UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-400-LA
CAROLINA POWER & LIGHT	)	
COMPANY	)	ASLBP No. 99-762-02-LA
	)	
(Shearon Harris Nuclear Power Plant)	)	
	)	

AFFIDAVIT OF CHRISTOPHER GRATTON IN SUPPORT OF NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON TECHNICAL CONTENTION 3

Christopher Gratton, having first been duly sworn, does hereby state as follows:

1. My name is Christopher Gratton. I am employed as a Reactor Systems Engineer for Plant Systems Branch, Division of Systems Safety and Analysis in the Office of Nuclear Reactor Regulation. I am responsible for reviews involving aspects of spent fuel storage, including spent fuel pool cooling, under 10 CFR Part 50. I have been conducting such reviews since 1994. A statement of my professional qualifications is attached. (Attachment A).

2. The purpose of this testimony is to address a safety concern raised in the Licensing Board's Memorandum and Order (Ruling on Standing and Contentions) (*Carolina Power & Light Co (*Shearon Harris Nuclear Power Plant), LBP-99-25, 50 NRC 25, 37 (1999). Specifically, the Board made the following statement:

> And, of course, if CP&L's plea is that the proposed alternatives provide an acceptable level of safety, we will need to confront directly the question of whether a failure of quality control could lead to a hazard, a question about which there is clearly a dispute

between CP&L and [Board of Commissioners of Orange County] BCOC.

3. Concerns have been raised by BCOC that given a failure in quality control that results in the degradation of components in the north-end fuel pool cooling system, a hazard may result that will have consequences to public health and safety. The components of concern are piping welds embedded during construction in the concrete that forms the structure of fuel pools' C and D. The documentation that supports the quality of these welds was destroyed several years after the pools' construction. The concern is that without the appropriate documentation, the welds may not have been properly constructed and may fail causing a loss of inventory and forced cooling to fuel pools C and D. This, in turn, may lead to fuel uncovering and overheating. In addition to the lack of documentation, BCOC is contending that the components have not been maintained properly during their layup period, have experienced degradation, and are subject to premature failure.

4. I examined the design of fuel pools C and D and the north-end fuel pool cooling system (supporting fuel pools C and D) using the Shearon Harris Final Safety Analysis Report (FSAR) to determine whether degradation of system components could result in unacceptable consequences to public health and safety. While some passive failures of the spent fuel pool cooling system piping would render the spent fuel pool cooling and cleanup system inoperable and reduce spent fuel pool coolant inventory, as discussed in the following paragraphs, it is unlikely that the stored fuel would ultimately become exposed to air, overheat, and become damaged resulting in a hazard.

5. Unless otherwise noted, information contained herein is based on the Harris FSAR, Sections 9.1.1 through 9.1.3 (Exhibit 1). The spent fuel storage system at Harris consists of four storage pools located in a single Fuel Handling Building (FHB). Two of the four pools are currently in use. Fuel pools A and B are currently in use and are located at the south end of the FHB, while fuel pools C and D are being prepared for use and are located at the north end of the FHB. Initially, the fuel pools were intended to service four units at the Harris site. After the cancellation of Units 2, 3, and 4, the licensee obtained permission to store spent fuel from their other nuclear power plants (Brunswick 1 and 2, and H. B. Robinson) in the Harris spent fuel pools (Exhibit 2). The pools are constructed of concrete to seismic standards and lined with stainless steel to be compatible with the pool water.

6. Two fuel pool cooling systems are provided to remove decay heat from the four fuel pools. Fuel pools A and B share one cooling system at the south end of the FHB, and fuel pools C and D share the other cooling system at the north end of the FHB. Each fuel pool cooling system has two 100% capacity trains that consist of a pump, heat exchanger, filter and the requisite piping, valves, and instrumentation to support the train's operation. The fuel pool cooling system servicing fuel pools A and B has been operational since the initial licensing of Harris Unit 1 reactor. The fuel pool cooling system servicing fuel pools C and D was partially completed, and is now being completed in preparation for use.

7. If a failure of a piping weld occurs, regardless of the mode of failure, the failure will breach the system's pressure boundary and will result in a loss of coolant from the system. The actual size and location of the weld failure will determine the actions that must be taken to

mitigate the problem, the time available to take those actions, and the consequence for public health and safety.

8. Should a failure occur in a section of pipe that is embedded in concrete with no available path for the leaking coolant (i.e., the treated water inventory contained in the spent fuel pool and the spent fuel pool cooling system) to exit the pool structure, leakage should not affect the cooling system's operation. The coolant exiting the pipe would fill any available voids in the concrete in the vicinity of the failure. Once the local voids are filled, leakage from the pipe would slow dramatically and any further inventory loss from the pool into cracks in the concrete, should any exist, would have minimal impact on the operation of the cooling system.

9. If a weld failure occurs in a location where the coolant could flow freely from the pool's concrete structure along the embedded pipe or from cracks in the concrete, the inventory in the pool would decrease until the leak was identified by the plant operators, or until the pool level fell below the suction piping penetration in the pool's liner. The fuel pool cooling system piping penetrates the fuel pool at an elevation approximately five feet below the normal coolant level in the fuel pool. This level is approximately eighteen feet above the stored fuel and will still provide a considerable heat sink and acceptable shielding to the stored fuel.

10. Prior to the coolant reaching this level in the fuel pool, however, spent fuel storage design features and plant administrative procedures provide the plant operators with several opportunities to detect the system failure and take mitigating actions. First, coolant level in the fuel pool is maintained at a reference level of 284 feet 6 inches, which is approximately 23 feet above the top of the stored fuel. Low and low-low fuel pool coolant level alarms which sound locally and in the control room are set at 284 feet and 282 feet, respectively, to warn operators of

the change in coolant level, giving them an opportunity to take mitigating actions. In addition to level alarms, if an embedded weld failed and the leaking coolant exited the pool's structure, plant operators and workers in the vicinity of the leak would observe the coolant exiting the structure and take mitigating actions. If no personnel were in the vicinity of the failed pipe, the coolant would accumulate in the floor and equipment drain sumps and trigger a sump high level alarm in the control room, alerting operators of the problem (Exhibit 1, page 9.1.3-6b).

