maximum CCW Capability without system upgrades" shown on the viewgraph "Projected CCW Heat Loads" (apparently paginated as "10" and dated 1/2/98 in Carolina Power & Light's 7/16/98 "10 CFR 50.55a Alternative Plan for Harris Spent Fuel Pool [sic] 'C' and 'D' " a set of viewgraphs from a CP&L/NRC Staff meeting on 7/16/98).
2) actual or projected heat removal capacity from any and/or all spent fuel pools at the Shearon Harris reactor site near Bonsal NC.
3) actual RHR, Service Water System, and/or CCW heat removal capacity under operating, shutdown or accident conditions at Harris, or any calculations or projections of any and/or all of these, including any of the calculations, numbers, charts, or projections of RHR, SWS and/or CCW system heat removal capacity and/or heat transfer capability, including any and all supporting calculations, assumptions, referenced documents, information or procedures for calculating any and/or all of the above, including all such information relating to the 1999 CP&L heat removal calculations presented or referred to at the May 9, 1999 Prehearing Conference of the ASLB held in Chapel Hill NC, and/or any other calculations or projections of any of the above performed by NRC, CP&L or anyone else, the dates thereof, and/or any criteria or procedures or memoranda related to such calculation(s), projection(s) and/or estimate(s)..
4) any methodology used, or rejected for use, in any of the above calculations, projections or estimates; and any documents related to such methodology and/or decision(s) concerning what methodology is or was used, or is not or was not used.

This request covers but is not limited to all draft and final reports, correspondence, viewgraphs (vu-graphs, etc.) or copies thereof, memoranda, notes, records of telephone contacts, electronic communications including fax transmissions and Email, or other written records, whether in paper or computer files including CDs.

Pursuant to our request, please provide all documents and communications prepared or utilized by, in the possession of or routed through the NRC related to the above items 1-4 above.

For any portion of the request that you deem appropriate to deny, NC-WARN requests that you describe the information that is denied, identify the exception to the FOIA on which you rely, and explain how that exception applies to the withheld information.

Pursuant to NRC regulations at 10 CFR 9.41, NC-WARN requests that any searching and copying fees incurred as a result of this search be waived, and provides the following information in response to the eight criteria listed in Section 9.41(b):

1) **Purpose of request:**

The purpose of the request is to gather information on the heat loads from spent fuel stored, and/or proposed to be stored, at the Harris nuclear plant site, and the ability to meet those heat loads and how such ability was and/or has been calculated, including the reasons for any re-calculations and all methods and data used.

The requested information is currently not available in the NRC's Public Document Room.

2) **Extent to which NC-WARN will extract and analyze the substantive content of the records:**

NC-WARN is qualified to make use of the requested information. Our staff and cooperating experts have demonstrated the ability to interpret information and communicate that information in a form comprehensible to the general public. Members of NC-WARN have published articles in news media of general circulation in North Carolina including the Raleigh News & Observer, Chapel Hill (NC) Herald, . NC-WARN is quoted as a reliable source of information on nuclear issues in newspapers and on radio and television across the North Carolina. NC-WARN was a key source of information to the public concerning low-level radioactive waste issues during the State of North Carolina's now-withdrawn membership in the Southeast LLRW Compact. NC-WARN was first to inform the general public of plans for a significant expansion of spent nuclear powerplant fuel storage proposed on 1998 for the Shearon Harris nuclear plant near Apex NC.

NC-WARN has a working relationship with attorneys, physicists, nuclear engineers, medical doctors, and other respected professionals who contribute to the full understanding of technical records.

3) Nature of the specific activity or research in which the records will be used and NC-WARN's qualifications to utilize the information for the intended use in such a way that it will contribute to public understanding:

NC-WARN seeks the requested information solely to contribute to and help shape the public debate on adequate worker and public health and safety. NC-WARN intends to use the information in order to advance the concerns for public understanding and safety.

4) Likely impact on the public's understanding of the subject as compared to the level of understanding of the subject prior to disclosure:

Since complete information on the above-referenced items is not available to the general public, NC-WARN will be able to provide the first comprehensive review of this information, which can impact public health and safety.

5) Size and nature of the public to whose understanding a contribution will be made:

NC-WARN's information is regularly reported in news media reaching millions of citizens in North Carolina ranging from Winston-Salem, Greensboro, Apex, Chapel Hill, Pittsboro, Raleigh, Fayetteville and Charlotte for examples, including over 3 million people and including the general public as well as public officials and specialists in health and safety related issues.

6) Means of distribution of the requested information:

NC-WARN provides information via reports, news releases, press conferences, newsletters, e-mail and other means. NC-WARN has been a key provider of information on nuclear, toxics and other health- and safety-related issues since 1990.

7) Whether free access to information will be provided:

NC-WARN will provide access to information received under this request freely.

8) No commercial interest by NC-WARN or any other party:

NC-WARN is a nonprofit organization (501(c)3) and has zero commercial interest, nor is party to any commercial interest in the above-requested information. To NC-WARN's best knowledge, no other commercial interest is involved with this request.

Sincerely,

Jim Warren

Jim Warren
Executive Director

9/20/99

cc: Sen. John Edwards
Rep. David Price
Rep. Bobby Etheridge
Rep. Eva Clayton
Orange County Board of Commissioners
Diane Curran

10CFR50.55a Alternative Plan for Harris Spent Fuel Pool ‘C’ and ‘D’

July 16, 1998

CP&L

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

The new fuel pool is designed for the storage of both new and spent fuel. Consequently, it is designed for both wet and dry storage. The maximum storage capacity of this pool is 480 PWR fuel assemblies, which is more than 3 cores. The fuel is stored in 6x10 PWR rack modules, which are designed for underwater removal and installation. The new fuel storage racks are of identical design to the spent fuel storage racks and can be used both wet and dry.

In the event additional space is needed for the storage of spent fuel from other nuclear plants in the CP&L system, the new fuel pool is designed for the storage of both PWR and BWR fuel. Spent BWR fuel will be stored in 11x11 BWR rack modules which are designed for underwater removal and installation.

The fuel racks consist of individual vertical cells fastened together through top and bottom supporting grid structures to form integral modules. A neutron absorbing material is encapsulated into the stainless steel walls of each storage cell. The PWR rack modules have a center-to-center spacing of 10.5 inches between cells. The BWR rack modules have a center-to-center spacing of 6.25 inches between cells. These free-standing, self-supporting modules are sufficient to maintain a subcritical array even in the event the fuel pool is flooded with unborated water. Table 9.1.2-1 shows the parameters for the SHNPP spent fuel racks, which may also be used to store new fuel.

The actual number and type of assemblies, the number, type and arrangement of storage modules may vary based on fuel storage needs provided structural analysis shows the proposed module arrangement to be acceptable.

The new fuel inspection pit may be used for storage of new fuel during and after receipt inspection. This facility provides only dry storage conditions.

9.1.1.2 Facilities Description

The new fuel storage pool is located in the south end of the Fuel Handling Building as shown on Figures 1.2.2-55 through 1.2.2-59.

The new fuel pool is interconnected with the three spent fuel pools by means of a transfer canal which runs the length of the Fuel Handling Building. These pools are normally isolated by means of removable gates.

The new fuel pool is a concrete structure with a stainless steel liner for compatibility with the pool water. There is no built-in drain connection in the new fuel pool, thus eliminating the possibility of draining the pool when spent fuel is being stored. The new fuel pool is provided with a sump to be used with a portable sump pump for drainage. Provisions are made to limit and detect leakage of the new fuel pool as discussed in Section 9.1.3. A description of the pool liner is given in Section 9.1.3.

The new fuel inspection pit is a concrete structure located in the north end of the Fuel Handling Building at elevation 261'. It has a concrete floor with no steel liner. It is not usable for wet storage, due to an open stairwell leading down to the 216' elevation, with a non-waterproof door into the pit.

TABLE 9.1.1-1 WAS DELETED BY AMENDMENT NO. 43.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

The maximum storage capacity of the three spent fuel pools is 3704 PWR Assemblies. The total storage capacity of both the new and spent fuel pools is 4184 PWR assemblies. Fuel is stored in a combination of 6x10, 6x8, 7x10, and 7x7 PWR rack modules designed for underwater removal and installation should rack rearrangements be desired. Rearrangement of the racks would have no effect on maximum stored fuel criticality. Module arrangement may vary based on changing fuel storage needs, provided structural analysis shows the proposed module arrangement to be acceptable.

In the event additional space is needed for the storage of spent fuel from other nuclear plants in the CP&L system the spent fuel pools are designed for the storage of both PWR and BWR fuel. The 7x7 PWR rack modules are interchangeable with 11x11 BWR rack modules as these racks cover the same floor area. The actual number and type of assemblies being stored will vary.

The fuel racks consist of individual vertical cells fastened together through top and bottom supporting grid structures to form integral modules. A neutron absorbing material is encapsulated into the stainless steel walls of each storage cell. The PWR rack modules have a center-to-center spacing of 10.5 in. between cells. The BWR rack modules have a center-to-center spacing of 6.25 in. between cells. These free-standing, self-supporting modules are sufficient to maintain a subcritical array of $K_{eff} \leq 0.95$ even in the event the fuel pools are flooded with unborated water. Table 9.1.2-1 shows the parameters for the SHNPP spent fuel racks.

The design of the spent fuel storage racks precludes fuel insertion in other than prescribed locations, thereby preventing any possibility of accidental criticality. A lead-in opening is provided for each PWR storage location, and the storage cells provide full length guidance for the fuel assembly. BWR storage locations do not have a lead-in since the lower nozzle design eliminates the need for lead-in. PWR fuel assemblies will not fit in a BWR spent fuel rack. Insertion of a BWR fuel assembly into a PWR spent fuel rack will result in a subcritical array of $K_{eff} \leq 0.95$.

9.1.2.2 Facilities Description

The spent fuel storage facility is located in the Fuel Handling Building as shown in Figures 1.2.2-55 through 1.2.2-59. The spent fuel is transferred from the Containment to the fuel transfer canal through the fuel transfer tube. The spent fuel bridge crane is used to transport the spent fuel to the spent fuel racks and later to the spent fuel cask. This procedure is carried out with the spent fuel assemblies totally submerged.

There are three spent fuel pools. These pools are interconnected by means of the main fuel transfer canal which runs the length of the Fuel Handling Building. These pools are normally isolated by means of removable gates.

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, that latter case is more limiting from a fuel rod damage potential. In previous accident analyses it was assumed the 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, the potential for damage to the fuel racks was analyzed. Five different locations on the top of a standard PWR poison rack assembly were analyzed for straight drop BPRA tool impact. 25

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 SHNPP stored fuel.

All materials used in construction are compatible with the storage pool environment, and all surfaces that come into contact with the fuel assemblies are made of annealed austenitic steel. All the materials are corrosion resistant and will not contaminate the fuel assemblies or pool environment.

Shielding considerations are discussed in Section 12.3. Radiological conditions associated with the fuel handling accident are discussed in Section 15.7.

TABLE 9.1.2-1

SHEARON HARRIS SPENT FUEL RACK DIMENSIONS*

Fuel Type: W 17x17, W 15x15, Ex 17x17, Ex 15x15, GE 8x8, GE 7x7, and GE 8x8R

RACK ITEM	PWR	BWR
C-C SPACING	10.500	6.250
CELL I.D.	8.750	6.050
POISON CAVITY	0.090	0.060-0.080
POISON WIDTH	7.500	5.100
CELL GAP (NOMINAL)	1.330	---
POISON THICKNESS	0.075	0.045-0.075
WALL THICKNESS	0.075	0.075
WRAPPER THICKNESS	0.035	0.035
POISON (GM-B10/SQ.CM)	0.020	0.0103-0.015

* All Dimensions in Inches

9.1.3 FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Basis

The Fuel Handling Building (FHB) is split into two storage facilities. The storage facility on the north end of the FHB consists of two spent fuel pools. The storage facility on the south end of the FHB consists of a spent fuel pool and a new fuel pool. The design bases for the Fuel Pool Cooling and Cleanup System (FPCCS) are as follows:

a) Each of the two fuel storage facilities consists of two 100 percent cooling systems, and cleanup equipment to remove the particulate and dissolved fission and corrosion products resulting from the spent fuel.

b) Fuel can be transferred between the storage facilities, as shown on Figure 1.2.2-55.

c) The FPCCS is designed to maintain water quality in the fuel storage pools and remove residual heat from the spent fuel.

d) The cooling system serving one fuel storage facility has been designed to remove heat loads generated by the quantities of fuel stored in the pool as indicated in Tables 9.1.3-1A, 9.1.3-1B, and 9.1.3-1C. 25

The fuel pool loading schedule, heat loads, equilibrium temperatures, heat-up rates and equipment design data are presented in Tables 9.1.3-1A, 9.1.3-1B, 9.1.3-1C, and 9.1.3-2. 25

e) The maximum pool temperature with the maximum normal heat load occurring simultaneously with a loss of a single fuel pool cooling loop will be 137 F. The maximum pool temperature with maximum abnormal heat load is 142 F. See Table 9.1.3-2 for fuel pool equilibrium temperatures. The pool concrete design temperature is 150 F. 25

The determinations of the fuel pool heatup rates indicated in Table 9.1.3-2 were calculated using the following assumptions:

1) No credit for operation of the FPCCS.

2) No evaporative heat losses.

3) No heat absorption by concrete or liner.

4) No heat absorption by spent fuel racks or fuel in pool. 25

f) The cleanup loop pumps have the capacity to provide makeup water at a rate greater than the loss of water due to normal system leakage and evaporation.

g) Safe water level (and thus sufficient radiation shielding) is maintained in the new and spent fuel pools since the cooling connections are at the tops of the pools.

Piping in contact with fuel pool water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pumps, heat exchangers and control valves to facilitate maintenance.

Control Room and local alarms are provided to alert the operator of high and low pool water level, and high temperature in the fuel pool. A low flow alarm, based on measured flow to the fuel pool, is provided to warn of interruption of cooling flow.

The Fuel Pool Cooling and Cleanup System is comprised of the following components. The component parameters are presented in Table 9.1.3-2.

a) Fuel Pool Heat Exchanger - Four fuel pool heat exchangers are provided, two per Fuel Pool Cooling and Cleanup System. The fuel pool heat exchangers are of the shell and straight tube type. Component cooling water supplied from the Component Cooling Water System (Section 9.2.2) circulates through the shell, while fuel pool water circulates through the tubes. The installation of two heat exchangers per FPCCS assures that the heat removal capacity of the cooling system is only partially lost if one heat exchanger fails or becomes inoperative.

b) Fuel Pool Cooling Pump - Four horizontal centrifugal pumps are installed, two per Fuel Pool Cooling and Cleanup System. The use of two pumps installed in separate lines assures that pumping capacity is only partially lost should one pump become inoperative. This also allows maintenance on one pump while the other is in operation.

c) Fuel Pool Demineralizer - Two demineralizers are installed, one for each Fuel Pool Cooling and Cleanup System. Each demineralizer is sized to pass five percent of the loop circulation flow to provide adequate purification of the fuel pool water and to maintain optical clarity in the pool.

d) Fuel Pool Demineralizer Filter and Fuel Pool and Refueling Water Purification Filter - Four filters are installed. Each FPCCS has one fuel pool demineralizer filter and one fuel pool and refueling water purification filter. The filters remove particulate matter from the fuel pool water.

e) Fuel Pool Cooling and Cleanup System Skimmers - Twenty-three skimmers
are installed; five each for the two largest spent fuel pools, two for each 25
fuel transfer canal, three each for the new fuel pool and the smallest spent fuel pool, two for the main fuel transfer canal, and one for the cask loading pool. A fuel pool skimmer pump and filter are provided for surface skimming of the fuel pool water. Flow from the pump is routed through the skimmer filter and returned to the fuel pools.

f) Fuel Pool and Refueling Water Purification Pump - Four fuel pool and refueling water purification pumps are provided, two for each fuel storage facility. Each pump can take suction from and return fluid to the refueling water storage tank via the Safety Injection System, the transfer canal, the new and spent fuel pools, or the refueling cavity. Fluids from these systems are purified by the fuel pool demineralizer and filter. Each pump can also take suction from the demineralized water storage tank for line flushing.

9.1.3.3 Safety Evaluation. All fuel pools are cooled by two independent cooling loops, either of which can remove the decay heat loads generated by the quantities of fuel on Tables 9.1.3-1A, 9.1.3-1B, and 9.1.3-1C.

Table 9.1.3-1A represents a full core unload case with the three spent fuel pools and flooded new fuel pool filled to capacity. The heat loads were calculated using Branch Technical Position ASB 9-2 with an uncertainty factor K equal to 0.20 for cooling times (t_s) less than 10^3 seconds and 0.10 for t_s greater than 10^3 seconds.

The heat load calculated represents the highest reasonable heat load possible for any combination of off-site and on-site fuel. It is assumed here that fuel discharged from Unit 1 will remain in the Southend pools and that off-site fuel from H. B. Robinson Unit 2 and Brunswick Units 1 and 2 is shipped to Harris in order to maintain full core reserve capability at these other reactors. Heat load values are dependent on inputs such as assembly power, EFPD, and decay time. Therefore, there is no dependence on BWR fuel type and the current heat load calculations (Reference 9.1.3-1) is bounding for those BWR fuel assemblies with burnups not exceeding 45,000 MWd/MTU.

In the event of a single failure in one of these Spent Fuel Cooling Loops, the other loop will provide adequate cooling. The pool temperature with one Fuel Pool Cooling Loop in operation will be equal to or less than 137°F.

The maximum normal heat load which would exist in the spent fuel pools concurrent with a LOCA would be 18.8 MBTU/hr. The maximum abnormal heat load value of 44.4 MBTU/hr given in FSAR Table 9.1.3-1A is not used because a LOCA is not required to be considered concurrent with the abnormal condition (complete core unload).

The maximum abnormal heat load has been recalculated for the Southend pools for fuel discharged from Unit 1 with batch average burnups up to 55,000 MWd/MTU and for offsite fuel shipped to Harris. The offsite fuel stored at Harris at the time of the heat load calculation was modeled using actual burnups and decay times. All fuel scheduled to be shipped to Harris was assumed to have a batch average burnup of 45,000 MWd/MTU and a decay time of 2.5 years prior to shipment. The BWR fuel may be channeled or unchanneled. The analysis is documented in the Nuclear Fuel Section QA files as Design Activity 93-0003, file NF 2493.0003 (Reference 9.1.3-1). The maximum heat load for these conditions remains below the heat load presented in Table 9.1.3-2, therefore; the maximum pool temperatures presented remain bounding for batch average burnups of 55,000 MWd/MTU for Harris fuel and 45,000 MWd/MTU for offsite fuel.

With this load, the amount of CCW flow required to maintain the fuel pool temperature less than 150 F is less than 3500 gpm. One train of CCW has sufficient capacity to carry the heat loads from the applicable RHR pump (5 gpm) and RHR heat exchanger (5600 gpm). This leaves 3545 gpm available to

The skimmer system for the new and spent fuel pools consists of surface skimmers, a skimmer pump and filter. The surface skimmers float on the water surface and are connected via flexible hose to the pump suction piping at various locations on the perimeter of the pools. Flow from the pump is routed through the skimmer filter and returned to the fuel pools below the water level.

Syphoning of the pools is prevented by limiting the skimmer hose length to approximately five (5) feet. In addition the skimmer system return piping enters the pool at a point five (5) feet below the normal pool water level and terminates flush with the pool liner. Therefore, water loss due to failures in the skimmer system piping would be limited to five (5) feet.

A failure of the skimmer system piping would not uncover spent fuel nor interrupt fuel pool cooling since the fuel pool cooling water suction connections are located more than five (5) feet below the normal water level.

Draining or syphoning of the spent and new fuel pools via piping or hose connections to these pools or transfer canals is precluded by the location of the penetrations, limitations on hose length, and termination of piping penetrations flush with the liner. Hoses connected to temporary equipment used in the new and spent fuel pools are administratively controlled to prevent syphoning. The fuel pools cooling water return piping terminate at elevation 279 ft., 6 in. The spent fuel pool suction piping exists at 278 ft., 6 in. and the new fuel pool exits at 277 ft., 6 in.. Normal pool water level is 284 ft., 6 in, with the top of the spent fuel at approximately 260 ft. Skimmer suction piping exits the pools at elevation 285 ft., 3 in.

The reduction of the normal pool water level by approximately 5 ft. due to any postulated pipe failure will have no adverse impact on the capability of the cooling system to maintain the required temperature and it does not effect the required shield water depth for limiting exposures from the spent fuel. The slow heatup rate of the fuel pool would allow sufficient time to take any necessary action to provide adequate cooling using the backup provided while the cooling capability for the fuel pool is being restored.

Technical Specification 3.9.11 requires a minimum 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks whenever irradiated fuel assemblies are in a pool. The ability of 23 feet of water to remove 99 percent of the assumed 10 percent iodine gas activity released during the postulated fuel accident forms the bases of this Technical Specification. Technical Specification 3.9.11 requires all movement of fuel assemblies and crane operations with loads in the affected pool area be suspended and the water level restored to within its limit within four hours if the water depth over the stored fuel assemblies falls below 23 feet. These actions would prevent a fuel handling accident with less than 23 feet of water over fuel assemblies stored in the spent fuel pools.

There are two alarms associated with low spent fuel pool water level. The first alarm occurs with approximately 24 feet of water over the stored fuel assemblies and provides the operators with sufficient warning and time to place all loads in a safe position prior to the water level decreasing to 23 feet.

The FPCCS undergoes preoperational and startup test as described in Section 14.2.12. Periodic tests are required as described in the Technical Specifications. Inservice inspection requirements are described in Section 6.6 and pump and valve testing will be performed as described in Section 3.9.6.

The spent fuel pool liners have a vacuum box test performed on the major liner field joints normally exposed to water prior to initial fill.

Components of the system are cleaned and inspected prior to installation. Demineralized water is used to flush the entire system. Instruments are calibrated and alarm functions checked for operability and setpoints during testing. The system will be operated and tested initially with regard to flow points, flow capacity and mechanical operability.

