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 RICHARD J. LAUFER 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 2 AND 3

I, Richard J. Laufer, having first been duly sworn, do hereby state as follows:

1. My name is Richard J. Laufer. I am employed as a Project Manager in Project Directorate II - Section 2, Division of Licensing Project Management in the Office of Nuclear Reactor Regulation (NRR). I serve as the NRR focal point for my assigned facility, currently the Shearon Harris Nuclear Power Plant (HNP). I am the principal point of contact for NRR with the Region and the licensee for those activities related to HNP. I have been an NRR Project Manager since July 1993, and have been assigned to HNP since February 1999. A statement of my professional qualifications is attached. (Attachment 1).

2. The purpose of this testimony is to provide background information related to the design and licensing of HNP, and to Carolina Power & Light Company's (CP&L's) December 23, 1998 amendment request (Exhibit 1) which prompted the petition to intervene

that resulted in this hearing granted by the Atomic Safety and Licensing Board in its 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).

3. HNP is a three-loop Westinghouse pressurized water reactor (PWR) operated by CP&L in Wake and Chatham Counties, North Carolina. Shearon Harris Nuclear Power Plant Final Safety Analysis Report (Exhibit 2, page 1.1-1). The NRC issued the construction permit for HNP, which was originally planned as a four nuclear unit site, on January 27, 1978. NUREG-1038, "Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983 (Exhibit 3, page 1-1). In order to accommodate four units at HNP, the fuel handling building (FHB) was designed and constructed with four separate pools capable of storing spent fuel. The two pools at the south end of the FHB, now known as spent fuel pools (SFPs) A and B were to support HNP Units 1 and 4. The two pools at the north end of the building were to support HNP Units 2 and 3. The multi-unit design included a SFP cooling and cleanup system to service SFPs A and B and a separate cooling and cleanup system to support SFPs C and D (Exhibit 1, Enclosure 1, page 1).

4. HNP Units 3 and 4 were canceled in late 1981 (Exhibit 3, page 1-1) and HNP Unit 2 was canceled in late 1983. Letter From Thomas A. Baxter to the Atomic Safety and Licensing Board dated December 21, 1983 (Exhibit 4). The Construction Permits for HNP Units 2, 3, and 4 expired on June 1, 1986, June 1, 1990, and June 1, 1987, respectively.

Federal Register Notice of Issuance of Construction Permits, 43 FED. REG 4465 (1978). (Exhibit 7) The FHB, all four SFPs (including liners), and the cooling and cleanup system to support SFPs A and B were completed (Exhibit 1, Enclosure 1, page 1). The construction on the SFP cooling and cleanup system for SFPs C and D was not completed by the time Unit 2 was canceled. However, the majority of the mechanical piping and equipment associated with the operation of SFPs C and D was already installed, including all of the embedded and most of the exposed portions of ASME Section III piping associated with the cooling system (Exhibit 1, Enclosure 8, page 2).

5. The staff's Safety Evaluation Report, NUREG-1038, issued in November 1983 (prior to the cancellation of HNP Unit 2), evaluates the design of all four pools SFPs and the associated cooling systems (Exhibit 3, Chapter 9). The SFP cooling system was designed to consist of one cooling system for each unit. Each cooling system was designed to have two trains of cooling; each train consisting of a heat exchanger, strainer, and cooling pump (Exhibit 3, page 9-5). The cooling for the SFP heat exchangers was to be provided by the component cooling water (CCW) system for the respective unit (Exhibit 3, page 9-6). The pools were designed to store both PWR and boiling water reactor (BWR) fuel (Exhibit 3, page 9-3). The cooling system was designed to handle the heat load of "... a full inventory of fuel assemblies, including PWR and BWR spent fuel from the H. B. Robinson and Brunswick plants" (Exhibit. 3, page 9-6). The current SFP cooling system is described in Chapter 9 of the HNP Final Safety Analysis Report (Exhibit 2). It states that "[s]pent fuel

may not be loaded into Pools C or D until they are completed and made operational" (Exhibit 2, page 9.1.3-1).

6. The NRC issued Facility Operating License No. NPF-63 for the operation of HNP Unit 1 on January 12, 1987 (full-power license) (Exhibit 5). Paragraph 2.B.(8) of the license authorized HNP to receive and store spent fuel from its other nuclear plants (Brunswick Units 1 and 2, and H. B. Robinson, Unit 2) (Exhibit 5, page 3). As permitted by its Operating License, CP&L has implemented a spent fuel shipping program. Spent fuel from Brunswick (2 BWR units) and Robinson (1 PWR unit) is shipped to HNP for storage in SFPs A and B. CP&L ships fuel to HNP in order to maintain full core offload capability at Brunswick and Robinson. As a result of the operation of HNP, shipping program requirements, and the unavailability of a Department of Energy (DOE) storage facility, CP&L has determined that it would be necessary to activate SFPs C and D and the associated cooling and cleanup systems by early in the year 2000. Activation of these two pools will provide storage capacity for all four CP&L nuclear units through the end of their current licenses (Exhibit 1, Enclosure 1, page 1 of 6).

7. On December 23, 1998, CP&L submitted a license amendment request to support placing SFPs C and D in service (Exhibit 1). The proposed action consists of three parts:

a. A revision to Technical Specification (TS) 5.6 to identify pressurized water reactor (PWR) burnup restrictions, boiling water reactor (BWR) enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D' (Exhibit 1, Enclosure 1, page 2).

CP&L is proposing to use higher density fuel racks in SFPs C and D than are currently used in SFPs A and B. The use of the higher density racks requires additional administrative controls on PWR burnup and BWR enrichment to ensure Keff is maintained less than or equal to 0.95 (Exhibit 1, Enclosure 1, page 1, 4; Enclosure 5, proposed TS page 5-7).

CP&L is also proposing to limit the heat load from fuel stored in SFPs C and D to 1.0 MBtu/hr (Exhibit 1, Enclosure 5, proposed TS page 5-7a). This value is consistent with the assumption used in the analysis to support the tie-in of the Unit 1 CCW system to the heat exchangers of the SFP C and D fuel pool cooling and cleanup system (Exhibit 1, Enclosure 9, page 1 of 4).

b. An alternative plan in accordance with the requirements of 10 CFR 50.55a to demonstrate an acceptable level of quality and safety in completion of the CCW and SFP 'C' and 'D' cooling and cleanup system piping (Exhibit 1, Enclosure 1, page 2).

In order to activate SFPs C and D, it is necessary to complete construction of the cooling and cleanup system for these pools and to install tie-ins to the existing HNP Unit 1 CCW system to provide heat removal capabilities. Approximately 80%

of the SFP cooling and cleanup system piping and the majority of the CCW piping was installed during the original plant construction. At the time that construction on the SFP cooling system was discontinued following cancellation of HNP Unit 2, a formal turnover of the partial system was not performed and CP&L has since discontinued its N Certificate program. Also, some of the field installation records for the completed piping are no longer available. As a result, the system when completed will not satisfy ASME Section III code requirements (i.e., will not be N stamped). Therefore, CP&L submitted an Alternative Plan in accordance with 10 CFR 50.55a(a)(3) to demonstrate that the completed system will provide an acceptable level of quality and safety (Exhibit 1, Enclosure 1, page 5).

c. An unreviewed safety question for additional heat load on the CCW system, which is not in contention in this proceeding (Exhibit 1, Enclosure 1, page 2).

8. On January 13, 1999, a Notice titled, "Carolina Power & Light; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Determination, and Opportunity for a Hearing," was published in the Federal Register, 64 FED. REG. 2237 (1999) (Exhibit 6).

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

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

Richard J. Laufer

RICHARD J. LAUFER

Sworn and Subscribed before me
this 4 day of January, 2000.

Circe E. Martin

Notary Public

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

My commission expires March 1, 2003

Richard J. Laufer

Experience:

2/99 - Present: NRC Project Manger - Shearon Harris Nuclear Power Plant

Serve as the Headquarters Focal Point for Information and Communication on all issues concerning the Shearon Harris Nuclear Power Plant. Maintain nearly daily communication with the licensee, the resident inspectors, and the regional staff. Participate in all significant licensee meetings in the region and on-site. Serve as Back-up Project Manager (PM) for another plant in the Project Directorate (currently H.B. Robinson).

Prepare and coordinate the numerous documents generated to support the licensing activities of the assigned plant. These documents include license amendments and exemptions and their associated environmental assessments and Federal Register Notice, Task Interface Agreement Responses, controlled correspondence, and numerous letters to the licensee associated with closing out Generic Letters, relief requests, and requests for additional information.

Coordinate, participate, and manage meetings and briefings by ensuring that the appropriate NRC contacts are informed, that meeting notices are prepared, and by preparing an accurate and concise meeting summary in a timely manner.

2/98 - 2/99: NRC Project Manager - Duane Arnold Energy Center

7/93 - 2/98: NRC Project Manager - Kewaunee Nuclear Power Plant

2/93 - 7/93: NRC Project Engineer - Division of Reactor Projects

5/89 - 2/93: NRC Operator Licensing Examiner - Operator Licensing Branch

- Certified NRC Operator Licensing Examiner on Westinghouse pressurized water reactors and non-power reactors

3/86 - 5/89: Engineering Division Officer on Navy nuclear submarine USS Vallejo (SSBN 658)

(Qualified as Engineering Officer of the Watch, Engineering Duty Officer)

Training:

1/90 Completed NRC's Westinghouse Technology Full Series Course

5/84- 3/86: Navy nuclear power training

<u>Education</u>:

5/84: B.S. Degree in Systems Engineering; U. S. Naval Academy, Annapolis, MD

Laufer Exhibit 1

CP&L

Carolina Power & Light Company
PO Box 165
New Hill NC 27562

James Scarola
Vice President
Harris Nuclear Plant

DEC 23 1998

SERIAL: HNP-98-188
10CFR50.90
10CFR50.59(c)
10CFR50.55(a)

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
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

Dear Sir or Madam:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) requests a license amendment to place spent fuel pools 'C' and 'D' in service. Specifically, Harris Nuclear Plant (HNP) proposes to revise TS 5.6 "Fuel Storage" to increase the spent fuel storage capacity by adding rack modules to pools 'C' and 'D'. The enclosures to this letter support the proposed license amendment.

Enclosure 1 provides background information, a description of the proposed changes, and the basis for the changes.

Enclosure 2 details, in accordance with 10 CFR 50.91(a), the basis for the CP&L's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment is required for approval of this amendment request.

Enclosure 4 provides page change instructions for incorporating the proposed revisions.

Enclosure 5 provides the proposed Technical Specification pages.

Enclosure 6 provides a report entitled "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D'" which contains supporting technical documentation. Please note that Enclosure 6 contains information which is considered proprietary pursuant to 10 CFR 2.790. In this regard, CP&L requests Enclosure 6 be withheld from public viewing.

Enclosure 7 is identical to Enclosure 6, except that the proprietary information has been removed and replaced by highlighting and/or a note of explanation at each location where the information has been omitted. CP&L provides this additional version for the purposes of public review.

5413 Shearon Harris Road New Hill NC Tel 919 362-2502 Fax 919 362-2095

Enclosure 8 provides a detailed description of the proposed alternatives to demonstrate compliance with ASME B&PV Code requirements for the cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i).

Enclosure 9 provides results of the thermal hydraulic analysis of the cooling water systems that support placing pools 'C' and 'D' in service. The analysis resulted in changes to previously reviewed and approved cooling water flow requirements. These changes have been identified as an unreviewed safety question and are being submitted for NRC review and approval pursuant to the requirements of 10 CFR 50.59(c) and 10 CFR 50.90.

CP&L requests the issuance date for this amendment be no later than December 31, 1999. This issuance date is necessary to support loading of spent fuel in pool 'C' starting in early 2000. CP&L also requests the proposed amendment be issued such that implementation will occur within 60 days of issuance to allow time for procedure revision and orderly incorporation into copies of the Technical Specifications.

Please refer any questions regarding this submittal to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

James Scarola

RSE/KWS/kws

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Technical Specification Pages
6. Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D' (proprietary version)
7. Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D' (non-proprietary version)
8. 10 CFR 50.55a(a)(3) Alternative Plan
9. Unreviewed Safety Question Analysis

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

Lisa M. Randall
Notary (Seal)

LISA M. RANDALL NOTARY PUBLIC WAKE COUNTY, N.C.

My commission expires: 6-7-2003

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. S. C. Flanders, NRC Project Manager
Mr. Mel Fry, Director, N.C. DRP
Mr. L. A. Reyes, NRC Regional Administrator

bc:
Ms. D. B. Alexander
Mr. K. B. Altman
Mr. G. E. Attarian
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
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. G. J. Kline
Ms. W. C. Langston (PE&RAS File)
Mr. R. D. Martin
Mr. J. W. McKay
Mr. P. M. Odom (RNP)
Mr. W. S. Orser
Mr. P. M. Sawyer (BNP)
Mr. J. M. Taylor
Nuclear Records
Licensing File
File: H-X-0512
File: H-X-0642

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

BASIS FOR CHANGE REQUEST

BASIS FOR CHANGE REQUEST

Background:

The Harris Plant was originally planned as a four nuclear unit site (Harris 1, 2, 3 and 4). In order to accommodate four units at Harris, the Fuel Handling Building (FHB) was designed and constructed with four separate pools capable of storing spent fuel. The two pools at the south end of the FHB, now known as Spent Fuel Pools (SFPs) 'A' and 'B', were to support Harris Units 1 and 4. The two pools at the north end of the FHB, now known as Spent Fuel Pools 'C' and 'D', were to support Harris Units 2 and 3. The multi-unit design included a spent fuel pool cooling and cleanup system to service SFPs 'A' and 'B' and a separate cooling and cleanup system to support SFPs 'C' and 'D'.

Harris Units 3 and 4 were canceled in late 1981. Harris Unit 2 was canceled in late 1983. The FHB, all four pools (including liners), and the cooling and cleanup system to support SFPs 'A' and 'B' were completed and turned over. However, construction on the spent fuel pool cooling and cleanup system for SFPs 'C' and 'D' was discontinued after Unit 2 was canceled and the system was not completed. Harris Unit 1 began operation in 1987 with SFPs 'A' and 'B' in service. The need to eventually activate SFPs 'C' and 'D' (depending on the availability of a permanent DOE spent fuel storage facility) was anticipated at the time the operating license for Harris Unit 1 was issued. The spent fuel storage capacity currently identified in Section 5.6.3 of the Harris Plant Technical Specifications (1832 PWR assemblies and 48 interchangeable (7 x 7 cell) PWR or (11 x 11 cell) BWR racks) assumes installation of racks in all four of the spent fuel pools.

Since the time that construction of the spent fuel pool cooling and cleanup system for SFPs 'C' and 'D' was halted, CP&L has implemented a spent fuel shipping program because DOE spent fuel storage facilities are not available and are not expected to be available for the foreseeable future. Spent fuel from Brunswick (2 BWR units) and Robinson (1 PWR unit) is shipped to Harris for storage in the Harris SFPs. Shipment of spent fuel to Harris is necessary in order to maintain full core offload capability at Brunswick and Robinson. As a result of the operation of the Harris Plant, shipping program requirements, and the unavailability of DOE storage, it will be necessary to activate SFPs 'C' and 'D' and the associated cooling and cleanup system by early in the year 2000. Activation of these two pools will provide storage capacity for all four CP&L nuclear units (Harris, Brunswick 1 and 2, and Robinson) through the end of their current licenses.

SFP 'A' now contains six Region 1 flux trap style (6 x 10 cell) PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. SFP 'A' has been, and will continue to be, used to store fresh (unburned) and recently discharged Harris fuel.

SFP 'B' now contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) PWR Region 1 style racks. SFP 'B' also currently contains seventeen (11 x 11 cell) BWR racks. SFP 'B' is licensed to store one more (11 x 11 cell) BWR rack, which would increase the total pool storage capacity to 2946 assemblies. Harris is postponing installation of the last BWR rack and prefers to reserve the pool open area for fuel examination and repair. Therefore, the total installed capacity in SFP 'B' will temporarily remain as 768 PWR cells and 2,057 BWR cells for a total of 2,825 storage cell locations.