11. The staff completed a generic action plan for ensuring the safety of spent fuel storage pools in response to two postulated event sequences involving spent fuel pools at two separate plants (Exhibit 3). The principle safety concerns addressed by the action plan involved the potential for a sustained loss of SFP cooling and the potential for a substantial loss of spent fuel coolant inventory that could expose the fuel. The study reviewed several factors that affected the safety of the stored fuel including spent fuel storage pool coolant inventory and temperature requirements and concluded that adequate cooling of the stored fuel and cladding is established by maintaining coolant level above the top of the stored fuel. In addition, a coolant level several feet above the top of the stored fuel serves as acceptable shielding. The study found that coolant temperature had a less direct effect on the safe storage of irradiated fuel than coolant inventory. The study found that forced cooling is not required to protect the fuel cladding integrity when adequate water is supplied to makeup for coolant inventory loss due to boiling.

12. If a postulated weld failure resulted in a small leak in Harris spent fuel pool C or D, one within the capacity of the available coolant make up systems, coolant make up to the fuel pool would be able to maintain normal inventory level and the capability to operate the fuel pool cooling and cleanup system. Coolant make up to the fuel pools is provided by the Refueling

Water Storage Tank and from the demineralized water system. Emergency make up water can also be provided from the Emergency Service Water System. Because the stored spent fuel would remain covered with coolant and the forced cooling system would remain operable, the event would have minimal impact of public health and safety.

13. If a postulated weld failure resulted in a large leak in a portion of the fuel pool cooling and cleanup system piping that cannot be isolated, one that exceeds the capacity of the available make up systems, such as the failure of the fuel pool cooling system suction piping, the fuel pool would drain to the elevation where the failed piping penetrates the fuel pool liner (i.e., approximately 5 feet below the normal water level). With the remainder of the fuel pool intact, the leakage from the pool would stop once the pool level reaches the penetration elevation. In this scenario, the capability to provide forced cooling to the fuel pool will be lost.

14. In Section 9.1.3.3 of the FSAR (Exhibit 1), the licensee evaluated the potential for draining or siphoning the pool through system penetrations and temporary connections. The licensee concluded that if a leak were to occur that drained the spent fuel pool to the level of the cooling system penetrations, because the remaining coolant inventory provides a large heat sink for the stored spent fuel, the system failure would have no adverse impact on the capability of the cooling system to maintain the required temperature because there would be sufficient time to take any necessary actions to provide adequate cooling to the stored fuel.

15. In a letter dated August 5, 1999, the staff requested additional information regarding the licensee's amendment to increase the fuel storage capacity at Harris (Exhibit 4, Letter from CP&L to the USNRC, "Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Response to NRC Request for additional Information Regarding Amendment Request to

Increase Fuel Storage Capacity," dated September 3, 1999, Requested Information Item 5). Specifically, the staff requested that the licensee perform an analysis to determined the rate at which the spent fuel pool coolant would rise given a loss of all forced cooling. For the decay heat rate proposed in the licensee's amendment, the licensee estimated that 300 hours would be required to heat the pool's coolant from its normal operating temperature to boiling (Exhibit 4). I performed an independent review of the licensee's estimations and found them to be acceptable. Given the low decay heat rate which results in a low heat up rate, I believe there is adequate time for plant operators to detect a loss of cooling and align a coolant make up system before any significant reduction in pool level occurs due to evaporation or boiling.

16. Even if the fuel pool cooling system could not be restored before the pool reaches boiling, I estimated that the coolant would boil at a rate of only a few gallons per minute, given the proposed maximum decay heat load of 1.0 Mbtu/hr in the fuel pool. This boil off rate is well within the capacity of the Harris coolant makeup systems and allows the plant operators to maintain the coolant inventory until the forced cooling system can be repaired and the pool refilled to its normal operating level. As previously stated, forced cooling is not required to protect the fuel cladding integrity.

17. In summary, I have reviewed the design and considered the operation of the spent fuel storage system at Harris and concluded that in the event that a failure of quality control results in the failure of a passive component, it is unlikely that a hazard that affects public health and safety will result. If a failure of an embedded weld occurs where the leakage cannot flow out of the pool's concrete structure, this failure will have minimal effect on the operation of the fuel pool cooling and cleanup system, the coolant inventory in the spent fuel pool, or the safety of the

stored fuel. If a failure of an embedded weld occurs where the leakage is able to flow out of the pool's concrete structure but whose leakage is within the capacity of the coolant make up systems, once detected by the plant operators, the failure would be mitigated by plant operators who would maintain the spent fuel pool at its normal operating level and repair the damaged piping. This type of leak may have a temporary effect on the operability of the fuel pool cooling system, but would not affect the safety of the stored fuel. If a leak greater than the capacity of the coolant makeup systems developed in an embedded portion of the fuel pool cooling and cleanup system, the spent fuel pool would drain to level equal to the fuel pool cooling and cleanup system piping penetration (approximately 18 feet above the stored fuel) causing a loss of all forced cooling to the affected spent fuel pool. The pool would gradually heat up, and if repairs to the damaged pipe could not be made in sufficient time, the pool would begin to boil. However, due to the low decay heat rate of the stored fuel, the rate of boiling would be low and within the capacity of the available coolant makeup systems to maintain the coolant inventory. Maintaining the coolant inventory ensures the fuel cladding will not overheat, become damaged and create a possible hazard that affects public health and safety. Therefore, in each postulated scenario where a degraded weld fails resulting in a leak from the spent fuel pool cooling system, the stored fuel remains covered and cooled with only a minimal impact on public health and safety.

19. The attached documents are true and correct copies of the documents relied upon in this affidavit.

20. The foregoing statements made by me are true and correct to the best of my knowledge, information and belief.

Christopher Gratton

CHRISTOPHER GRATTON

Sworn and Subscribed before me
this day of December, 1999.

CIRCE E. MARTIN
NOTARY PUBLIC
MONTGOMERY CO., MD.

Circe E. Martin

Notary Public

My commission expires March 1, 2003

Christopher Gratton
Reactor Systems Engineer

Education

M.B.A. University of Maryland, 1992
B.S., Engineering, University of Maryland, 1980

Employment

U.S. Nuclear Regulatory Commission, Reactor Systems Engineer, 1992 - present
Performs safety evaluations of reactor license applications, technical specifications, and topical reports for various balance-of-plant systems including wet spent fuel storage facilities. Performed risk assessments and safety evaluations regarding the adequacy of spent fuel storage issues as part of the Spent Fuel Storage Pool Action Plan.