Data will be taken periodically during normal system operation to confirm heat transfer capabilities, purification efficiency, and differential pressures across components.

TABLE 9.1.3-1A

MAXIMUM ABNORMAL HEAT LOAD - ALL FUEL POOLS 25

Cooling Time (Days)	Operating Time (Days)	Source	No. of Assemblies	Fuel Type	Core Operating Power (Mwt)	Heat Load BTU/hr
11315	985	HBR	52	PWR	2300	114712
10950	985	HBR	52	PWR	2300	117520
10585	985	HBR	53	PWR	2300	122695
10585	1230	BSEP	360	BWR	2436	270144
10220	985	HBR	52	PWR	2300	123292
9855	985	HBR	52	PWR	2300	126256
9855	1230	BSEP	360	BWR	2436	283392
9490	985	HBR	53	PWR	2300	131811
9490	930	SHNPP	52	PWR	2900	159640
9125	985	HBR	52	PWR	2300	132444
9125	930	SHNPP	52	PWR	2900	163488
9125	1230	BSEP	360	BWR	2436	297288
8760	985	HBR	52	PWR	2300	135668
8760	930	SHNPP	53	PWR	2900	170660
8395	985	HBR	53	PWR	2300	141616
8395	930	SHNPP	52	PWR	2900	171496
8395	1230	BSEP	360	BWR	2436	311832
8030	985	HBR	52	PWR	2300	142272
8030	930	SHNPP	52	PWR	2900	175656
7665	985	HBR	52	PWR	2300	145756
7665	930	SHNPP	53	PWR	2900	183380
7665	1230	BSEP	360	BWR	2436	327132
7300	985	HBR	53	PWR	2300	152163
7300	930	SHNPP	52	PWR	2900	184288
6935	985	HBR	52	PWR	2300	152880
6935	930	SHNPP	52	PWR	2900	188708
6935	1230	BSEP	360	BWR	2436	343152
6570	985	HBR	52	PWR	2300	156572
6570	930	SHNPP	53	PWR	2900	197001
6205	985	HBR	53	PWR	2300	163452
6205	930	SHNPP	52	PWR	2900	197964
6205	1230	BSEP	360	BWR	2436	359964
5840	985	HBR	52	PWR	2300	164268
5840	930	SHNPP	52	PWR	2900	202748
5475	985	HBR	52	PWR	2300	168220
5475	930	SHNPP	53	PWR	2900	211682
5475	1230	BSEP	360	BWR	2436	377640
5110	985	HBR	53	PWR	2300	175642
5110	930	SHNPP	52	PWR	2900	212732
4745	985	HBR	52	PWR	2300	176488
4745	930	SHNPP	52	PWR	2900	217880
4745	1230	BSEP	360	BWR	2436	396000
4380	985	HBR	52	PWR	2300	180752

TABLE 9.1.3-1A (cont'd) 25

MAXIMUM ABNORMAL HEAT LOAD - ALL FUEL POOLS

Cooling Time (Days)	Operating Time (Days)	Source	No. of Assemblies	Fuel Type	Core Operating Power (Mwt)	Heat Load BTU/hr
4380	930	SHNPP	53	PWR	2900	227423
4015	985	HBR	4	PWR	2300	14244
4015	930	SHNPP	52	PWR	2900	228592
4015	1230	BSEP	360	BWR	2436	415440
3650	930	SHNPP	52	PWR	2900	234208
3285	930	SHNPP	53	PWR	2900	244648
3285	1230	BSEP	360	BWR	2436	436320
2920	930	SHNPP	52	PWR	2900	246376
2920	1230	BSEP	33	BWR	2436	44784
2555	930	SHNPP	52	PWR	2900	253552
2190	930	SHNPP	53	PWR	2900	267915
1825	930	SHNPP	52	PWR	2900	277264
1460	930	SHNPP	52	PWR	2900	304356
1095	930	SHNPP	53	PWR	2900	370788
730	930	SHNPP	52	PWR	2900	508768
365	930	SHNPP	52	PWR	2900	968760
5.8	930	SHNPP	53	PWR	2900	10737800
5.8	620	SHNPP	52	PWR	2900	10332400
5.8	310	SHNPP	52	PWR	2900	9796800
					TOTAL	4.44E+07

TABLE 9.1.3-1B

MAXIMUM NORMAL HEAT LOAD - ALL FUEL POOLS

25

Cooling Time (Days)	Operating Time (Days)	Source	No. of Assemblies	Fuel Type	Core Operating Power (Mwt)	Heat Load BTU/hr
11315	985	HBR	52	PWR	2300	114712
10950	985	HBR	52	PWR	2300	117520
10585	985	HBR	53	PWR	2300	122695
10585	1230	BSEP	360	BWR	2436	270144
10220	985	HBR	52	PWR	2300	123292
9855	985	HBR	52	PWR	2300	126256
9855	1230	BSEP	360	BWR	2436	283392
9490	985	HBR	53	PWR	2300	131811
9490	930	SHNPP	52	PWR	2900	159640
9125	985	HBR	52	PWR	2300	132444
9125	930	SHNPP	52	PWR	2900	163488
9125	1230	BSEP	360	BWR	2436	297288
8760	985	HBR	52	PWR	2300	135668
8760	930	SHNPP	53	PWR	2900	170660
8395	985	HBR	53	PWR	2300	141616
8395	930	SHNPP	52	PWR	2900	171496
8395	1230	BSEP	360	BWR	2436	311832
8030	985	HBR	52	PWR	2300	142272
8030	930	SHNPP	52	PWR	2900	175656
7665	985	HBR	52	PWR	2300	145756
7665	930	SHNPP	53	PWR	2900	183380
7665	1230	BSEP	360	BWR	2436	327132
7300	985	HBR	53	PWR	2300	152163
7300	930	SHNPP	52	PWR	2900	184288
6935	985	HBR	52	PWR	2300	152880
6935	930	SHNPP	52	PWR	2900	188708
6935	1230	BSEP	360	BWR	2436	343152
6570	985	HBR	52	PWR	2300	156572
6570	930	SHNPP	53	PWR	2900	197001
6205	985	HBR	53	PWR	2300	163452
6205	930	SHNPP	52	PWR	2900	197964
6205	1230	BSEP	360	BWR	2436	359964
5840	985	HBR	52	PWR	2300	164268
5840	930	SHNPP	52	PWR	2900	202748
5475	985	HBR	52	PWR	2300	168220
5475	930	SHNPP	53	PWR	2900	211682
5475	1230	BSEP	360	BWR	2436	377640
5110	985	HBR	53	PWR	2300	175642
5110	930	SHNPP	52	PWR	2900	212732
4745	985	HBR	52	PWR	2300	176488
4745	930	SHNPP	52	PWR	2900	217880
4745	1230	BSEP	360	BWR	2436	396000
4380	985	HBR	52	PWR	2300	180752

TABLE 9.1.3-1B (cont'd)

MAXIMUM NORMAL HEAT LOAD - ALL FUEL POOLS 25

Cooling Time (Days)	Operating Time (Days)	Source	No. of Assemblies	Fuel Type	Core Operating Power (Mwt)	Heat Load BTU/hr
4380	930	SHNPP	53	PWR	2900	227423
4015	985	HBR	4	PWR	2300	14244
4015	930	SHNPP	52	PWR	2900	228592
4015	1230	BSEP	360	BWR	2436	415440
3650	930	SHNPP	52	PWR	2900	234208
3285	930	SHNPP	53	PWR	2900	244648
3285	1230	BSEP	360	BWR	2436	436320
2920	930	SHNPP	52	PWR	2900	246376
2920	1230	BSEP	33	BWR	2436	44784
2555	930	SHNPP	52	PWR	2900	253552
2190	930	SHNPP	53	PWR	2900	267915
1825	930	SHNPP	52	PWR	2900	277264
1460	930	SHNPP	52	PWR	2900	304356
1095	930	SHNPP	53	PWR	2900	370788
730	930	SHNPP	52	PWR	2900	508768
365	930	SHNPP	52	PWR	2900	968760
32	930	SHNPP	53	PWR	2900	5200000
					TOTAL	1.88E+07

25

TABLE 9.1.3-1C

MAXIMUM NORMAL/ABNORMAL HEAT LOAD - NORTH END POOLS 25

Cooling Time (Days)	Operating Time (Days)	Source	No. of Assemblies	Fuel Type	Core Operating Power (Mwt)	Heat Load BTU/hr
11315	985	HBR	52	PWR	2300	114712
10950	985	HBR	52	PWR	2300	117520
10585	985	HBR	53	PWR	2300	122695
10585	1230	BSEP	360	BWR	2436	270144
10220	985	HBR	52	PWR	2300	123292
9855	985	HBR	52	PWR	2300	126256
9855	1230	BSEP	360	BWR	2436	283392
9490	985	HBR	53	PWR	2300	131811
9125	985	HBR	52	PWR	2300	132444
9125	1230	BSEP	360	BWR	2436	297288
8760	985	HBR	52	PWR	2300	135668
8395	985	HBR	53	PWR	2300	141616
8395	1230	BSEP	360	BWR	2436	311832
8030	985	HBR	52	PWR	2300	142272
7665	985	HBR	52	PWR	2300	145756
7665	1230	BSEP	360	BWR	2436	327132
7300	985	HBR	53	PWR	2300	152163
6935	985	HBR	52	PWR	2300	152880
6935	1230	BSEP	360	BWR	2436	343152
6570	985	HBR	52	PWR	2300	156572
6205	985	HBR	53	PWR	2300	163452
6205	1230	BSEP	360	BWR	2436	359964
5840	985	HBR	52	PWR	2300	164268
5475	985	HBR	52	PWR	2300	168220
5475	1230	BSEP	360	BWR	2436	377640
5110	985	HBR	53	PWR	2300	175642
4745	985	HBR	52	PWR	2300	176488
4745	1230	BSEP	24	BWR	2436	26400
4380	985	HBR	22	PWR	2300	76472
					TOTAL	5.42E+06

TABLE 9.1.3-2

FUEL POOL COOLING AND CLEANUP SYSTEM PARAMETERS

	South End	North End	
Fuel Pool Heat Loads			
Maximum abnormal (based on fuel distribution shown in Tables 9.1.3-1A & 1C) Btu/hr	39.02×10^6	5.417×10^6	27
Maximum normal (based on fuel distribution shown in Tables 9.1.3-1B & 1C) Btu/hr	13.35×10^6	5.417×10^6	
Fuel Pool Equilibrium Temperature			
Maximum abnormal load, one cooling loop operating each FPCCS, °F	142	110	
Maximum normal load, one cooling loop operating both FPCCS, °F	137	126	
Combined Spent and New Fuel Pool Water Heat Inertia, No Heat Removal			
Maximum abnormal load, rate of temperature increase, °F/hr	12.9	1.7	
Maximum normal load, rate of temperature increase, °F/hr	4.4	1.7	
Fuel Pool Heat Exchanger			
Quantity (per FPCCS)	2		
Type	Shell and Two Pass Straight Tube		
UA (Design per Heat Exchanger), Btu/hr.-F	21.1×10^5		27
Shell Side (Component Cooling Water) - Design			
Inlet temperature, F	105		
Outlet temperature, F	110		
Flowrate, lb./hr.	2.68×10^6		
Design pressure, psig	150		
Design temperature, F	200		
Material	Carbon Steel		
Tube Side (Fuel Pool Water) - Design			27
Inlet temperature, F	120		
Outlet temperature, F	113		
Flowrate, lb./hr.	2.256×10^6		
Design pressure, psig	150		
Design temperature, F	200		
Material	Stainless Steel		

TABLE 9.1.3-2 (Continued)

Fuel Pool Cooling Pump			
Quantity (per FPCCS)			2
Type			Horizontal Centrifugal
Design flowrate, gpm			4560
TDH, ft. H2O			98
Motor horsepower			150
Design pressure, psig			150
Design temperature, °F			200
Material			Stainless Steel
Spent Fuel Pools	Pool 1	Pool 2	Pool 3
Volume gals.	403,920	403,920	191,480
Boron concentration, ppm (minimum)*	2,000	2,000	2,000
Liner material	Stainless Steel	Stainless Steel	Stainless Steel
New Fuel Pool			
Volume, gals.			147,804
Boron concentration, ppm (minimum)*			2,000
Liner material			Stainless Steel
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

*The actual boron concentration will be determined by the plants' Technical Specifications for Refueling.

TABLE 9.1.3-2 (Continued)

FUEL POOL COOLING AND CLEANUP SYSTEM PARAMETERS

Fuel Pool Demineralizer	
Quantity (per FPCCS)	1
Type	Flushable
Design pressure, psig	400
Design temperature, F	200
Design flowrate, gpm	325
Volume of resin (each), ft^3	85
Fuel Pool and Refueling Water Purification 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
Fuel Pool Strainer	
Quantity (per FPCCS)	1
Type	Basket
Flowrate, gpm	4560 | 40
Design pressure, psig	150
Design temperature, F	200
Maximum differential pressure across the strainer element at above flow (clean), psi	1.4
Mesh	40
Fuel Pool Skimmer Pump Suction Strainer	
Quantity (per FPCCS)	1
Type	Duplex Basket
Design pressure, psig	150
Design temperature, F	200
Flowrate, gpm	385

TABLE 9.1.3-2 (Continued)

FUEL POOL COOLING AND CLEANUP SYSTEM PARAMETERS

Fuel Pool Skimmer Pump Suction Strainer (Continued)		
Maximum differential pressure across filter element at rated flow (clean), psi	5	
Maximum differential pressure across filter element prior to removing, psi	60	
Mesh	100	
Fuel Pool Skimmer Filter		
Quantity (per FPCCS)	1	15
Type	Flushable	
Design pressure, psig	400	
Design temperature, F	200	
Flowrate, gpm	400	
Maximum differential pressure across filter element at rated flow (clean), psi	5	
Maximum differential pressure across filter element prior to removing, psi	60	
Fuel Pool Skimmer Pump		
Quantity (per FPCCS)	1	15
Flowrate, gpm	385	
TDH, ft. H20	210	
Motor horsepower	40	
Design pressure, psig	150	
Design temperature, F	200	
Material	Stainless Steel	
Fuel Pool and Refueling Water Purification Pump		
Quantity (per FPCCS)	2	
Type	Vertical In-line Centrifugal	
Design flowrate, gpm	325	
TDH, ft. H20	320	
Motor horsepower	60	
Design pressure, psig	150	
Design temperature, F	200	
Material	Stainless Steel	

TABLE 9.1.3-2 (Continued)

FUEL POOL COOLING AND CLEANUP SYSTEM PARAMETERS

Fuel Pool Cooling and Cleanup System Piping and Valves			
Material	Stainless Steel		
Design pressure, psig	150		
Design temperature, F	200		
Fuel Pool Skimmers	Quantity	gpm each	
Large Spent Fuel Pool (per FPCCS)	5	35	
Small Spent Fuel Pool	3	30	25
New Fuel Pool	3	30	
Fuel Transfer Canal (per FPCCS)	2	25	
Main Fuel Transfer Canal (per FPCCS)	1	20	
Cask Loading Pool	1	50	25

Table 9.1.3-3 Deleted by Amendment No. 43

Activation of the 'C' and 'D' Spent Fuel Pools

Originally, SHNPP was intended to be a four unit site with four fuel pools (A,B,C, and D) and two Fuel Pool Cooling and Cleanup systems (FPCCS). Although three of the four units were canceled, the construction of all four pools and one of the FPCCS was completed. Also, a portion of the piping for the other FPCCS was installed. Currently pools 'A' and 'B' are in service and not only store SHNPP fuel, but also store spent fuel from other CP&L plants (Brunswick Units 1& 2, and Robinson). Pools 'C' and 'D' are not in service.

CP&L has determined that pools 'C' and 'D' will be needed to ensure all 4 units maintain a prudent operating reserve for core off loads. According to CP&L, pool 'C' is needed by early 2000 to support fuel shipments from Brunswick and Robinson. In order to place pools 'C' and 'D' in service, the second FPCCS must be completed. In addition, CP&L intends to install higher density racks in pools 'C' and 'D' to assure sufficient storage capacity for all four units through the end of their current license. CP&L intends to submit a license amendment before the end of December 1998.

The amendment would address three specific licensing issues associated with the completion of pools 'C' and 'D'. The first is a potential unreviewed safety question (USQ) associated with the modification of the Unit 1 Component Cooling Water (CCW) System. Although the Unit 1 CCW system was not originally designed to cool the FPCCS for pools 'C' and 'D', CP&L has determined that the Unit 1 CCW system has sufficient margin to accept the 'C' and 'D' FPCCS load. The original design was for the Unit 1 CCW to cool the FPCCS for pools 'A' and 'B', and for the Unit 2 CCW system to cool the FPCCS for pools 'C' and 'D'. The second issue involves piping certification for the 'C' and 'D' FPCCS. A portion of the piping for the 'C' and 'D' FPCCS is already installed, however, because some of the installed piping is embedded in concrete approximately 14 field welds are inaccessible. For much of the installed piping, CP&L cannot supply the records which demonstrates that the piping satisfies the design and construction requirements of 10 CFR 50.55a because they inadvertently disposed of the piping certification records for the installed piping. CP&L intends to propose an alternative that demonstrates the piping is constructed such that it provides an acceptable level of quality and safety. Finally, CP&L intends to submit a Technical Specification (TS) change to allow higher density racks to be installed in pools 'C' and 'D'. The TS change would modify SHNPP spent fuel capacity.

This activity has generated a great deal of public interest. A member of the Union of Concerned Scientist (UCS) attended the July 16, 1998, NRC-CP&L public meeting. A local public interest group NC WARN (Waste Awareness and Reduction Network) has also expressed interest in this activity. NC WARN issued a letter to the CP&L CEO which expressed its concerns about the storage of additional spent fuel at SHNPP, and requested that CP&L be more open about the process and allow public participation. NC WARN is also running ads in the local newspapers protesting the activation of the 'C' and 'D' pools. Also, local governments (Board of County Commissioners Chatam County and Mayor of Chapel Hill) have requested that the State of North Carolina Department of Environment and Natural Resources to complete a thorough review of this activity and forward questions and comments to the NRC.

CP&L planning an open house meeting at Harris to discus issues on 2/4/99.

E/2

CAROLINA POWER & LIGHT COMPANY

NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

UPDATED FACILITY OPERATING LICENSE

(AMENDMENT NO. 84)

License No. NPF-63

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

   A. The application for license filed by the Carolina Power & Light Company acting for itself, and the North Carolina Eastern Municipal Power Agency (the licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Shearon Harris Nuclear Power Plant, Unit 1, (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-158 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. Carolina Power & Light Company* is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

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*Carolina Power & Light Company is authorized to act for the North Carolina Eastern Municipal Power Agency, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

E/3

F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Facility Operating License No. NPF-63, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied;

I. The receipt, possession and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70.

2. Based on the foregoing findings and the Partial Initial Decisions issued by the Atomic Safety and Licensing Board dated February 20, 1985, August 20, 1985, December 11, 1985, and April 28, 1986, regarding this facility and pursuant to approval by the Nuclear Regulatory Commission at a meeting on January 8, 1987, Facility Operating License No. NPF-63, which supersedes the license for fuel loading and low power testing, License No. NPF-53 issued on October 24, 1986, is hereby issued to the Carolina Power & Light Company and the North Carolina Eastern Municipal Power Agency (the licensees) as follows:

A. This license applies to the Shearon Harris Nuclear Power Plant, Unit 1, a pressurized water reactor and associated equipment (the facility) owned by the North Carolina Eastern Municipal Power Agency and the Carolina Power & Light Company, and operated by the Carolina Power & Light Company. The facility is located on the licensees' site in Wake and Chatham Counties, North Carolina, approximately 16 miles southwest of the nearest boundary of Raleigh, and is described in Carolina Power & Light Company's Final Safety Analysis Report, as supplemented and amended, and in its Environmental Report, as supplemented and amended;

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, Carolina Power & Light Company to possess, use, and operate the facility at the designated location in Wake and Chatham Counties, North Carolina, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 50, North Carolina Eastern Municipal Power Agency to possess the facility at the designated location in Wake and Chatham Counties, North Carolina, in accordance with the procedures and limitations set forth in the license;

(3) Pursuant to the Act and 10 CFR Part 70, Carolina Power & Light Company to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Carolina Power & Light Company to receive, possess, and use at any time any byproduct, source and special nuclear material such as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Carolina Power & Light Company to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Carolina Power & Light Company to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein;

(7) Pursuant to the Act and 10 CFR Parts 30 and 40, Carolina Power & Light Company to receive, possess and process for release or transfer to the Shearon Harris site such byproduct material as may be produced by the Shearon Harris Energy and Environmental Center;

(8) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Carolina Power & Light Company to receive and possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Brunswick Steam Electric Plant, Units 1 and 2, and H. B. Robinson Steam Electric Plant, Unit 2.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 84, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)*

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval of a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(6) Detailed Control Room Design Review (Item I.D.1, Section 18)

Carolina Power & Light shall submit the final results of the control room surveys prior to startup following the first refueling outage.

(7) Safety Parameter Display System (Section 18.2.1)

Carolina Power & Light Company shall submit to the NRC for review prior to startup following the first refueling:

(a) The final Validation Test Report,

(b) The resolution of additional human engineering deficiencies identified on the safety parameter display system.

(8) Deleted

(9) Formal Federal Emergency Management Agency Finding

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(10) Fresh Fuel Storage

The following criteria apply to the storage and handling of new fuel assemblies in the Fuel Handling Building:

(a) The minimum edge-to-edge distance between a new fuel assembly outside its shipping container or storage rack and all other new fuel assemblies shall be at least 12 inches.

(b) New fuel assemblies shall be stored in such a manner that water would drain freely from the assemblies in the event of flooding and subsequent draining of the fuel storage area.