Proposed Changes:

The proposed changes will allow CP&L to increase the spent fuel storage capacity at the Harris plant by placing SFPs 'C' and 'D' in service. In order to activate the pools, CP&L requests that the NRC review and approve the following changes:

1. Revised Technical Specification 5.6 to identify PWR burnup restrictions, BWR enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D'.

   The use of the high density region 2 racks has been shown to be acceptable based on the analysis performed by Holtec International.

2. 10CFR50.55a Alternative Plan to demonstrate acceptable level of quality and safety in the completion of the component cooling water (CCW) and SFP 'C' and 'D' cooling and cleanup system piping.

   The cooling system for SFPs 'C' and 'D' cannot be N stamped in accordance with ASME Section III since some installation records are not available, a partial turnover was not performed when construction was halted following the cancellation of Unit 2 and CP&L's N certificate program was discontinued following completion of Unit 1. The Alternative Plan demonstrates that the originally installed equipment is acceptable for use and that the design and construction on the remaining portion of the cooling system piping (estimated at about 20%) maintains the same level of quality and safety through the use of the CP&L Appendix B QA program supplemented by additional QA requirements integrated into the plant modification package which completes the system

3. Unreviewed safety question for additional heat load on the component cooling water (CCW) system.

   The acceptability of the 1.0 MBtu/hr heat load from SFPs 'C' and 'D' was demonstrated by the use of thermal-hydraulic analyses of the CCW system under various operating scenarios. The dynamic modeling used in the thermal-hydraulic analyses identified a decrease in the minimum required CCW system flow rate to the RHR heat exchangers. This change has not been previously reviewed by the NRC and is deemed to constitute an unreviewed safety question.

Basis for Change

Installation of spent fuel storage racks in SFPs 'C' and 'D':

The FHB and SFPs 'C' and 'D' (including pool liners) were fully constructed and turned over as part of the construction and licensing of Harris Unit 1. However, the decision was made to not place SFPs 'C' and 'D' in service until needed (depending on the availability of DOE spent fuel storage). SFPs 'C' and 'D' are flooded but have not been previously used for spent fuel storage. CP&L proposes to expand the storage capacity at Harris by installing Region 2 (non-flux trap style) rack modules in Pools 'C' and 'D' in incremental phases (campaigns), on an as needed basis. SFP 'C' will provide the initial storage expansion for both PWR and BWR fuel. In its fully implemented storage configuration, SFP 'C' can accommodate 927 PWR and 2763 BWR assemblies. Expansion of storage capacity by installing racks in SFP 'D' will occur once SFP 'C' is substantially filled. SFP 'D' will contain only PWR fuel and can accommodate 1025 maximum density storage cells.

Following this proposed change, Spent Fuel Pool capacities will be as follows:

Pool	PWR spaces	BWR spaces	Total
'A'	360	363	723
'B'	768	2178	2946
'C'	927	2763	3690
'D'	1025	0	1025
Total	3080	5304	8384

Racks in SFP 'C' and 'D' will be installed in the following phases:

SFP 'C' - 1st Campaign - install by early 2000
4 PWR racks → 360 PWR spaces
10 BWR racks → 1320 BWR spaces

SFP 'C' - 2nd Campaign - install approximately 2005
4 PWR racks → 324 PWR spaces
6 BWR racks → 936 BWR spaces

SFP 'C' - 3rd Campaign - install approximately 2014
3 PWR racks → 243 PWR spaces
3 BWR racks → 507 BWR spaces

SFP 'D' - 1st Campaign - install approximately 2016
6 PWR racks → 500 PWR spaces

SFP 'D' - 2nd Campaign - installation date to be determined
6 PWR racks → 525 PWR spaces

(Note: The projected rack installation dates listed above are based on the current spent fuel shipping schedule. These dates may change as the shipping schedule is revised).

This configuration represents the mixture of PWR and BWR storage which will accommodate future storage requirements based on currently identified needs. Within SFP 'C', eighteen (18) of the racks are sized to allow interchangeability between BWR and PWR storage if required in the future. The dimensions of the (9 x 9 cell) PWR rack and the (13 x 13 cell) BWR rack are virtually identical. Therefore, rack configurations other than those identified above are possible.

Enclosure 6 of this license amendment request provides a report developed in conjunction with Holtec International which describes the evaluations performed to show the acceptability of the proposed change to install the racks in pools 'C' and 'D'. (Enclosure 7 is a non-proprietary version of enclosure 6). The report includes listings of the applicable regulations, codes and standards, descriptions of the evaluation methodology, acceptance criteria, and evaluation results. The licensing report also includes discussions on the need for the proposed change and considerations of other alternatives. Technical Specification Section 5.6, Fuel Storage, will be revised to identify PWR burnup restrictions, BWR enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D' (See Enclosure 5).

Completion of Cooling and Cleanup System for SFPs 'C' and 'D':

In order to activate Spent Fuel Pools 'C' and 'D', it is necessary to complete construction of the cooling and cleanup system for these pools and to install tie-ins to the existing Harris Unit 1 component cooling water system to provide heat removal capabilities. Approximately 80% of the SFP cooling and cleanup system piping and the majority of the CCW piping was installed during the original plant construction. In addition, other major system components such as the SFP cooling heat exchangers and pumps were also installed before original construction was discontinued. The cooling and cleanup system for pools 'C' and 'D' will be completed such that system design and operation is consistent with the design and operation of the cooling and cleanup system for pools 'A' and 'B'. The spent fuel pool cooling system for pools 'C' and 'D' is nuclear safety related with two fully redundant 100% capacity trains.

At the time that construction on the SFP cooling system was discontinued following cancellation of Harris Unit 2, a formal turnover of the partial system was not performed and CP&L has since discontinued its N certificate program. Also, some of the field installation records for the completed piping are no longer available. As a result, the system when completed will not satisfy ASME Section III code requirements (i.e. will not be N stamped). Therefore, an Alternative Plan in accordance with 10CFR50.55a(a)(3) is provided as Enclosure 8 to demonstrate that the completed system will provide an acceptable level of quality and safety. The majority of the ASME Section III piping was already installed when original construction was discontinued. As identified in the Alternative Plan, that piping to the extent that it was completed, was designed, constructed and inspected to Section III requirements. The remainder of the system will also be designed, constructed, inspected and tested to Section III requirements to the extent practical considering CP&L no longer has an N certificate program. Work will be performed in accordance with CP&L's 10CFR50 Appendix B QA program with any differences between Section III requirements and Appendix B requirements conservatively dispositioned. Supplemental QA requirements will be integrated into the modification package(s) as appropriate.

Calculations have been performed to verify that the existing CCW system is adequate to provide heat removal for near-term pool operation. The Spent Fuel Pool 'C' and 'D' heat loads will be limited to 1.0 MBtu/hr for near-term operation. Technical Specification section 5.6.3 will be revised to identify this heat load limit (Enclosure 5). This heat load limit is being established since additional CCW heat loads resulting from the power uprate project (potential to increase post-accident containment temperature resulting in an increased containment sump temperatures and increased load on RHR during long term recirculation phase) are not quantified at this time. Therefore, it has been determined that the most prudent action is to establish limiting heat loads based on current system loads. Additional heat load analysis will be performed concurrent with the power uprate project to establish the maximum heat loads on the CCW system that will exist at the end of plant licensed life when all spent fuel pools are expected to be full. Any CCW modifications necessary to increase system heat removal capability will be identified and implemented at that time. As part of the licensing required to support the power uprate project (currently planned for implementation concurrent with the steam generator replacement in late 2001), the technical specification heat load limit will either be revised or removed completely.

The plant design change package and supporting analyses for the CCW tie-in demonstrated that adequate capacity exists on the CCW system to add the 1.0 MBtu/hr for the near-term operation of SFPs 'C' and 'D'. The thermal-hydraulic analysis performed in support of this plant design change package modeled the dynamic RHR heat exchanger performance based on fluid property changes. Previous analyses evaluated RHR heat exchanger performance at a fixed data sheet value. This results in a reduction in the required CCW flow to the RHR heat exchanger. While technically valid, the lower required flow rate has not been previously reviewed by the NRC and, therefore, is deemed to constitute an unreviewed safety question. Included in Enclosure 9 are the results of the 10CFR50.59 evaluation for the unreviewed safety question identified by the tie-in to Unit 1 CCW.

Conclusion:

CP&L has concluded that placing SFPs 'C' and 'D' in service at this time to provide spent fuel storage is the safe and prudent alternative for increasing spent fuel storage capacity in the nuclear generating system. This option has been shown to be safe and in conformance with the appropriate regulations, codes and standards. Expansion of storage capacity by using Pools 'C' and 'D' will support continued operation of the Harris, Brunswick and Robinson facilities until the end of their current operating licenses.

Enclosure 2 to Serial: HNP-98-188

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

10CFR50.92 EVALUATION
SIGNIFICANT HAZARDS CONSIDERATION

10CFR50.92 EVALUATION

The commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists with regard to a proposed license amendment. A change involves no significant hazards consideration if it would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light has reviewed the proposed change and determined that it does not involve a significant hazards consideration. The following safety assessment summarizes the results of this review. Responses to the three significant hazard consideration questions follow at the end of this evaluation.

Safety Assessment

The planned expansion of storage capacity involves installing up to 30 storage rack modules in Pool 'C' and up to 12 storage rack modules in Pool 'D'. The implementation of the storage capacity increase in pools 'C' and 'D' will be performed on an as needed basis through incremental phases (campaigns), as follows:

Pool	Campaign	Number of Racks	Total Storage Locations
'C'	I	14	1680
	II	10	1260
	III	6	750
'D'	I	6	500
	II	6	525

The cells of the new racks will contain a fixed neutron absorber for primary reactivity control. To maximize storage capacity, the new racks will be "Region 2" style racks, which are designed without the usual flux trap design associated with "Region 1" style racks. The effective enrichment of the stored fuel will be controlled administratively to maintain reactivity within acceptable limitations. Acceptable effective enrichment will be ensured prior to placement of spent fuel into the pools.

Rack modules in both pools will be freestanding and self-supporting. The new modules will be separated by a gap of approximately 0.625 inch from one another. Along the pool walls, a nominal gap will also be provided which will vary from approximately 2.5 inches to 6.1 inches.

The proposed cooling system modifications for Pools 'C' and 'D' have been designed to ensure that sufficient heat removal capability exists to maintain the temperature in the pools below the design limit. For the initial installation of racks into Pool 'C', the maximum heat load will be limited to 1.0 MBtu/hr consistent with revised Technical Specification 5.6. In conjunction with the planned implementation of power uprate, additional analyses will be performed and any required system upgrades will be made to ensure the adequacy of the cooling system to dissipate the heat loads associated with the end of plant life. A comprehensive multi-system thermal-hydraulic analysis was performed in support of the plant design change package for the initial rack installation campaign. This analysis facilitates a reduction in this CCW flow requirement currently stated in the FSAR as being a basis for acceptance of postulated post-LOCA consequences. While the analysis methods are technically valid, this lower flowrate has been deemed to constitute an unreviewed safety question and requires NRC review and approval.

The predominant pool heat load typically develops from the residual heat associated with the most recent reactor core offload. Transient heat loads are not a significant concern for Pools 'C' and 'D' due to the spent fuel cooling time required prior to placement within these two pools. Satisfactory spent fuel cooling time will be ensured through administrative controls of fuel decay time subsequent to reactor discharge.

In order to activate Spent Fuel Pools 'C' and 'D', it is necessary to complete construction of the cooling and cleanup system for these pools and to install tie-ins to the existing Harris Unit 1 component cooling water system (CCW) to provide heat removal capabilities. The majority of the ASME Section III piping was already installed when original construction was discontinued. An alternative plan in accordance with 10CFR50.55a(a)(3)(i) is provided to demonstrate that the completed system will provide an acceptable level of quality and safety.

The Spent Fuel Pool thermal performance, completion of construction, criticality, and seismic response have been analyzed considering the increased storage capacity and fuel enrichment. The results of these analyses have shown that the pool structure and proposed cooling systems (within the limitation of the new technical specifications) are adequate to support storage of spent fuel within Pools 'C' and 'D'.

Significant Hazards Consideration Determination

In accordance with 10CFR50.92, Carolina Power & Light has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the threshold for the three criteria of 10CFR50.92(c) are not reached. The proposed activity does not involve a SHC because it would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   In the analysis of the safety issues concerning the expanded pool storage capacity within Harris' Fuel Handling Building, the following previously postulated accident scenarios have been considered:

   a. A spent fuel assembly drop in a Spent Fuel Pool
   b. Loss of Spent Fuel Pool cooling flow
   c. A seismic event
   d. Misloaded fuel assembly

   The probability that any of the accidents in the above list can occur is not significantly increased by the activity itself. The probabilities of a seismic event or loss of Spent Fuel Pool cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misloadings are primarily influenced by the methods used to lift and move these loads. The method of handling loads during normal plant operations is not significantly changed, since the same equipment (i.e., Spent Fuel Handling Machine and tools) and procedures as those in current use in pools 'A' and 'B' will be used in pools 'C' and 'D'. Since the methods used to move loads during normal operations remain nearly the same as those used previously, there is no significant increase in the probability of an accident. Current shipping activities at the Harris Nuclear Plant will continue as previously licensed. The consequences of an accident involving shipping activities is not changed and there is no significant increase in the probability of an accident.

   During rack installation, all work in the pool area will be controlled and performed in strict accordance with specific written procedures. Any movement of fuel assemblies which is required to be performed to support this activity (e.g., installation of racks) will be performed in the same manner as during normal refueling operations.

   Accordingly, the proposed activity does not involve a significant increase in the probability of an accident previously evaluated.

   The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Spent Fuel Pool have been re-evaluated for the proposed change. The results show that such the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, K_{eff} less than or equal to 0.95, will be maintained. The structural damage to the Fuel Handling Building, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed activity from those considered previously, the structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will not be increased from those previously considered, since the pertinent fuel parameters remain unchanged. These dose levels remain "well within" the levels required by 10CFR100, paragraph 11, as defined in Section 15.7.4.II.1 of the Standard Review Plan. Thus, the results of the postulated fuel drop accidents remain acceptable and do not represent a significant increase in consequences from any of the same previously evaluated accidents that have been reviewed and found acceptable by the NRC.

The consequences of a loss of Spent Fuel Pool cooling have been evaluated and found to have no increase. The concern with this accident is a reduction of Spent Fuel Pool water inventory from bulk pool boiling resulting in uncovering fuel assemblies. This situation would lead to fuel failure and subsequent significant increase in offsite dose. Loss of spent fuel pool cooling at Harris is mitigated in the usual manner by ensuring that a sufficient time lapse exists between the loss of forced cooling and uncovering fuel. This period of time is compared against a reasonable period to re-establish cooling or supply an alternative water source. Evaluation of this accident usually includes determination of a time to boil, which in the case of pools 'C' and 'D' is in excess of 13 hours based on a consideration of end of plant life heat loads. This evaluation neglects any possible cooling from the connection to pools 'A' and 'B' through the transfer canal. The 13 hour period is much shorter than the onset of any significant increase in offsite dose, since once boiling begins it would have to continue unchecked until the pool surface was lowered to the point of exposing active fuel. The time to boil represents the onset of loss of pool water inventory and is commonly used as a gauge for establishing the comparison of consequences before and after a refueling project. The heatup rate in the Spent Fuel Pool is a nearly linear function of the fuel decay heat load. Subsequent to the proposed changes, the fuel decay heat load will increase because of the increase in the number assemblies from those considered from Pools 'A' and 'B' alone. The methodology used in the thermal-hydraulic analysis determined the maximum fuel decay heat loads. In the unlikely event that pool cooling is lost to pools 'C' and 'D', sufficient time will still be available for the operators to provide alternate means of cooling before the onset of pool boiling. Therefore, the proposed change represents no increase in the consequences of loss of pool cooling.

The consequences of a design basis seismic event are not increased. The consequences of this accident are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The new racks have been analyzed in their new configuration and found safe during seismic motion. The fuel stored in these racks has been determined to remain intact and the racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the appropriate combinations of dead weight, thermal, and seismic loads. The Fuel Handling Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, storage array, and pool moderator/coolant. Thus, the consequences of a seismic event are not increased.