U.S. Nuclear Regulatory Commission, Operator Licensing Examiner, 1987 - 1992
Qualified operator licensing examiner for boiling water, test and research reactors. Developed and administered licensing examinations to candidates applying for certification. Examinations included written, oral, and performance based tests administered on plan simulators. Topics included physics, reactor system and auxiliary system design and operation, emergency operating and offsite accident response procedures.

Norfolk Naval Shipyard, Assistant Chief Test Engineer, 1980 - 1987
Responsible for the overall safety of the nuclear power plant aboard submarines undergoing overhaul. Qualified Shift Test Engineer (STE) for reactors designed by Westinghouse and General Electric. Responsible for setting electrical and mechanical isolations, changing plant conditions to support maintenance and testing, and directing post-maintenance tests.

Gratton Exhibit 1

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.

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 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.

Tool drop accidents involving the RCCA change tool, BPRA tool, thimble plug tool, PWR spent fuel handling tools, the BWR spent fuel handling tool, refueling trash baskets and items carried by the spent fuel handling tools (vendor supplied refueling trash basket, failed fuel rod storage basket and dummy spent fuel assembly) have been evaluated. If the consequences of dropping a tool from the maximum height which the tool can be raised by the spent fuel bridge crane is not acceptable, then a tool lift limit is indicated on the tool. Tool lift limit marks are placed on tools, and are only applicable when the tools are located in pools A and B. During tool operation, verification that tool lift limits have not been exceeded, is determined by observing that the lift limit marks on the tools are not raised above the upper hand rail of the spent fuel bridge crane. If the thimble plug tool, a fuel pool trash basket (including the specimen basket) with its handling tool or the failed fuel rod storage basket with its handling tool is dropped from the full height that can be achieved by the spent fuel bridge crane; or the other tools are dropped from their lift limits, the consequences will be less severe than for a dropped spent fuel assembly and its handling tool. PWR spent fuel racks have been evaluated for a tool drop which develops 6677 ft-lbs of kinetic energy. BWR spent fuel racks have been evaluated for tool drop which develops 3800 ft-lbs of kinetic energy.

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[1]
POISON (GM-B10/SQ.CM)	0.020	0.0103-0.015

Notes:

(1) Racks manufactured by Holtec International have boundary wrapper thicknesses of 0.075" and inner wrapper thicknesses of 0.035".

* 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 9.

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 9 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 9 (Reference 9.1.3-3). 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 Pumps - 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 make-up to the fuel pools and 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 9.

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 9. 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 8. A bounding heat load from the additional spent fuel to be received during Cycle 9 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.

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.

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 amount of water coverage in the fuel pools to reduce the potential doses resulting from a fuel handling accident. This minimum water depth provides sufficient iodine removal capability to maintain both the whole body and thyroid doses well within the acceptable limits of 10CFR100 which forms the basis for this Technical Specification and the fuel handling accident doses described in Chapter 15. 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 level falls below the minimum required.

The fuel handling accident described in Section 15.7.4 was evaluated with a dropped PWR fuel assembly impacting a stored PWR fuel assembly and ultimately coming to rest in a horizontal position on top of BWR fuel assemblies seated in the BWR fuel storage racks. This scenario results in the minimum water depth above the dropped fuel assembly, which is utilized to determine conservative decontamination factors used for the removal of iodines assumed in the accident evaluation. Assumptions and inputs supporting the fuel handling accident evaluation are located in Section 15.7.4. Maintaining water level in accordance with Technical Specification 3.9.11 assures that water coverages and decontamination factors used in the Chapter 15 fuel handling accident analysis remain bounding.

Alarms are provided for the indication of fuel pool water levels. Alarms for both high and low water levels indicate changing conditions in the pools. The fuel pool low level alarm indicates the minimum required water depth. An additional alarm set at a lower fuel pool water level indicates degraded pool water capacity conditions. The high level alarm provides equipment protection as well as inventory control during pool makeup and water transfer activities.

Normal makeup for evaporative losses and small amounts of system leakage from the fuel pools is accomplished using the Demineralized Water System (DWS), although other sources, such as from the reactor makeup water storage tank or the recycle holdup tank, may also be used. The DWS connects to the fuel pools and refueling water purification pumps, spent fuel pools cooling pumps, and fuel pools skimmer pumps to permit makeup to the fuel pools, or may be directly added to the pools via hoses. The seismic Category I Refueling Water Storage Tank (RWST) may also be aligned to provide borated makeup water to the fuel pools, and a seismic Category I source of emergency makeup water is available from the Emergency Service Water (ESW) system, by connecting flexible hoses to connections on the ESW and fuel pool cooling and cleanup system piping.

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 9.1.3-1A deleted by Amendment No. 48

Table 9.1.3-1A deleted by Amendment No. 48

Table 9.1.3-1B deleted by Amendment No. 48

Table 9.1.3-1B deleted by Amendment No. 48

Table 9.1.3-1C deleted by Amendment No. 48

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	16.84×10^6 Btu/hr
Full Core Offload Shuffle	35.06×10^6 Btu/hr
Post Outage Full Core Offload	35.87×10^6 Btu/hr
Fuel Pool Equilibrium Temperature**	
Incore Shuffle	≤137°F
Full Core Offload Shuffle	≤137°F
Post Outage Full Core Offload	≤137°F
Combined Spent and New Fuel Heat Pool Heat Inertia	
Incore Shuffle	4.37°F hr
Full Core Offload Shuffle	9.09°F hr
Post Outage Full Core Offload	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 9 with the bounding heat load from post RFO-8 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 9.1.3-3 Deleted by Amendment No. 43

Gratton Exhibit 2

January 12, 1987

Docket No. 50-400

DISTRIBUTION (*w/tech specification)

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	J. Guillen	

Mr. E. E. Utley, Senior Executive
Vice President
Power Supply and Engineering
and Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

Subject: Issuance of Facility Operating License No. NPF-63
Shearon Harris Nuclear Power Plant, Unit 1

The NRC has issued the enclosed Facility Operating License No. NPF-63 together with the Technical Specifications and Environmental Protection Plan for the Shearon Harris Nuclear Power Plant, Unit 1. The license authorizes operation of the Shearon Harris Nuclear Power Plant, Unit 1, at reactor power levels not in excess of 2775 megawatts thermal (100% of rated core power). Also enclosed is a Safety Evaluation which resolves several new issues or issues that remained to be resolved from the previous issuance of the Shearon Harris Safety Evaluation Report (NUREG-1038) and Supplements 1 through 4.