D. Exemptions

The facility requires an exemption from Appendix E, Section IV.F.1, which requires that a full participation exercise be conducted within one year before the issuance of a license for full power operation. This exemption is authorized by law and will not endanger life or property or the common defense and security, and certain special circumstances are present. This exemption is, therefore, hereby granted pursuant to 10 CFR 50.12 as follows:

> Shearon Harris Nuclear Power Plant, Unit 1, is exempt from the requirement of 10 CFR Part 50, Appendix E, Section IV.F.1 for the conduct of an offsite full participation exercise within one year before the issuance of the first operating license for full power and prior to operation above 5 percent of rated power, provided that a full participation exercise is conducted before or during March 1987.

The facility is granted an exemption from Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50 (see SER Section 6.2.6). This exemption is authorized by law and will not endanger life or property or the common defense and security, and certain special circumstances are present. In addition, the facility was previously granted an exemption from the criticality alarm requirements of paragraph 70.24 of 10 CFR Part 70 insofar as this section applies to this license. (See License Number SNM-1939 dated October 28, 1985, which granted this exemption).

E. Physical Security (Section 13.6.2.10)

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Shearon Harris Nuclear Power Plant Security Plan," with revisions submitted through September 23, 1987; "Shearon Harris Nuclear Power Plant Security Personnel Training and Qualification Plan," with revisions submitted through October 2, 1985; and "Shearon Harris Nuclear Power Plant Safeguards Contingency Plan," with revisions submitted through October 2, 1985. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Fire Protection Program (Section 9.5.1)

Carolina Power & Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility as amended and as approved in the Safety Evaluation Report (SER) dated November 1983 (and supplements 1 through 4), and the Safety Evaluation dated January 12, 1987, subject to the following provision below.

The licensees may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Reporting to the Commission

Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, Carolina Power & Light Company shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within twenty-four (24) hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73 (b), (c) and (e).

H. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. This license is effective as of the date of issuance and shall expire at midnight on October 24, 2026.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.a The spent fuel storage racks are designed and shall be maintained with a k_{eff} less than or equal to 0.95 when flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3.2.6 of the FSAR. This is assured by maintaining:

1. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks.

2. The maximum core geometry K_∞ for PWR fuel assemblies less than or equal to 1.470 at 68°F.

5.6.1.b The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The new and spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pools below elevation 277.

CAPACITY

5.6.3 The new and spent fuel storage pools are designed for a storage capacity of 1832 PWR fuel assemblies and a variable number of PWR and BWR storage spaces in 48 interchangeable 7x7 PWR and 11x11 BWR racks. These interchangeable racks will be installed as needed. Any combination of BWR and PWR racks may be used.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

E/4

1.2.2.6 Steam and Power Conversion System. The Steam and Power Conversion Systems transform the thermal output of the reactor into electrical power via a turbine-generator. This is accomplished by the transfer of heat from the primary coolant loop to a secondary coolant loop through the steam generators. In the secondary loop, feedwater enters the steam generators and is heated to produce steam. This steam drives the turbine generator, is condensed by rejecting heat to the Circulating Water System, and is recirculated to the Feedwater System. The secondary coolant loop provides an additional barrier to the release of radioactivity to the environment.

The SHNPP has a Westinghouse turbine-generator, rated at approximately 950 MWe, which converts the potential energy of the steam into electrical energy. The steam supply path has the necessary flexibility, relief, and isolation valves to ensure integrity and safety.

The turbine is a three-element, tandem-compound, four flow-exhaust, 1800 rpm unit with moisture separation and single stage reheat between the high-pressure and low-pressure elements. The AC generator is directly connected to the turbine generator shaft.

The Steam Dump System provides the capability to sustain sudden large load decreases up to and including full load loss down to auxiliary loads, concurrent with the loss of external auxiliary power.

The Feedwater and Condensate System is a closed system that deaerates the condensate and pumps it from the condenser hotwell through the feedwater heaters to the steam generators. In the event of the loss of normal feedwater from any cause the safety related Auxiliary Feedwater System will provide water to the steam generators. The safety related Auxiliary Feedwater System includes two 100 percent capacity motor driven pumps and one 200 percent capacity steam turbine driven pump. See Section 10.4 for additional information.

1.2.2.7 Nuclear Fuel Handling and Storage Systems. The Fuel Handling and Storage System provides for the safe handling of fuel assemblies and control element assemblies and for the required assembly, disassembly, and storage of the reactor vessel head and internals. The Nuclear Fuel Storage System is designed to store new fuel, and spent fuel produced at the SHNPP, H. B. Robinson Steam Electric Plant, and Brunswick Steam Electric Plant in the new fuel pool and spent fuel pool.

The new fuel pool can be filled with water so that it can be used to store spent fuel. The storage system is designed such that the integrity of the fuel is maintained under normal and abnormal conditions. The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a fuel storage pool. The spent fuel is then placed in a cask for shipment from the site.

E/5

Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat.

The fuel handling structure may be generally divided in two areas: the refueling cavity which is flooded only during shutdown for refueling; and the spent fuel pools and fuel transfer canal system, which are kept full of water and are always accessible to operating personnel. The refueling cavity and the fuel transfer canal are connected by the fuel transfer tube through which an underwater conveyor transfers the fuel.

The manipulator crane, fuel transfer tube, manual tools and spent fuel bridge crane facilitate the transfer of fuel from the refueling cavity to the spent fuel racks.

The Spent Fuel Pool Cooling and Cleanup System removes decay heat and impurities from the fuel pools.

New fuel assemblies may be stored in the new or spent fuel pool. New fuel is delivered to the reactor by lowering it into the appropriate spent fuel pool and taking it through the fuel transfer system. Some of the fuel for the initial core loading may be temporarily stored in the Spent Fuel Pool. See Section 9.1 for further information.

1.2.2.8 Cooling Water and Other Auxiliary Systems.

1.2.2.8.1 Circulating and Service Water System. The Circulating Water System provides the main condenser with a continuous supply of cooling water for removing the heat rejected by the main turbine. Three 33-1/3 percent capacity circulating water pumps, sized for the maximum heat rejection and the required system head, take suction from the cooling tower basin and deliver water to the condenser inlet waterboxes through two large reinforced concrete pipes. After leaving the condenser, the heated circulating water returns to the cooling tower. In addition to circulating water, service water for cooling of auxiliary equipment in the secondary portions is provided during normal operation from the cooling tower basin by means of service water pumps located in a separate intake pump structure. The service water is returned to the Circulating Water System at the outlet from the condenser, for cooling by the Cooling Tower. The Unit has a Service Water System designed to provide redundant cooling water to those components necessary for safety either during normal operation or under accident conditions.

1.2.2.8.2 Component Cooling Water System. The Component Cooling Water System (CCWS) is an intermediate cooling water system serving components and systems important to the safety of the plant. The CCWS is designed to meet all assigned plant component cooling loads during normal operation, assuming the highest possible service water temperature (95°F). At that temperature, there are no limitations placed on normal plant operation.

3. Spent Fuel Pool Cooling and Cleanup System 9.1.3

4. Process Auxiliaries 9.3

3.1.53 Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

CRITERION - Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

DISCUSSION - Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction in conjunction with the use of mechanical neutron poisons. This is accomplished by fixing the minimum separation between assemblies, inserting neutron poison between assemblies, and requiring an initial minimum amount of integral burnable absorber for fuel assemblies with high enrichments. Fuel elements are limited by rack design to only top-loading and to specific fuel assembly positions.

Spent fuel is stored under water in the new and/or spent fuel pool. The racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Each rack is composed of individual vertical cells which are fastened together in any number to form a module.

The combination of maintaining a minimum separation between assemblies, inserting neutron poison between assemblies, and requiring an initial minimum amount of integral burnable absorber for fuel assemblies with high enrichments is sufficient to ensure a $K_{eff} \leq 0.95$ even if unborated water is used to fill the pool.

The fuel racks are designed to withstand shipping, handling, normal operating loads (impact and dead loads) of fuel assemblies as well as SSE and OBE seismic loads meeting ANS Safety Class 3 and AISC requirements. The fuel racks are also designed to meet Seismic Category I requirements of Regulatory Guide 1.13.

Refueling interlocks include circuitry which senses conditions of the refueling equipment. These interlocks reinforce operational procedures that prevent making the reactor critical. The Fuel Handling System is designed to provide a safe, effective means of transporting and handling fuel and is designed to minimize the possibility of mishandling or malfunction.

For further discussion, see the following sections:

a)	All Other Instrumentation Systems Required for Safety	7.6
b)	Fuel Storage and Handling	9.1

3.1.54 CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

CRITERION:

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

DISCUSSION:

Instrumentation is provided in the fuel pool cooling and purification system which will detect a loss of residual heat removal capability. Appropriate safety actions are initiated by operator responses. The Waste Management System does not require the capacity for residual or decay heat removal.

Process ventilation and area radiation monitors are provided to detect and alarm excessive radiation levels in the Fuel Handling Building and Waste Management Systems areas. High radiation level in the fuel pool area will automatically shutoff the normal ventilation system and will automatically start the emergency exhaust system.

For further discussion, see the following sections:

a)	Process and Effluent Radiological Monitoring and Sampling System	11.5
b)	Fuel Storage and Handling	9.1
c)	Air Conditioning, Heating, Cooling, and Ventilation Systems	9.4
d)	Liquid Waste Management Systems	11.2
e)	Gaseous Waste Management Systems	11.3
f)	Solid Waste Management System	11.4
g)	All Other Instrumentation Systems Required for Safety	7.6

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Design Bases. The new fuel pool, referred to as Pool A or New Fuel Pool Unit 1, is designed for the storage of both new and spent fuel. Consequently, it is designed for both wet and dry storage. The maximum storage capacity of this pool is 480 PWR fuel assemblies, which is more than 3 cores. The fuel is stored in 6x10 PWR rack modules, which are designed for underwater removal and installation. The new fuel storage racks are of identical design to the spent fuel storage racks and can be used both wet and dry.

In the event additional space is needed for the storage of spent fuel from other nuclear plants in the CP&L system, the new fuel pool is designed for the storage of both PWR and BWR fuel. Spent BWR fuel will be stored in 11 x 11 BWR rack modules which are designed for underwater removal and installation. The actual number and type of assemblies, the number, type and arrangement of storage modules may vary based on fuel storage needs provided structural analysis shows the proposed module arrangement to be acceptable.

The fuel racks consist of individual vertical cells fastened together through top and bottom supporting grid structures to form integral modules. A neutron absorbing material is encapsulated into the stainless steel walls of each storage cell. Certain PWR rack modules have designated cells that do not contain the neutron absorbing material in one cell wall. These cells are utilized for an absorber material coupon surveillance program. The PWR rack modules have a center-to-center spacing of 10.5 inches between cells. The BWR rack modules have a center-to-center spacing of 6.25 inches between cells. These free-standing, self-supporting modules are sufficient to maintain a subcritical array even in the event the fuel pool is flooded with unborated water. Table 9.1.2-1 shows the parameters for the SHNPP spent fuel racks, which may also be used to store new fuel.

The new fuel inspection pit may be used for storage of new fuel during and after receipt inspection. This facility provides only dry storage conditions.

9.1.1.2 Facilities Description. The new fuel storage pool is located in the south end of the Fuel Handling Building as shown on Figures 1.2.2-55 through 1.2.2-59.

The new fuel pool is interconnected with the three spent fuel pools by means of a transfer canal which runs the length of the Fuel Handling Building. These pools can be isolated by means of removable gates.

The new fuel pool is a concrete structure with a stainless steel liner for compatibility with the pool water. There is no built-in drain connection in the new fuel pool, thus eliminating the possibility of draining the pool when spent fuel is being stored. Provisions are made to limit and detect leakage from the fuel pools through the use of liner leak detection channels which are placed in various locations outside the stainless steel liner and pool gates. These channels funnel any leakage to drain lines which are checked periodically to determine the structural integrity of the pools and gates. A description of the pool liner is given in Section 9.1.3.

The new fuel inspection pit is a concrete structure located in the north end of the Fuel Handling Building at Elevation 261'. It has a concrete floor with no steel liner. It is not usable for wet storage, due to an open stairwell leading down to the 216' elevation, with a non-waterproof door into the pit.

9.1.1.3 Safety Evaluation. The Fuel Handling Building is designed in accordance with Regulatory Guide 1.13, Rev. 1, "Spent Fuel Storage Facility Design Basis," and provides protection to the fuel racks and other pieces of equipment against natural phenomena such as tornadoes, hurricanes, and floods as discussed in Sections 3.3, 3.4, and 3.5.

The design and safety evaluation of the fuel racks is in accordance with the NRC position paper, "Review and Acceptance of Spent Fuel Storage and Handling Applications."

The racks, being ANS Safety Class 3 and Seismic Category I structures, are designed to withstand normal and postulated dead loads, live loads, loads due to thermal effects, and loads caused by the operating bases earthquakes and safe shutdown earthquake events in accordance with Regulatory Guide 1.29, and stress allowables defined by ASME Code, Section III. The racks can withstand an uplift force equal to the maximum uplift capability of the spent fuel bridge crane.

The design of the fuel racks is such that for PWR assemblies with a maximum core geometry K-infinity less than or equal to 1.470 at 68°F, and the pool flooded with unborated water at optimum moderation K_{eff} is ≤ 0.95.

The design of the spent fuel racks is such that for BWR assemblies with reactivity bounded by the 8 x 8R, 3.2 w/o U235 assembly, the K_{eff} for the racks will not exceed 0.95 with the spent fuel pool flooded with unborated water. With this limit on assembly reactivity, all fuel assemblies loaded in BSEP Unit 1 through reload 5 and all fuel assemblies located in BSEP Unit 2 through reload 6 are conservatively bounded and may be stored at SHNPP.

Consideration is given to the inherent neutron absorbing effect of the materials of construction. Fuel handling accidents will not alter the rack geometry to the extent that the criticality acceptance criteria is violated. The criticality safety analysis is discussed in Section 4.3.2.6.

Materials used in construction are compatible with the storage pool environment, and surfaces that come in contact with the fuel assemblies are made of annealed austenitic stainless steel.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases. The maximum storage capacity of the three spent fuel pools is 3704 PWR Assemblies. The total licensed storage capacity of both the new and spent fuel pools is 4184 PWR assemblies. Fuel is stored in a combination of 6 x 10, 6 x 8, 7 x 10, and 7 x 7 PWR rack modules designed for underwater removal and installation should rack rearrangements be desired. Rearrangement of the racks would have no effect on maximum stored fuel criticality. Module arrangement may vary based on changing fuel storage needs, provided structural analysis shows the proposed module arrangement to be acceptable.

In the event additional space is needed for the storage of spent fuel from other nuclear plants in the CP&L system the spent fuel pools are designed for the storage of both PWR and BWR fuel. The 7 x 7 PWR rack modules are interchangeable with 11 x 11 BWR rack modules as these racks cover the same floor area. The actual number and type of assemblies being stored will vary.

The fuel racks consist of individual vertical cells fastened together through top and bottom supporting grid structures to form integral modules. A neutron absorbing material is encapsulated into the stainless steel walls of each storage cell. Certain PWR rack modules have designated cells that do not contain the neutron absorbing material in one cell wall. These cells are utilized for an absorber material coupon surveillance program. The PWR rack modules have a center-to-center spacing of 10.5 in. between cells. The BWR rack modules have a center-to-center spacing of 6.25 in. between cells. These free-standing, self-supporting modules are sufficient to maintain a subcritical array of $K_{eff} \leq 0.95$ even in the event the fuel pools are flooded with unborated water. Table 9.1.2-1 shows the parameters for the SHNPP spent fuel racks.

The design of the spent fuel storage racks precludes fuel insertion in other than prescribed locations, thereby preventing any possibility of accidental criticality. A lead-in opening is provided for each PWR storage location, and the storage cells provide full length guidance for the fuel assembly. BWR storage locations do not have a lead-in since the lower nozzle design eliminates the need for lead-in. PWR fuel assemblies will not fit in a BWR spent fuel rack. Insertion of a BWR fuel assembly into a PWR spent fuel rack will result in a subcritical array of $K_{eff} \leq 0.95$.

9.1.2.2 Facilities Description. The spent fuel storage facility is located in the Fuel Handling Building as shown in Figures 1.2.2-55 through 1.2.2-59. The spent fuel is transferred from Containment to the Fuel Handling Building through the fuel transfer tube. The spent fuel bridge crane is used to transfer the spent fuel between the storage racks, fuel pools, transfer canals, and the spent fuel cask. This procedure is carried out with the spent fuel assemblies totally submerged.

There are three spent fuel pools. The spent fuel pool at the south end of the FHB is referred to as Pool B or Spent Fuel Pool Unit 1. The north end of the FHB contains two additional spent fuel pools. The larger of these two pools is referred to as Pool C or Spent Fuel Pool Unit 2. The smaller north end pool is referred to as Pool D, Spent Fuel Pool, or New Fuel Pool Unit 2. These pools are interconnected by means of the main fuel transfer canal which runs the length of the Fuel Handling Building. These pools can be isolated by means of removable gates.

The spent fuel pools are concrete structures with a stainless steel liner for compatibility with the pool water. Provisions are made to limit and detect leakage from the fuel pools through the use of liner leak detection channels which are placed in various locations outside the stainless steel liner and pool gates. These channels funnel any leakage to drain lines which are checked periodically to determine the structural integrity of the pools and gates. A description of the pool liner is given in Section 9.1.3.

9.1.2.3 Safety Evaluation. The Fuel Handling Building is designed in accordance with Regulatory Guide 1.13, Rev. 1, "Spent Fuel Storage Facility Design Basis," and provides protection to the fuel racks and other pieces of equipment against natural phenomena such as tornadoes, hurricanes and floods as discussed in Sections 3.3, 3.4, and 3.5.

The design and safety evaluation of the fuel racks is in accordance with the NRC position paper, "Review and Acceptance of Spent Fuel Storage and Handling Applications."

The racks, being ANS Safety Class 3 and Seismic Category I structures, are designed to withstand normal and postulated dead loads, live loads, loads due to thermal effects, loads caused by the operating bases earthquakes, and safe shutdown earthquake events in accordance with Regulatory Guide 1.29, and stress allowables defined by ASME Code, Section III.

Consideration is given to the inherent and fixed neutron absorbing effect of the materials of construction. The design of the racks is such that $K_{eff} \leq 0.95$ under all conditions, including fuel-handling accidents. Due to the close spacing of the cells, it is impossible to insert a fuel assembly in other than design locations. Inadvertent insertion of a fuel assembly between the rack periphery and the pool wall is considered a postulated accident and, as such, realistic initial conditions such as boron in the water can be taken into account. This condition has an acceptable $K_{eff} \leq 0.95$. A discussion of the criticality analysis is provided in Section 4.3.2.6.

The racks are also designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of the spent fuel bridge crane. Handling equipment capable of carrying loads heavier than a fuel assembly is prevented by interlocks or administrative controls, or both, from traveling over the fuel storage area. When such loads must travel over the spent fuel storage area, redundant holding systems as described in Table 9.1.4-1 are used. The racks can withstand an uplift force equal to the maximum uplift capability of the spent fuel bridge crane.

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, that latter case is more limiting from a fuel rod damage potential. In previous accident analyses it was assumed the 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, the potential for damage to the fuel racks was analyzed. Five different locations on the top of a standard PWR poison rack assembly were analyzed for straight drop BPRA tool 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 SHNPP stored fuel.

Materials used in construction are compatible with the storage pool environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic steel. The materials are corrosion resistant and will not contaminate the fuel assemblies or pool environment.

Shielding considerations are discussed in Section 12.3. Radiological conditions associated with the fuel handling accident are discussed in Section 15.7.

TABLE 9.1.2-1

SHEARON HARRIS SPENT FUEL RACK DIMENSIONS*

Fuel Type: W 17 x 17, W 15 x 15, Ex 17 x 17, Ex 15 x 15, GE 8 x 8, GE 7 x 7, GE 8x8R, SPC 17 x 17, and SPC 15 x 15,

RACK ITEM	PWR	BWR
C-C SPACING	10.500	6.250
CELL I.D.	8.750	6.050
POISON CAVITY	0.090	0.060-0.080
POISON WIDTH	7.500	5.100
CELL GAP (NOMINAL)	1.330	---
POISON THICKNESS	0.075	0.045-0.075
WALL THICKNESS	0.075	0.075
WRAPPER THICKNESS	0.035	0.035
POISON (GM-B10/SQ.CM)	0.020	0.0103-0.015

* All Dimensions in Inches

9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Basis. The Fuel Handling Building (FHB) is split into two storage facilities. The storage facility on the south end of the FHB consists of a new fuel pool, also referred to as Pool A or New Fuel Pool Unit 1 and a spent fuel pool, also referred to as Pool B or Spent Fuel Pool Unit 1. Both new fuel and spent fuel may be stored in either of the pools in this facility, as described in Sections 9.1.1 and 9.1.2. The storage facility on the north end of the FHB consists of a spent fuel pool, also referred to as Pool C or Spent Fuel Pool Unit 2 and a New Fuel Pool, also referred to as Pool D or New Fuel Pool Unit 2. By design, both of the pools in this facility may accommodate both new and spent fuel. Spent fuel may not be loaded into Pools C or D until they are completed and made operational. The design bases for the Fuel Pool Cooling and Cleanup System (FPCCS) for the operational pools, Pools A and B, are as follows:

a) The fuel storage facility consists of two 100 percent cooling systems in addition to cleanup equipment for removing the particulate and dissolved fission and corrosion products resulting from the spent fuel.

b) Fuel can be transferred within the operational storage facility as shown on Figure 1.2.2-55. Fuel handling is described in detail in Section 9.1.4.

c) The FPCCS is designed to maintain water quality in the fuel storage pools and remove residual heat from the spent fuel.

d) The current and typical refueling practice at SHNPP of transferring the entire core to the storage facility is referred to herein as the Full Core Offload Shuffle. The refueling practice of transferring only that portion of the core to be discharged to the storage facility is referred to herein as the Incore Shuffle. Both of these practices are reported as Normal Cases when meeting the requirements of the Standard Review Plan. The Abnormal Case is reported as the transfer of the entire core to the storage facility following startup of the next operating cycle. This case is referred to herein as the Post Outage Full Core Offload.

e) The cooling system serving the operational fuel storage facility has been designed to remove the heat loads generated by the quantities of fuel to be stored in the pools through operation to the end-of-Cycle 8.

f) The Standard Review Plan pool temperature requirement for the Normal Case, assuming a single active failure, is 140°F. The minimum decay time prior to movement of irradiated fuel in the reactor vessel will address both radiological and decay heat considerations. Administrative controls are placed on the minimum cooling time before transfer of spent fuel to the pools, to limit the fuel pool temperature to less than or equal to 137°F. The pool temperature requirement for the Abnormal Case is to be below boiling. The pool concrete design temperature is 150°F.

g) Calculations of the maximum amount of thermal energy to be removed by the spent fuel cooling system are made in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." An uncertainty factor K equal to 0.20 for cooling times (t_s) less than 10^3 seconds and 0.10 for t_s greater than 10^3 seconds was used.

h) The fuel pool heatup rates were calculated using the following assumptions:

1) No credit for operation of the FPCCS.