Fuel misloading and mislocation accidents were previously credible occurrences, since fuel could be placed at an unintended storage location or could have been lowered outside and adjacent to a storage rack in Pools 'A' or 'B'. However, neither of these two scenarios previously represented any concern because of the flux trap style of the rack designs in these two pools. Similar procedures, equipment and methods of fuel movement will be used for Pools 'C' and 'D' as those used previously for Pools 'A' and 'B'. Therefore, the proposed activity does not represent any increase in the probability of occurrence. The proposed non-flux trap design racks for Pools 'C' and 'D' require administrative controls to ensure that fuel assemblies meet effective enrichment criteria prior to storage. Under these conditions, misloading of a fuel assembly by placement in an unintended storage cell has no significant consequences. Therefore, the only remaining potential mislocation of a fuel assembly is for an assembly to be lowered outside of and directly adjacent to a storage rack. This accident occurring in Pools 'C' or 'D' has been analyzed for the worst possible storage configuration subsequent to the proposed activity and it has been shown that the consequences remain acceptable with respect to the same criteria used previously. Thus, there is no increase in consequences for fuel mislocation or misloading.

Therefore it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

To assess the possibility of new or different kind of accidents, a list of the important parameters required to ensure safe fuel storage was established. Safe fuel storage is defined here as providing an environment, which would not present any significant threats to workers or the general public (i.e., meeting the requirements of 10CFR100 and 10CFR20). Any new events, which would modify these parameters sufficiently to place them outside of the boundaries analyzed for normal conditions and/or outside of the boundaries previously considered for accidents would be considered to create the possibility of a new or different accident. The criticality and radiological safety evaluations were reviewed to establish the list of important parameters. The fuel configuration and the existence of the moderator/coolant were identified as the only two parameters, which were important to safe fuel storage. Significant modification of these two parameters represents the only possibility of an unsafe storage condition. Once the two important parameters were established, an additional step was taken to determine what events (which were not previously considered) could result in changes to the storage configuration or moderator/coolant presence during or subsequent to the proposed changes. This process was adopted to ensure that the possibility of any new or different accident scenario or event would be identified. Due to the proposed activity, an accidental drop of a rack module during construction activity in the pool was considered as the only event which might represent a new or different kind of accident.

A construction accident resulting in a rack drop is an unlikely event. The proposed activity will utilize the defense-in-depth approach for these heavy loads. The defense-in-depth approach is intended to meet the requirements of NUREG-0612 and preclude the possibility of a rack drop. All movements of heavy loads over the pool will comply with the applicable administrative controls and guidelines (i.e. plant procedures, NUREG-0612, etc.). A temporary hoist and rack lifting rig will be introduced to lift and suspend the racks from the bridge of the Auxiliary Crane. These items have been designed in accordance with the requirements of NUREG-0612 and ANSI N14.6 and will be similar to those used recently to install storage rack modules in Pool 'B'.

The postulated rack drop event is commonly referred to as a "heavy load drop" over the pools. Heavy loads will not be allowed to travel over any racks containing fuel assemblies. The danger represented by this event is that the racks will drop to the pool floor and the pool structure will be compromised leading to loss of moderator/coolant, which is one of the two important parameters identified above. Although the analysis of this event has been performed and shown to be acceptable, the question of a new or different type of event is answered by determining whether heavy load drops over the pool have been considered previously. As stated above, heavy loads (storage rack modules) were recently installed in Pool 'B' using similar methods. Therefore, the rack drop does not represent a new or different kind of accident.

The proposed change does not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents. The proposed change does not affect any of the important parameters required to ensure safe fuel storage. Therefore, the potential for a new or previously unanalyzed accident is not created.

3. Involve a significant reduction in the margin of safety.

The function of the Spent Fuel Pool is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with Pools 'C' and 'D'.

CP&L has addressed the safety issues related to the expanded pool storage capacity in the following areas:

1. Material, mechanical and structural considerations

   The mechanical, material, and structural designs of the new racks have been reviewed in accordance with the applicable provisions of the NRC Guidance entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications". The rack materials used are compatible with the spent fuel assemblies and the Spent Fuel Pool environment. The design of the new racks preserves the proper margin of safety during normal and abnormal loads. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

2. Nuclear criticality

   The methodology used in the criticality analysis of the expanded Spent Fuel Pool meets the appropriate NRC guidelines and the ANSI standards (GDC 62, NUREG 0800, Section 9.1.2, the OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Reg. Guide 1.13, and ANSI/ANS 8.17). The margin of safety for subcriticality is maintained by having the neutron multiplication factor equal to, or less than, 0.95 under all accident conditions, including uncertainties. This criterion is the same as that used previously to establish criticality safety evaluation acceptance and remains satisfied for all analyzed accidents.

3. Thermal-hydraulic and pool cooling

   The thermal-hydraulic and cooling evaluation of the pools demonstrated that the pools can be maintained below the specified thermal limits under the conditions of the maximum heat load and during all credible accident sequences and seismic events. The pool temperature will not exceed 137°F during the highest heat load conditions. The maximum local water temperature in the hot channel will remain below the boiling point. The fuel will not undergo any significant heat up after an accidental drop of a fuel assembly on top of the rack blocking the flow path. A loss of cooling to the pool will allow sufficient time (>13 hours) for the operators to intervene and line up alternate cooling paths and the means of inventory make-up before the onset of pool boiling. The thermal limits specified for the evaluations performed to support the proposed activity are the same as those that were used in the previous evaluations. It has also been demonstrated that adequate margin exists in the Unit 1 CCW system to support near term operation of the pools subject to the requirements of the proposed changes to the Technical Specifications.

Based on the preceding discussion it is concluded that this activity does not involve a significant reduction in the margin of safety.

The NRC has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (51FR7751, March 6, 1986) of amendments that are considered not likely to involve a SHC. The proposed changes for Harris are similar to Example (x): an expansion of the storage capacity of Spent Fuel Pool when all of the following are satisfied:

(1) The storage expansion method consists of either replacing existing racks with a design that allows closer spacing between stored spent fuel assemblies or placing additional racks of the original design on the pool floor if space permits.

The Harris storage expansion involves installation of storage racks for PWR and BWR fuel assemblies with a design that allows closer spacing of stored PWR spent fuel assemblies.

(2) The storage expansion method does not involve rod consolidation or double tiers.

The Harris rack installation does not involve fuel consolidation. The racks will not be double tiered; no fuel assemblies will be stored above other assemblies.

(3) The K_{eff} of the pool is maintained less than, or equal to, 0.95.

The design of the new racks integrates Boral as a neutron absorber within each rack cell to allow close storage of spent fuel assemblies while ensuring that K_{eff} remains less than 0.95 under all conditions. Additionally, the water in the Spent Fuel Pool does contain boron as further assurance that K_{eff} remains less than 0.95. The boron that is contained in the pool is not credited under normal or accident conditions.

(4) No new technology or unproven technology is utilized in either the construction process or the analytical techniques necessary to justify the expansion.

The rack vendor has successfully participated in the licensing of numerous other racks of a similar design. The construction process and the analytical techniques of the Harris pool expansion are substantially the same as in the other completed rerack projects. Thus, no new or unproven technology is used in the Harris rack installation.

The similarities of the proposed activity to the above example and the previously discussed satisfaction of the three criteria from 10 CFR 50.92(c) confirm the conclusion stated above that the modification does not represent a Significant Hazards Consideration (SHC).

Enclosure 3 to Serial: HNP-98-188

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

ENVIRONMENTAL EVALUATION

ENVIRONMENTAL EVALUATION

CP&L has reviewed activities described in the proposed license amendment for environmental considerations. Each of the proposed changes was evaluated against the criteria of 10CFR51.22 to ascertain whether the criteria for categorical exclusion were satisfied or if formal environmental impact statement would be required.

Significantly, this review identified that the newly activated spent fuel pools will be similar in design with that originally conceived and approved for construction for this portion of the Harris Plant. All four pools were included in the original four unit design of the Harris Nuclear Plant, and the completion and operation of these pools continued to be reflected in plant licensing documentation up to and including the issuance of the operating license for Units 1 & 2 (ref. NUREG-1038, dated Nov. 1983) and the associated environmental report (ref. Shearon Harris Nuclear Power Plant Environmental Report, Amendment 5, dated Dec. 1982). The most notable difference between the previously licensed and currently proposed designs is that, rather than having a separate operating unit to provide auxiliaries such as CCW for cooling and RWST for makeup, the current design will utilize Unit 1 facilities for those functions. Nonetheless, the design of the fuel pools themselves, including cooling and cleanup systems, will be essentially the same as that previously reviewed, and the differences which do exist between the current design and that originally licensed are not of a scope or nature as to have a significant bearing on environmental impact.

Since the design and operation of the 'C' & 'D' Spent Fuel Pools and supporting systems is essentially identical to that originally licensed in NUREG-1038 and the associated environmental report, no increase in occupational exposure is anticipated with regard to new equipment design or operating constraints. On the contrary, the operating experience of the 'A' and 'B' spent fuel pools is being utilized to ensure that the new design is as ALARA friendly as possible. For instance, local flow indicators for the new systems are being located in areas know to have lower dose rates than their counterparts already in operation. In addition to an ALARA friendly design, existing fuel handling and ALARA and procedures will continue to be utilized, and fuel handling equipment reliability is not diminished. Spent fuel pool shielding levels are not decreased, and no appreciable increase in area dose rates is expected. Based on these considerations, it can be concluded that this activity will not result in a significant increase in individual or cumulative occupational exposures.

The issues which were evaluated to reach this determination also include an evaluation of the thermal impact on the plant environs resulting from the additional spent fuel heat load. Calculations assessing the impact of spent fuel pool activation predict that an increase in UHS temperature of less than 0.01 °F would result from an additional 1.0 MBtu/hr heat input This increase is insignificant relative the available margin in the UHS to its design temperature and considering the uncertainties existing in the analyses.

Finally, it is easily seen that the thermal impact on the environment of a single operating unit with four spent fuel pools is bounded by that of the two unit - four spent fuel pool configuration which was previously evaluated and licensed by NUREG-1038 and the associated environmental report. It is concluded that no additional assessment is required regarding to thermal impacts on the UHS.

In summary, the licensing activities associated with the activation of the 'C' & 'D' spent fuel pools as described herein do not significantly increase the types and amounts of effluents that may be released offsite, nor significantly increases individual or cumulative occupational exposures nor constitutes any other type of new and appreciable environmental impact. It is concluded that these activities are essentially environmentally benign and that no additional impact studies are necessary in support of this submittal.

Enclosure 4 to Serial: HNP-98-188

SHEARON HARRIS NUCLEAR POWER PLANT
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PAGE CHANGE INSTRUCTIONS

Removed Page	Inserted Page
5-7	5-7
	5-7a
	5-7b

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TECHNICAL SPECIFICATION PAGES

BEFORE

5.6 FUEL STORAGE

CRITICALITY

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

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

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

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

DRAINAGE

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

CAPACITY

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

5.6 FUEL STORAGE

CRITICALITY

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

1. The reactivity margin is assured for pools 'A' and 'B' by maintaining a nominal 10.5 inch center-to-center distance between fuel assemblies placed in the flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks.

2. The reactivity margin is assured for pools 'C' and 'D' by maintaining a nominal 9.017 inch center-to-center distance between fuel assemblies placed in the non-flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks. The following restrictions are also imposed through administrative controls:

   a. PWR assemblies must be within the "acceptable range" of the burnup restrictions shown in Figure 5.6.1 prior to storage in Pools 'C' or 'D'

   b. BWR assemblies are acceptable for storage in Pool 'C' provided that the maximum planar average enrichments is less than 4.6 wt% U235 and K_{inf} is less than or equal to 1.32 for the standard cold core geometry (SCCG).

DRAINAGE

5.6.2 The pools 'A', 'B', 'C' and 'D' are designed and shall be maintained to prevent inadvertent draining of the pools below elevation 277.

CAPACITY

5.6.3.a Pool 'A' contains six (6 x 10 cell) flux trap type PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. Pool 'B' contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) flux trap style PWR racks and seventeen (11 x 11 cell) BWR racks and is licensed for one additional (11 x 11 cell) BWR rack that will be installed as needed. The combined pool 'A' and 'B' licensed storage capacity is 3669 assemblies.

5.6.3.b Pool 'C' is designed to contain a combination of PWR and BWR assemblies. Pool 'C' can contain two (11 x 9 cell) and nine (9 x 9 cell) PWR racks for storage of 927 PWR assemblies. Pool 'C' can contain two (8 x 13 cell), two (8 x 11cell), six (13 x 11 cell), and nine (13 x 13 cell) BWR racks for storage of 2763 BWR assemblies. The (9 x 9 cell) PWR racks and the (13 x 13 cell) BWR racks are dimensioned to allow interchangeability between PWR or BWR storage rack styles as required. The racks in pool 'C' will be installed as needed.

5.6.3.c Pool 'D' contains a variable number of PWR storage spaces. These racks will be installed as needed. Pool 'D' is designed for a maximum storage capacity of 1025 PWR assemblies.

5.6.3.d The heat load from fuel stored in Pools 'C' and 'D' shall not exceed 1.0 MBtu/hr.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

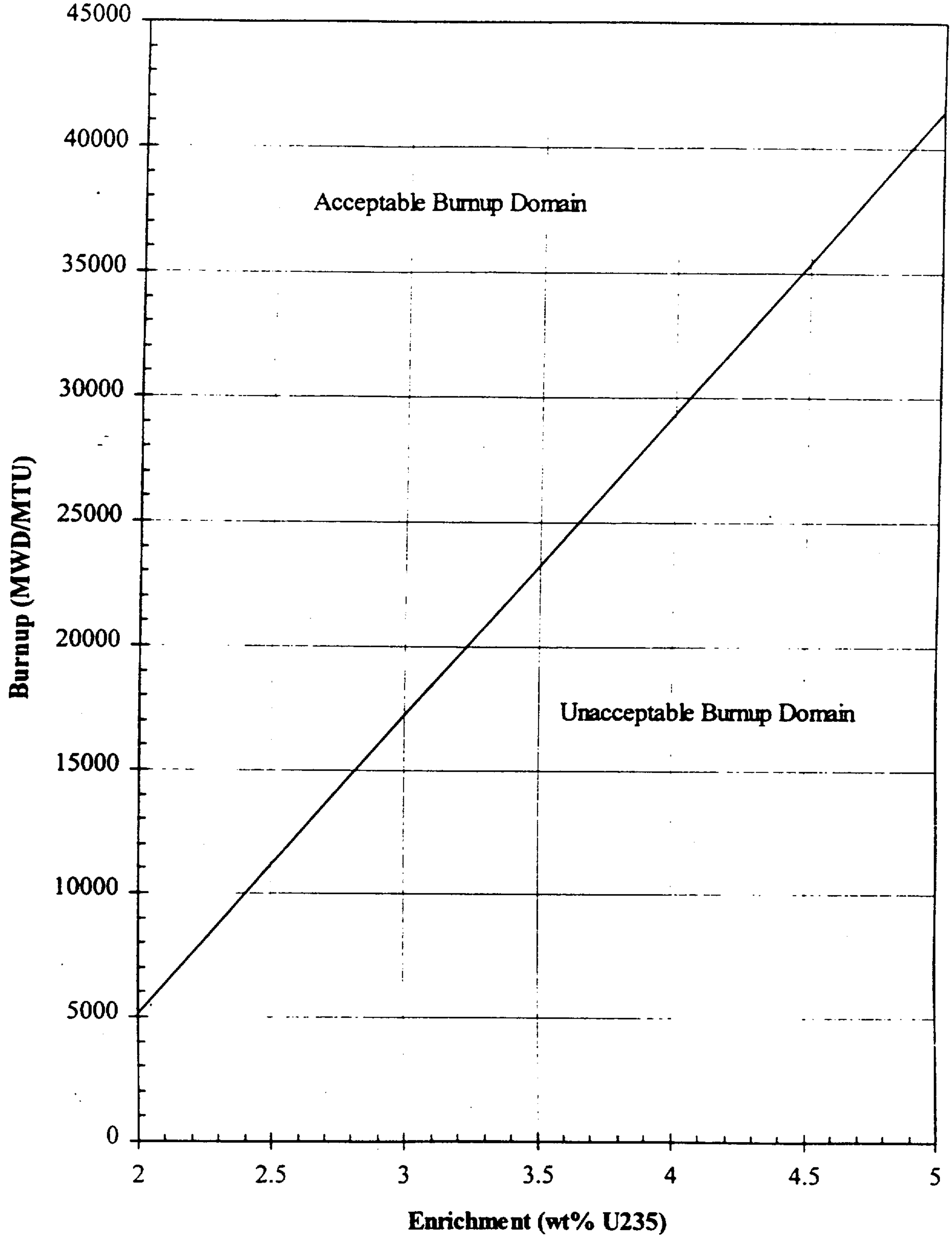

Figure 5.6.1: Burnup Versus Enrichment for PWR Fuel

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SPENT FUEL STORAGE

10CFR50.55a ALTERNATIVE PLAN

10CFR50.55a ALTERNATIVE PLAN

I. Introduction

Regulatory Background
10CFR50.55a (Codes and Standards) requires that nuclear power facilities be subject to the licensing condition that (1) structures, systems and components are designed, fabricated, erected, constructed and inspected to quality standards commensurate with the importance of the safety function to be performed, and (2) that certain systems and components of nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code. 10CFR50.55a(a)(3) allows alternatives to these requirements with the permission of the Office of Nuclear Reactor Regulation if it can be demonstrated that the proposed alternative would provide an acceptable level of quality and safety, or if compliance with the requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The following is an outline of a "10CFR50.55a Alternative Plan" for licensing plant systems originally intended for use in cooling and storage of Harris Units 2 and 3 spent fuel. This portion of the plant was only partially completed under the Harris Plant construction program at the time that Unit 1 was completed and was never turned over as a part of the licensed and operating facility. The completion of this spent fuel storage capacity is now needed for long term storage of spent fuel from the Harris, Brunswick and Robinson Nuclear Plants in support of continued operation of these CP&L facilities. However, continuing its construction on the basis of the original site construction program is not viable since (1) CP&L has discontinued its N certificate holder program, and (2) certain code required construction records associated with the field installation of this piping are no longer available. This 10CFR50.55a Alternative Plan is intended to provide the basis for construction requirements for the completion of this portion of the Harris Plant and to justify the acceptability of previously constructed equipment in light of missing documentation.