A copy of a related notice, the original of which has been forwarded to the Office of the Federal Register for publication, is also enclosed.

Three signed copies of Amendment No. 2 to Indemnity Agreement No. B-103, which covers the activities authorized under License No. NPF-63, are enclosed. Please sign all copies and return one copy to this office. License condition 2.B.8 provides that 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 may be received and possessed at the Shearon Harris plant. It is our understanding that such byproduct and special nuclear materials will not be received on the Shearon Harris site until the appropriate indemnity agreement amendment has been resolved.

Sincerely,
/s/
Thomas M. Novak, Acting Director
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Facility Operating License No. NPF-63
2. Safety Evaluation
3. Federal Register Notice
4. Amendment No. 2 to Indemnity Agreement No. B-103

cc w/enclosures: See next page *PREVIOUS CONCURRENCE SEE DATE

LA:PAD#2*	PM:PAD#2*	FOB*	EB*	RSB*	PSB*	EISCB*
DMiller	BBuckley:hc	VBenaroya	RBallard	CBerlinger	JMilhoan	FRosa
1/2/87	1/2/87	1/6/87	1/7/87	1/6/87	1/6/87	1/6/87
AD/PWR-A*	OGC*	P&RABS*	SP*	D:PAD#2	AD:DPLA	DD/NRR	D/NRR
ERossi	CBarth	WLambe	DNash	LRubenstein	TNovak	RVollmer	HDenton
1/12/87	1/7/87	1/7/87	1/7/87	1/12/87	1/12/87	/ /87	1/12/87

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

CAROLINA POWER & LIGHT COMPANY

NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

FACILITY OPERATING LICENSE

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;

*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.

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. , 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. 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.

Amdt. #6, 5-25-88

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.

FOR THE NUCLEAR REGULATORY COMMISSION

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachment 1 -
   TDI Diesel Engine Requirements
2. Appendix A - Technical
   Specifications
3. Appendix B - Environmental
   Protection Plan
4. Appendix C - Antitrust Conditions

Date of Issuance: January 12, 1987

Gratton Exhibit 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 26, 1996

MEMORANDUM TO: Chairman Jackson
Commissioner Rogers
Commissioner Dicus

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL ACTION PLAN ISSUES

In a meeting with Chairman Jackson on February 1, 1996, regarding spent fuel pool issues, the staff committed to prepare a course of action for resolving significant issues developed through the staff's Task Action Plan for Spent Fuel Storage Pool Safety. The significant issues examined within the framework of that plan were the reliability of spent fuel pool decay heat removal and the maintenance of an adequate spent fuel coolant inventory in the spent fuel pool. The staff was also directed to identify plant-specific and generic areas for regulatory analyses in support of further regulatory action.

The staff has completed its review and evaluation of design features related to the spent fuel pool associated with each operating reactor. Details of the staff's review and evaluation are presented in the attached report. The staff classified operating reactors on the basis of specific design features associated with the spent fuel pool in the following areas: coolant inventory control, coolant temperature control, and fuel reactivity control.

In comparing design features with NRC design requirements and guidance, the staff determined that design features related to coolant inventory control and reactivity control were more consistent with NRC guidance than were design features associated with coolant temperature control. The staff concluded that coolant inventory control design features were more consistent with present guidance because the staff had issued explicit guidance for prevention of coolant inventory loss in the form of design criteria before it issued most construction permits for currently operating reactors. These criteria are documented in plant specific AEC Design Criteria in each affected facility's safety analysis report; in the General Design Criteria of Appendix A to 10 CFR Part 50, which became effective in 1971; and in Safety Guide 13 (now Regulatory Guide 1.13), "Spent Fuel Storage Facility Design Basis," which was issued in March 1971. The staff concluded that reactivity control provisions are consistent because nearly all operating reactors have increased their spent fuel pool storage capacity since the NRC issued specific guidance for reactivity control, and such increases involve design and analysis of new fuel storage racks for criticality prevention. Conversely, the NRC staff did not issue specific guidance on the design of spent fuel pool cooling systems until the issuance of the Standard Review Plan (NUREG-75/087) in 1975, which was

CONTACT: Steven Jones, NRR
415-2833

Enclosure

plan for spent fuel storage pool safety, the staff determined that three of the seven license amendments for spent fuel storage capacity increases were approved on the basis of substantial hardware modification to the SFP cooling system. Despite the hardware modifications necessary to satisfy the staff acceptance criteria at the time of the increase in spent fuel storage capacity, the staff did not identify the fuel storage issue as an SEP "lessons learned" issue.

3.0 PARAMETERS AFFECTING THE SAFE STORAGE OF IRRADIATED FUEL

3.1 Coolant Inventory

The coolant inventory in the SFP protects the fuel cladding by cooling the fuel, protects operators by serving as shielding, decreases fission product releases from postulated fuel handling events by retaining soluble and particulate fission products, and supports operation of forced cooling systems by providing adequate net positive suction head. Adequate cooling of the fuel and cladding is established by maintaining a coolant level above the top of the fuel (however, this condition does not ensure that the SFP structure and other non-fuel components will not be degraded by high temperature). A water depth of several feet above the top of irradiated fuel assemblies stored in racks serves as acceptable shielding, but additional water depth is necessary to provide adequate shielding during movement of fuel assemblies above the storage racks and to maintain operator dose as low as is reasonably achievable (ALARA). Consequence analyses for fuel handling accidents typically assume a water depth of 23 feet above the top of irradiated fuel storage racks, and this value is specified as a minimum depth for fuel handling operations in the NRC's Standard Technical Specifications. Because cooling system suction connections to the SFP are typically located well above the top of stored fuel to prevent inadvertent drainage, a substantial depth of water above the top of fuel storage racks is necessary to provide adequate net positive suction head for forced cooling system pumps.