2) No evaporative heat losses.

3) No heat absorption by concrete or liner.

4) No heat absorption by spent fuel racks or fuel in pool.

i) The cleanup loop pumps have the capacity to provide makeup water at a rate greater than the loss of water due to normal system leakage and evaporation.

j) Safe water level (and thus sufficient radiation shielding) is maintained in the new and spent fuel pools since the cooling connections are at the tops of the pools.

k) Components and structures of the system are designed to the safety class and seismic requirements indicated in Table 3.2.1-1.

l) The FPCCS will perform its safety related function assuming a single active failure (Reference 9.1.3-1).

9.1.3.2 System Description. The Fuel Pool Cooling and Cleanup System is provided as shown on Figures 9.1.3-1, 9.1.3-2, 9.1.3-3 and 9.1.3-4. The FPCCS is comprised of the two operational fuel pools, Pools A and B; the Cask Loading/Unloading Pool; the Main Fuel Transfer Canal; the south Fuel Transfer Canal; the north Fuel Transfer Canal; two fuel pool heat exchangers; two fuel pool cooling pumps; two fuel pool strainers; a fuel pool demineralizer; a fuel pool demineralizer filter; a fuel pool and a refueling water purification filter; two fuel pool and refueling water purification pumps; provisions for skimmer connections as follows: three fuel Pool A skimmers; five Pool B skimmers; two south transfer canal skimmers; two north transfer canal skimmers, one main transfer canal skimmer, one cask loading/unloading pool skimmer; a fuel pool skimmer pump, a fuel pool skimmer strainer, and a fuel pool skimmer filter.

The new fuel pool, Pool A, and the spent fuel pool, Pool B, are interconnected by the south Fuel Transfer Canal. The Cask Loading/Unloading Pool, the non-operational Pool C, and the non-operational Pool D are interconnected by the north Fuel Transfer Canal. The Main Fuel Transfer Canal connects the south and north Fuel Transfer Canals. Gates are provided to isolate the pools, as needed. Spent fuel is placed in the operational pools during refueling or from shipments of off-site fuel and stored until it is shipped to a reprocessing facility or otherwise disposed. Fuel handling is discussed in detail in Section 9.1.4. The overall arrangement of the pools is shown on Figure 1.2.2-55. Cooling of spent fuel can be accomplished in the operational fuel pools since they are serviced by the fuel pool cooling system. The location of the inlet and outlet connections to the pools precludes the possibility of coolant flow "short circuiting" the pool.

The Fuel Handling Building is designed to Seismic Category I requirements and to the tornado criteria as stated in Section 3.3.

The fuel pools in the Fuel Handling Building will not be affected by any loss of coolant accident in the Containment Building. The water in the pools is isolated from that in the refueling cavity during most of the refueling operation. Only a very small amount of interchange of water will occur as fuel assemblies are transferred during refueling.

The FPCCS is designed for the removal of sensible heat from the fuel pools. Current analyses have evaluated this function for a decay heatload equivalent to that generated by fuel discharged at HNP through operation to the end-of-Cycle 8 and from additional fuel assemblies planned to be shipped from H. B. Robinson Unit 2 and Brunswick Units 1 and 2 through end-of-Cycle 8. For this mode of operation, the equilibrium temperatures are as shown in Table 9.1.3-2.

The clarity and purity of the fuel pool water is maintained when desired or necessary by passing approximately five percent of the cooling system flow through a cleanup loop consisting of two filters and a demineralizer. The fuel pool cooling pump suction line, which can be used to lower the pool water level, penetrates the fuel pool wall approximately 18 ft. above the fuel assemblies. The penetration location precludes uncovering the fuel assemblies as a result of a postulated suction line rupture.

Piping in contact with fuel pool water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pumps, heat exchangers and control valves to facilitate maintenance.

Control Room and local alarms are provided to alert the operator of high and low pool water level, and high temperature in the fuel pool. A low flow alarm, based on measured flow to the fuel pool, is provided to warn of interruption of cooling flow.

The Fuel Pool Cooling and Cleanup System is comprised of the following components. The component parameters are presented in Table 9.1.3-2.

a) Fuel Pool Heat Exchanger - Two fuel pool heat exchangers are provided. The fuel pool heat exchangers are of the shell and straight tube type. Component cooling water supplied from the Component Cooling Water System (Section 9.2.2) circulates through the shell, while fuel pool water circulates through the tubes. The installation of two heat exchangers assures that the heat removal capacity of the cooling system is only partially lost if one heat exchanger fails or becomes inoperative.

b) Fuel Pool Cooling Pump - Two horizontal centrifugal pumps are installed. The use of two pumps installed in separate lines assures that pumping capacity is only partially lost should one pump become inoperative. This also allows maintenance on one pump while the other is in operation.

c) Fuel Pool Demineralizer - One demineralizer is installed. The demineralizer is sized to pass approximately five percent of the loop circulation flow to provide adequate purification of the fuel pool water and to maintain optical clarity in the pool.

d) Fuel Pool Demineralizer Filter and Fuel Pool and Refueling Water Purification Filter - Two filters are installed - one fuel pool demineralizer filter and one fuel pool and refueling water purification filter. The filters remove particulate matter from the fuel pool water.

e) Fuel Pool Cooling and Cleanup System Skimmers - Provisions for fourteen skimmers are installed; three for Pool A, five for Pool B, two for each fuel transfer canal, one for the main fuel transfer canal, and one for the cask loading/unloading pool. A fuel pool skimmer pump, fuel pool skimmer pump suction strainer, and filter are provided for surface skimming of the fuel pool water. Flow from the pump is routed through the skimmer filter and returned to the fuel pools.

f) Fuel Pool and Refueling Water Purification Pump - Two fuel pool and refueling water purification pumps are provided. Each pump can take suction from and return fluid to the refueling water storage tank via the Safety Injection System, the transfer canal, the new and spent fuel pools, or the refueling cavity. Fluids from these systems are purified by the fuel pool demineralizer and filter. Each pump can also take suction from the demineralized water storage tank for line flushing.

g) Fuel Pool Cooling and Cleanup System Valves - Manual stop valves are used to isolate equipment and lines and manual throttle valves provide flow control. Valves in contact with fuel pool water are of austenitic stainless steel or of equivalent corrosion resistant material.

h) Fuel Pool Cooling and Cleanup System Piping - All piping in contact with fuel pool water is of austenitic stainless steel construction. The piping is welded except where flanged connections are used at the pumps, heat exchanger, and control valve to facilitate maintenance. Also, flanged joints with line blanks are installed at locations to provide isolation capabilities for non-operational portions of Unit 2 (Pools C and D) system flow paths.

i) Fuel Pool Gates - The vertical steel gates on the new fuel pool, spent fuel pools, fuel transfer canals, main fuel transfer canal and cask loading pools allow the spent fuel to be immersed at all times while being moved to its destination. They also allow each area to be isolated for drainage, if necessary, and enable new fuel to be stored dry in the new fuel pool.

Fuel Pool water chemistry limits and guidelines are specified in plant chemistry procedures. These procedures insure the fuel pool water chemistry is consistent with current specifications and guidelines established by the NSSS vendor, fuel manufacturer and EPRI standards. The plant Chemistry subunit routinely monitors the fuel pools water by chemical and radiochemical analysis of grab samples. When chemistry exceeds plant procedure limits, appropriate corrective actions are implemented to restore the parameter within its limit. The performance of the Fuel Pool Demineralizer is routinely monitored and when the ion exchange media is depleted, the resin is replaced.

The Spent Fuel Pool fission and corrosion product activities are discussed in FSAR Section 11.1.7. Design and normal operating specific activities are given in FSAR Table 11.1.7-1.

Radiological monitoring of the various samples for the subject system is described in detail in FSAR Sections 11.5.2.5 and 11.5.2.6.

The differential pressure across the flushable filter is measured with on line instrumentation. Before the differential pressure approaches 60 psig, the filter being deposited with maximum amount of crud requires a back-flushing treatment.

9.1.3.3 Safety Evaluation. All fuel pools are cooled by two independent cooling loops, either of which can remove the decay heat loads generated by the quantities of fuel through operation to the end-of-Cycle 8.

Table 9.1.3-2 provides the fuel pool heat load, equilibrium temperature, and water heat inertia for the Incore Shuffle, Full Core Offload Shuffle and Post Outage Full Core Offload cases. These three cases were evaluated based on operation through end-of-Cycle 8. For cases assuming a single active failure, a single CCW train supplies both essential and non-essential loads, resulting in reduced CCW flow to the fuel pool cooling system heat exchanger. Heat loads were calculated for the three cases above. Each of these cases modeled the spent fuel received from previous plant operation and from spent fuel from H. B. Robinson Unit 2 and Brunswick Units 1 and 2 received through end-of-Cycle 7. A bounding heat load from the additional spent fuel to be received during Cycle 8 was also addressed.

Administrative controls are placed on the minimum cooling time prior to transfer of irradiated fuel from the core to the storage facility in order to maintain the pools at less than or equal to 137°F (Reference 9.1.3-2). The minimum cooling time prior to movement of irradiated fuel in the reactor vessel addresses both radiological and decay heat considerations. The most conservative of these two are used in determining the actual required cooling time.

In the event of a single failure in one of these Spent Fuel Cooling Loops, the other loop will provide adequate cooling. The pool temperature with one Fuel Pool Cooling Loop in operation will be equal to or less than 137°F.

The maximum normal heat load which would exist in the spent fuel pools concurrent with a LOCA would be 16.84 MBTU/hr. The maximum heat load values given in FSAR Table 9.1.3-2 for the Full Core Offload Shuffle and the Post Outage Full Core Offload are not used because a LOCA is not required to be considered concurrent with these conditions (complete core unload).

When the Emergency Core Cooling System is aligned to recirculate from the containment sump to the Reactor Coolant System, the CCW trains are separated from each other and from the nonessential header to maintain protection against single passive failure and to provide sufficient flow to their respective RHR trains. Once separated, each train provides flow to its respective essential header composed of heat loads from the RHR pump and RHR Heat Exchanger. In this alignment, each CCW train is balanced to provide greater than 5 gpm to the RHR pump and 6050 gpm to the RHR Heat Exchanger.

When the CCW trains are isolated from the nonessential header, CCW flow to the Spent Fuel Pool Heat Exchanger is also isolated. At 5.56 hours from the time of LOCA initiation, the heat load in the containment sump will be low enough to permit the realignment of CCW to the spent fuel pool heat exchanger. The pools will heat up to 137°F in 5.56 hours assuming an initial temperature of 112.7°F and a normal maximum heat load subsequent to a LOCA of 16.84 Mbtu/hr. With this heat load, 2.97 hours is available for manual actions to restore CCW to the spent fuel pool heat exchanger prior to reaching 150°F in the pools. The CCW flow required to maintain the pool temperature at 150°F assuming this same heat load is 1789 gpm.

The minimum CCW flow which must be maintained through the RHR Heat Exchanger and the RHR pump subsequent to alignment to recirculation is 5600 gpm and 5 gpm, respectively. Subsequent to alignment to recirculation, operators are directed by Operating Procedures to restore sufficient CCW cooling from one CCW train to the spent fuel pools to maintain temperature less than 150°F. Based on the CCW flows established through the RHR pump and RHR Heat Exchanger when the nonessential header is isolated, each train is capable of individually providing the required 5600 gpm and 5 gpm through the RHR Heat Exchanger and RHR pump and 1789 gpm through the spent fuel pool heat exchanger assuming that all other nonessential loads are isolated. The spent fuel pool heat up time of 2.97 hours from 137°F to 150°F is sufficient to allow operators to isolate any non-essential loads and to throttle the CCW flow through the spent fuel pool heat exchanger as required. All local manual manipulations are performed in areas which are accessible subsequent to a LOCA.

To assure reliability, each of the fuel pool cooling pumps is powered from separate buses so that each pump receives power from a different source. If a total loss of offsite power should occur, the operator has the option of transferring the pumps to the emergency power source.

In addition, emergency cooling connections are provided in the loops to permit the installation of portable pumps to bypass the fuel pool cooling pumps should they become inoperable when cooling is required in either pool.

Compliance of the Fuel Pool Cooling and Cleanup System to the guidance of NRC Regulatory Guide No. 1.13, "Fuel Storage Facility Design Basis," is addressed in Section 1.8.

The cooling loop piping and components are designed to Seismic Category I criteria. The cleanup loop is not designed to Seismic Category I criteria; however, suitable valving is provided between the cooling loop and the cleanup loop to permit isolation of the cleanup loop. The cooling loop portion of the FPCCS is protected against externally generated missiles. The fuel pool cooling pumps and associated piping are located in an area of the plant where there are no postulated internally generated missiles. The fuel pool cooling pumps have not been considered credible sources of internally generated missiles. The no-load speed of the pumps is equal to the synchronous speed of the electric motors; consequently, there are no pipe-break plus single failure combinations which could result in a significant increase in pump suction or discharge header. In addition, the FPCCS is protected against the effects of high energy and moderate energy fluid system piping failures (Section 3.6).

The FPCCS is manually controlled and may be shut down safely for reasonable time periods for maintenance or replacement of malfunctioning components.

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to a fuel pool, a small quantity of fission products may enter the fuel pool cooling water. The cleanup loop is provided to remove fission products and other contaminants from the water.

The cleanup loop will normally be run on an intermittent basis as required by fuel pool water conditions. It will be possible to operate the purification system with either the ion exchanger or filter bypassed. Local sample points are provided to permit analysis of ion exchanger and filter efficiencies.

In the event of a high radiation alarm in the Fuel Handling Building, the purification system will be manually started. The cleanup loop is not started automatically since the short delay to manually initiate purification would not significantly speed the reduction of contamination in the pool.

The skimmer system for the new and spent fuel pools consists of surface skimmers, a fuel pool skimmer pump, a fuel pool skimmer pump suction strainer and a fuel pool skimmer filter. The surface skimmers float on the water surface and are connected via flexible hose to the pump suction piping at various locations on the perimeter of the pools. Flow from the pump is routed through the skimmer filter and returned to the fuel pools below the water level.

Siphoning of the pools is prevented by limiting the skimmer hose length to approximately five (5) feet. In addition the skimmer system return piping enters the pool at a point five (5) feet below the normal pool water level and terminates flush with the pool liner. Therefore, water loss due to failures in the skimmer system piping would be limited to five (5) feet.

A failure of the skimmer system piping would not uncover spent fuel nor interrupt fuel pool cooling since the fuel pool cooling water suction connections are located more than five (5) feet below the normal water level.

Draining or siphoning of the spent and new fuel pools via piping or hose connections to these pools or transfer canals is precluded by the location of the penetrations, limitations on hose length, and termination of piping penetrations flush with the liner. Hoses connected to temporary equipment used in the new and spent fuel pools are administratively controlled to prevent siphoning. The fuel pool cooling water return piping terminate at elevation 279 ft., 6 in. The spent fuel pool suction piping exists at 278 ft., 6 in. and the new fuel pool exits at 277 ft., 6 in.. Normal pool water level is 284 ft., 6 in, with the top of the spent fuel at approximately 260 ft. Skimmer suction piping exits the pools at elevation 285 ft., 3 in.

The reduction of the normal pool water level by approximately 5 ft. due to any postulated pipe failure will have no adverse impact on the capability of the cooling system to maintain the required temperature and it does not effect the required shield water depth for limiting exposures from the spent fuel. The slow heatup rate of the fuel pool would allow sufficient time to take any necessary action to provide adequate cooling using the backup provided while the cooling capability for the fuel pool is being restored.

Technical Specification 3.9.11 requires a minimum 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks whenever irradiated fuel assemblies are in a pool. The ability of 23 feet of water to remove 99 percent of the assumed 10 percent iodine gas activity released during the postulated fuel accident forms the bases of this Technical Specification. Technical Specification 3.9.11 requires all movement of fuel assemblies and crane operations with loads in the affected pool area be suspended and the water level restored to within its limit within four hours if the water depth over the stored fuel assemblies falls below 23 feet. These actions would prevent a fuel handling accident with less than 23 feet of water over irradiated fuel assemblies seated in the storage racks.

There are two alarms associated with low spent fuel pool water level. The first alarm occurs with approximately 24 feet of water over the stored fuel assemblies and provides the operators with sufficient warning and time to place all loads in a safe position prior to the water level decreasing to 23 feet. The second alarm occurs with approximately 22 feet of water over the stored fuel assemblies.

Normal makeup water to the fuel pool is supplied by the Seismic Category I refueling water storage tank. A backup system for filling the fuel pool is available through flexible hoses, ESW and RWST lines and their existing vent lines for emergency connection to the Emergency Service Water System, the Seismic Category I source of emergency makeup water.

As shown on Figure 9.1.3-2, valving and blind flange connections are provided at the suction and discharge side of the fuel pool cooling pumps for emergency connection of a spare cooling pump.

Makeup water is normally pumped to the pool by the fuel pool cooling pumps and may be pumped by the fuel pool cleanup loop pumps. Each of these pumps has the capacity to provide makeup water at a rate greater than the loss of water resulting from normal system leakage and evaporation.

Floor and equipment drain sumps and pumping systems are provided to collect and transfer FPCCS leakage to the Waste Management System. High level alarms are annunciated in the Control Room when high sump level is reached.

Fuel handling equipment is designed such that the equipment cannot fall into the pool under SSE conditions (Section 9.1.4). In addition, the Fuel Handling Building is tornado missile resistant (Section 3.5).

The new fuel pool and spent fuel pools are furnished with stainless steel liners. Although they are classified as non-Nuclear Safety, the fuel pool liners are designed and constructed to the applicable portions of the ASME Code, Section III and they are subject to the Quality Assurance Criteria of 10 CFR 50, Appendix B. Other portions of the fuel transfer system in the Fuel Handling Building which are in communication with the new and spent fuel pools; namely, the fuel transfer canal, the main fuel transfer canal and the fuel cask loading pit, are also furnished with stainless steel liners.

Although these liners are qualified to the same requirements as the fuel pool liners, it is impossible for leakage in these portions of the fuel transfer system to jeopardize the inventory of cooling water in the fuel pools due to a difference in floor elevation. These areas may also be isolated from the fuel pools by gates.

A Permanent Cavity Seal Ring (PCSR) has been installed in the annulus of the reactor cavity adjacent to the refueling cavity. The PCSR is furnished with eight hatch covers which are closed and tested prior to flood-up for refueling. The PCSR is classified as nuclear safety related, subject to the quality assurance provisions of 10CFR50 Appendix B. It is designed and constructed to the applicable portions of the ASME Code Section III, Subsection ND, but is not code stamped by an ANI.

Piping and components of the Fuel Pool Cooling and Cleanup System are designed to the applicable codes and standards listed in Section 3.9. Those portions of the FPCCS required to ensure cooling of the fuel pool are Safety Class 3, since their prolonged failure could result in the release to the environment of normally retained gaseous radioactivity. Piping in contact with fuel pool water is austenitic stainless steel.

Fuel pool nozzles shall be stainless steel Seismic Category I designed and fabricated to ASME Section III, Subsection No. ND. However, they are classified as NNS.

9.1.3.4 Inspection and Testing Requirements. Provisions are incorporated in the layout of the system to allow for periodic inspection, using visual and monitoring instrumentation. Equipment is arranged and shielded to permit inspection with limited personnel exposure.

Preoperational and startup tests as described in Section 14.2.12 were conducted in the FPCCS. Periodic tests are required as described in the Technical Specifications. Inservice inspection requirements are described in Section 6.6 and pump and valve testing will be performed as described in Section 3.9.6.

Prior to initial fill, vacuum box testing was performed on the major liner field joints normally exposed to water.

Components of the system were cleaned and inspected prior to installation. Demineralized water was used to flush the entire system. Instruments were calibrated and alarm functions checked for operability and setpoints during testing. The system was operated and tested initially with regard to flow points, flow capacity and mechanical operability.

Data will be taken periodically during normal system operation to confirm heat transfer capabilities, purification efficiency, and differential pressures across components.

TABLE 9.1.3-2

FUEL POOL COOLING AND CLEANUP SYSTEM PARAMETERS

Fuel Pool Heat Load, Equilibrium Temperature and Heat Inertia*

Fuel Pool Heat Load	
Incore Shuffle (RFO-7 Normal Case)	16.84×10^6 Btu/hr
Full Core Offload Shuffle (RFO-7 Normal Case)	35.06×10^6 Btu/hr
Post Outage Full Core Offload (Post-RFO-7 Abnormal Case)	35.87×10^6 Btu/hr
Fuel Pool Equilibrium Temperature**	
Incore Shuffle (RFO-7 Normal Case)	≤137°F
Full Core Offload Shuffle (RFO-7 Normal Case)	≤137°F
Post Outage Full Core Offload (Post-RFO-7 Abnormal Case)	≤137°F
Combined Spent and New Fuel Heat Pool Heat Inertia	
Incore Shuffle (RFO-7 Normal Case)	4.37°F hr
Full Core Offload Shuffle (RFO-7 Normal Case)	9.09°F hr
Post Outage Full Core Offload (Post-RFO-7 Abnormal Case)	9.30°F hr
Fuel Pool Heat Exchanger	
Quantity (per FPCCS)	2
Type	Shell and Two Pass Straight Tube
UA (Design per Heat Exchanger), Btu/hr.-F	21.1×10^5
Shell Side (Component Cooling Water) - Design	
Inlet temperature, F	105
Outlet temperature, F	110
Design flowrate, lb./hr.	2.68×10^6
Design pressure, psig	150
Design temperature, F	200
Material	Carbon Steel

*Based on operation through end-of-Cycle 8 with the bounding heat load from post RFO-7 plus additional spent fuel shipments.