Construction History / Chronology
Carolina Power & Light filed an application with the Atomic Energy Commission in 1971 for licenses to construct and operate its proposed Shearon Harris Nuclear Power Plant Units 1, 2, 3 and 4, in Wake County, NC. After completion of preconstruction reviews and hearings, the AEC issued Construction Permit Nos. CPPR-158, CPPR-159, CPPR-160 and CPPR-161 on January, 1978. Construction proceeded on the four unit site until December 1981, when CP&L informed the NRC that Units 3 and 4 had been canceled, and requested that Units 1 and 2 be considered concurrently for operating licenses. NUREG-1038 was issued in November 1983 for Unit 1, and reflected ongoing construction and eventual completion of Unit 2. However, Unit 2 was canceled soon

afterward in December 1983, leaving Unit 1 as the only Unit to be completed and licensed. The Unit 1 Full Power Operating License was issued in January 1987, with commercial operation beginning in May 1987.

The original design of the four unit Harris Nuclear Plant located Units 1 and 4 at the south end of the plant, and Units 2 and 3 on the north end. These four units were to share a common fuel handling building to serve the purposes of loading and offloading fuel, as well as storage of spent fuel. Two sets of fuel storage pools were located in the fuel handling building, each set containing a spent fuel pool and a new fuel pool. The spent fuel pools were intended to function primarily as spent fuel storage capacity, while the new fuel pools were provided for staging new fuel and offloading spent fuel from the reactor. In the initial design, Units 1 and 4 shared the south ('A' and 'B') fuel pools, while the north ('C' and 'D') fuel pools were intended to service Unit 2 and 3.

The Fuel Handling Building was a common feature to all units, and completion of the building itself was requisite for operation of the first unit placed into service. Logical progression of the Fuel Handling Building construction dictated that major pieces of equipment be installed early in the schedule. As a result, the full complement of Spent Fuel Pool Cooling pools, heat exchangers and pumps initially associated with four unit construction was installed. Many of the smaller pumps, filters, strainers and lesser pieces of equipment were installed as well. Fuel Handling Building construction also dictated that all of the piping to be embedded in concrete be installed at the logical interval as the building was erected. Since the pools were encased in concrete, the adjoining portions of piping providing cooling connections and auxiliaries were necessarily constructed, inspected and tested prior to the encasement concrete being poured.

Subsequent to the cancellation of Units 3 and 4, work on the 'C' and 'D' Spent Fuel Pools continued in support of the planned completion of Unit 2. By the time that Unit 2 was canceled, the majority of the mechanical piping and equipment associated with operation of the 'C' and 'D' end pools was already installed, including all of the embedded and most of the exposed portions of ASME Section III piping associated with these fuel pools' cooling system. Work on the remaining equipment associated with the 'C' and 'D' pools in the Fuel Handling Building was suspended when Unit 2 was canceled. Plant documents from that time describe plans to eventually complete the 'C' and 'D' spent fuel pools and place them into service.

<u>Construction Records Issue</u>
The completed portion of the Unit 2 Fuel Pool Cooling and Cleanup System (FPCCS) and supporting facilities were constructed to the same codes and standards and using the same procedures and personnel as was Unit 1, which was fully completed and licensed. Appropriate records documenting field activities were generated at the time of construction as required by the construction codes and plant procedures, and maintained in storage under the control of the construction Quality Assurance (QA) program pending system completion and turnover. When construction on Unit 2 was halted, these records

were transferred to temporary storage facilities maintained by the Harris Nuclear Plant Document Control. They were not microfilmed since they were associated with systems which were not fully completed and accepted under the site's N Certificate Program, and later were inadvertently discarded during a document control records cleanup effort.

Notably, these discarded records include the piping isometric packages for field installation of the completed portion of Unit 2 Fuel Pool Cooling and Cleanup System and Component Cooling Water System (CCWS) piping within Code boundaries. As a result, Code required records are no longer available for approximately 40 of the nearly 200 large bore welds in the completed ASME Section III portions of the Unit 2 FPCCS and CCWS.

II. Alternative Plan for Missing Construction Records (Piping Pedigree Plan)

The plan for addressing the missing construction documentation associated with the portion of the piping initially installed during plant construction and intended for the 'C' and 'D' Spent Fuel Pools' cooling systems consists of four elements. These are: (1) scoping, (2) records retrieval and review, (3) examination and testing, and (4) reconciliation. The intent of this plan is to develop the body of evidence which supports the quality of the previously completed constructed piping. Consistent with 10CFR50.55a, any deficiencies identified will be evaluated to determine whether a acceptable level of quality and safety can be provided through alternate methods, or if not, whether attaining full compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

(1) The scoping portion of the Piping Pedigree Plan defines the boundaries of piping within the plan, and basically consists of a review of the extent of existing construction vs. that required for completion of the system. The extent of previously completed construction is determined by conducting and documenting detailed field walkdowns. Identification markings such as spoolpiece numbers, welder identification numbers, heat numbers, etc. are recorded at this time for use later in the records review and retrieval phase. Accessibility (both external and internal) are assessed for planning the examination / testing phase.

(2) The records review and retrieval phase of the project is an investigation of construction era documents to compile the archived body of evidence which substantiates the quality of the Unit 2 Spent Fuel Cooling piping. Specific sources of this information are discussed as follows:

A) Procurement documents for piping spool pieces. Requirements to which these spool pieces were fabricated were delineated on Purchase Order NY 435035, which invoked piping spec CAR-SH-M-30. Vendor Data Packages were supplied to the requirements of the pipe spool vendor's NPT program, and include records of material certification, welding activities and Nondestructive Examination (NDE) and hydrotesting. These records were retained by the Harris Nuclear Plant Document Control Program and are available on microfilm.

B) Construction era documents which defined requirements associated with the procurement, storage, handling and installation of the piping. Work procedures fall into this category, and include those for welding, weld material control, piping installation, concrete placement, hydrotesting, etc. Development of the sequence of installation through controlling procedures establishes the activities related to quality (tests, inspections, reviews, etc.) which by procedure would have to be satisfactorily completed in order to meet specific documented construction milestones, such as concrete placement and hydrotest.

C) Review of records which are available through the Harris Nuclear Plant Document Control System relating to construction of the Spent Fuel Pools and related equipment. Record types which fall into this category include, hydrotest records, concrete placement tickets, records relating to pipe spool modifications, etc. In many cases records may be found which do not directly establish quality, but rather serve to demonstrate that the construction of this piping was subject to the same level of scrutiny as was comparable Unit 1 piping, for which the appropriate quality records do exist.

D) Review of construction era records which are not quality assurance records, but which do serve to substantiate the quality of construction. This category would include documents such as engineering files, or quality control inspector log books which note specific inspections or records review.

(3) An examination and test phase will recreate, to the extent possible, any inspections or records which would have originally been required by plant procedures and the construction code and for which documentation is no longer available. The primary focus of this phase will consist of inspection and NDE of field welds for which weld data records are not available. Accessible ASME Section III welds will be subject to 100% surface examination, and ANSI B31.1 welds will receive a visual examination. Where feasible, internal weld inspections will be performed to verify fitup and adequacy of shielding gas purge. Notably, this will include an internal remote camera inspection of a substantial portion of the embedded FPCCS piping. Alternate methods of attaining comparable assurance will be developed whenever code required inspections cannot be performed, or deficiency in code required records cannot be otherwise addressed. For example, since filler material traceability cannot be established by weld data records, examination and testing of weld filler material will be performed to verify the composition of filler material is consistent with weld requirements. Finally, system hydrotesting will be performed upon completion of the piping systems using ASME Section III hydrotest criteria.

(4) The reconciliation phase of the Piping Pedigree Plan is a review of the data collected in previous phases and assessment of the level to which original construction documentation requirements were met. This is accomplished by compiling the body of records retrieved from document control and those generated by the examination / testing effort, then reviewing this record set against code documentation requirements to determine the extent to which code requirements are met. For instances wherein deficiencies are identified, the body of evidence (alternate tests or inspections, construction procedures, etc) which substantiates the quality of the component would be evaluated to determine if comparable assurance of quality and safety exists.

Piping Pedigree Plan - Implementation

ASME Section III Piping:

The elements of the Piping Pedigree Plan as described above are essentially complete for the ASME Section III piping associated with the 'C' and 'D' pools' FPCCS.
The following is a summary of the results of this effort to date:

Scope Definition - The ASME Section III piping associated with the 'C' and 'D' SPF Cooling System has been walked down by CP&L engineering and Harris Nuclear Plant Quality Control personnel to compare the plant configuration with construction isometric drawings and ensure that all welds, both vendor and field constructed, have been identified. Pipe spool identification numbers and welder symbols were inspected and recorded for review and comparison against vendor data packages. The scope of the ASME Section III piping within the plan has been defined based on field walkdowns, a review of modification design and results of the records retrieval effort. Basically, the plan will cover the large bore ASME Section III piping in the FPCCS and CCWS, leaving the small bore pipe welds (vents, drains, etc.) to be cut out and redone as part of the modification effort. A total of 40 large bore piping field welds and 12 pipe hanger attachment welds are being addressed within this portion of the Alternative Plan scope. Of this total, 37 are FPCCS piping welds (15 of which are embedded in concrete) and 3 are CCWS piping welds. All 12 hanger attachment welds are in the FPCCS piping.

Vendor Data Package review - All of the 44 vendor data packages associated with the ASME Section III portions of the 'C' and 'D' FPCCS have been retrieved and reviewed to ensure that the requisite paperwork is in hand. These packages account for approximately 80% of the large bore piping welds in the previously constructed portions of this system . Of the nearly 200 existing large bore (12" and 16") ASME Section III FPCCS piping welds, approximately 160 are vendor welds for which all required records exist. As noted above, these vendor data packages also account for all but 12 of the hanger attachments welds existing in the FPCCS piping. Only 2 vendor data packages are associated with the portion of the previously installed Unit 2

CCW System which will be used in the design to tie in Unit 1 CCW to the 'C' and 'D' Spent Fuel Pool Cooling Heat Exchangers. These packages account for all but 3 of the existing large bore piping welds in this piping.

Review of other documentation - A review of other Construction Quality Control (QC) documentation in the document control system has identified that some construction information does exist for the piping in question. Notably, hydrotest records were located which show that all of the embedded piping was in fact subject to hydrotest. Completion of weldments within the hydrotest boundary and review of Weld Data Reports (WDRs) was a procedural prerequisite for conducting these hydrotests. Of these 15 embedded field welds, hydrotest records contain specific signoffs attesting to satisfactory review of completed WDRs for 9. An additional 4 embedded welds are specifically identified as being within the hydrotest boundary with a general signoff attesting to satisfactory review of weld records, while the remaining 2 can be shown to be within a hydrotest boundary with a signoff for review of welding documentation, although not specifically identified by name.

Additional information pertaining to the quality of the 15 embedded field welds can be found in QC reports (ie., nonconformance reports or deficiency disposition reports*) associated with construction of this piping. Notably, several of these records contain WDR and repair WDRs for embedded welds, providing information pertaining to welder id, filler material and / or NDE for those welds. Pipe Spool Modification packages were located on microfilm; these have been reviewed to determine if any field changes had been made to the pipe spools as supplied from the vendor. Construction era procedures and specifications have been reviewed to identify programmatic requirements pertinent to construction quality.

(* Note - These QC records address routine construction issues which were satisfactorily resolved, and do not have any adverse implications on overall construction quality. On the contrary, the existence of such records serves to strengthen the position that construction was subject to the appropriate level of QC scrutiny.)

Field inspections - Reinspection and NDE of the 37 piping field welds and 12 hanger attachment field welds within the ASME Section III SFP Cooling System portion of the plan scope has been completed. WDRs were generated to document the inspection results; these will be reviewed by both Harris Nuclear Plant Quality Control personnel and the site Authorized Nuclear Inspector (ANI). These inspections also located and recorded weld symbols from each field weld to verify which welds were performed by the pipe spool vendor and to identify the specific welder responsible for field welds. This information was reviewed against pipe spool modification records and vendor data packages to determine that the original vendor welds were intact (had not been replaced or altered by field work), and to ensure that all welds had been identified and their origin accounted for. A total of 4 externally accessible field welds were also subject to internal examination by engineering and welding craft supervisory personnel, with no anomalies being identified which might indicate substandard weld quality.

The internal examination of externally inaccessible field welds is an integral component of the Piping Pedigree Plan These inspections will be completed prior to post-modification acceptance testing. CP&L has contracted with a specialty vendor to provide remote camera inspections of a substantial portion of the embedded piping and field welds. An inspection procedure will be developed specifically for this activity and will include detailed inspection and acceptance criteria. Based on a feasibility walkdown with the vendor, it is anticipated that greater than one third of the embedded field welds will be subject to an internal inspection in this manner. These inspections will take place at the appropriate interval in the modification process, when pool levels are lowered and the welded piping blanks are removed. Any discrepancies will be appropriately dispositioned at that time, including any necessary supplemental submittals to this 10CFR50.55a Alternative Plan.

Filler Material Analysis - All of the accessible large bore FPCCS piping field welds were subject to examination and/or testing to ascertain the composition of filler material. Generally, this was done using a nondestructive x-ray diffraction "alloy analyzer". In addition, chip samples were taken from three welds at random to support the validity of the alloy analyzer results. The results of this effort support that filler material alloy used in these field welds is consistent with that required by site specifications and welding procedures. The carbon steel CCWS piping welds do not lend themselves to conclusive identification using an x-ray diffraction analyzer, so the three field welds in this piping will either be subject to chemical analysis of chip samples, or as an alternative, cut out and replaced.