Design features to reduce the potential for a loss of coolant inventory are common. On the basis of the staff's design review, all operating reactors have a reinforced-concrete SFP structure designed to retain their function following the design-basis seismic event (i.e., seismic Category I or Class 1) and a welded, corrosion-resistant SFP liner. Only one operating reactor lacks leak detection channels positioned behind liner plate welds to collect leakage and direct the leakage to a point where it can easily be monitored. Nearly all operating reactors have passive features preventing draining or siphoning of the SFP to a coolant level below the top of stored, irradiated fuel. Excluding paths used for irradiated fuel transfer, passive features at nearly all operating reactors prevent draining or siphoning of coolant to a level that provides inadequate shielding for fuel seated in the storage racks.

In the event that SFP coolant inventory decreases significantly, several indications are available to alert operators of that condition. The primary indication is a low-level alarm. A secondary indication of a loss of coolant level is provided by area radiation alarms. These alarms indicate a loss of shielding that occurs when SFP coolant inventory is lost. Except for the SFP located inside the containment building, the area radiation alarms are set to after the issuance of most construction permits for currently operating reactors, and spent fuel storage capacity increases have seldom involved a sufficient increase in decay heat generation that an expanded cooling system was warranted.

The staff has found that existing structures, systems, and components related to storage of irradiated fuel provide adequate protection for publ[illegible] health and safety. Prctection has been provided by several layers of d[illegible]ses that perform accident prevention functions (e.g., quality controls on design, construction, and operation), accident mitigation functions (e.g., multiple cooling systems and multiple makeup water paths), radiation protection functions, and emergency preparedness functions. Design features addressing each of these areas for spent fuel storage have been reviewed and approved by the staff. In addition, the limited risk analyses available for spent fuel storage suggest that current design features and operational constraints cause issues related to spent fuel pool storage to be a small fraction of the overall risk associated with an operating light water reactor. Notwithstanding this finding, the staff has reviewed each operating reactor's spent fuel pool design to identify strengths and weaknesses, and to identify potential areas for safety enhancements.

The staff plans to address certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. We intend to pursue regulatory analyses for safety enhancement backfits on a plant-specific basis pursuant to 10 CFR 50.109 at the small number of operating reactors possessing each particular identified design feature. The specific plans for safety enhancement backfits and their bases are described in the attached report. Because of the relatively low safety significance of these issues, the staff recognizes that some, or all, of these potential enhancements may not pass the backfit tests.

The staff will provide the attached report to the licensees of all operating reactors. The staff intends to request that those licensees identified in the report for plant-specific regulatory analysis verify the applicability of the staff's findings and conclusions. The staff will also request that licensee's provide, on a voluntary basis, their perspective on the potential increase in the overall protection of public health and safety and information regarding the cost of potential modifications to address the design features identified in the staff report. Staff reviews of potential plant-specific or generic backfits will be appropriately coordinated with the Committee to Review generic Requirements (CRGR).

The staff also plans to address issues relating to the functional performance of spent fuel pool decay heat removal, as well as the operational aspects related to coolant inventory control and reactivity control, through expansion of the proposed, performance-based rule, "Shutdown Operations at Nuclear Power Plants" (10 CFR 50.67), to encompass fuel storage pool operations.

Concurrent with the regulatory analyses for the potential safety enhancements, the staff will develop guidance for implementing the proposed rule for fuel storage pool operations at nuclear power plants. The staff will also develop plans to improve existing guidance documents related to design reviews of spent fuel pool cooling systems. In addition, the staff will issue an information notice as a mechanism for distributing information in areas where regulatory analyses do not support rulemaking or plant-specific backfits.

Attachment: Plan for Resolving Spent Fuel Storage Pool Action Plan Issues alarm at a level low enough to detect a loss of coolant inventory early enough to allow for recovery before radiation levels could make such a recovery difficult.

The staff noted five categories of operating reactors that warrant further review based on specific design features that are contrary to guidance in Regulatory Guide 1.13. These categories are described in the next five sections.

3.1.1 Spent Fuel Pool Siphoning via Interfacing Systems

The SFPs serving four operating reactors lack passive anti-siphon devices for piping systems that could, through improper operation of the system, reduce coolant inventory to a level that provides insufficient shielding and eventually exposes stored fuel. These four operating reactors, all issued construction permits preceding the issuance of Safety Guide 13, have piping that penetrates the SFP liner several feet above the top of stored fuel, but the piping extends nearly to the bottom of the SFPs. Because, for each of these reactors, this piping is connected to the SFP cooling and cleanup system through a normally locked closed valve and lacks passive anti-siphon protection, mispositioning of the normally locked-closed valve coincident with a pipe break or refueling water transfer operation could reduce the SFP coolant inventory by siphon flow to a level below the top of the stored fuel.

This concern is related to a 1988 event at San Onofre Unit 2, which involved a partial loss of SFP coolant inventory due to an improper purification system alignment and inadequate anti-siphon protection. The NRC issued Information Notice 88-65, "Inadvertent Drainages of Spent Fuel Pools," to alert holders of operating licenses and construction permits of this event and similar system misalignments. Although the coolant inventory loss at San Onofre Unit 2 was not significant in this instance, the piping extended deep enough in the pool that failure of operator action to halt the inventory loss would have been of concern. Corrective action for this event included removing the portion of piping that extended below the technical specification limit on SFP level and strengthening administrative controls on system alignment.

Reduction in coolant inventory to an extremely low level is unlikely because of the low probability of the necessary coincident events, the long time period necessary for significant inventory loss through small siphon lines, and the many opportunities afforded operators to identify the inventory loss (e.g., SFP low-level alarm, SFP area high-radiation alarms, building sump high-level alarms, observed low level in SFP, and accumulation of water in unexpected locations). However, the staff believes that a design modification to introduce passive anti-siphon protection for the SFP could be easily implemented at the plants currently lacking this protection. Therefore, the staff will conduct a regulatory analysis to determine if such modifications are justified.