**Administrative controls are placed on the minimum cooling time prior to transfer of irradiated fuel from the core to the storage facility to maintain the pools at less than or equal to 137°F. The minimum decay time prior to movement of irradiated fuel in the reactor vessel will address both radiological and decay heat considerations.

TABLE 9.1.3-2 (Continued)

Tube Side (Fuel Pool Water) - Design	
Inlet temperature, F	120
Outlet temperature, F	113
Design flowrate, lb./hr.	1.88×10^6
Design pressure, psig	150
Design temperature, F	200
Material	Stainless Steel
Fuel Pool Cooling Pump	
Quantity	2
Type	Horizontal Centrifugal
Design flowrate, gpm	4560
TDH, ft. H20	98
Motor horsepower	150
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless Steel
New Fuel Pool (Pool A or New Fuel Pool Unit 1)	
Volume, gallons (at normal level, elevation 284.5 feet)	142,272
Boron concentration, ppm (minimum)*	2,000
Liner material	Stainless Steel
Spent Fuel Pool (Pool B or Spent Fuel Pool Unit 1)	
Volume, gallons, (at normal level, elevation 284.5 feet)	388,800
Boron concentration, ppm (minimum)*	2,000
Liner material	Stainless Steel

*The actual boron concentration will be determined by the plants' Technical Specifications for Refueling.

TABLE 9.1.3-2 (Continued)

Fuel Pool Demineralizer Filter	
Quantity (per FPCCS)	1
Type	Back 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
Fuel Pool Demineralizer	
Quantity	1
Type	Flushable
Design pressure, psig	400
Design temperature, F	200
Design flowrate, gpm	325
Volume of resin (each), ft^3	85
Fuel Pool and Refueling Water Purification Filter	
Quantity	1
Type	Back Flushable
Design pressure, psig	400
Design temperature, F	200
Design flowrate, 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

TABLE 9.1.3-2 (Continued)

Fuel Pool Strainer	
Quantity	1
Type	Basket
Design flowrate, gpm	4560
Design pressure, psig	150
Design temperature, F	200
Maximum differential pressure across the strainer element above flow (clean), psi	1.4
Mesh	40
Fuel Pool Skimmer Pump Suction Strainer	
Quantity	1
Type	Duplex Basket
Design pressure, psig	150
Design temperature, F	200
Design flowrate, gpm	385
Maximum differential pressure across strainer element at rated flow (clean), psi	5
Maximum differential pressure across strainer element prior to removing, psi	60
Mesh	100
Fuel Pool Skimmer Filter	
Quantity	1
Type	Back Flushable
Design pressure, psig	400
Design temperature, F	200
Design flowrate, gpm	400
Maximum differential pressure across filter element at rated flow (clean), psi	5
Maximum differential pressure across filter element prior to removing, psi	60

TABLE 9.1.3-2 (Continued)

Fuel Pool Skimmer Pump		
Quantity	1	
Design flowrate, gpm	385	
TDH, ft. H20	210	
Motor horsepower	40	
Design pressure, psig	150	
Design temperature, F	200	
Material	Stainless Steel	
Fuel Pool and Refueling Water Purification Pump		
Quantity	2	
Type	Vertical In-line Centrifugal	
Design flowrate, gpm	325	
TDH, ft. H20	320	
Motor horsepower	60	
Design pressure, psig	150	
Design temperature, F	200	
Material	Stainless Steel	
Fuel Pool Cooling and Cleanup System Piping and Valves		
Material	Stainless Steel	
Design pressure, psig	150	
Design temperature, F	200	
Fuel Pool Skimmers	Quantity	gpm each
Spent Fuel Pool (Pool B)	5	35
New Fuel Pool (Pool A)	3	30
Fuel Transfer Canal (2 each canal - south and north)	4	25
Main Fuel Transfer Canal (per FPCCS)	1	20
Cask Loading Pool	1	50

9.1.4 Fuel Handling System

9.1.4.1 Design Bases. The Fuel Handling Building contains one new fuel pool and three spent fuel pools with a transfer canal system that permits transfer of fuel between pools and the reactor cavity in the Containment Building. References to "new and spent fuel pools" in this section refer to the operable new fuel pool and operable spent fuel pool located in the south end of the Fuel Handling Building. The new fuel pool and one spent fuel pool are currently completed and inservice. The additional two spent fuel pools will be made operational when required. The Fuel Handling System (FHS) is designed in conformance with Regulatory Guide 1.13 as detailed in Section 1.8.

The Fuel Handling System will provide the following services on SHNPP:

a) provides the means for safely moving the fuel as necessary to accomplish receipt and storage of new and spent fuel, refueling, receiving shipments of offsite spent fuel, and shipment of spent fuel to offsite locations.

b) provides the means for safely preparing the plant facilities for fuel movement, such as placement of fuel transfer canal gates in appropriate positions, dismantling and replacing reactor vessel components to allow for refueling and placement of portable barriers for safe spent fuel cask handling.

c) provides the means for safely transferring spent fuel among all fuel pools.

d) provides shielding for protection of personnel from excessive radiation exposure during refueling, inspection, and fuel storage.

e) provides that either:

1) a load drop resulting from a single electrical or lifting cable failure is precluded, or;

2) the consequences of a load drop can be accommodated without affecting the ability to bring the plant to a safe shutdown condition or to control the release of significant amounts of radioactive material.

f) is designed such that maximum design load on the wire rope hoisting cables shall not exceed 1/5 ultimate strength of the cables.

g) provides appropriate containment isolation boundaries for containment penetration.

h) is designed such that lifting devices have appropriate administrative controls, interlocks and stopping capability.

i) is designed such that fuel lifting and handling equipment and structures will not fail in such a manner as to damage Seismic Category I equipment or structures in the event of an SSE.

Structures, systems, and components designed as Seismic Category I are shown in Table 3.2.1-1. Structures, systems, and components which could damage safety-related equipment upon failure are designed to withstand an SSE event without causing such damage. Components that are designed for Safety Class 1, 2, or 3 are shown in Table 3.2.1-1.

9.1.4.1.1 Fuel transfer decay heat. The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. The operable fuel pools are connected to the pool cooling and clean-up systems, which are discussed in detail in Section 9.1.3.

9.1.4.1.2 Fuel transfer radiation shielding. Adequate shielding from radiation is provided during reactor refueling by transferring and storing spent fuel underwater and maintaining a safe shielding depth of water above the fuel assemblies during refueling. This permits visual control of the operation at all times while maintaining acceptable radiation levels for periodic occupancy of the area by operating personnel.

9.1.4.2 System Description.

9.1.4.2.1 System. The Fuel Handling System consists of the equipment and associated structures used to handle fuel from the time of receipt until it leaves the plant and the handling equipment used to prepare the reactor to discharge and receive fuel.

The equipment consists of:

a) Containment building overhead polar crane.

b) Manipulator crane.

c) Spent fuel bridge crane.

d) Spent fuel cask handling crane.

e) Auxiliary crane.

f) Fuel handling tools and fixtures.

g) Fuel transfer system.

h) Fuel racks.

i) New fuel elevator.

The following areas are associated with the fuel handling equipment:

a) Refueling cavity.

b) Fuel pools and fuel transfer canal system.

c) New fuel storage area.

d) Spent fuel cask loading pool and decontamination area.

Refer to Figures 1.2.2.55 through 1.2.2-59 for general arrangements of the Fuel Handling Building.

The associated fuel handling structures may be divided into two areas:

a) The refueling cavity which is flooded only during shutdown for refueling.

b) The new and spent fuel pools and fuel transfer canal system.

9.1.4.2.2 Components.

9.1.4.2.2.1 Reactor vessel head lifting device. The reactor vessel head lifting device consists of a welded and bolted structural steel frame which enables the overhead polar crane to lift the head and store it during refueling operations. This device is part of the Integrated Reactor Vessel Head (IRVH).

9.1.4.2.2.2 Reactor internals lifting device. The reactor internals lifting device is a structural frame mechanism which provides the means of gripping the upper and lower internal packages to transmit the lifting load to the crane (refer to Figure 9.1.4-1). By the use of auxiliary brackets, the assembly is guided onto the internal packages. Attachment is accomplished by manually connecting the assembly to the internals with handling tools operated from the internals lifting rig platform. The upper internals are stored in the flooded refueling cavity during refueling. Although their removal is not required for refueling, the lower internals may be stored in the flooded refueling cavity when required.

9.1.4.2.2.3 Manipulator crane. The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water (refer to Figure 9.1.4-2). The bridge spans the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position.

Controls for the manipulator crane are mounted on a console on the trolley. The bridge and trolley are positioned by a combination of a video position indication system and visual observation.

This system consists of the following: two video cameras, (one mounted on the bridge truck and the other on the trolley), two sets of position plates, (one set mounted near the guide rail for bridge position and the other on the bridge for the trolley position), a CRT monitor and selector switch, (both mounted in the control console). The drives for the bridge, trolley, and hoist are variable speed and include separate slow speed jog switches for each.

The manipulator crane will not collapse nor become disengaged as a consequence of an SSE.

9.1.4.2.2.4 Spent fuel bridge crane. The spent fuel bridge crane, as shown on Figure 9.1.4-3, is a wheel-mounted walkway spanning the width of the Fuel Handling Building, which carries an electric monorail hoist on an overhead structure. The monorail hoist has access to all spent and new fuel pools, as well as interconnecting transfer canals. The fuel assemblies are moved within the fuel pools by means of a long-handled tool (refer to Figure 9.1.4-4) suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth. The spent fuel bridge crane will not drop its load nor leave the rails as a consequence of an SSE. The capacity of the crane is 2,500 lbs; the approximate weight of the handling tool and a fuel assembly is 2,000 lbs. The capacity of the spent fuel bridge crane precludes excessive loads from being carried over spent fuel storage area.

The spent fuel bridge crane hoist is equipped with a load monitor preset to prevent hoist operation at a load of 200 lbs above the weight of a fuel assembly with RCCA and handling tool. Changing of the set points to lift heavier loads will be under administrative control.

9.1.4.2.2.5 Fuel Transfer System. The Fuel Transfer System includes an underwater conveyor car running on tracks extending from the refueling cavity through the transfer tube and into the fuel transfer canal and an upending frame at each end of the transfer tube (refer to Figure 9.1.4-5). To remove a fuel assembly from the reactor, the upending frame in the refueling cavity receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is then lowered to a horizontal position for passage through the transfer tube and raised to a vertical position by the upending frame in the fuel transfer canal. The hoist on the spent fuel bridge then takes the fuel assembly to a position in the spent fuel racks via the fuel transfer canals.

To seal the reactor Containment during Unit operation, a blind flange is bolted on the end of the transfer tube in the refueling cavity inside containment, and a manually operated valve is locked closed in the fuel transfer canal in the Fuel Handling Building (Section 6.2.4). The transfer tube and the blind flange are designed to Seismic Category I requirements.

9.1.4.2.2.6 Rod cluster control assembly (RCCA) changing fixture. The following description applies when performing incore shuffles. If the complete core is off-loaded to the Fuel Handling Building, this equipment is not utilized. An RCCA changing fixture is mounted on the refueling cavity wall for transferring RCCA's from fuel assemblies removed from control positions and inserting RCCA's into the fuel assemblies to be placed in the control positions (Figure 9.1.4-6). The fixture consists of two main components: a guide tube mounted to the wall for containing and guiding the RCCA and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch which grips the RCCA and lifts it out of the fuel assembly. By repositioning the carriage, another fuel assembly is brought under the guide tube; and the gripper lowers and releases the RCCA. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

9.1.4.2.2.7 Spent fuel cask handling crane and auxiliary crane. The spent fuel cask handling crane (150-ton) transfers the spent fuel cask between the railroad car and the spent fuel cask loading pool, (refer to Figures 1.2.2-55 through 1.2.2-59). Design of the Fuel Handling Building and the spent fuel cask handling crane prevents the possibility of the cask passing over or falling into any fuel pool. Permanent mechanical stops, which will withstand the impact of the crane at maximum operating speed, are provided to limit the crane movement so that travel of the center of the main hook is limited to 12 in. south of the centerline of the cask loading pool. Additionally, only the micro drives will be functional in the last 5 ft. of crane travel in the southerly direction as controlled by limit switch. In the unlikely event that the crane comes in contact with the mechanical stops while at maximum operating speed, the maximum swing of the bottom of the cask from its normal position in the vertical plane will be 14.5 inches in the southerly direction. When the cask reaches this deflected position, it is still entirely over the cask pool. Therefore, if dropped while in this extended position, the cask will not come in contact with spent fuel in the fuel pools.

The spent fuel cask handling crane is equipped with limit switches which limit main hook vertical travel to ensure the shipping cask could never fall more than 30 feet through air to any load-bearing surface and will not be raised more than 6 in. above the operating floor.

Two independent systems are provided to prevent the centerline of the main hook of the spent fuel cask crane from coming within 10 ft. 6 in. of the west edge and 15 ft. of the north edge of the nearest spent fuel pool. A combination of limit switches and mechanical stops restrict the crane from the spent fuel pool area. Travel of the center line of the main hook on the cask crane is restricted to the shaded area as shown on Figure 9.1.4-7 by a combination of limit switches and mechanical stops. During cask handling, the center line of the main hook is further restricted under administrative control to the path cross hatched on the figure. A removable barrier is provided with its west face in line with the east edge of the cask unloading pool on top of the dividing wall between that pool and the cask head storage area. The function of the removable barrier is to prevent the cask, in the remote chance of being dropped on top of the dividing wall between the cask loading pool and cask head and yoke storage area, from toppling over and falling into a currently inoperable fuel pool. The dropping cask, after landing on the dividing wall, may start to topple over and strike the barrier. The barrier is designed to withstand the striking force, thus preventing the cask from falling into a currently inoperable fuel pool. The removable barrier is 21 feet 6 inches in overall height. It is set in place by being lowered into a 4 feet deep recess in the concrete floor; therefore, the installed height is 17 feet 6 inches measured from the operating floor. The removable barrier is not used as a mechanical stop.

The auxiliary crane will be used for the installation and re-removal of this barrier.

Figures 9.1.4-7 through 9.1.4-12 show the envelope of travel of the main hook of the spent fuel cask handling crane as controlled by design and administrative control of the crane, and within the main hook envelope, the area to which cask travel will be restricted by administrative control.

There is no safety-related equipment within the possible area of main hook (and, therefore, fuel cask) travel, either on the operating floor level or on floors beneath.

Additionally, a review for the consequences on the building structure due to dropping of the cask crane load block has been performed. The floors within the load block travel envelope will withstand a postulated drop of the load block from the maximum height to which it can be raised with the following exceptions:

a) The stairs near column line 73 will fail, but no safety-related components will be affected.

b) The floor at elevation 286.00 ft. north of column line 73Z and the floor at elevation 261.00 ft. in the new fuel containers storage area will sustain damage, however, the effect is considered to be local. Only non-safety related components will be affected by a load drop in this vicinity.

The auxiliary crane (design capacity of 12 tons) operates on the same runway as the spent fuel cask handling crane as shown in Figures 1.2.2-55 through 1.2.2-59. Two independent systems are provided to prevent the two cranes from coming in contact with each other. Design of each system provides that the auxiliary crane can operate in the common operating area only when the cask crane is in its parking position which is at a safe distance away from the end of travel of the auxiliary crane. While the cask crane is operating, the auxiliary crane is limited to operate at a safe distance away from the common area.

A redundant supporting system is provided on the auxiliary crane in regard to hook, reeving, and braking mechanisms. Provisions are made to manually move the crane to a laydown area for emergency manual lowering of the load. A detailed description of the auxiliary crane is given in Table 9.1.4-1.

Both cranes are capable of retaining the maximum load during an SSE although the crane may not be operable after the seismic event. The bridge and trolley are provided with means for preventing them from leaving their runways with or without hook load during operation or under any postulated seismic event. There is no other lifting device that can carry excessive loads over the fuel storage areas.

9.1.4.2.2.8 Containment Circular Bridge Crane

The overhead crane in the Containment (250 ton/50 ton) used for reactor servicing operations is of the polar configuration, and is seated on a girder bracketed off the containment wall. The crane is capable of retaining a 177.5-ton lifted load (weight of integrated reactor vessel head with lifting rig, which is the weight used by Westinghouse in the head drop analysis and is the heaviest component to be lifted during refueling operations) during an OBE or SSE, although the crane may not be operable after the seismic event. The bridge and trolley are prevented from leaving their runways with or without the 177.5 ton lifted load during operation or under any seismic event.

The centerline of the crane is offset from the reactor vessel centerline to assure the alignment of lifting devices with all possible loads and to provide clearance for containment spray header piping risers which run vertically along the containment liner.

The consequences of dropping the Integrated Reactor Vessel Head (IRVH) during preparation for or after completion of fuel handling has been analyzed. A summary of the assumption and results of the analysis follows:

The worst case drop scenario is evaluated; normally this is the concentric drop of the IRVH onto the vessel. It is pointed out that the fuel in the RPV will not suffer significant impact damage affecting the integrity of the fuel rods in this event. Also, it is determined that the RPV and primary shield wall (PSW) supporting the RPV (see Section 3.8.3.4.1) would remain intact, the reactor vessel primary nozzles would not be stressed above allowable limits and the reactor coolant loop piping and the essential auxiliary piping connected thereto remain capable of continued circulation of borated water at the specified flow rate. Therefore, the offsite doses would not approach 1/4 of the 10 CFR 100 limits in the concentric drop scenario and would be characteristically less for a non-concentric drop since the refueling cavity concrete floor would absorb part of the load.

During preparation for, and after completion of fuel handling, the IRVH will be transported to and from the laydown area in accordance with the pre-determined load path. The lifting of the IRVH to and from the laydown area is limited by administrative controls to a maximum of 12 in. above the operating floor. (In the laydown area the IRVH is raised above 12 in. to be placed upon its stand.) A drop from this strictly limited height onto the massive concrete and steel structure is not likely to have any serious consequences. Nevertheless, additional protection for required safe shutdown equipment is afforded by redundancy since the dropped IRVH could only damage the limited amount of equipment which is directly below it.

In all of the IRVH drops scenarios considered, the integrity of the vessel was never jeopardized nor was adequate make up water and cooling capacity interrupted such that the fuel could be uncovered. Furthermore, rapid containment isolation is provided with prompt automatic actuation on high radiation so that any unexpected releases result in doses well below 1/4 of the 10 CFR 100 limits taking into account delay times in detection and actuation.

9.1.4.2.2.9 Spent fuel basket storage. The north fuel transfer canal (former Unit No. 2 and No. 3 fuel transfer canal) is utilized as a permanent storage area for spent fuel shipping baskets. The baskets are inserted into the IF 300 cask for shipping spent fuel assemblies. Maximum number of baskets to be stored in the canal is 2 PWR and 4 BWR. The baskets will be stored under water and may be positioned anywhere within the canal except during movement of fuel assemblies anywhere within through the canal. At which time the baskets will be stored east of 4'-6" east of column Line M. Storage of the basket within the canal will be freestanding. If the baskets topple over during a seismic event, no damage to the structure or liner will result. The fuel baskets may be damaged as a result of toppling over. However, when stored in the transfer canal, the baskets do not perform any safety-related function. The auxiliary crane will be utilized to transfer the baskets between the cask loading pool and fuel transfer canal and anywhere within the canal.

9.1.4.2.3 Fuel handling description (new and spent). New fuel assemblies received for initial refueling are removed one at a time from the shipping container and moved to the new fuel assembly inspection area. After inspection, the acceptable new fuel assemblies are stored in the racks in the new or spent fuel pools.

New fuel assemblies received for subsequent refueling may, if found acceptable by inspection, be stored either dry in racks in the new fuel inspection area or in racks in any operational fuel pool.

Should the need exist in the future, spent fuel from other nuclear plants in the CP&L system would be brought to the SHNPP site in an approved shipping cask. The cask would be placed in the flooded shipping cask pool. The spent fuel would then be removed from the cask and transported to the storage racks. This procedure would be carried out with the spent fuel assemblies totally submerged.

The fuel handling equipment handles the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a container for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat.

The associated fuel handling structures are generally divided into two areas: the refueling cavity which is flooded only during a plant shutdown for refueling, and the fuel pools and fuel transfer canals. The refueling cavity and the Fuel Handling Building are connected by a fuel transfer tube which is fitted with a blind flange on the Containment end and a gate valve on the Fuel Handling Building end. The blind flange is in place except during refueling to ensure containment integrity. Fuel is carried through the tube on an underwater transfer car.

Fuel is moved between the reactor vessel and the refueling cavity by the manipulator crane. A rod cluster control changing fixture is located in the refueling cavity for transferring control elements from one fuel assembly to another. The fuel transfer system is used to move fuel assemblies between the Containment Building and the Fuel Handling Building. After a fuel assembly is placed in the fuel upender, the lifting arm pivots the fuel assembly to the horizontal position for passage through the fuel transfer tube. After the transfer car transports the fuel assembly through the transfer tube, the lifting arm at the end of the tube pivots the assembly to a vertical position so that the assembly can be lifted out of the fuel upender.

In the Fuel Handling Building, fuel assemblies are moved about by the spent fuel bridge crane or (new fuel only) the auxiliary crane. When lifting irradiated fuel assemblies, the spent fuel bridge crane uses a long-handled tool to ensure that sufficient radiation shielding is maintained. Initially, a short tool is used to handle new fuel assemblies, (see Figure 9.1.4-13) but the new fuel elevator is used to lower the assembly to a depth at which the bridge crane, using the long-handled tool, can place the new fuel assemblies into the storage racks.