B31.1 Piping:

The non-safety related piping and equipment providing skimmer, purification and other support functions for the 'C' and 'D' spent fuel pools was very nearly completed at the time of original construction. All of this piping which will be retained in the final design is considered in the scope of the piping pedigree plan. As with the ASME Section III piping, vendor records can be located for this piping, but not the construction records associated with field installation. Under B31.1 and plant welding procedures, this piping would have been subject to external visual inspection at the time of construction. Reinspections have been performed on a large number of these field welds, with none being rejected. A complete reinspection of this piping will be accomplished as part of the modification effort, and a full system hydrotest to original construction requirements will be completed as part of post-modification acceptance testing.

Piping Pedigree Plan Conclusion - an acceptable level of quality and safety

10CFR50.55a(a)(3) allows for the development of an alternative plan with the permission of the Office of Nuclear Reactor Regulation if it can be demonstrated that the proposed alternative would provide an acceptable level of quality and safety, or if compliance with the requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. In the case of unavailable Unit 2 construction records, a great deal of evidence can be compiled to demonstrate that this piping was indeed constructed to the quality requirements consistent with the construction codes. These are summarized as follows:

Design - CP&L held the N certificate over the ASME Section III portion of Harris Nuclear Plant Construction. A single N Certificate program was developed and implemented uniformly to ensure code compliance for the entire site. All materials were specified to a common program using the same procurement specifications. The same welder qualification program and weld procedures, weld engineering, NDE program, and QC program were common to the site.

Work and Document Control - The Harris Nuclear Plant was designed and constructed (to the extent that it was completed) under a single construction program. Common work control procedures, document control, warehousing and storage facilities were used throughout the site. Generally, the same pool of craft and supervisory personnel, QC personnel and engineering staff was available for construction of all four units.

Welder Qualification - Welder identification symbols have been identified at each of the externally accessible field welds, and can be traced to welders qualified to perform that weld. The chronology of precisely when a welder was qualified vs. when the weld was made is difficult to establish since the precise time the weld was performed cannot be determined, but the work control procedures ensure that the appropriate qualifications were established prior to performing weld, particularly with regard to welds within ASME Section III boundaries.

Obviously, welder identification symbols cannot be inspected and recorded for the 15 embedded welds, but again, the same program and procedures would have applied. Work procedures specifically directed the creation of WDR packages for all welds within code boundaries and required that the supervisor ensure that welders were appropriately qualified. Besides the craft supervisor, welder qualification would have been subject to scrutiny by QC and the ANI upon review of the weld records. Of the 15 embedded field welds, QC construction reports provide the identification of welders associated with at least 3 of these welds. No direct records of welder identification have yet been located for the remaining 12 embedded field welds, but hydrostatic test records have been located which attest to the existence of completed WDR packages for these welds at the time of construction. These records contain signatures individually attesting to satisfactory review of completed WDRs for 9 of the 15 embedded field welds, with an additional 4 welds being specifically identified as being within the test boundary with a general signoff attesting to satisfactory review of weld records. The remaining 2 embedded field welds were also shown to be within a hydrotest boundary, although not specifically identified by name.

Generally, the same pool of welders was available for work on Unit 2 as was for the completed Unit 1 at any point during construction. A programmatic lack of appropriate welder qualification would have represented a quality assurance breakdown in the welder qualification program for the site, not just for a given unit. Thus, the satisfactory completion and subsequent operation of Unit 1 using a common craft pool qualified under a single welder qualification program provides strong assurance that the Unit 2 welders were also appropriately qualified.

Filler Material Identification - The WDR package generated for each field weld contained the heat number of weld filler metal which provided the traceability for this material. Since the WDRs are typically the only historical source of this information, material certification cannot be directly established for field welds without these records. However, assurance that the filler material was procured to ASME Section III requirements and supplied with traceability records is provided in Site Specification SS-021 (Purchasing Welding Materials for Permanent Plant Construction). Per this procedure, austenetic stainless steel weld filler material procured for permanent plant welding (such as would have been used in the embedded FPCCS piping) was purchased to ASME Section III requirements, including those requirements associated with traceability and certification.

Issuance and control of weld filler material was strictly controlled through the site materials control program. This program and its implementing procedures were common to all Harris units under construction. The site materials control program was regularly subject to QC audit to ensure compliance with the site ASME Section III Program Manual.

An examination and testing program has been completed for the accessible large bore piping welds in the ASME Section III portion of the 'C' and 'D' pools' FPCCS, as well as 12 hanger welds on this piping. Each of these welds was tested either by use of a non-destructive alloy analyzer or by removing chip samples for chemical assay. In each case, the results supported that the filler material alloy was consistent with that required by site specifications and welding procedures. Such inspections cannot be performed for the inaccessible welds, but the quality of filler metal in these welds is supported by the existence of hydrotest records as discussed above, the existence of QC records for several of these welds which do provide certification and traceability information, the procurement requirements of Site Specification SS-021, as well as satisfactory test results from the 22 accessible welds. The 3 carbon steel CCW field welds in the Piping Pedigree Plan will also be subject to chemical analysis of chip samples to verify composition.

NDE - The WDR package generated for each field weld contained the record of code required inspections and non-destructive examination. The specification of required NDE was a line item on the WDR, and completion of these examinations was affirmed by signature on the WDRs and supported by NDE records included in the respective piping isometric package. Site work control procedures required that these examinations be performed and appropriately documented, and it is clear from interviewing plant personnel that these piping isometric packages were generated and did exist until recently discarded. Since the WDRs are again the only source of this information, the completion of original construction NDE cannot be directly established for the field welds in question.

To address the issue of NDE records, each of the accessible field welds identified as being in the Piping Pedigree Plan scope has been subjected to reinspection and NDE consistent with that which would have been originally performed and found to be acceptable. Obviously, this level of NDE cannot be reperformed on the field welds embedded in concrete, but the existence of hydrotest records attesting to review of completed WDR, QC records for several of these welds which do contain the appropriate NDE records, and the satisfactory NDE of accessible field welds with no rejections provides assurance that the NDE was satisfactorily completed for the embedded welds as well.

The internal camera inspection of a large percentage of embedded field welds will also be performed against inspection criteria developed to provide both subjective examination of weld quality and, to the extent feasible, objective compliance with code and procedural requirements. While an inspection of this nature is not a Code requirement, it is significant in that it will provide direct physical evidence of quality for the embedded field welds. These inspections will take place at the appropriate interval in the modification process, when pool levels are lowered and the welded piping caps are removed. Any discrepancies will be appropriately dispositioned at that time, including any necessary supplemental submittals to this 10CFR50.55a Alternative Plan.

In summary, the portion of the 'C' and 'D' FPCCS which were installed at the time of original plant construction were constructed under CP&L's N Certificate program, using sitewide programs and controls for quality assurance and a common pool of craft, quality control and engineering resources. There is no evidence to support that the level of quality in this portion of Harris plant construction is any less than that of Unit 1, and indeed, it would be difficult to conceive of an unacceptable deficiency which might exist in the partially completed Spent Fuel Cooling facilities without implicating the possibility of its existence in Unit 1 as well. That Unit 1 was completed, licensed and has been in commercial operation for approximately 12 years without cause to suspect construction quality provides strong assurance of that the quality assurance programs for the site were suitably comprehensive and fully implemented. It follows that a comparable level of quality exists in the partially completed Unit 2 facilities, including those for spent fuel storage.

Beyond programmatic assurances, a large body of evidence has been compiled which directly attest to quality of construction. Vendor data packages, hydrostatic test records, QC records and other construction era documentation has been retrieved which constitute substantial proof of compliance with site programs and procedures. An examination effort has been completed in which code required external NDE of accessible welds has been reperformed with no rejectable indications, and material examinations provide proof that the filler metal used in field welds was appropriate for the weldment. These results provide direct evidence of the quality of accessible field welds, and by extension, the smaller group of welds which are embedded. Internal examination of a significant percentage of these embedded field welds provides an additional measure of quality assurance beyond that required by the Code.

There is no evidence that supports that the missing records were never generated, and to the contrary, document control records indexes indicate that these piping isometric packages were transferred to QA storage and maintained there until they were inadvertently discarded in a document control "cleanup effort". Adverse Condition Report 93-354 was generated at that time which specifically identifies that installation documentation for the 'C' and 'D' FPCCS, including installation verification data and field weld records, was inadvertently discarded during Sept. 1993.

It is concluded that the Piping Pedigree Plan outlined above provides ample evidence exists to support that the portion of the Harris plant associated with the 'C' and 'D' Spent Fuel Pools which was completed during the original site construction effort was indeed constructed to the appropriate level of quality and safety and in compliance with construction code requirements. It follows that the issue of missing code documentation is simply that, a documentation issue, and does not infer a physical lack of quality in the field.

III. Alternative Plan for Continuance of Design and Construction

The original construction of the Harris Nuclear Plant was subject to the full requirements of ASME Section III of the ASME Boiler and Pressure Vessel Code under the authorization of a single N Certificate program maintained by CP&L. This site ASME Section III QA program was discontinued shortly after completion and turnover of Unit 1, and a corporate QA program meeting 10CFR50 Appendix B requirements was implemented as required to address plant operation, including Section XI requirements regarding inspection, repair and replacement activities. Thus, the original construction program no longer exists and it is not possible to complete construction of the 'C' and

'D' FPCCS as a continuance of this program. Further, since a Code data report was not prepared by CP&L for this partially completed piping and equipment under its N certificate holder program at the time it was constructed, responsibility for its construction cannot be now assumed by another N certificate holder under a current program. It follows that it is not possible to N stamp the previously completed portion plant associated with the 'C' and 'D' Spent Fuel Pools. Given this, and considering that the majority of construction has been completed, it is the opinion of CP&L and code authorities within the Hartford Steam Boiler Inspection and Insurance Co. and Bechtel Power Corporation that there is no benefit with invoking an N certificate program to govern the completion of the relatively small outstanding portion of construction vs. using another suitable quality assurance program of comparable rigor.

Since this portion of the plant was never turned over at the time of construction, it is not considered part of the operating facility from the perspective of the ASME code and its completion could not be interpreted as a replacement activity as defined in Section XI. However, the site Section XI Repair and Replacement Program as implemented under the Corporate 10CFR50, Appendix B QA Program does contain many elements of quality control (ie., welder qualification, weld procedures, inspections, documentation, etc.) consistent with the original construction program. Therefore, CP&L proposes to complete the design of this portion of the plant to appropriate ASME Section III requirements, but utilize the Corporate 10CFR50, Appendix B QA Program and site procedures for those elements of quality assurance for which it is appropriate to provide. Generally, any conflicts between the ASME Section III requirements and that of the Corporate 10CFR50, Appendix B QA Program (and the corporate and site procedures which invoke it) would be conservatively dispositioned, such as the use of ASME Section III hydrotest requirements vs. those requirements found in Section XI.

A set of supplemental quality assurance requirements has also been developed to augment the Corporate 10CFR50, Appendix B QA Program in completion of the Code portions of the plant associated with the 'C' and 'D' Spent Fuel Pools. These requirements were obtained by a close review of the requirements in the approved ASME Section III Construction QA Program Manual as it existed at the time of completion of construction vs. those of the currently existing Corporate 10CFR50, Appendix B QA Program, and are specifically intended to identify and conservatively reconcile deficiencies in the corporate program with ASME Section III requirements. For instance, the supplemental requirements specify a level of ANI involvement commensurate with ASME Section III requirements, including review of work packages prior to field issuance, integration of ANI involvement into the work control process, and final review and approval of documentation subsequent to work completion. Other highlights of the supplemental quality assurance requirements include integration of comparable requirements for design specifications and a process for system documentation review and turnover similar to that of N Stamping. These supplemental quality assurance requirements will be implemented by integration into the modification package, or when necessary, by procedure revision.

Since the current Corporate 10CFR50, Appendix B QA Program is sufficient to govern ongoing operation of the Harris Plant (including Section XI repair and replacement activities), it follows that it is of sufficient rigor for the construction effort to complete and activate the portion of the plant associated with the 'C' and 'D' spent fuel pools. There are instances wherein the Corporate 10CFR50, Appendix B QA Program does not address specific ASME Section III quality assurance requirements, and a set of supplemental quality assurance requirements has been developed specifically for the purpose of addressing these items. This approach for continuance of construction is both technically acceptable and commercially viable, and will ensure the requisite level of quality and safety in the completed systems as discussed in 10CFR50.55a(a)(3)(i).

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
SPENT FUEL STORAGE

UNREVIEWED SAFETY QUESTION ANALYSIS

CCW UNREVIEWED SAFETY QUESTION DISCUSSION

As part of the preparation of the design change package for the tie-in of the existing Component Cooling Water (CCW) system, a 10CFR50.59 Safety Evaluation was prepared. The scope of the evaluation addressed the tie-in of the Unit 1 CCW system to the heat exchangers of the 'C' and 'D' Fuel Pool Cooling and Cleanup System (FPCCS). This evaluation considered a heat load of no more than 1.0 MBtu/hr[1] in the 'C' and 'D' Spent Fuel Pools (SFP). In support of this design change package, a thermal-hydraulic model was created to analyze the overall impact of this additional heat load, including its impact on the Emergency Service Water (ESW) system and the Ultimate Heat Sink (UHS). This analysis demonstrated that adequate thermal margin exists in the CCW system to accommodate the proposed additional heat load in Spent Fuel Pools 'C' and 'D'. However, it was determined that while the post-modification configuration was safe it was potentially an Unreviewed Safety Question (USQ). The following discussion delineates the methodology used in this analysis and the reasoning behind its classification as a USQ.

CURRENT SYSTEM CONFIGURATION

The CCW system serves as an intermediate closed cooling water system between the radioactive or potentially radioactive systems and the non-radioactive service water system. The FPCCS rejects its heat via the CCW system which in turn rejects its heat via the station service water system to the Ultimate Heat Sink. The Ultimate Heat Sink is comprised of three separate possible cooling sources that are used independently: the main cooling towers for normal service and the auxiliary or main reservoir for emergency service.

The CCW system provides cooling to various safety related (RHR Heat Exchangers, RHR Pump, and Spent Fuel Pool Heat Exchangers) and non-safety related heat loads. The CCW system contains two separate trains, each containing a component cooling water system heat exchanger. There are three component cooling water pumps for the two trains. Two pumps are normally operated during cooldown, with each pump supplying half of the total component cooling water flow. Normal power operation only requires one pump for operation with another on standby. In the event of a LOCA, only one pump is required although two CCW pumps start to ensure cooling flow to the safeguards loads in the event of a single failure.

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 non-essential header to maintain protection against a single passive failure and to provide sufficient flow to their respective RHR trains. In this alignment, each CCW train

[1] Controlled by revised Technical Specification 5.6

is balanced to provide greater than 5 gpm to the RHR pump for cooling the pump and 6050 gpm is available to the RHR heat exchanger.

The minimum CCW flow that 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 the operators are directed by Operating Procedures to restore sufficient CCW flow from one CCW train to the SFP heat exchangers to maintain the temperature of the spent fuel pools to less than 150°F. Based on the CCW flows established to the RHR heat exchanger and the RHR pump when the non-essential header is isolated, each train is capable of individually providing the specified 5600 gpm and 5 gpm in addition to the minimum flow of 1789 gpm through the SFP heat exchangers 'A' and 'B'.

10CFR50.59 SAFETY EVALUATION OVERVIEW

Performance of the 10CFR50.59 Safety Evaluation requires that certain questions must be answered to determine if the proposed activity will require the completion of an Unreviewed Safety Question Determination (USQD). Since this design change involved a change to the Technical Specifications (to facilitate the control of the heat loads in Spent Fuel Pools 'C' and 'D') it could not be implemented without prior NRC approval. Nonetheless it was determined that a USQD be performed since this modification involves a change to the facility, a change to procedures described in the SAR, a change to the licensed operator training program, etc. and no previously approved USQ determination fully bounds this activity.

UNREVIEWED SAFETY QUESTION DETERMINATION

The USQD analysis performed yielded an affirmative answer to the question concerning whether the proposed activity may reduce the margin of safety as defined in the basis for any Technical Specification. The portion of the design change which triggered this affirmative response centered on the analysis methodology used in the thermal-hydraulic analysis to verify that adequate excess thermal capacity existed in the CCW system to accommodate the additional heat loads from Spent Fuel Pools 'C' and 'D'. The following is a discussion of the subject thermal-hydraulic analysis and the logic that prompted the decision to categorize this activity as a USQ.