3.1.5 Absence of Isolation Capability for Leakage Collection System

The absence of isolation capability for leakage identification systems could allow water to leak at a rate in excess of make-up capability for certain events that cause failure of the SFP liner. The staff identified four operating reactors with this characteristic, but this item was not included in our previous information collection efforts. However, the staff also has not collected the information necessary to evaluate makeup capability relative to credible leakage through the leakage detection channels. To address this omission, the staff will examine previous licensing reviews to determine if the staff had previously evaluated makeup capability relative to credible coolant inventory loss through the leakage detection channels. Because the four plants identified with this characteristic were not evaluated for inventory control using the SRP guidance, the staff believes that the depth of review for these plants would be indicative of the depth of review at other operating reactors. If this issue has not been previously addressed by the staff at the four operating reactors, the staff will initiate additional information collection activities for this design characteristic and conduct a regulatory analysis to determine if modification to the leakage detection system is justified.

3.2 Coolant Temperature

Coolant temperature has a less direct effect on safe storage of irradiated fuel than coolant inventory. Coolant temperature at the pool surface is limited by evaporative cooling from the free surface of the pool to a value of about 100°C [212°F], and the design of the pool storage racks provides adequate natural circulation to maintain the coolant in a subcooled state at the fuel cladding surface assuming the coolant inventory is at its normal level. Therefore, forced cooling is not required to protect the fuel cladding integrity when adequate water is supplied to makeup for coolant inventory loss. The temperature of the SFP does have an effect on structural loads, the operation of SFP purification systems, operator performance during fuel handling, and the environment around the SFP.

3.2.1 Structural Considerations

The SFP structure is evaluated to ensure that its structural integrity and leak tightness are retained under various operating, accidental, and environmental loadings. The reinforced concrete SFP walls and floors are required to withstand the loadings without exceeding the corresponding allowables set forth in the American Concrete Institute Code requirements for Nuclear Structures (ACI 349) as modified by Regulatory Guide 1.142. Appendix A, "Thermal Consideration," of ACI 349 limits the long-term temperature exposure of concrete surfaces to 150° F, and short term exposures temperature (under accident conditions) to 350° F. It permits long term temperature exposures higher than 150° F, provided tests are performed to evaluate reductions in the concrete strengths and elastic modulus, and these reductions are applied to design allowables. During the approval of Amendments related to reracking of SFPs, the staff reviews the structural, thermal and seismic loadings on the SFPs and the proposed storage racks to ensure their compliance with the regulatory provisions (relevant SRPs and Regulatory Guides).

Gratton Exhibit 4

TAC MA4432

CP&L

Carolina Power & Light Company
Harris Nuclear Plant
P.O. Box 165
New Hill NC 27562

SEP 3 1999

SERIAL: HNP-99-129

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING AMENDMENT REQUEST TO INCREASE FUEL
STORAGE CAPACITY

Dear Sir or Madam:

By letter HNP-98-188, dated December 23, 1998, Carolina Power & Light Company (CP&L) submitted a license amendment request to increase fuel storage capacity at the Harris Nuclear Plant (HNP) by placing spent fuel pools C & D in service. The U. S. Nuclear Regulatory Commission (NRC) issued letters dated March 24, 1999, April 29, 1999, and June 16, 1999 requesting additional information regarding our license amendment application. HNP letters HNP-99-069, dated April 30, 1999, HNP-99-094, dated June 14, 1999, and HNP-99-112, dated July 23, 1999 provided our respective responses.

By letter dated August 5, 1999, the NRC issued a fourth request for additional information (RAI) regarding our license amendment application to place spent fuel pools C & D in service. The Enclosures to this letter provides the HNP response to the NRC staff's August 5, 1999 RAI.

The enclosed information is provided as supplement to our December 23, 1998 amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

for D. B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosures:

1. HNP Responses to NRC Request For Additional Information (RAI)
2. Calculation SF-0040, entitled "Spent Fuel Pools C and D Activation Project Thermal-Hydraulic Analysis" (w/o Attachments)
3. Calculation SF-0041, entitled "Harris Fuel Pool Heatup Calculation"
4. Attachment Z to Calculation SF-0040 - Evaluation of CCW System LOCA-Containment Sump Recirculation (RHR and SFP) Alignment Thermal Performance

c: Mr. J. B. Brady, NRC Senior Resident Inspector (w/ Enclosure 1)
Mr. Mel Fry, N.C. DEHNR (w/ Enclosure 1)
Mr. R. J. Laufer, NRC Project Manager (w/ all Enclosures)
Mr. L. A. Reyes, NRC Regional Administrator - Region II (w/ Enclosure 1)

bc: (all w/ Enclosure 1)

Mr. K. B. Altman
Mr. G. E. Attarian
Mr. R. H. Bazemore
Mr. C. L. Burton
Mr. S. R. Carr
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
Mr. W. J. Dorman (BNP)
Mr. R. S. Edwards
Mr. R. J. Field
Mr. K. N. Harris
Ms. L. N. Hartz
Mr. W J. Hindman
Mr. C. S. Hinnant
Mr. W. D. Johnson
Mr. G. J. Kline
Mr. B. A. Kruse
Ms. T. A. Head (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. J. H. O'Neill, Jr.
Mr. J. S. Scarola
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File
Files: H-X-0511
H-X-0642

Requested Information Item 5: Chris

Section 5.4.1 of Enclosure 6 discusses time-to-boil assuming a complete loss of cooling to spent fuel pools C and D. The analysis assumes a decay heat load of 15.63 MBtu/hr, which results in a heat up rate of 5.4 °F/hr. Given that the storage pools are limited to 1.0 MBtu/hr by the proposed TS, provide a pool heat up analysis using a decay heat rate of 1.0 MBtu/hr. In addition, discuss the available makeup sources to spent fuel pools C and D and their capacities relative to the calculated boil off rate.

Response 5:

The time to boil and pool heatup analyses for the 1.0 MBtu/hr scenario are well bounded by the time to boil and pool heatup analyses for the 15.63 MBtu/hr scenario presented in Section 5.4.1 of Enclosure 6 to the license amendment request. Analyses specific to 1.0 MBtu/hr have been performed and are documented in HNP Fuel Pool Heatup Calculation SF-0041, provided herein as Enclosure 3. These analyses calculate an estimated pool heatup rate of approximately 0.33 °F /hr, and conclude that the pools would not heat up to 140 °F until approximately 100 hours into the event, and an additional 200 hours would be required to reach boiling conditions.