Administrative procedures will ensure that no irradiated fuel (outside of sealed casks) will be handled or transported inside the FHB unless the operating floor equipment hatch to the unloading area is in place.

Decay heat, generated by the spent fuel assemblies in the fuel pools, is removed by the Spent Fuel Pool Cooling and Cleanup System, which is described in Section 9.1.3.

After a sufficient decay period, the spent fuel assemblies may be removed from the fuel racks and loaded into the spent fuel shipping cask for removal from the site.

9.1.4.2.4 New Fuel Receiving and Inspecting Procedure

a) New fuel arrives at the north end of the Fuel Handling Building on truck or rail car.

b) The airtight door is opened to admit the carrier and closed behind it.

c) The equipment hatch cover is removed.

d) The new fuel containers are lifted from the carrier by the auxiliary crane or the cask crane and placed on the operating level or in the new fuel inspection area. If the outside airtight door is to be open at any time, the equipment hatch cover will be replaced first.

e) The new fuel container lids are unbolted and removed by the auxiliary crane and stored. This may be performed on the operating level or in the new fuel inspection area.

f) The fuel assemblies are prepared for upending.

g) The fuel assemblies are upended using the auxiliary crane and lifted with the spent fuel bridge crane or auxiliary crane to the operating level and inspected.

h) After inspection, acceptable new fuel assemblies are transported to any operational fuel pool, or stored dry in rack(s) in the new fuel inspection area.

i) Unacceptable new fuel may be temporarily stored dry in a rack/racks in the new fuel inspection area or returned to the shipping containers and stored in the new fuel inspection area until corrective action is taken.

j) The new fuel container lids are returned to the containers by the auxiliary crane and bolted.

k) The equipment hatch cover is removed.

l) The empty new fuel containers are lifted by the auxiliary crane and loaded back on the carrier. The equipment hatch cover is replaced.

m) The airtight door is opened and the carrier, loaded with empty new fuel containers, leaves the building and the airtight door is closed.

9.1.4.2.5 Offsite Spent Fuel Receiving Procedure

a) Offsite spent fuel arrives at the north end of the Fuel Handling Building in approved shipping containers by truck or railcar.

b) The airtight door is opened to admit the vehicle and closed behind it.

c) The equipment hatch cover is removed.

d) The cask is prepared for lifting.

e) The cask is lifted by spent fuel cask handling crane and transported to the decontamination or work area. The equipment hatch cover is replaced.

f) Cask is prepared for pool entry.

g) Cask loading pool is flooded and the gate removed.

h) Removable barrier is put in place by the FHB auxiliary crane, or verified to be in place. The removable barrier is designed to withstand DBE seismic loads in accordance with Positions C.2 and C.4, Regulatory Guide 1.29.

i) Cask is transported and lowered into the cask loading pool.

j) Cask head is removed and stored in an appropriate location.

k) Move all fuel baskets that are being stored in the north fuel transfer canal to a position 4'-6" east of column line 'M'.

l) Fuel is removed from the cask using the appropriate long-handled spent fuel tool (PWR or BWR).

m) The spent fuel is transported, using the spent fuel bridge crane, to its pre-assigned storage location.

n) After the cask is emptied the head is returned to the cask and replaced.

o) Cask is lifted by the spent fuel cask handling crane and placed in the decontamination area.

p) Removable barrier is placed in storage, or left in place.

q) Cask is prepared for shipment and decontaminated to acceptable levels.

r) The equipment hatch cover is removed.

s) Cask is lifted from the decontamination area and returned to the truck or railcar for removal and the equipment hatch cover is replaced.

t) The airtight door is opened and the vehicle, loaded with the empty cask, leaves the building and the airtight door is closed.

9.1.4.2.6 Spent fuel shipping procedure.

a) A truck or railcar arrives at the north end of the Fuel Handling Building carrying an empty approved spent fuel shipping container.

b) The airtight door is opened to admit the vehicle and closed behind it.

c) The equipment hatch cover is removed.

d) The cask is prepared for lifting.

e) The cask is lifted by spent fuel cask handling crane and transported to the decontamination or work area. The equipment hatch cover is replaced.

f) Cask is prepared for pool entry.

g) Cask loading pool is flooded and the gate removed.

h) Removable barrier is put in place by the FHB auxiliary crane, or verified to be in place.

i) Cask is transported and lowered into the cask loading pool.

j) Cask head is removed and stored in an appropriate location.

k) Move all fuel baskets that are being stored in the north fuel transfer canal to a position 4'-6" east of column Line 'M'.

l) Spent fuel is loaded into the cask using the appropriate long-handled spent fuel tool (PWR or BWR) and the spent fuel bridge crane.

m) After cask is loaded, the head is returned to the cask and replaced.

n) Cask is lifted by the spent fuel cask handling crane and placed in the decontamination area.

o) Removable barrier is placed in storage, or left in place.

p) Cask is prepared for shipment and decontaminated.

q) The equipment hatch cover is removed.

r) Cask is lifted from the decontamination area and returned to the truck or railcar for removal and the equipment hatch cover is replaced.

s) The airtight door is opened and the vehicle, loaded with the full cask, leaves the building and the airtight door is closed.

9.1.4.2.7 Refueling procedure.

9.1.4.2.7.1 Preparation.

a) The reactor is shut down and cooled to ambient conditions with a final $K_{eff} \leq 0.95$.

b) A radiation survey is made and if the levels are sufficiently low, the containment is entered.

c) The reactor vessel coolant level is lowered slightly below the reactor vessel flange.

d) IRVH cables are disconnected and removed to storage.

e) Reactor vessel head insulation and instrument leads are removed.

f) The fuel transfer tube blind flange is removed and the refueling cavity drain valves are closed.

g) Checkout of the fuel transfer system and manipulator crane is started.

h) The reactor vessel head nuts are loosened with the hydraulic tensioner.

i) The rector vessel head studs and nuts are removed to storage. (Stuck studs are a possible exception.) (See Section 1.8)

j) Guide studs are installed in at least two and typically three reactor vessel flange holes, and the remainder of the holes are plugged.

k) The refueling cavity underwater lights are installed.

l) The reactor vessel permanent cavity seal ring hatch covers are closed and tested.

m) Final preparation of tools is made. Checkout of the fuel transfer system is completed. Manipulator crane is parked.

n) The reactor vessel head is unseated, raised approximately one inch, and checked for levelness.

o) The reactor vessel integrated head is lifted slowly clear of the refueling pool cavity.

p) The refueling cavity is filled.

q) The control rod drive shafts are unlatched from the spider.

r) The reactor vessel internals lifting rig is lowered into position and latched to the upper internals package.

s) The reactor vessel upper internals package and drive shafts are lifted out of the vessel and placed in the underwater storage rack.

t) Checkout of manipulator crane is complete (Core Index).

u) The core is now ready for refueling.

9.1.4.2.7.2 Refueling reassembly. The refueling sequence is now started with the manipulator crane. Refueling may be accomplished either by the transfer of the entire core to the storage facility, referred to herein as the Full Core Offload Shuffle or the transfer of only that portion of the core to be discharged to the storage facility, referred to herein as the Incore Shuffle. For the Full Core Offload, some partially spent fuel assemblies and the new fuel assemblies are added to the core. For the Incore Shuffle, some partially spent fuel assemblies have their positions changed and new assemblies are added to the core.

For fuel assemblies containing rod cluster control assemblies (RCCA), the refueling sequence is modified as required. For the Incore Shuffle, if a transfer of the RCCA between fuel assemblies is necessary, the assemblies are taken to the RCCA changing fixture for the exchange. For the Full Core Offload Shuffle, if a transfer of the RCCA between fuel assemblies is necessary, the assemblies are placed in storage racks in the fuel pools and the RCCA is exchanged using the portable RCCA change fixture attached to the spent fuel bridge crane. Such an exchange may be required whenever a spent fuel assembly containing an RCCA is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during refueling rearrangement.

9.1.4.2.7.3 Reactor reassembly.

a) The fuel transfer conveyor car is parked, and the refueling cavity is isolated from the fuel transfer canal by closing the manual gate valve on the FHB side.

b) The manipulator crane is parked.

c) The reactor vessel upper internals package is placed in the vessel. The reactor vessel internals lifting rig is unlatched and removed to storage.

d) The full-length control rod drive shafts are relatched to the RCCA spiders.

e) The old seal rings are removed from the reactor vessel head, the grooves cleaned, and new rings installed.

f) The water level in the refueling cavity is lowered just below the flange.

g) The flange surface is cleaned.

h) The reactor cavity is drained.

i) The reactor vessel head is positioned and seated onto the vessel flange.

j) The guide studs are removed to their storage rack. The stud hole plugs are removed.

k) The head studs are placed and retorqued.

l) The refueling cavity drain holes are opened, and the flange for the fuel transfer tube is replaced.

m) Electrical leads are reconnected to the IRVH.

n) Vessel head insulation and instrumentation leads are replaced.

o) The permanent cavity seal ring hatch covers are opened.

p) A hydrostatic test is performed on the reactor vessel.

q) Control rod drives are checked.

r) Pre-start-up tests are performed.

9.1.4.2.8 Codes and standards.

a) Cranes - Crane Manufacturers Association of America (CMAA) Specification No. 70 and/or AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

b) Structures - ASME Code, Section III, Appendix XVII.

c) Electrical - Applicable standards and requirements of the National Electric Code, NFPA 70, and NEMA Standards MAI and ICS for design installation and manufacturing.

d) Materials - Main load-bearing materials to conform to the specifications of the ASTM, ASME, or AISC Standards.

e) Safety - i -OSHA Standards, 29CFR1910 and 29CFR1926, including load-testing requirements.

ii - ANSI N18.2

iii - Regulatory Guide 1.29 and GDC 61 and 62.

iv - ANSI B30.2, "Safety Standards for Overhead and Gantry Cranes."

f) Fuel Transfer Tube: ASME Section III, Code Class 2.

9.1.4.3 Safety Evaluation. The extent of compliance of the fuel handling system with Regulatory Guide 1.13 is discussed in Section 1.8.

Movement of heavy loads over safety-related equipment is controlled in accordance with plant procedure MMM-020 "Operation, Testing, Maintenance and Inspection of Cranes and Special Lifting Equipment." This procedure was reviewed as part of the requested actions for NRC Bulletin 96-02 "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment" (Reference 9.1.4-1) to confirm that the program continues to be implemented within the licensing basis.

9.1.4.3.1 Fuel handling equipment. Electrical interlocks and limit switches on the bridge and trolley drives of the manipulator crane protect the equipment. In an emergency, the bridge, trolley, and winch can be operated manually using a hand-wheel on the motor shaft. Manual operation of the bridge and trolley, with appropriate administrative controls in place, is also acceptable when used to move fuel to and from open water to avoid fuel assembly interaction and possible damage. (See Westinghouse Specification F-5, "Instructions, Precautions, and Limitations for Handling New and Partially Spent Fuel Assemblies.")

The manipulator crane design includes the following provisions to ensure safe handling of fuel assemblies.

Safety Interlocks

Operations which could endanger the operator or damage the fuel are prohibited by mechanical or fail-safe electrical interlocks or by redundant electrical interlocks. All other interlocks are intended to provide equipment protection and may be implemented either mechanically or by electrical interlock, not necessarily fail-safe.

Fail-safe electrical design of a control system interlock may be applied according to the following rules.

a) Fail-safe operation of an electrically operated brake is such that the brake engages on loss of power.

b) Fail-safe operation of an electrically operated clutch is such that the clutch disengages on loss of power.

c) Fail-safe operation of a relay is such that the de-energized state of the relay inhibits unsafe operation.

d) Fail-safe operation of a switch, termination, or wire is such that breakage or high resistance of the circuit inhibits unsafe operation. The dominant failure mode of the mechanical operation of a cam-operated limit switch is sticking of the plunger in its depressed position. Therefore, use of the plunger-extended position (on the lower part of the operating cam) to energize a relay is consistent with fail-safe operation.

e) Fail-safe operation of an electrical comparator or impedance bridge is not defined.

Those parts of a control system interlock required to be fail-safe which are not or cannot be operated in a fail-safe mode as defined in these rules, may be supplemented by a redundant component or components to provide the requisite protection.

a) When the gripper is engaged, the machine shall not traverse unless the guide tube is either in its full up position or jog permissive have been selected. Then the machine is allowed to traverse in a controlled manner when used to avoid fuel assembly interaction and possible damage as described above.

b) When the gripper is disengaged, the machine shall not traverse unless the gripper is withdrawn into the mast.

c) Vertical motion of the guide tube shall be permitted only in a controlled area over the reactor (avoiding the vessel guide studs), a fuel transfer system, or rod cluster control changing fixture.

d) Traverse of the trolley and bridge shall be limited to the areas of item c and a clear path connecting those areas.

e) A key-operated interlock bypass switch shall be provided to defeat interlocks a through d to allow operation of an inspection camera on the gripper.

f) The gripper shall be monitored by limit switches to confirm operation to the fully engaged or fully disengaged position. An audible and a visual alarm shall be actuated if both engaged and disengaged switches are actuated at the same time or neither is actuated. A time delay may be used to allow for recycle time of normal operation.

g) The loaded fuel gripper shall not release unless it is in its down position in the core, or in the fuel transfer system or rod cluster control changing fixture, and the weight of the fuel is off the mast.

h) Raising of the guide tube shall not be permitted if the gripper is disengaged and the load monitor indicates that it is still attached to the fuel assembly.

i) Raising of the guide tube shall not be permitted if the hoist loading exceeds the allowable limit set in accordance with the Westinghouse Specification F-5 "Instructions, Precautions, and Limitations for Handling New and Partially Spent Fuel Assemblies."

j) Lowering of the guide tube shall not be permitted if slack cable exists in the hoist.

k) The guide tube shall be prevented from rising to a height where there is less than the safe shielding depth of water over the fuel assemblies.

l) The guide tube shall travel only at a controlled speed of about 2 fpm when: 1) the bottom of the fuel begins to enter the core, and 2) the gripper approaches the top of the core. In addition, just above those points, the guide tube shall automatically stop lowering, and shall require acknowledgement from the operator before proceeding.

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:

1. 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.

2. 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.

3. Deleted by Amendment No. 46.

4. 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.

5. 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 engaged gripper is in the full up position or the disengaged gripper is withdrawn into the mast, or the manipulator crane is over the core.

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

9.1.4.3.2 Overhead cranes. Overhead cranes used in refueling and fuel handling operations include the 250/50-ton overhead polar crane, the 150-ton spent fuel cask handling crane, and the 12-ton (design load) auxiliary crane. These cranes are classified as non-nuclear safety (NNS) since they neither provide nor support any safety system function.

a) Overhead Polar Crane

The crane is used for removal of the Integrated Reactor Vessel Head and the upper internals package during the refueling shutdown. This crane is provided with seismic restraints to prevent derailment in the event of an SSE or OBE.

A discussion of consequences from dropping the Integrated Reactor Vessel Head is noted in Section 9.1.4.2.2.8. The consequences of various postulated accidents involving the dropping of the reactor vessel upper internals are discussed in Appendix 9.1A, Heavy Loads Analysis.

b) Spent Fuel Cask Handling Crane

This 150-ton crane is provided for handling the spent fuel shipping cask. Crane design and building arrangement preclude travel of this crane hook over the fuel pools. This crane will maintain its structural integrity and hold its load under the dynamic loading conditions of the SSE as described in Section 9.1.4.2.2.7. A postulated drop of the fuel cask will not cause damage to spent fuel and safety-related equipment.

The consequences of load dropping is noted in Section 9.1.4.2.2.7.

c) Auxiliary Crane

The auxiliary crane is used for handling of the removable barrier, pool gates, fuel racks and other miscellaneous items weighing less than 10 tons.

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. 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 percent of drive rated speed to set the holding brake and stop the load.

In the unlikely event of a phase reversal, a time-delay reverse phase relay actuates such that the crane drives cannot operate. In the event of a phase reversal during hoist motor operation, the time-delay reverse phase relay will 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, fuel-handling equipment to be used during the refueling outage 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:

1. Manipulator crane, spent fuel bridge crane, rod cluster control changing fixture (if used), and new fuel elevator.

The minimum acceptable initial test shall include the following:

a. Manipulator Crane and Spent Fuel Bridge Crane shall be load tested at 125 percent of the rated load.

b. The equipment shall be checked for proper functional and running operation.

The following maintenance and checkout tests are recommended to be performed prior to using the equipment:

a. Visually inspect for loose or foreign parts. Keep free of dirt and grease.

b. Lubricate exposed gears with proper lubricant.

c. Inspect hoist cables for worn or broken strands.

d. Perform operational checks of limit switches and limit switch actuators for proper functional operation.

e. Check the equipment for proper functional and running operation.

2. Reactor vessel head lifting device and reactor internals lifting device.

The minimum acceptable test shall include the following:

a. The devices shall be load tested to 125 percent of the rated load.

b. The devices shall be assembled to ensure proper component fit up.

The following maintenance and checkout tests are recommended to be performed prior to using the tools:

a. Visually inspect for loose or foreign parts or damaged surfaces.

b. Visually inspect all engagement surfaces and lubricate with proper lubricant.

c. On the reactor internals lifting device, check for the proper functioning of the engagement and protective rig operators.

3. New fuel assembly handling tool and spent fuel assembly handling tool

The minimum acceptable test shall include the following:

a. The tools shall be load tested to 125 percent of the rated load.

b. The tools shall be checked for proper functional operation.

The following maintenance and checkout tests are recommended to be performed prior to using the tools.

a. Visually inspect the tools for dirt, loose hardware, and for any signs of damage such as nicks and burns.

b. Check the tools for proper functional operation.

4. Fuel transfer system

The minimum acceptable test shall include the following:

a. The system shall be checked for proper functional and running operation.

The following maintenance and checkout tests are recommended to be performed prior to using the tools.

a. Visually inspect for loose or foreign parts. Keep free of dirt and grease.

b. Lubricate exposed gears with proper lubricant.

c. Perform operational checks of limit switches and limit switch actuators for proper functional operation.

d. Check the system for proper functional and running operation.

5. Reactor vessel stud tensioner

The minimum acceptable test shall include the following:

a. The tensioner shall be checked for proper functional and running operation.

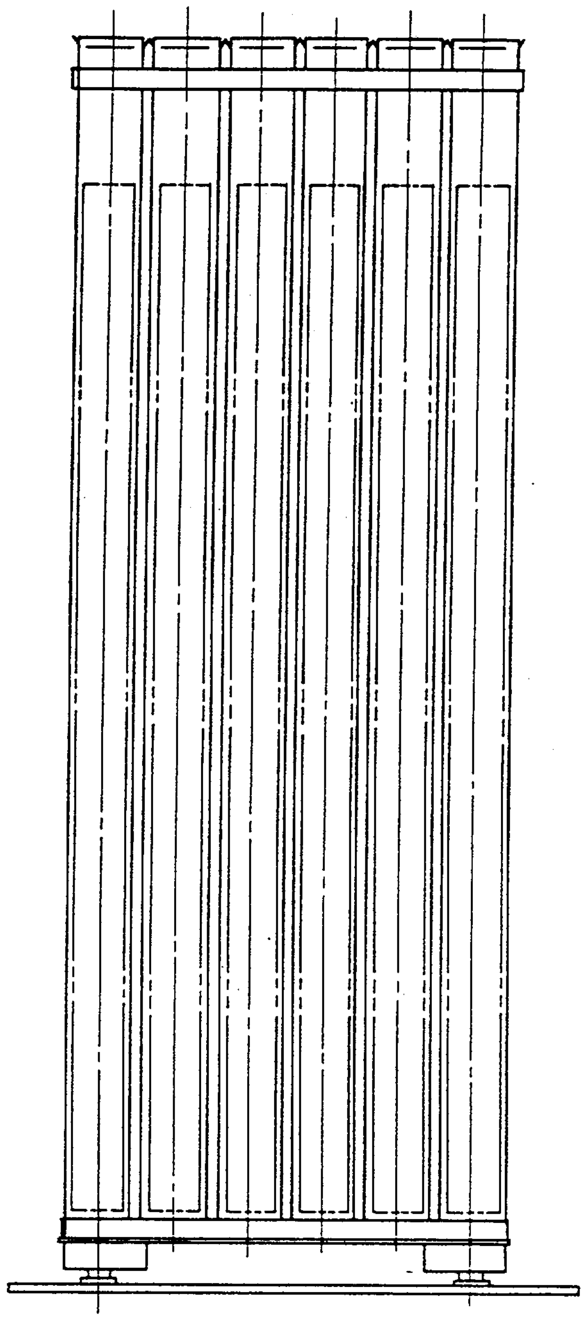
The following maintenance and checkout tests are recommended to be performed prior to using the equipment.

a. Visually inspect for loose or foreign parts.

b. Inspect hydraulic lines for wear or damage.