The new thermal-hydraulic analysis was performed to evaluate the 1.0 MBtu/hr heat load that would be added to Spent Fuel Pools 'C' and 'D' as a result of this activity. This thermal-hydraulic analysis includes an assessment of Core Shuffle and Abnormal Full Core Offload scenario heat loads to satisfy the analysis requirements of NUREG-0800 (Standard Review Plan). The analysis demonstrates that adequate margin exists during all normal and accident modes of system operation and that the CCW system has adequate thermal-hydraulic capacity to provide the minimum flow required by the fuel pool heat exchangers after the activation of Pools 'C' and 'D'. As a result of the analysis, the minimum CCW flow to the RHR heat exchangers and the minimum ESW flow to the CCW heat exchanger change from the current requirements.

The analysis considered the additional spent fuel pool cooling heat load well as a 6% modeling uncertainty and degraded IST pump performance. The new analysis also accounts for the change in RHR heat exchanger performance as it relates to the variation in fluid properties. This is a departure from the current licensing basis with regard to RHR heat exchanger performance. Current analyses assume that the performance of the RHR heat exchanger is fixed based on the design values associated with the heat exchanger data sheet. The data sheet fixes the tubeside inlet temperature to the RHR heat exchanger to 139°F, however, during the development of the new thermal-hydraulic analysis it was noted that RHR tube side inlet temperature is postulated to rise to 244.1°F during the initial phase of containment sump recirculation. This increase in the tube side fluid temperature is predicted to increase the overall heat transfer coefficient approximately 10% due to the change in tube side fluid viscosity. These conditions tend to increase heat transfer through the RHR heat exchanger and might otherwise increase CCW system supply temperatures above the maximum of 120°F under limiting conditions of minimum CCW heat exchanger ESW flow and maximum ESW supply temperature. The two previously mentioned changes in minimum CCW flow to the RHR heat exchangers and the minimum ESW flow to the CCW heat exchanger are specified to address this issue.

The minimum specified CCW system flow to the RHR heat exchanger is reduced to a level consistent with a heat rejection of 111.1 MBtu/hr under the new analysis. It is important to note that this heat rejection rate is consistent with the existing post-LOCA containment pressure/temperature calculations, such that no change in containment heat removal is prescribed. The thermal-hydraulic calculation includes an analysis of RHR heat exchanger performance to determine the minimum shell side flow rate to maintain 120°F shell side inlet temperature, 244.1°F tube side inlet temperature and 1.846E6 lbm/hr tube side flow rate to maintain the aforementioned consistency. It was shown that a minimum CCW system flow rate of 4874 gpm at 120°F is required at the beginning of the sump recirculation phase. The specified CCW system flow to the RHR heat exchanger under these conditions; assuming 6% model uncertainty consistent with previously developed hydraulic models is 5166 gpm, or approximately 5200 gpm. As the containment sump temperature decreases, the minimum required CCW system flow rate decreases based on maintaining a maximum RHR heat exchanger tube side outlet temperature of 180°F. The CCW system was initially rebalanced in the model in the LOCA recirculation (RHR only) alignment, with a 10% degraded CCW pump curve. When the nominal CCW pump curve is applied to this alignment CCW system flow to the RHR heat exchanger increases to approximately 5440 gpm, resulting in an increased RHR heat exchanger heat duty of 118 MBtu/hr. Under the most limiting postulated conditions, the increased RHR heat exchanger duty could increase CCW system supply temperature marginally above its 120 °F design limit. This concern is addressed by increasing the current minimum required ESW flow to the CCW system heat exchanger from 8250 gpm to a slightly higher value of 8500 gpm.

Summarizing the preceding discussion, a reduction in the minimum specified RHR heat exchanger CCW system flow from 5600 gpm to 5200 gpm and an increase in the minimum specified CCW heat exchanger ESW system flow from 8250 gpm to 8500 gpm are prescribed by the new thermal-hydraulic analysis in order to maintain all thermal/hydraulic assumptions which are used in the HNP containment analysis. A minimum specified ESW system flow of 8500 gpm to the CCW heat exchangers was verified to be within the capacity of the current system even considering the most limiting ESW system single failure.

Per CP&L's Draft SER OI 365 - ASB Question 9.2.2(1) Revised Response, 5600 gpm was the number specified to the NRC as that which was "...sufficient capacity..." from one train of CCW "...to carry the heat loads from the ... RHR heat exchanger". Section 9.2.2 of the SER (NUREG-1038) states that *"5600 gpm would be <u>required</u> for the RHR heat exchanger"* and that *"...flow remaining from one operating CCW train would be sufficient to keep the Unit 1 SFP at a temperature of 150°F or less"*. In this context, it follows that the NRC's acceptance of the CCW system is based, in part, on ensuring that 5600 gpm CCW system flow is provided to the RHR heat exchangers under these conditions. Therefore, the decrease in minimum required CCW system flow to the RHR heat exchangers is deemed to be a reduction in the acceptance limit. The change in the minimum specified RHR heat exchanger CCW system flow from 5600 gpm to 5200 gpm as a result of the new thermal-hydraulic analysis does not prevent the CCW system from meeting the previously defined criteria in any way. The addition of Spent Fuel Pools 'C' and 'D' to the CCW system does not directly result in changing the minimum specified RHR heat exchanger CCW system flow. As previously discussed, an increase in the minimum specified CCW heat exchanger ESW system flow from 8250 gpm to 8500 gpm also results from the new thermal-hydraulic analysis but unlike the minimum specified RHR heat exchanger CCW system flow, this value is not mentioned in the SER.

<u>SUMMARY</u>

In determining whether or not the proposed activity reduces the margin of safety, as defined in the basis of any Technical Specification, the only item which could not be ruled out was that associated with the reduction in the minimum CCW flow to the RHR heat exchanger. Since this is deemed to be a change in the acceptance limit, this activity is considered to be a USQ.

Laufer Exhibit 2

CP&L

Carolina Power & Light Company
PO Box 165
New Hill NC 27562

James Scarola
Vice President
Harris Nuclear Plant

MAY 25 1999

SERIAL: HNP-99-087
10 CFR 50.71(e)

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
FSAR AMENDMENT NO. 49

Dear Sir or Madam:

In accordance with 10 CFR 50.71(e), Carolina Power & Light Company (CP&L) hereby submits the periodic update to the Final Safety Analysis Report (FSAR) for the Harris Nuclear Plant (HNP). Amendment No. 49 to the FSAR includes changes to the plant since the submittal of Amendment No. 48 and analyses prepared pursuant to NRC requirements. Amendment No. 49 is current through November 28, 1998 (end of RFO 8). Some changes and analyses completed after November 28, 1998 have also been included in this amendment. Enclosed are one original and ten copies of the amendment.

Questions regarding this submittal may be referred to Mr. J. H. Eads at (919) 362-2646.

Sincerely,

James Scarola

MGW

Enclosure

c: Mr. J. B. Brady (NRC Senior Resident Inspector, HNP)
Mr. Rich Laufer (NRR Project Manager, HNP)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)

9906030074

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

1.1.1 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) is submitted in support of the Carolina Power & Light Company's (CP&L) application for a Class 103 facility operating license for the Shearon Harris Nuclear Power Plant (SHNPP). This FSAR has been organized in accordance with the guidelines contained in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 3 dated November 1978), and the regulations of the NRC set forth in 10CFR50.

Lists of acronyms, abbreviations and names of major buildings and structures used throughout this FSAR are given in Tables 1.1.1-1, 1.1.1-2, and 1.1.1-3,
respectively. Figures 1.1.1-1 and 1.1.1-2 provide flow diagram symbols while 26
Figure 1.1.1-1a provides piping and instrumentation symbols used on engineering drawings throughout the FSAR.

Carolina Power & Light Company and the North Carolina Eastern Municipal Power 26
Agency (NCEMPA) own the plant. Carolina Power & Light Company has the overall responsibility to ensure that it is designed, constructed, and operated without undue risk to the health and safety of the public. Ebasco Services, Incorporated is the architect/engineer responsible for the design, engineering, and equipment and material procurement for SHNPP. This includes all plant structures, systems, and components except for those provided by Westinghouse Electric Corporation, the Nuclear Steam Supply System (NSSS) Supplier. Daniel Construction Company, Inc., as the constructor, performed the major part of the plant construction. Selected portions of the work, however, were performed by other contractors under direct supervision of CP&L.

1.1.2 STATION LOCATION

The SHNPP site is located in the extreme southwest corner of Wake County, North Carolina, and the southeast corner of Chatham County, North Carolina. The city of Raleigh, North Carolina, is approximately 16 miles northeast and the city of Sanford, North Carolina, is about 15 miles southwest.

1.1.3 NUCLEAR STEAM SUPPLIER

The Nuclear Steam Supply System (NSSS) for the Unit is a pressurized water reactor (PWR) consisting of three closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to the "hot" leg of one of the loops. The NSSS, along with the design and fabrication of the initial fuel core, is supplied by Westinghouse Electric Corporation.

1.1.4 Containment

The Containment is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel. The Containment is designed by Ebasco Services Incorporated, architect/engineer for SHNPP.

1.1.5 Core Thermal Power

The Unit is licensed for a core thermal power output of 2775 megawatts thermal (Mwt). The total unit thermal output is approximately 2787 Mwt, which includes 12 Mwt from the reactor coolant pumps. The thermal output corresponds to an electrical output of approximately 860 megawatts electric (Mwe) net or 917 Mwe gross. The NSSS stretch rating has been estimated at 2910 Mwt, all safety systems, including containment and engineered safety features have been analyzed for operation at 2775 Mwt or greater. Some analyses have been performed at as high as 2910 Mwt.

1.1.6 Schedule

The construction schedule for SHNPP is based on a commercial operation date in the fourth quarter of 1986. This schedule requires that an operating license be issued in time for fuel loading by June 1986.

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

Laufer Exhibit 3

NUREG-1038

Safety Evaluation Report
related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2

Docket Nos. STN 50-400 and STN 50-401

Carolina Power and Light Company
North Carolina Eastern Municipal Power Agency

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1983

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Carolina Power and Light Company (hereinafter referred to as the applicant) filed with the Atomic Energy Commission (AEC) an application docketed on September 7, 1971 for licenses to construct and operate its proposed Shearon Harris Nuclear Power Plant Units 1, 2, 3, and 4 (Harris plant or facility). The facility is located in Wake County, North Carolina, approximately 16 miles southwest of the nearest boundary of Raleigh, the state capital.

The AEC (now the Nuclear Regulatory Commission (NRC or Commission)) reported the results of its preconstruction review in a Safety Evaluation Report (SER) dated December 22, 1972, and in Supplements 1 through 4 of the SER, dated April 27, 1973; May 6, 1974; July 30, 1977; and September 20, 1977.

Following a public hearing before an Atomic Safety and Licensing Board, Construction Permits Nos. CPPR-158, CPPR-159, CPPR-160, and CPPR-161 were issued on January 27, 1978.

The applicant submitted an application for an Operating License (OL) by letter dated June 26, 1980. The NRC performed a pre-docketing acceptance review and determined that sufficient information was available to initiate detailed environmental and safety reviews. Docketing occurred on December 22, 1981. The applicant informed the NRC on December 18, 1981 that Units 3 and 4 had been cancelled, and on January 7, 1982 requested that Units 1 and 2 be considered concurrently for operating licenses.

Prior to issuing an OL for a nuclear power plant, the NRC staff is required to conduct a review of the effects of the plant on public health and safety. The staff safety review has been based on the Final Safety Analysis Report (FSAR) that accompanied the OL application through Amendment 10. During the course of its review, the staff held a number of meetings with representatives of the applicant to discuss the design, construction, and proposed operation of the plant. The staff requested additional information, which the applicant provided partly in response to questions and partly in amendments to the FSAR. This information is available to the public for review at the NRC Public Document Room at 1717 H Street, NW, Washington, D.C. and at the Local Public Document Room at the Wake County Public Library, Fayetteville Street, Raleigh, North Carolina.

Following the accident at Three Mile Island Unit 2 (TMI-2), the Commission paused in its licensing activities to assess the impact of the accident. During this pause, the recommendations of several groups established to investigate the lessons learned from TMI-2 became available. All available recommendations were correlated and assimilated into a "TMI Action Plan," now published as NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." Additional guidance relating to implementation of the Action Plan is in NUREG-0737, "Clarification of TMI Action Plan Requirements," and Supplement 1 to NUREG-0737. These licensing requirements have been established to ensure that

9 AUXILIARY SYSTEMS

FSAR Chapter 9, "Auxiliary Systems," has been reviewed in accordance with the SRP (NUREG-0800). The staff has reviewed the design of the auxiliary systems necessary for safe reactor operation, shutdown, and fuel storage.

The auxiliary systems necessary for safe reactor operation or shutdown include the essential service water system (ESWS), the component cooling water system (CCWS), the ultimate heat sink (UHS), the condensate storage facility, the essential services chilled water system, the control room area ventilation systems, and the engineered safety feature ventilation system.

The auxiliary systems necessary to ensure the safety of the fuel storage facility include new fuel storage, spent fuel storage, the spent fuel pool cooling and cleanup system, fuel-handling systems, and the spent fuel pool area ventilation system.

The staff has also reviewed other auxiliary systems to verify that their failure will not prevent safe shutdown of the plant or result in unacceptable release of radioactivity to the environment. These systems include the nonessential service water system, the demineralized water makeup system, potable and sanitary water system, the nonessential services chilled water system, the waste processing building cooling water system, the compressed air systems, the equipment and floor drainage systems, the turbine building area ventilation system, and the control rod drive mechanism ventilation system. This review also included nonessential portions of the essential systems discussed above.

9.1 <u>Fuel Storage and Handling</u>

9.1.1 New Fuel Storage

The new fuel storage facility was reviewed in accordance with SRP 9.1.1.

The acceptance criteria for the new fuel storage facility include meeting ANS 57.1, "Design Requirements for Light-Water Reactor Fuel Handling System," and ANS 57.3, "Design Requirements for New LWR Storage Facilities." The SRP guidelines were used in lieu of ANS 57.1 and ANS 57.3.

[redacted]. [redacted]ed [redacted] fuel. The storage to [redacted] facility consists of two new fuel pools, one for each unit. The pools are interconnected by means of a transfer canal whose length is that of the fuel-[redacted] by handling building. H[redacted]

Each new [redacted], this constitutes enough fuel for more than three fuel cores. The new fuel is stored in rack modules of several designs that can be removed and installed under water. There are separate new fuel pools for each unit. Nevertheless, the new fuel storage system is designed so that new fuel for either unit may be stored in either new fuel pool, space permitting. However, no failure resulting from sharing of the storage area prevents safe shutdown of either or both units. Therefore, the requirements of GDC 5 are satisfied.

The[redacted]ks and[redacted]. This building is also designed against flooding [redacted] (see Sections 3.4.1 and 3.5.2 of this SER). Thus, GDC 2 and 4 and RG 1.29, Position C.1, are satisfied.

The new fuel storage facility is not [redacted] moderate- [redacted]achinery. Separation from such potential missile sources protects the new fuel from internally generated missiles and the effects of pipe breaks (see Sections 3.5.1.1 and 3.6.1 of this SER).

Accidental damage to the new fuel would release relatively minor amounts of radioactivity that would be accommodated by the spent fuel pool area ventilation system. Thus, GDC 61 is satisfied.

The applicant stated that the new fuel storage racks are designed so that with f[redacted] un[redacted] If the unborated water is replaced [redacted] 0.95 or less. The staff is making an independent evaluation of new fuel reactivity under the most adverse conditions to ensure that the new fuel storage facility complies with GDC 62 with regard to criticality as a result of storage of new fuel assemblies.

The storage racks can withstand an uplift force equal to the maximum uplift capability of the spent fuel bridge crane and are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. Thus, GDC 62 is satisfied.

Based on its review, the staff concludes that the new fuel storage facility is in conformance with GDC 2, 4, 5, and 61 as they relate to new fuel protection against natural phenomena, missiles, shared functions, and radiation protection and prevention of criticality, and with 1.29, Position C.1, as it relates to seismic classification. The staff is making an independent evaluation as to whether the design of the new fuel storage facility complies with the requirements of GDC 62. P[redacted]udes that the design of the new fuel storage facility meets SRP 9.1.1.