Requested Information Item 6: Hsii

The USQ analysis results of Enclosure 9 indicate that a minimum CCW system (CCWS) flow rate of 4874 gpm at 120°F is required at the beginning of the containment sump recirculation phase of a LOCA and that, assuming a 6% model uncertainty and degraded inservice test (IST) pump performance, the specified CCW flow to the residual heat removal (RHR) HX would be 5166 gpm, which is less than 5600 gpm in the existing analysis. This result is based on (1) the RHR HX heat rejection rate of 111.1 MBtu/hr, which is said to be consistent with the existing post-LOCA containment pressure/temperature calculations, and (2) the use of a "dynamic" RHR HX performance model, in which the tube side inlet temperature is postulated to rise to 244.1°F during the initial phase of sump recirculation, rather than a fixed 139°F assumed in the existing analysis, resulting in an increase of the overall RHR HX heat transfer coefficient (HTC) of approximately 10% due to change in tube side viscosity.

Provide the dynamic RHR HX heat transfer analysis during and subsequent to the recirculation phase of a LOCA. Important parameters to be provided include the time-dependent decay heat rate, the containment sump water temperature, and the HTCs, heat transfer rates and flow rates (both tube and shell sides) of the RHR HX, and CCW HX, etc. Also describe how the effects of HX degradation mechanism such as fouling and tube plugging of the RHR and CCW systems are accounted for in the RHR and CCW HX heat transfer calculations.

ENCLOSURE 3 to SERIAL: HNP-99-129

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO INCREASE FUEL STORAGE CAPACITY

Calculation SF-0041

Harris Fuel Pool Heatup Calculation

Title/Approval Sheet

SYSTEM# 7110

CALC. TYPE Mechanical

CAROLINA POWER & LIGHT COMPANY

SF-0041
(CALCULATION #)

FOR

Harris Fuel Pool Heatup Calculation
(TITLE INCLUDING STRUCTURE/SYSTEM/COMPONENT)

FOR

SHEARON HARRIS NUCLEAR POWER PLANT X

NUCLEAR ENGINEERING DEPARTMENT

QUALITY CLASS X A ☐ B ☐ C ☐ D ☐ E

REV. NO.	RESPONSIBLE ENGINEER DATE	☒ DESIGN VERIFIED BY ☐ ENGINEERING REVIEW BY DATE	APPROVED BY RESPONSIBLE SUPERVISOR DATE
0	Jeff Lundy 10/26/98	Merid Aboye 10/26/98	R. Steven Edwards 11-16-98
REASON FOR CHANGE			
REASON FOR CHANGE			

Computed by: Jeff Lundy	Date: 10/26/98	CAROLINA POWER & LIGHT COMPANY	Calculation ID: SF-0041	
Checked by: Merid Aboye	Date: 10/26/98	CALCULATION SHEET	Pg i of 5	Rev 0
Project No.:			File:	
Project Title: Spent Fuel Pools C and D Activation Project				
Calculation Title: Harris Fuel Pool Heatup Calculation				

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Computed by: Jeff Lundy	Date: 10/26/98	CAROLINA POWER & LIGHT COMPANY	Calculation ID: SF-0041	
Checked by: Merid Aboye	Date: 10/26/98		Pg ii of 5	Rev 0
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RSL 11/11/98

Computed by: Jeff Lundy	Date: 10/26/98	CAROLINA POWER & LIGHT COMPANY	Calculation ID: SF-0041	
Checked by: Merid Aboye	Date: 10/26/98		Pg 1 of 5	Rev 0
Project No.:		CALCULATION SHEET	File:	
Project Title: Spent Fuel Pools C and D Activation Project				
Calculation Title: Harris Fuel Pool Heatup Calculation				

1.0 PURPOSE

The purpose of this calculation is to determine the rates of temperature increase in the spent fuel pool for the heat loads associated with an assumed inventory of Spent Fuel Pool C which generates 1.0 MBTU/hr of decay heat. The times required to raise Spent Fuel Pool C bulk temperature from 105°F to 137°F, 105°F to 150°F and 105° to 212°F will also be calculated based on the above heat load. For this calculation, "spent fuel pool" shall mean pool C only.

2.0 REFERENCES

(1) Harris Nuclear Plant Calculation SF-0038 Revision 0, Spent Fuel Pool Heat Up Rate / Time to Boil Calculation, dated 2/21/97

(2) Harris Nuclear Plant Operating Procedure, OP-116

(3) Harris Nuclear Plant Calculation HNP-F/NFSA-0026 Revision 0, Maximum Decay Heat Load for Spent Fuel Pools A,B&C Through the End of the Year 2001, dated 4/16/98

(4) ASME Steam Tables, Fifth Edition

(5) Holtec International Drawing 1994, Revision 1, "Pool Layout for Pool "C" – PWR & BWR Spent Fuel Storage Racks", dated 3/18/98

(6) Harris Nuclear Plant Drawing CAR 2165-G-411 Revision 9

3.0 ENGINEERING ANALYSIS SOFTWARE

None Used.

4.0 CALCULATION

This calculation will be performed by:

1) Calculating the net water volume in Spent Fuel Pool C,

2) Calculating the fuel pool heat up rate associated with a fuel pool inventory generating 1.0 MBTU/hr and

3) Calculating the time to reach 137°F, 150°F and 212°F using the calculated heat up rate.

4.1 Assumptions

4.1.1 Water Densities for 137°F, 150°F and 212°F will be used for conservatism.

ρ_{137F} =1/0.016279 = 61.43 lbm/ft^3 for 137°F, Reference (4)

ρ_{150F} =1/0.016343 = 61.19 lbm/ft^3 for 150°F, Reference (4)

ρ_{212F} =1/0.016719 = 59.81 lbm/ft^3 for 212°F, Reference (4)

4.1.2 Only the volume of Spent Fuel Pool C will be considered.

4.1.3 Evaporative heat losses to the Fuel Handling Building ambient and conduction losses through the fuel pool liner are NOT credited for heat removal for this analysis.

4.1.4 Fuel Pool Water heat capacity (Cp) = 1.0 BTU/lbm-°F.

4.1.5 The spent fuel racks are constructed of 304 stainless steel with a specific weight of 0.29 lbm/ft^3, Reference (1)

4.1.6 Pool C contains 539 BWR spent fuel assemblies, Reference (3).

4.1.7 The spent fuel assembly decay heat is 1.0 MBTU/hr for Spent Fuel Pool C, Reference (3).

4.1.8 The initial steady state pool equilibrium temperature is 105°F, Reference (2).

4.1.9 This calculation only considers Phase I of the Spent Fuel Pool C activation which includes fuel rack modules A1, A2, B1, B2, C1, C2, D1, D2, E1, E2, E3, F1, F2, and F3 per CP&L Project Engineering direction.