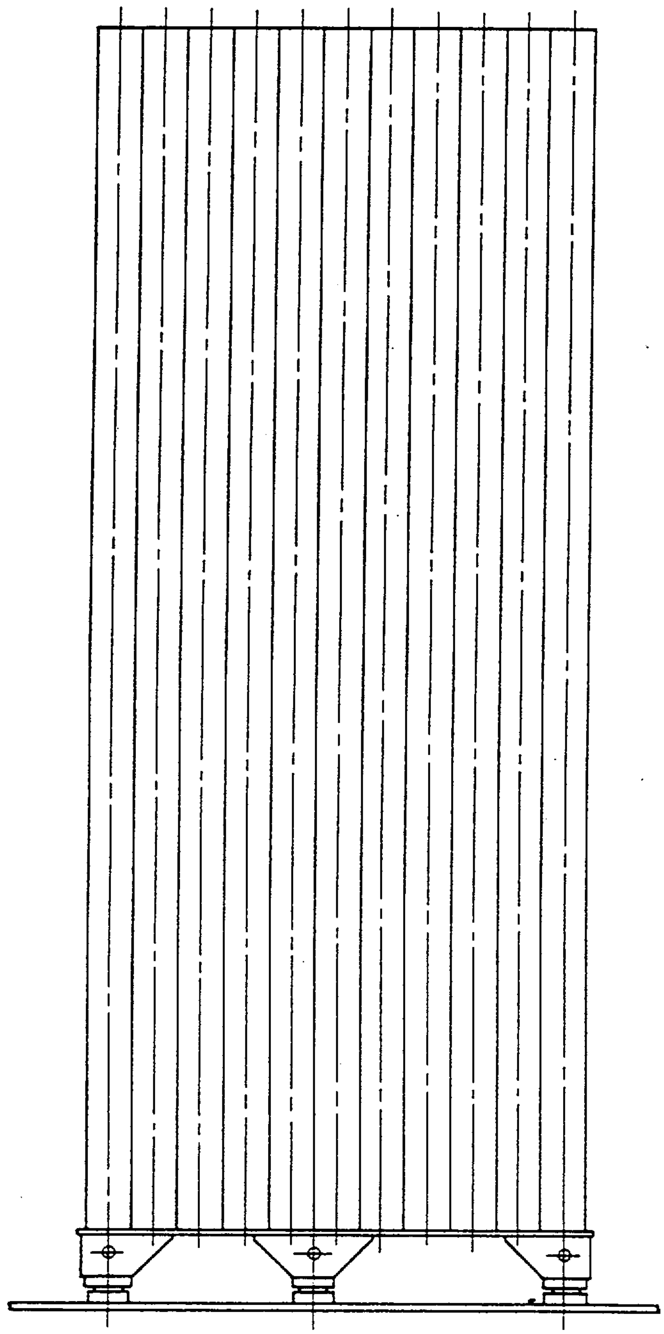
c. Check the hydraulic unit for proper pressurization and if any leaks occur at operating pressure.

9.1.4.5 Instrumentation Requirements. Instrumentation requirements of equipment, including interlocks, are discussed in Sections 9.1.4.2 and 9.1.4.3.



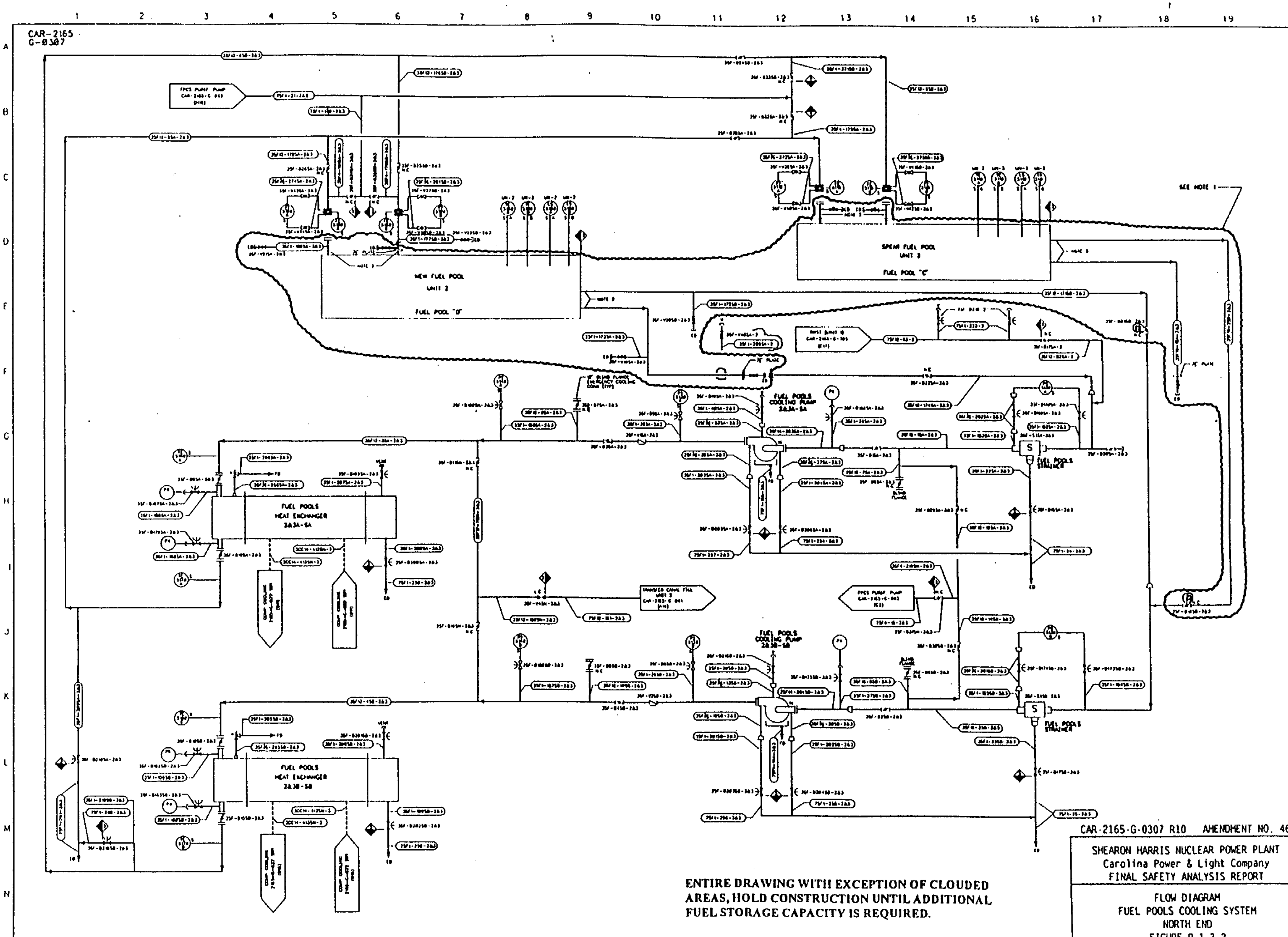
Amendment No. 7

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	PWR FUEL RACK	Figure 9.1.1-1.



Amendment No. 7

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	BWR FUEL RACK	Figure 9.1.1-2.

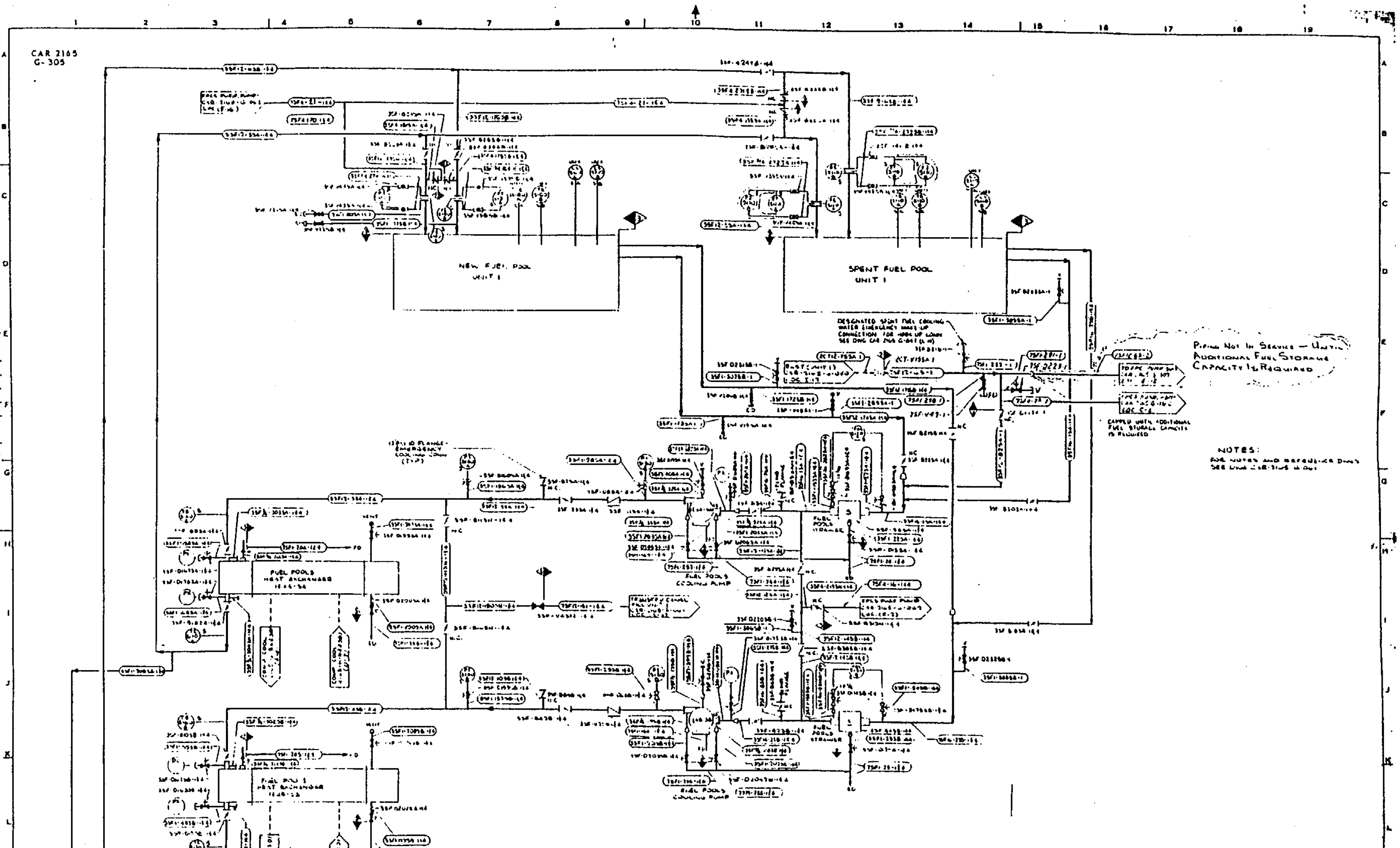


CAR-2165-G-0307 R10 AMENDMENT NO. 46

ENTIRE DRAWING WITH EXCEPTION OF CLOUDED AREAS, HOLD CONSTRUCTION UNTIL ADDITIONAL FUEL STORAGE CAPACITY IS REQUIRED.

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

FLOW DIAGRAM
FUEL POOLS COOLING SYSTEM
NORTH END
FIGURE 9.1.3-2



CAR-2165-G-0305 R14 AMENDMENT NO. 46

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

FLOW DIAGRAM
FUEL POOLS COOLING SYSTEM
SOUTH END
FIGURE 9.1.3-1

9.2.2 Component Cooling System

The Component Cooling Water System (CCWS) consists of two component cooling water heat exchangers, three component cooling water pumps, a component cooling water surge tank, cooling lines to various components being cooled, and the associated piping, valves, and instrumentation. It provides cooling water to various plant components during all phases of plant operation and shutdown, serving as an intermediate system between the Reactor Coolant System (RCS) and the Service Water System.

9.2.2.1 Design Basis. The CCWS is designed to operate during all phases of plant operation including startup, power operation, shutdown, refueling, and the injection and recirculation phases following a loss-of-coolant accident (LOCA).

The design of the CCWS is based on a maximum service water supply temperature of 95F.

The CCWS is designed to supply cooling water at a maximum temperature of 120F to the components being cooled when the Residual Heat Removal System is first placed in operation during plant shutdown, this being the maximum permissible temperature of the cooling water supply to the reactor coolant pumps. During normal plant operation, the temperature of the cooling water supplied to all components is a maximum of 105F.

The design cooldown rate, based on reducing the temperature of the reactor coolant from 350F to 140F in 17 hours is achieved using two CCWS pumps and two CCWS heat exchangers. Failure of a heat exchanger would increase the time required for shutdown but would not affect the safe operation of the plant. Failure of a pump would not affect the time required for shutdown since a standby pump is available.

The CCWS is also required for engineered safeguards operations to remove decay heat from the RCS and to provide cooling water to various engineered safeguards components. For this reason, the CCWS is designed to meet the single failure criterion by providing two completely independent parallel trains. Each CCWS train consists of a pump and heat exchanger. The surge tank is also separated into two parts by a baffle. Each train services all of the safeguards components in the appropriate safeguards train and can be isolated from the redundant CCWS train for long term post-accident recirculation. Isolation provisions are discussed in the sections on valves and on post-accident operation.

All Class 2 and 3 components of the CCWS are designed to meet Seismic Category I requirements. Equipment that is necessary for shutdown and equipment that is required to mitigate the effects of an accident is supplied with emergency diesel power, should normal and offsite power sources fail.

9.2.2.2 System Description. The CCWS is shown in Figures 9.2.2-1 through 9.2.2-4. Design parameters for the CCWS are given in Table 9.2.2-1.

The CCWS serves as an intermediate system between the RCS and the Service Water System to ensure that leakage of radioactive fluid from the components being cooled is contained within the plant.

The system consists of two component cooling water heat exchangers, three component cooling water pumps, a component cooling water surge tank, cooling lines to the various components being cooled, and associated piping, valves and instrumentation. The component cooling water flows from the pumps, through the shell side of the component cooling water heat exchangers, through the components being cooled, and back to the pumps.

The surge tank is connected to the suction side of the component cooling water pumps. It accommodates surges resulting from component coolant water thermal expansion and contraction and accommodates water which may leak into the system from components which are being cooled. The surge tank also contains a supply of water to provide component cooling water supply until a leaking cooling line can be isolated. The surge tank water level is adjusted manually from the Control Room by delivering makeup water from the Demineralized Water System to the tank.

Water chemistry control of the CCWS is accomplished by additions to the chemical addition tank or to the surge tank. Mixing with the loop water can be accomplished by recirculation through either tank. Corrosion coupon racks are installed on the system to monitor the effectiveness of the corrosion inhibitor. Periodic grab samples will be analyzed by Chemistry to ensure adequate protection is being provided to maintain the integrity of the system. Corrosion will be minimized through control of impurities and chemical additives. Adjustments will be made based upon results obtained from analyses.

Table 9.2.2-3 presents typical CCWS flow rates during normal operation.

9.2.2.2.1 Equipment served. The Component Cooling Water System provides cooling for the following heat sources:

a) reactor coolant pump motor bearing oil cooler

b) reactor coolant pump thermal barrier

c) letdown heat exchanger (Chemical and Volume Control System)

d) seal water heat exchanger (Chemical and Volume Control System)

e) excess letdown heat exchanger (Chemical and Volume Control System)

f) residual heat removal pumps (Residual Heat Removal System)*

*Heat loads from these components are considered to be safety-related.

g) residual heat exchangers (Residual Heat Removal System) *

h) recycle evaporator package (Boron Recycle System)

i) reactor coolant drain tank heat exchanger (Waste Processing System)

j) spent fuel pool heat exchangers (Spent Fuel Pool Cooling and Cleanup System) *

k) sample heat exchangers (Process Sampling System)

l) gross failed fuel detector (Process Sampling System)

The residual heat removal pump coolers and heat exchangers are the essential loop defined in the Technical Specifications, and their flowpaths are the only flowpaths subject to periodic surveillances. 41

9.2.2.2.2 Component Description

The codes and standards to which the individual components of the Component Cooling Water System are designed and selected according to the most severe condition expected for each component either during normal operation or during operation in conjunction with the Emergency Core Cooling System. The codes selected are appropriate for the limiting conditions of operation and are consistent with the safety classifications for these components. Codes and standards applicable to the CCWS are listed in Table 3.2.1-1.

Component Cooling Water Heat Exchangers

The CCWS heat exchangers are shell and straight tube type units. Component cooling water circulates through the shell side. The shell is constructed of carbon steel and the tubes are made of 90-10 copper nickel alloy.

Each heat exchanger is designed to remove one half of the heat removal load during the period of reducing the reactor coolant temperature from 350F to 140F. The heat removal load during normal full-power operation is accommodated by one component cooling water heat exchanger with the additional exchanger providing 100 percent standby capacity. The provisions of two component cooling water heat exchangers provides redundancy required for compliance with safeguards single failure criteria, ensures that heat removal capacity is only partially lost if one exhcanger fails or becomes inoperative, and permits maintenance or replacement of one exchanger while the other unit is in service.

Component Cooling Water Pumps

There are three CCWS pumps. Each pump is a horizontal, centrifugal unit with carbon steel casings, internals, and shafts. The design capacity of each pump equals or exceeds the required capacity for single pump operation during normal operation, safety injection, and post-accident recirculation on a redundant train basis.

* Heat loads from these components are considered to be safety-related.

During plant cooldown two pumps are operated; each pump circulates one-half of the total component cooling water flow. During normal operation, only one of the component cooling water pumps is operating although two pumps are required to be available in order to supply the normal cooling water flow demands and to maintain 100 percent standby flow capacity in case the operating pump trips off. A minimum of two pumps is also required for compliance with safeguards single failure criteria. The provision of a third pump provides redundancy should one of the three pumps require preventive or corrective maintenance.

Provisions to ensure that electrical power is available to the pumps are discussed in Section 9.2.2.3.

<u>Component Cooling Water Surge Tank</u>

The CCWS surge tank is cylindrical, horizontal and is made of carbon steel.

The surge tank capacity permits:

a) The surge tank to accommodate changes in CCWS water volume due to changes in operating temperature.

b) A volume to accommodate, for 20 minutes, the maximum flow from either makeup water supply.

c) A reservoir of water to provide time to locate and terminate a system leak should one develop.

d) A volume to accommodate, for about 2 hours, the Technical Specification maximum identified reactor coolant leakage of 10 gpm.

The CCWS surge tank is located at an elevation that ensure adequate net positive suction head (NPSH) to the CCWS pumps.

The tank is exposed to atmospheric pressure via two vent lines, which also protects the tank from overpressurization. Orifices in the demineralized water and reactor makeup water addition lines limit the flow into the system to a maximum of 50 gpm, ensuring that the capacity of the vent lines to handle an accident mass influx from these two sources is not exceeded. The tank is connected to the system by two four (4) inch lines, both equipped with locked-open valves.

<u>Component Cooling Water System Piping</u>

Carbon steel is used for piping since it has good corrosion resistance when in contact with the inhibited component cooling water. All piping joints and connections are welded. Flanged connections are used at pumps, heat exchangers, valves and instrumentation connections, to facilitate removal for maintenance.

The piping of the Component Cooling Water System does not require thermal insulation since it rarely reaches a temperature higher than 150F.

The relief valve downstream of the excess letdown heat exchanger is sized to relieve flow should a tube rupture occur in the excess letdown heat exchanger. In this case the valve must relieve a water steam mixture.

A relief valve is also provided on each CCWS reactor coolant pump thermal barrier line.

The cooling water headers to the reactor coolant pumps and to the excess letdown and reactor coolant drain tank heat exchangers are provided with check valves and motor operated gate valves on the inlet lines and motor operated gate valves on the outlet lines. The gate valves on the lines to and from the heat exchangers close automatically on a Phase A containment isolation (T) signal; the gate valves on the lines to and from the reactor coolant pumps close automatically on a Phase B containment isolation (P) signal. For protection in the unlikely event of a ruptured thermal barrier cooling coil in the reactor coolant pump, each of the feedlines to the pump thermal barriers is provided with check valves and the discharge header is provided with a third motor operated gate valve that would close automatically on high discharge flow.

In order to isolate the non-essential header, four motor operated valves (two each in the CCW pump suction and discharge header) are provided so that the CCW can be remotely separated into two independent loops upon the initiation of post accident recirculation. It is not necessary that these valves be closed during the injection phase of a LOCA.

The component cooling water to the non-safety processing sampling system (sample heat exchangers and gross failed fuel detector) is provided with two air operated valves on the inlet lines and two check valves on the outlet lines. The air operated valves on the inlet lines will close automatically on a Safety Injection (S) signal thus isolating the CCW system from non-safety related systems.

9.2.2.2.3 System Operation

Normal Operation

Normal operation includes the power generation and hot standby operating conditions when the reactor plant is at normal operating temperature and pressure.

Only one component cooling water heat exchanger and one pump are required for operation. One component cooling water pump is placed on standby to start automatically on low pressure in the component cooling water pump discharge header. The four manual valves in the component cooling pump suction and discharge cross connects are kept closed during normal operation. These valves separate the normally operating component cooling pump from the pump providing backup. By closing them, separability is maintained in the suction and discharge headers such that complete redundancy and separation of cooling water trains for post-accident recirculation can be completed remotely from the Control Room by remotely closing the non-essential header inlet and outlet isolation valves.

Periodically, a sample of the component cooling water is taken by the plant operator to ascertain that water chemistry specifications are met. Chemicals are then added and mixed by recirculation as necessary.

The CCWS System is capable of supplying cooling water to each of the two spent fuel pool heat exchangers. This is established by appropriate flow alignment of manual isolation and throttle valves in the spent fuel pool heat exchanger cooling water supply, return, and interconnecting headers. When two spent fuel pool heat exchangers are placed in service, they are typically operated in parallel. In this alignment, one heat exchanger is capable of being aligned to a separate pool. The CCW system is capable of supplying two spent fuel pool heat exchangers from a single CCW pump provided that the system is balanced to provide the required flow to the RHR system and the other required CCW cooling loads.

<u>Plant Shutdown</u> - Plant shutdown is defined as the operations which bring the reactor plant from normal operating temperature and pressure to cold shutdown for maintenance or refueling.

As discussed in Section 5.4.7.1, the Residual Heat Removal System is placed in operation approximately four hours after reactor shutdown when indicated temperature and pressure of the RCS are less than approximately 350°F and 363 psig, respectively. The standby component cooling heat exchanger and pump are placed in operation, component cooling water flow is initiated and both residual heat exchangers of the Residual Heat Removal System are placed in operation.

The rate of heat removal from the reactor coolant is controlled by regulating the reactor coolant flow rate through the residual heat exchangers. The cooldown rate is limited by the allowable cooling rate based on stress limits of the reactor vessel and steam generator, and the limits set on the operating temperature of the Component Cooling Water System by other components using the system. During the cooldown period, the cooling water inlet temperature to the various components is permitted to increase to 120F.