9.1.2 Spent Fuel Storage

The spent fuel storage facility was reviewed in accordance with SRP 9.1.2.

The acceptance criteria for the spent fuel storage facility include meeting various portions of the guidelines of ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." The guidelines contained in the SRP were used in lieu of ANS 57.2. The acceptance criteria also include RG 1.115, "Protection Against Low-Trajectory Turbine Missiles." Turbine missiles are evaluated in Section 3.5.1.3 of this SER.

A spent fuel storage facility is provided for each unit. These pools are inter-connected by the main fuel transfer canal, but they are normally isolated by removable gates designed to seismic Category I standards. The maximum storage capacity of the two spent fuel pools is 3024 PWR assemblies, which is more than 19 full PWR cores. Fuel may be stored in a combination of 6 x 10, 6 x 8, and 7 x 7 PWR rack modules. The applicant stated that rack rearrangement would have no effect on stored fuel criticality. The spent fuel pools may also be used for the storage of BWR fuel; the 7 x 7 PWR fuel storage modules may be interchanged with 11 x 11 BWR fuel storage modules because both cover the same floor area. The PWR racks have a center-to-center distance of 10.5 inches between cells, while the BWR racks have a center-to-center distance of 6.25 inches. PWR fuel assemblies cannot be inserted into BWR rack modules, but BWR fuel assemblies can be inserted into PWR rack modules. Introduction of a BWR fuel assembly into a PWR rack module, however, will result in a subcritical array of fuel assemblies with $K_{eff} \leq 0.95$. The applicant has provided sufficient information regarding fuel enrichment, geometry of spent fuel and racks in pool, and calculational methodology to permit the staff to make an independent evaluation of spent fuel pool reactivity under the most adverse conditions. The staff has reviewed the information provided and concludes that $K_{eff} \leq 0.95$ under the most adverse conditions. Therefore, the staff concludes that the spent fuel facility complies with a GDC 62.

The structure housing the spent fuel storage facility (the fuel-handling building) is designed to seismic Category I criteria, as are the storage racks, pool liners, gates, canals, and storage pools. The building is also designed against flooding and tornado missiles (see Sections 3.4.1 and 3.5.2 of this SER). The staff concludes that GDC 2 and RGs 1.13, Position C.3; 1.29, Positions C.1 and C.2; and 1.117, Positions C.1 through C.3, are satisfied for the spent fuel storage facility.

The spent fuel storage facility is not in the vicinity of any high energy lines or rotating machinery. Therefore, physical protection by means of separation is utilized to protect the spent fuel from internally generated missiles and the effects of pipe breaks (see Sections 3.5.1.1 and 3.6.1 of this SER). Thus, GDC 4 and RG 1.13, Position C.3, are satisfied.

The shared portion of the facility has sufficient redundancy of services and is of seismic Category I, Quality Group C design, so that an accident in one unit with loss of offsite power will not impair its ability to safely store the spent fuel. This satisfies GDC 5.

The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift forces exerted by the spent fuel bridge crane. In addition, loads greater than a fuel assembly are not carried over spent fuel, in compliance with the guidelines of Position C.5.c of RG 1.13 (see Sections 9.1.4 and 9.1.5 for discussion of protection of spent fuel in both new and spent fuel pools against damage from light and heavy loads).

The fuel-handling building has a ventilation system (the spent fuel pool area ventilation system) to limit the potential release of radioactivity in the event of an accident (see Section 9.4.2 of this SER for a discussion of the spent fuel pool area ventilation system) in accordance with Position C.4 of RG 1.13. In this way, the spent fuel storage facility complies with GDC 61.

Control room and local alarms are provided to alert the operator to high and low pool water level and high temperature in the fuel pool. The fuel-handling building has a radiation monitoring system. These features satisfy GDC 63.

The staff also has reviewed the compatibility and chemical stability of the materials of the spent fuel pool and storage racks wetted by the pool water. The pool liner, rack lattice structure, and fuel storage tubes are stainless steel. The pool contains oxygen-saturated demineralized water containing 2000 to 4000 ppm boron as boric acid.

In this environment of oxygen-saturated borated water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.00×10^{-5} in. in 100 years (Weeks, 1977), which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. Provisions are incorporated to allow for the periodic inspection of spent fuel pool components.

The staff, therefore, concludes that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate based on test data and actual service experience in operating reactors. The staff also concludes that the selection of appropriate materials by the applicant meets GDC 61 by having a capability to permit appropriate periodic inspection and testing of components, and GDC 62 by preventing criticality by maintaining structural integrity of components.

Based on its review, the staff concludes that the spent fuel storage facility is in conformance with GDC 2, 4, 5, 61, 62, and 63 as they relate to protection of spent fuel against natural phenomena, missiles, environmental effects, the facility's shared functions, radiation protection, periodic inspecting and testing of components, prevention of criticality, and performance monitoring, and with RG 1.13, Positions C.1, C.3 and C.4; 1.29, Positions C.1 and C.2; and 1.117, Positions C.1 through C.3, relating to the facility's design, seismic classification, and protection against tornado missiles. Therefore, the staff concludes that the spent fuel storage facility meets SRP 9.1.2.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system was reviewed in accordance with SRP 9.1.3.

The acceptance criteria for the cooling portion of the fuel pool cooling and cleanup system (FPCCS) include meeting RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," if the cooling portion of the system does not meet GDC 2. This does not apply to Shearon Harris because the fuel pool cooling portion of the FPCCS meets GDC 2, as discussed below.

The two new fuel storage pools (one for each unit) and two spent fuel storage pools (one for each unit) are housed in the fuel-handling building (FHB). The FPCCS consists of two fuel pool cooling systems (one for each unit); the fuel pool cleanup systems for both units are separate from the cooling systems. Each fuel pool cooling system services the new and spent fuel pool storage pools associated with that unit. Each FPCCS contains two fuel pools cooling trains; each train contains a heat exchanger, strainer, and fuel pool cooling pump, with each pump capable of being manually loaded onto a separate emergency power supply in case of loss of offsite power. Each cooling training is a 100% subsystem, servicing both the new and spent fuel storage pool in that unit.

The cleanup systems contain two fuel pool skimmer pumps, two demineralizers, filters, and skimmers for all the pools (including the cask loading pool in Unit 1), skimmers for the transfer canals, and two fuel pool and water purification pumps. The main fuel transfer canal, which runs almost the length of the FHB, interconnects the two spent fuel pools.

The new fuel storage pools, at opposite ends of the FHB, are each connected to the main fuel transfer canal via fuel transfer canals; each spent fuel pool is also connected to the fuel transfer canal in its unit. All pools and canals are kept separate by means of removable gates. Makeup to the pools may be provided from a seismic Category I source (the refueling water storage tank) by means of the fuel pool cooling pumps. This complies with Position C.8 of RG 1.13, "Spent Fuel Storage Design Basis."

The ~~fuel pool cooling portion of the FPCCS is designed to seismic Category I~~, Quality Group C standards, whi~~le the fuel pool cleanup systems portion of the~~ FPCCS ~~is designed to nonseismic Category~~ I, Quality Group D standards. The FPCCS is designed to remove the decay heat from the spent fuel assemblies stored in the pools and to maintain the clarity of the water in the pools.

The essential portions of the system are housed in the seismic Category I, flood- and tornado-protected fuel-handling building (see Sections 3.4.1 and 3.5.2 of this SER). The system itself, with the exception of the cleanup portion, is designed to Quality Group C and seismic Category I requirements. Failure of the nonseismic Category I, Quality Group D cleanup portion will not affect operation of the cooling train because isolation capability of that portion of the piping system is provided, and no adverse effect on safety-related equipment would result from such a failure.

The applicant reported the elevations at which the fuel pool cooling and cleanup system (FPCCS) pipelines entered and left the fuel pools. The applicant also noted that skimmer hose length was limited so that only 5 feet could be submerged. Based on this information, the staff concludes that the minimum level to which the water in the pool could be drawn down, assuming syphoning or the worst pipe failure, is about 274.5 feet, while the level of the top of the spent fuel would be 260 feet. This would leave at least 14.5 feet of water above the top of the fuel. Therefore, the design satisfies Positions C.1 and C.2 of RG 1.29; Positions C.1, C.2, and C.6 of RG 1.13; and Position C.2 of RG 1.26, thus meeting GDC 2.

The various components of the system are located in separate missile-shielded cubicles within the tornado-missile-protected fuel-handling building and are separated from other moderate- and high-energy piping systems (see Sections 3.5.1.1 and 3.6.1 of this SER). Thus, GDC 4 and RG 1.13, Position C.2, are satisfied.

The applicant had originally intended (FSAR Section 1.2.3c, page 1.2.3-1) to make the Unit 1 FPCCS operative with one cooling train. However, in response to a staff question regarding possible failure of the FPCCS, the applicant committed to install two fuel pool cooling pumps and heat exchangers for operation of Unit 1. The applicant has made a similar commitment for Unit 2.

The fuel storage pools and FPCCS have been designed to permit transfer of fuel assemblies between units; this does not affect operation of the FPCCS because it is intended to be used to remove the decay heat from the fuel pools with a full complement of spent fuel assemblies. Therefore, the FPCCS meets the requirements of GDC 5.

The applicant has designed the fuel pool cooling system to maintain the temperature of the Unit 2 spent fuel pool (the pool with the greatest estimated maximum heat load) at 124°F with both trains of cooling in operation, assuming a full inventory of fuel assemblies including PWR and BWR spent fuel from the H. B. Robinson and Brunswick plants. Under these same conditions, with only one train in operation, the pool temperature would reach 142°F. The staff has confirmed the applicant's estimate for the cooling heat load with a full complement of spent fuel assemblies, calculated in accordance with BTP ASB 9-2. Thus, the staff concludes that the FPCCS has been designed to meet GDC 44 with regard to cooling capability.

The safety-related component cooling water system provides cooling water to the fuel pool heat exchanger and transfers its heat to the ultimate heat sink (see Sections 9.2.2 and 9.2.5 of this SER). The spent fuel pool pumps can be powered from the emergency (Class 1E) power sources; thus the design meets GDC 44.

Normally, makeup water to the fuel pool is supplied from two seismic Category I refueling water storage tanks, one in each unit. The applicant has stated that the emergency service water system (ESWS) is available to fill the fuel pool through valved and flanged emergency connections as a backup seismic Category I water source. While Unit 2 is being built, only the Unit 1 refueling water storage tank (RWST) is available for normal spent fuel pool makeup. If an accident during that period incapacitates the Unit 1 RWST and causes failure of one of the two spent fuel pools cooling system trains of Unit 1 (single failure), the other train could maintain the pool water at a temperature of approximately 141°F. Water would be lost from the pool only through an evaporative process. In this case, makeup would not be required for several days and could easily be provided within 24 hours by the emergency connections to the ESWS. The staff finds this acceptable and in compliance with the requirements of GDC 44.

The design of the spent fuel pool cooling system and its accessible location are such that periodic testing and inservice inspection of the system can be accomplished. The active components of the spent fuel pool cooling system are either in continuous or intermittent operation during all plant operating conditions. Thus, GDC 45 and 46 are satisfied.

The system incorporates control room alarmed pool water level, temperature, and building radiation level monitoring systems. Although the liners for the new fuel storage pools and spent fuel storage pools are classified as nonnuclear safety, they are designed as seismic Category I. A low flow alarm is provided to warn the operator of any interruption or cessation of flow in each fuel pool cooling train. Thus, the requirements of GDC 63 are satisfied.

The spent fuel pool cleanup system is designed to maintain optical clarity of and to remove corrosion products, fission products, and impurities from the spent fuel pool water. The spent fuel pool water will be sampled weekly for chlorides, fluorides, pH, boron, calcium, magnesium, and radioactivity. The applicant has provided the chemical impurity limits to be maintained in the pool water in accordance with the chemistry criteria and specifications for Westinghouse PWRs (WCAP-7452, Revision 2, 1977). A decontamination factor for the demineralizer will be measured to determine when the resin is to be replaced. A high pressure drop in the filter is the basis for changing the filter cartridge. Area radiation monitors are provided.

The staff has determined that the spent fuel pool cleanup system (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool water, and thus meets the requirements of GDC 61 as it relates to appropriate filtering systems for fuel storage; (2) is capable of reducing occupational exposure to radiation by removing radioactive products from the pool water, and thus meets the requirements of 10 CFR 20.1(c), as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water in the demineralizer and filters, and thus meets Position C.2.f(2) of RG 8.8, as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters and thus meets Position C.2.f(3) of RG 8.8, as it relates to removing crud through physical actions.

Thus, the staff concludes that the spent fuel pool cleanup system meets GDC 61, 10 CFR 20.1(c), and the appropriate sections of RG 8.8.

Based on its review, the staff concludes that the FPCCS is in conformance with GDC 2, 4, 5, 45, 46, 61, and 63 as they relate to protection against natural phenomena, missiles and environmental effects, sharing of systems, cooling capability, inservice inspection, functional testing, fuel cooling and radiation protection, and monitoring provisions, and with the guidelines of RGs 1.13, 1.26, and 1.29 as they relate to the system's design and quality and seismic group classification. The applicant has provided assurance that the operation of the new and spent fuel storage pools for Unit 2 will not be initiated with less than two fuel pool cooling pumps and two heat exchangers operable. Therefore, the staff finds the FPCCS acceptable. The FPCCS meets SRP 9.1.3.

9.1.4 Fuel-Load Handling System

The light-load-handling system was reviewed in accordance with SRP 9.1.4. Except as noted below, conformance with the acceptance criteria, formed the basis for the staff evaluation of the light-load-handling system with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for for the light-load-handling system include meeting the guidelines of ANS 57.1, "Design Requirements for LWR Fuel Handling Systems." The SRP guidelines were used in lieu of ANS 57.1.

23 CONCLUSIONS

Based on its evaluation of the application as set forth above, the staff has determined that, upon favorable resolution of the outstanding matters described herein, it will be able to conclude that

(1) The application for facility licenses filed by the applicant dated June 26, 1980, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1.

(2) Construction of Shearon Harris Units 1 and 2 has proceeded, and there is reasonable assurance that it will be substantially completed, in conformity with Construction Permits Nos. CPPR-158 and 159, the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(4) There is reasonable assurance (a) that the activities authorized by the operating licenses can be conducted without endangering the health and safety of the public and (b) that such activities will be conducted in compliance with regulations of the Commission set forth in 10 CFR Chapter 1.

(5) The applicant is technically qualified to engage in the activities authorized by the licenses, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1.

(6) The issuance of these licenses will not be inimical to the common defense and security or to the health and safety of the public.

Before operating licenses are issued to the applicant for operation of the Shearon Harris Units 1 and 2, the units must be completed in conformity with the provisional construction permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the Commission before licenses are issued.

Furthermore, before operating licenses are issued, the applicant will be required to satisfy the applicable provisions of 10 CFR 140.

Laufer Exhibit 4

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December 21, 1983 822-1090

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In the Matter of
Carolina Power & Light Company and North
Carolina Eastern Municipal Power Agency
(Shearon Harris Nuclear Power Plant, Units 1 and 2)
Docket Nos. 50-400 and 50-401 OL

Administrative Judges Kelley, Bright and Carpenter:

The Board of Directors of Carolina Power & Light Company approved today the cancellation of Unit 2 of the Shearon Harris Nuclear Power Plant. The schedule for Unit 1 remains unchanged. In our opinion, the cancellation will not significantly change safety or environmental analyses, although certain structural modifications of the sort required as a result of the cancellation of Units 3 and 4 will be necessary. Applicants will file an appropriate amendment to their operating license application in the near future.

Respectfully submitted,

Thomas A Baxter

Thomas A. Baxter
Counsel for Applicants

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Q PDR

TAB:jah
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DS03

Laufer Exhibit 5

January 12, 1987

Docket No. 50-400

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

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

Shearon Harris

cc:

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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

Laufer Exhibit 6

Written data, views or comments for consideration by the committee may be submitted, preferably with 20 copies, to Joanne Goodell at the address provided below. Any such submissions received prior to the meeting will be provided to the members of the Committee and will be included in the record of the meeting. Because of the need to cover a wide variety of subjects in a period of time, there is usually insufficient time on the agenda for members of the public to address the committee orally. However, any such requests will be considered by the Chair who will determine whether or not time permits. Any request to make an oral presentation should state the amount of time desired, the capacity in which the person would appear, and a brief outline of the content of the presentation. Individuals with disabilities who need special accommodations should contact Theresa Berry (phone: 202–693–1999; FAX: 202–693–1641) one week before the meeting.