4.1.10 This analysis assumes a total loss of SFP C cooling, therefore the heat removal by the SFP heat exchangers is 0 BTU/hr.

4.2 Estimate of Water Volume in Spent Fuel Pool C

The volume for a BWR fuel assembly, 1.164 ft^3, is taken from Reference (1)

The volume for the fuel racks is calculated as follows:

Fuel Rack Specific Weight = 0.29 lbm/in^3, Reference (1)
Fuel Rack Volume = Empty Weight /Specific Weight/1728

Module ID	Rack Capacity	Rack Empty Weight (lb)	Specific Weight (lb/in^3)	Rack Volume (ft^3)
A1	11 x 9 (PWR)	14770	0.29	29.47
A2	11 x 9 (PWR)	15620	0.29	30.45
B1	9 x 9 (PWR)	12250	0.29	24.45
B2	9 x 9 (PWR)	12940	0.29	25.82
C1	8 x 13 (BWR)	9710	0.29	19.38
C2	8 x 13 (BWR)	9710	0.29	19.38
D1	8 x 11 (BWR)	8460	0.29	16.88
D2	8 x 11 (BWR)	8460	0.29	16.88
E1	13 x 13 (BWR)	15370	0.29	30.67
E2	13 x 13 (BWR)	15700	0.29	31.33
E3	13 x 13 (BWR)	15700	0.29	31.33
F1	13 x 11 (BWR)	13380	0.29	26.70
F2	13 x 11 (BWR)	13380	0.29	26.70
F3	13 x 11 (BWR)	13380	0.29	26.70
			Total Volume =	356.14

The minimum gross volume of the fuel pool is:
(Low Level Alarm - SFP Floor Elevation) x Pool Length x Pool Width
where:

Low Level Alarm = 284 ft, Reference (1)
SFP Floor Elevation = 246 ft, Reference (6)
Pool Length = 50 feet, Reference (5)
Pool Width = 27 feet, Reference (5)

Gross Volume = (284 ft - 246 ft) x 50 ft x 27 ft = 51300 ft^3

The net pool water volume is:
Net Pool Water Volume = Gross Volume - Fuel Assy Volume - Total Fuel Rack Volume
Net Pool Water Volume = 51300 - 539 x 1.164 – 356.14 = 50316.5 ft^3

The net pool water thermal mass is:
Net Pool Water Thermal Mass = ρ_{SFP} x V_{SFP} x C_p
Net Pool Water Thermal Mass at 137°F = 61.43 x 50316.5 x 1.0 = 3090942.6 BTU/hr-°F
Net Pool Water Thermal Mass at 150°F = 61.19 x 50316.5 x 1.0 = 3078866.6 BTU/hr-°F
Net Pool Water Thermal Mass at 212°F = 59.81 x 50316.5 x 1.0 = 3009429.9 BTU/hr-°F

4.3 Estimate of Transient Spent Fuel Pool Thermal Performance

An estimate of the transient thermal performance of spent fuel pool C was performed to determine the time to various fuel pool temperatures during a loss of CCW cooling scenario. The transient analysis conservatively assumes a decay heat load of 1.0 MBTU/hr while only accounting for the net water volume of the fuel pool and neglecting changes in the water thermal properties.

In general, fuel pool heatup thermal transients are calculated from:

$$\rho \cdot C_p \cdot V \frac{dT}{dt} = Q_{DecayHeat} - Q_{SFPHx} \qquad \text{Equation (1)}$$

where:

ρ = Pool Water Density (lbm/cuft) at the Specified Pool Temperature

C_p = Pool Water Specific Heat (BTU/lbm/F)

V = Pool Net Water Volume (cuft)

$Q_{DecayHeat}$ = 1.0 MBTU/hr

$Q_{SFPHx} = 0$

$$HeatUpRate = \frac{dT}{dt} = \frac{Q_{DecayHeat} - Q_{SFPHx}}{\rho_{SFP} \cdot C_p \cdot V_{SFP}} \qquad \text{Equation (2)}$$

It is conservatively assumed that the fuel pool is at the maximum temperature limit of 105°F, Reference (2), prior to the thermal transient. This analysis also assumes no operator action with respect to the fuel pools.

4.4 Estimate of Spent Fuel Pool C Heat Up Rates

For the given parameters for Spent Fuel Pool C, the estimated heat up rates are:

The heatup rate at 137°F is:

$$HeatUpRate_{137F} = \frac{1.0\text{E6 BTU/hr}}{3090942.6\ \text{BTU/hr} - \text{F}} = 0.324\ \text{F/hr}$$

The heatup rate at 150°F is:

$$HeatUpRate_{150F} = \frac{1.0\text{E6 BTU/hr}}{3078866.6\ \text{BTU/hr} - \text{F}} = 0.325\ \text{F/hr}$$

The heatup rate at 212°F is:

$$HeatUpRate_{212F} = \frac{1.0E6\ \text{BTU/hr}}{3009429.9\ \text{BTU/hr} - \text{F}} = 0.332\ \text{F/hr}$$

4.5 Estimate of Time to Reach 137°F

The time to reach 137°F is:

$$Time_{137F} = \frac{137F - 105F}{0.324} = 98.8\ \text{hr}$$

4.6 Estimate of Time to Reach 150°F

The time to reach 150°F is:

$$Time_{150F} = \frac{150F - 105F}{0.325} = 138.5\ \text{hr}$$

4.7 Estimate of Time to Reach 212°F

The time to reach pool boiling conditions of 212°F is:

$$Time_{212F} = \frac{212F - 105F}{0.332} = 322.3\ \text{hr}$$

5.0 CONCLUSIONS

This analysis concludes that the estimated Spent Fuel Pool C heat up rate at 137°F, 150°F and 212°F is 0.324°F/hr, 0,.325°F/hr and 0.332°F/hr, respectively, for a maximum decay heat load of 1.0 MBTU/hr. The time to reach 137°F, 150°F and 212°F from the 105°F administrative limit is 98.8 hours, 138.5 hours and 322.3 hours, respectively.