The spent fuel pool heat duty is supplied by the CCWS. The spent fuel pool heat exchanger is cooled by the CCWS; however, the plant cooldown time will be extended and may require termination of CCW to other non-essential equipment.

During a cold shutdown condition, residual heat from the reactor core is removed by the Component Cooling Water and Residual Heat Removal Systems. The number of pumps and heat exchangers in service varies depending upon the residual heat removal load, the spent fuel pool load and the operations in progress in the Boron Recycle System.

Post-Accident Operation

During the injection phase following a LOCA the component cooling water pumps receive an "S" signal to ensure that at least one pump is started and supplying cooling flow to the safeguards pumps.

As the Safety Injection System is switched over to the recirculation phase, special preparations have to be made in the CCWS. This system is designed to be operable with a single active or passive failure during the recirculation phase. Therefore, the system must be separated into two parts, each of which can function independently and remove the residual heat from the recirculated sump water.

The separation of the system into parts consists of closing two of the four motor operated valves in the cross-connecting headers downstream of the component cooling water heat exchangers and upstream of the component cooling water pump suction header. In the event of a passive failure in the system, only one half of the system will be affected, and the second half will remain operational. In the case of an active failure of one of the pumps, the valves in the suction header and in the discharge header can be repositioned to align the spare pump with the affected half of the system.

In order to separate the trains, four motor operated butterfly valves (two on upstream of the CCW pump suction header and two downstream of the CCW heat exchanger header) are provided so that the CCW can be remotely separated into two independent loops upon the initiation of post accident recirculation. It is not necessary to separate the trains during the injection phase of a LOCA.

The component cooling water to the non-safety processing sampling system (sample heat exchangers and gross failed fuel detector) is provided with two air operated valves on the inlet lines and two check valves on the outlet lines. The air operated valves on the inlet lines will close automatically on a Safety Injection (S) signal thus isolating the CCW system from non-safety related systems.

9.2.2.3 Safety Evaluation

Safety-related portions of the CCWS are Seismic Category I design, capable of withstanding adverse environmental occurrences such as postulated earthquakes, tornadoes, and tornado missiles (Seismic Category I structures are discussed in Chapter 3.)

Because the CCWS serves as an intermediate system between the Reactor Coolant System and the Service Water System, it ensures that leakage of radioactive fluid from the components being cooled is contained within the plant.

Leakage into the Component Cooling System can be detected by either of two radiation detectors, routine sampling, high temperature, and in the case of a large leak, a rise in the surge tank level which will initiate a high level alarm. Details of the radiation monitoring equipment are given in Section 11.5.

To assure reliability, the component cooling water pumps and the motor operated valves are connected to two separate buses to that pumps and valves performing similar functions will receive power from different sources, both when normal off-site power is available and when emergency on-site power is required. Therefore, two component cooling water pumps are connected to separate and redundant electrical power and control circuits.

See Section 8.3.1.1.2.4 for additional information on electrical connections, interlocks, and alarms.

Sufficient cooling capacity is provided to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operation of safety features equipment. A failure modes and effects analysis of the CCWS is provided in Table 9.2.2-4.

The component cooling water lines supplying the RCP motor oil coolers are ASME Class 3 and seismic Category I. During normal operation manually operated isolation and throttle valves on these lines do not require adjustment. Therefore to preclude inadvertent operator action, these valves will be locked in the throttled position. Also, RCP motor vibration detectors and alarms are available to the operators if excessive vibration were to occur.

Instantaneous seisure of a RCP motor due to a loss of CCW to the oil coolers is not considered to be a credible event. However, an evaluation of a locked rotor scenario has been completed for Shearon Harris (FSAR 15.3.3), and the radiological consequences are less than the 10 CFR 100 guideline values.

9.2.2.4 Tests and Inspections. Preoperational testing is described in Chapter 14. The performance and structural and leaktight integrity of all component cooling water system components is demonstrated by continuous operation.

The CCWS is testable through the full operational sequence that brings the system into operation for reactor shutdown and for accident mitigation, including operation of applicable portions of the protection system and the transfer between normal and standby power sources.

TABLE 9.2.2-3

TYPICAL SYSTEM FLOW RATES DURING NORMAL OPERATION

Component Cooling Water Main Headers	
Total CCW Flow	5000 to 11,000 gpm
RHR HX A OUT FLOW	5600 to 6150 gpm*
RHR HX B OUT FLOW	5600 to 6150 gpm*
*Values shown are for RHR in service. Normal flowrate is 0 gpm.	
Component Cooling Water to Reactor Coolant Drain Tank HX & Excess Letdown HX	
CCW Flow - Reactor Coolant Drain Tank HX	200 to 225 gpm
CCW Flow - Excess Letdown HX	230 to 270 gpm
Component Cooling Water to Reactor Coolant Pumps	
CCW Flow - Reactor Coolant Pumps A, B, C Thermal Barrier	40 to 45 gpm
RCP Thermal Barrier Total CCW Header Flow	120 to 135 gpm
CCW Flow - Reactor Coolant Pumps A, B, C Oil Cooler Header	165 to 180 gpm
CCW Flow - Reactor Coolant Pumps A, B, C Bearing Flow	5 to 10 gpm
Component Cooling Water to Gross Failed Fuel System and Primary Sample Panel	
CCW Flow - Gross Failed Fuel Detector	5 to 10 gpm
CCW Flow - Primary Sample Panel Heat Exchangers	160 gpm
Component Cooling Water to Seal Water HX, Letdown HX, and Recycle Evaporator Package	
CCW Flow - Seal Water HX	200 to 230 gpm
CCW Flow - Letdown HX	500 to 575 gpm
CCW Flow - Recycle Evaporator Package	700 to 780 gpm
Component Cooling Water to Fuel Pool HX	
CCW Flow - Fuel Pool HX	1500 to 5600 gpm
Component Cooling Water to RHR Pump Coolers	
CCW Flow - RHR A&B Train	5 to 10 gpm

This scaling is valid since no changes in the assumptions relating to transport mechanisms have been made. A flow chart of the calculation procedure is presented in Figure 15.7.4-1.

15.7.4.4.3 Description of SPC calculations. The radiological assessment for the Shearon Harris high burnup fuel followed the procedure presented in Section 15.7.4.4.2. The fuel and core fission product inventories were calculated for the Westinghouse fuel (base case) and the SPC extended burnup fuel using the ORIGEN code (Reference 15.7.4-3). The operating conditions assumed for the Fuel Handling Accident (FHA) are summarized in Table 15.7.4-5. Fixed fractions of the fission product inventory were assumed to be released in the FHA using Regulatory Guide 1.25 (Reference 15.7.4-4). Release fractions based on vendor models are not part of the licensing basis, and therefore, are not part of this analysis. The results of the FHA fuel inventory and release calculations are presented in Table 15.7.4-6.

The whole body and thyroid dose proportionality constants are calculated with the above two equations. Whole body doses are calculated based on an energy weighted summation of all fission product isotopes over the period of exposure. The thyroid doses, however, are calculated based only on the iodine isotopes. Whole body and thyroid doses are evaluated for 2 hour and 8 hour exposure for the FHA.

15.7.4.4.4 SPC calculated accident radiological consequences. The effect of a FHA on radiological releases is analyzed using similar assumptions as were made in the Westinghouse analysis. Table 15.7.4-7 presents the calculated doses for the FHA. The calculated doses are less than the specified limits in the Standard Review Plan.

<u>15.7.4.5 Other Fuel Handling Accidents</u>. FSAR Section 9.1.2.3 also describes a Fuel Handling Accident in the spent fuel pool where more than one fuel assembly is affected by the accident. The accident involves dropping one fuel assembly on top of another. This section states ". . . 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." It has been determined that the maximum number of failed rods from such an accident would be 314 failed fuel rods (Reference 15.7.4-5). Fifty fuel rods are expected to fail in the impacted assembly and all of the rods in the dropped assembly (264) would fail when the assembly falls over. This number of rods exceed the 264 rods (one assembly) assumed in the two accidents (FHA Inside Containment and a FHA Outside Containment) previously discussed in this chapter. When the Harris Plant procedural requirement of a minimum of 100 hours decay time prior to fuel movement is used in reanalyzing the previous accidents, the increased decay time more than compensates for the increase in failed rods, resulting in calculated doses which are bounded by the doses presented in Tables 15.7.4-2, 15.7.4-4, and 15.7.4-7.

Only a maximum of 2 PWR spent fuel assemblies are affected by the FHA's (Fuel Handling Accident) described in FSAR Section 9.1.2.3. If a spent fuel assembly (PWR or BWR) falls across the top of a loaded BWR spent fuel rack, then multiple BWR spent fuel assemblies could be damaged. This is because the upper bail handle of a BWR spent fuel assembly extends above the top of a BWR spent fuel rack. Fuel handling accidents involving a dropped PWR or BWR spent fuel assembly, which initially strikes a stationary PWR or BWR spent fuel assembly and then falls onto an adjacent loaded BWR spent fuel rack have been evaluated. The worst case dose consequences would result from a dropped PWR spent fuel assembly, which initially strikes a stationary PWR spent fuel assembly and then falls onto an adjacent loaded BWR spent fuel rack, which could result in the failure of up to 314 PWR spent fuel rods and up to 52 BWR spent fuel assemblies. The dose consequences from this FHA have been evaluated and found to be bounded by the doses provided in FSAR Table 15.7.4-7 for a FHA outside Containment. Likewise, the dose consequences of dropping a PWR or BWR spent fuel assembly onto a loaded BWR spent fuel rack have been evaluated, and the resulting dose are also bounded by the doses provided in FSAR Table 15.7.4-7 for a FHA outside Containment.

15.7.5 Spent Fuel Cask Drop Accidents

15.7.5.1 Cask Drop Into the New or Spent Fuel Pool. As discussed in Section 9.1, the cask handling crane is prohibited from traveling over the new and spent fuel pools or any unprotected safety related equipment. Thus, an accident resulting from dropping a cask or other major load into the new or spent fuel pools is not credible.

15.7.5.2 Cask Drop to Flat Surface. The cask has full integrity when the head is fully tensioned and the valve box covers are installed.

15.7.5.2.1 Cask with full integrity. As discussed in Section 9.1, the potential drop of a spent fuel cask is limited to less than an equivalent 30 ft. drop onto a flat, essentially unyielding, horizontal surface. Since the spent fuel cask, with the valve box covers installed and the head fully tensioned, is designed to withstand such loadings, the radiological consequences of dropping the cask in this condition are not evaluated.

15.7.5.2.2 Cask with less than full integrity. The loaded IF-300 series cask may be moved with the valve covers removed and, from the decon pit to the unloading pool, with only four cask head bolts installed. An evaluation of a 30-ft. drop during the movement from the decon pit to the unloading pool was performed and indicated that, while fuel components would be retained in the cask, the cask is not expected to be gas tight. Noble gas and iodine gap activity could be released to the Fuel Handling Building and subsequently to the environment. Damage to the valves caused by dropping the cask could cause the same type of release. The radiological consequences from this release were analyzed for an IF-300 series cask. This analysis utilized the worst case fuel types anticipated to be shipped in the cask, as shown in Table 15.7.5-1. The results of this analysis show that these consequences would be a small fraction of the 10CFR100 exposure guidelines. The analysis is documented in Harris Plant calculation HNP-M/FHB-1001.

HARRIS SFP PROJECT

Briefing

Background

HNP SFP Project

- The Harris Nuclear Plant (HNP) fuel handling building (FHB) was constructed with four separate pools.
- Only two of the four pools, 'A' and 'B', are currently in operation (i.e., storing fuel).
- HNP stores spent fuel from Robinson (HBR) and Brunswick 1&2 (BNP).

Background- continued

HNP SFP Project

- HBR and BNP must ship spent fuel to HNP to maintain full core offload capability.

- Based on the current CP&L shipping schedule, HNP will lose full core offload capability by 2000 if the other two spent fuel pools, 'C' and 'D', are not made operational.

Activation of 'C' and 'D' SFPs

HNP SFP Project

- CP&L has divided the technical work necessary to make the 'C' and 'D' SFP operational into 6 tasks:
  - Completion of the 'C' & 'D' SFP Cooling and Cleaning System (SFPCCS)
  - Storage optimization (Re-rack) for the 'C' & 'D'
  - 'D' SFP cleanup
  - Upgrade of shipping cask basket hoist rigging
  - Disposal of low density Boraflex racks stored in the 'C' pool
  - CCW upgrades

Licensing Activities for HNP SFP Project

HNP SFP Project

- The HNP License Amendment for the HNP SFP Project will include three distinct pieces:
    - TS change for the rack design and storage capacity
    - USQ for the Component Cooling Water System
    - Proposed alternative pursuant to 10 CFR 50.55a (a)(3)(i) to satisfy the ASME Code Section III design requirements for the SFPCCS piping.
- An environmental assessment will also be needed for this amendment

TS Change

HNP SFP Project

- TS 5.6 will be revised by:
  - adding the nominal center-to-center distance between PWR fuel assemblies for the racks in pools 'C' and 'D'. (9.0" pitch)
  - changing the fuel storage capacity
- A TS will also be added to limit the heat load of the fuel stored in the 'C' and 'D' pools to no more than 1 MBTU/hour

CCW USQ

HNP SFP Project

- Unit 1 CCW was originally designed to only provide cooling to the unit 1 SFPCCS.
- The licensee completed an engineering analysis which concludes that the Unit 1 CCW has sufficient margin to provide cooling to the SPFCCS for the expected heat load for the 'C' & 'D' pools through 2001.
- The licensee intends to upgrade the CCW as a part of power uprate.
- The power uprate CCW upgrade will ensure that sufficient capacity will be available to provide adequate cooling to C& D pools when they are at maximum capacity.

SFPCCS Piping

HNP SFP Project

- Piping certification documents for the Unit 2 SFPCCS were discarded.

- CP&L will propose an alternative plan to satisfy the requirements of 10 CFR 50.55a

- Proposed plan will consist of :

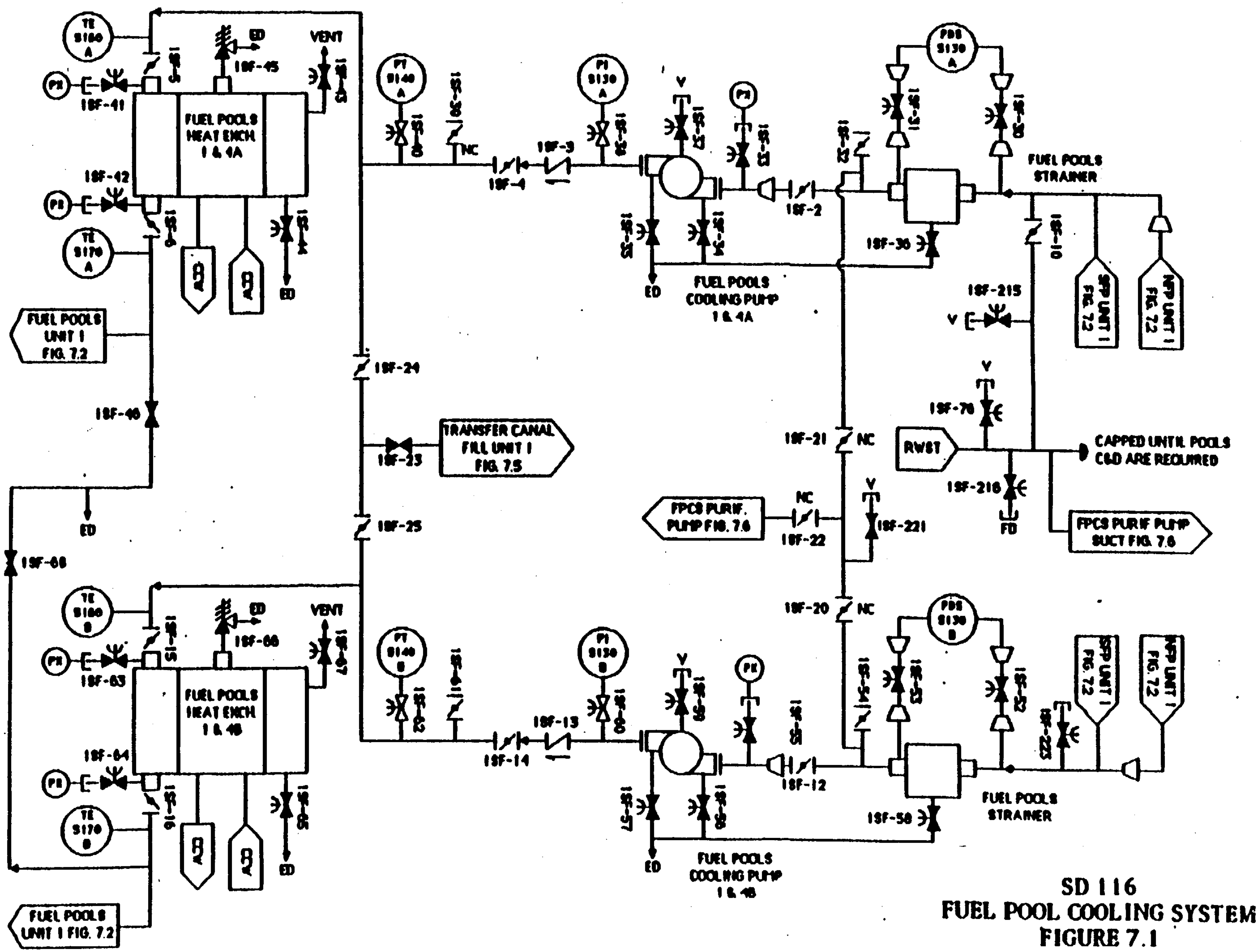
  - available documentation such as hydrotest records and piping field installation records.

  - External and internal inspections using visual and liquid penetrant

Status and Schedule

HNP SFP Project

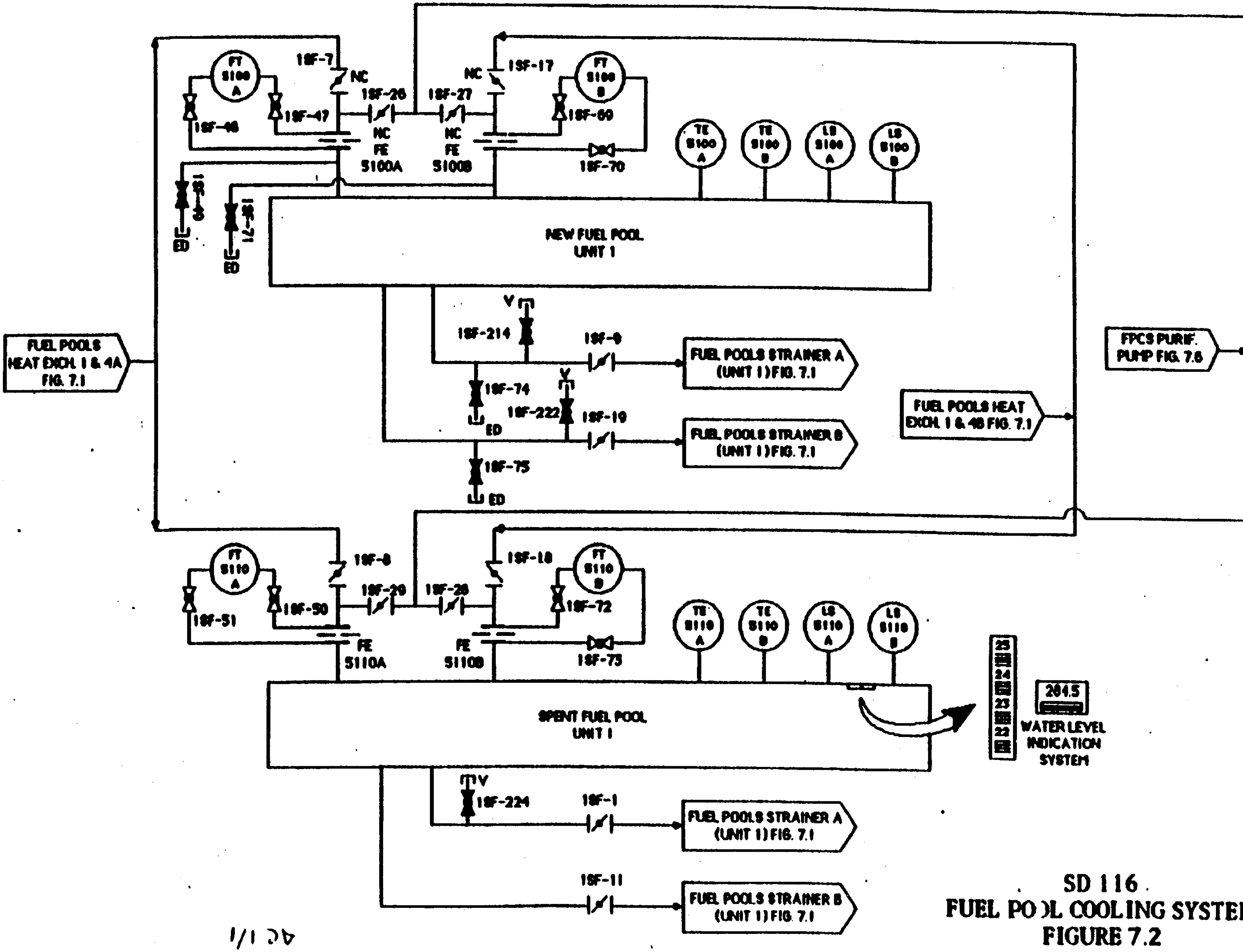
- Two public meetings have been held: 3/3/98 and 7/16/98
- Good Technical Staff attendance and participation at both meetings
- Submittal is scheduled for November 1998
- Public involvement is expected
  - D. Lochbaum attended 7/16/98 meeting
  - Local government official inquired about the project
  - State government official inquired about the project after an interest group approached the State about the project



SD 116
FUEL POOL COOLING SYSTEM
FIGURE 7.1

AS 1/1

F/8



SD 116
FUEL POOL COOLING SYSTEM
FIGURE 7.2