An official record of the meeting will be available for public inspection in the OSHA Technical Data Center (TDC) located in Room N2625 of the Department of the Labor Building (202–693–2350). For additional information contact: Joanne Goodell, Occupational Safety and Health Administration (OSHA); Room N–3641, 200 Constitution Avenue NW, Washington, D.C., 20210 (phone: 202–693–2400; FAX: 202–693–1641; e-mail joanne.goodell@osha-no.osha.gov; or at www.osha.gov).

Signed at Washington, D.C., this 7th day of January, 1999.

Charles N. Jeffress,

Assistant Secretary of Labor for Occupational Safety and Health.

[FR Doc. 99–744 Filed 1–12–99; 8:45 am]

BILLING CODE 4510–26–M

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice 99–013]

NASA Advisory Council, Minority Business Resource Advisory Committee; Meeting

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of meeting.

SUMMARY: In accordance with the Federal Advisory Committee Act, Pub. L. 92–463, as amended, the National Aeronautics and Space Administration announces a forthcoming meeting of the NASA Advisory Council, Minority Business Resource Advisory Committee.

DATES: Wednesday, January 27, 1999, 9:00 a.m. to 4:00 p.m. and Thursday, January 28, 1999, 9:00 a.m. to noon.

ADDRESSES: National Aeronautics and Space Administration, Lyndon B. Johnson Space Center, Building 1, Room 820, Houston, TX 77058–3696.

FOR FURTHER INFORMATION CONTACT: Mr. Ralph C. Thomas III, Code K, National Aeronautics and Space Administration, Washington, DC 20546, (202) 358–2088.

SUPPLEMENTARY INFORMATION: The meeting will be open to the public up to the seating capacity of the room. The agenda for the meeting is as follows:

—MBRAC Subpanel Reports
—Status of MBRAC Recommendations
—Special Issues
—Action Items
—Call to Order
—Reading of Minutes
—Agency Small Disadvantaged Business (SDB) Program
—Report of Chair
—Public Comment
—Center Directorate Reports
—Report on NASA FY 98 SDB Accomplishments

It is imperative that the meeting be held on these dates to accommodate the scheduling priorities of the key participants. Visitors will be requested to sign a visitors' register.

Dated: January 7, 1999.

Matthew M. Crouch,

Advisory Committee Management Officer, National Aeronautics and Space Administration.

[FR Doc. 99–741 Filed 1–12–99; 8:45 am]

BILLING CODE 7510–01–P

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice 99–012]

Notice of Prospective Patent License

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of prospective patent license.

SUMMARY: NASA hereby gives notice that Benick Brands, Inc., of Glastonsbury, Connecticut, has applied for an exclusive license to practice the inventions described and claimed in U.S. Patent No. 5,772,912, entitled "Environmentally Friendly Anti-Icing Fluid," and in NASA Case No. ARC–12069–9GE, entitled "Anti-Icing Fluid or Deicing Fluid." Both inventions are assigned to the United States of America as represented by the Administrator of the National Aeronautics and Space Administration. Written objections to the prospective grant should be sent to NASA Ames Research Center.

DATES: Responses to this notice should be received by March 15, 1999.

FOR FURTHER INFORMATION CONTACT: Kathleen Dal Bon, Patent Counsel, NASA Ames Research Center, Mail Stop 202A–3, Moffett Field, CA 94035–1000; telephone (650) 604–5104.

Dated: January 7, 1999.

Edward A. Frankle,

General Counsel.

[FR Doc. 99–742 Filed 1–12–99; 8:45 am]

BILLING CODE 7510–01–P

NATIONAL BIPARTISAN COMMISSION ON THE FUTURE OF MEDICARE

Public Meeting

The National Bipartisan Commission on the Future of Medicare will hold a public meeting on Tuesday, January 26, 1999 at the Cannon House Office Building, Cannon Caucus Room 340, Washington, DC. Please check the Commission's web site for additional information: *http://Medicare.Commission.Gov*

Tuesday, January 26, 1999, 9:00 a.m.

Tentative Agenda

Members of the Commission to discuss options to reform the Medicare program.

If you have any questions, please contact the Bipartisan Medicare Commission, ph: 202–252–3380.

I hereby authorize publication of the Medicare Commission meetings in the **Federal Register.**

Julie Hasler,

Office Manager, National Bipartisan Medicare Commission.

[FR Doc. 99–681 Filed 1–12–99; 8:45 am]

BILLING CODE 1132–00–M

UNITED STATES NUCLEAR REGULATORY COMMISSION

[Docket No. 50–400]

Carolina Power & Light; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF–63 issued to Carolina Power & Light (CP&L or the licensee) for operation of the Shearon Harris Nuclear Power Plant located in Wake and Chatham Counties, North Carolina.

The proposed amendment would support a modification to the plant to increase the spent fuel storage capacity by adding rack modules to spent fuel pools (SFPs) "C" and "D" and placing the pools in service. In order to activate the pools, CP&L requests that the NRC review and approve the following:

i. Revised Technical Specification 5.6 to identify PWR burnup restrictions, BWR enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D.'

ii. 10 CFR 50.55a Alternative Plan to demonstrate acceptable level of quality and safety in the completion of the component cooling water (CCW) and SFP 'C' and 'D' cooling and cleanup system piping.

The cooling system for SFPs 'C' and 'D' cannot be N stamped in accordance with ASME Section III since some installation records are not available, a partial turnover was not performed when construction was halted following the cancellation of Unit 2 and CP&L's N certificate program was discontinued following completion of Unit 1.

iii. Unreviewed safety question for additional heat load on the CCW system. The acceptability of the 1.0 MBtu/hr heat load from SFPs 'C' and 'D' was demonstrated by the use of thermal-hydraulic analyses of the CCW system under various operating scenarios. The dynamic modeling used in the thermal-hydraulic analyses identified a decrease in the minimum required CCW system flow rate to the residual heat removal heat exchangers. This change has not been previously reviewed by the NRC and is deemed to constitute an unreviewed safety question.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

In the analysis of the safety issues concerning the expanded pool storage capacity within Harris' Fuel Handling Building, the following previously postulated accident scenarios have been considered:

a. A spent fuel assembly drop in a Spent Fuel Pool.

b. Loss of Spent Fuel Pool cooling flow.

c. A seismic event.

d. Misloaded fuel assembly.

The probability that any of the accidents in the above list can occur is not significantly increased by the activity itself. The probabilities of a seismic event or loss of Spent Fuel Pool cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misloadings are primarily influenced by the methods used to lift and move these loads. The method of handling loads during normal plant operations is not significantly changed, since the same equipment (i.e., Spent Fuel Handling Machine and tools) and procedures as those in current use in pools 'A' and 'B' will be used in pools 'C' and 'D'. Since the methods used to move loads during normal operations remain nearly the same as those used previously, there is no significant increase in the probability of an accident. Current shipping activities at the Harris Nuclear Plant will continue as previously licensed. The consequences of an accident involving shipping activities [are] not changed and there is no significant increase in the probability of an accident.

During rack installation, all work in the pool area will be controlled and performed in strict accordance with specific written procedures. Any movement of fuel assemblies which is required to be performed to support this activity (e.g., installation of racks) will be performed in the same manner as during normal refueling operations.

Accordingly, the proposed activity does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Spent Fuel Pool have been re-evaluated for the proposed change. The results show that such the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, K_{eff} less than or equal to 0.95, will be maintained. The structural damage to the Fuel Handling Building, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed activity from those considered previously, the structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will not be increased from those previously considered, since the pertinent fuel parameters remain unchanged. These dose levels remain "well within" the levels required by 10 CFR 100, paragraph 11, as defined in Section 15.7.4.II.1 of the Standard Review Plan. Thus, the results of the postulated fuel drop accidents remain acceptable and do not represent a significant increase in consequences from any of the same previously evaluated accidents that have been reviewed and found acceptable by the NRC.

The consequences of a loss of Spent Fuel Pool cooling have been evaluated and found to have no increase. The concern with this accident is a reduction of Spent Fuel Pool water inventory from bulk pool boiling resulting in uncovering fuel assemblies. This situation would lead to fuel failure and subsequent significant increase in offsite dose. Loss of spent fuel pool cooling at Harris is mitigated in the usual manner by ensuring that a sufficient time lapse exists between the loss of forced cooling and uncovering fuel. This period of time is compared against a reasonable period to re-establish cooling or supply an alternative water source. Evaluation of this accident usually includes determination of a time to boil, which in the case of pools 'C' and 'D' is in excess of 13 hours based on a consideration of end of plant life heat loads. This evaluation neglects any possible cooling from the connection to pools 'A' and 'B' through the transfer canal. The 13 hour period is much shorter than the onset of any significant increase in offsite dose, since once boiling begins it would have to continue unchecked until the pool surface was lowered to the point of exposing active fuel. The time to boil represents the onset of loss of pool water inventory and is commonly used as a gauge for establishing the comparison of consequences before and after a refueling project. The heatup rate in the Spent Fuel Pool is a nearly linear function of the fuel decay heat load. Subsequent to the proposed changes, the fuel decay heat load will increase because of the increase in the number assemblies from those considered from Pools 'A' and 'B' alone. The methodology used in the thermal-hydraulic analysis determined the maximum fuel decay heat loads. In the unlikely event that pool cooling is lost to pools 'C' and 'D', sufficient time will still be available for the operators to provide alternate means of cooling before the onset of pool boiling. Therefore, the proposed change represents no increase in the consequences of loss of pool cooling.

The consequences of a design basis seismic event are not increased. The consequences of this accident are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The new racks have been analyzed in their new configuration and found safe during seismic motion. The fuel stored in these racks has been determined to remain intact and the racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the appropriate combinations of dead weight, thermal, and seismic loads. The Fuel

Handling Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, storage array, and pool moderator/coolant. Thus, the consequences of a seismic event are not increased.

Fuel misloading and mislocation accidents were previously credible occurrences, since fuel could be placed at an unintended storage location or could have been lowered outside and adjacent to a storage rack in Pools 'A' or 'B'. However, neither of these two scenarios previously represented any concern because of the flux trap style of the rack designs in these two pools. Similar procedures, equipment and methods of fuel movement will be used for Pools 'C' and 'D' as those used previously for Pools 'A' and 'B'. Therefore, the proposed activity does not represent any increase in the probability of occurrence. The proposed non-flux trap design racks for Pools 'C' and 'D' require administrative controls to ensure that fuel assemblies meet effective enrichment criteria prior to storage. Under these conditions, misloading of a fuel assembly by placement in an unintended storage cell has no significant consequences. Therefore, the only remaining potential mislocation of a fuel assembly is for an assembly to be lowered outside of and directly adjacent to a storage rack. This accident occurring in Pools 'C' or 'D' has been analyzed for the worst possible storage configuration subsequent to the proposed activity and it has been shown that the consequences remain acceptable with respect to the same criteria used previously. Thus, there is no increase in consequences for fuel mislocation or misloading.

Therefore it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

To assess the possibility of new or different kind of accidents, a list of the important parameters required to ensure safe fuel storage was established. Safe fuel storage is defined here as providing an environment, which would not present any significant threats to workers or the general public (i.e., meeting the requirements of 10 CFR 100 and 10 CFR 20). Any new events, which would modify these parameters sufficiently to place them outside of the boundaries analyzed for normal conditions and/or outside of the boundaries previously considered for accidents would be considered to create the possibility of a new or different accident. The criticality and radiological safety evaluations were reviewed to establish the list of important parameters. The fuel configuration and the existence of the moderator/coolant were identified as the only two parameters, which were important to safe fuel storage. Significant modification of these two parameters represents the only possibility of an unsafe storage condition. Once the two important parameters were established, an additional step was taken to determine what events (which were not previously considered) could result in changes to the storage configuration or moderator/coolant presence during or subsequent to the proposed changes.

This process was adopted to ensure that the possibility of any new or different accident scenario or event would be identified. Due to the proposed activity, an accidental drop of a rack module during construction activity in the pool was considered as the only event which might represent a new or different kind of accident.

A construction accident resulting in a rack drop is an unlikely event. The proposed activity will utilize the defense-in-depth approach for these heavy loads. The defense-in-depth approach is intended to meet the requirements of NUREG–0612 and preclude the possibility of a rack drop. All movements of heavy loads over the pool will comply with the applicable administrative controls and guidelines (i.e. plant procedures, NUREG–0612, etc.). A temporary hoist and rack lifting rig will be introduced to lift and suspend the racks from the bridge of the Auxiliary Crane. These items have been designed in accordance with the requirements of NUREG–0612 and ANSI N14.6 and will be similar to those used recently to install storage rack modules in Pool 'B'.

The postulated rack drop event is commonly referred to as a "heavy load drop" over the pools. Heavy loads will not be allowed to travel over any racks containing fuel assemblies. The danger represented by this event is that the racks will drop to the pool floor and the pool structure will be compromised leading to loss of moderator/coolant, which is one of the two important parameters identified above. Although the analysis of this event has been performed and shown to be acceptable, the question of a new or different type of event is answered by determining whether heavy load drops over the pool have been considered previously. As stated above, heavy loads (storage rack modules) were recently installed in Pool 'B' using similar methods. Therefore, the rack drop does not represent a new or different kind of accident.

The proposed change does not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents. The proposed change does not affect any of the important parameters required to ensure safe fuel storage. Therefore, the potential for a new or previously unanalyzed accident is not created.

3. Involve a significant reduction in the margin of safety.

The function of the Spent Fuel Pool is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with Pools 'C' and 'D'.

CP&L has Addressed the Safety Issues Related to the Expanded Pool Storage Capacity in the Following Areas:

1. Material, mechanical and structural considerations. The mechanical, material, and structural designs of the new racks have been reviewed in accordance with the applicable provisions of the NRC Guidance entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications". The rack materials used are compatible with the spent fuel assemblies and the Spent Fuel Pool environment. The design of the new racks preserves the proper margin of safety during normal and abnormal loads. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

2. Nuclear Criticality

The methodology used in the criticality analysis of the expanded Spent Fuel Pool meets the appropriate NRC guidelines and the ANSI standards (GDC 62, NUREG 0800, Section 9.1.2, the OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Reg. Guide 1.13, and ANSI/ANS 8.17). The margin of safety for subcriticality is maintained by having the neutron multiplication factor equal to, or less than, 0.95 under all accident conditions, including uncertainties. This criterion is the same as that used previously to establish criticality safety evaluation acceptance and remains satisfied for all analyzed accidents.

3. Thermal-hydraulic and Pool Cooling

The thermal-hydraulic and cooling evaluation of the pools demonstrated that the pools can be maintained below the specified thermal limits under the conditions of the maximum heat load and during all credible accident sequences and seismic events. The pool temperature will not exceed 137°F during the highest heat load conditions. The maximum local water temperature in the hot channel will remain below the boiling point. The fuel will not undergo any significant heat up after an accidental drop of a fuel assembly on top of the rack blocking the flow path. A loss of cooling to the pool will allow sufficient time (>13 hours) for the operators to intervene and line up alternate cooling paths and the means of inventory make-up before the onset of pool boiling. The thermal limits specified for the evaluations performed to support the proposed activity are the same as those that were used in the previous evaluations. It has also been demonstrated that adequate margin exists in the Unit 1 CCW system to support near term operation of the pools subject to the requirements of the proposed changes to the Technical Specifications.

Based on the preceding discussion it is concluded that this activity does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By February 12, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

The Commission hereby provides such notice that this is a proceeding on an application for a license amendment falling within the scope of section 134 of the Nuclear Waste Policy Act of 1982 (NWPA), 42 U.S.C. 10154. Under section 134 of the NWPA, the

Commission, at the request of any party to the proceeding, must use hybrid hearing procedures with respect to "any matter which the Commission determines to be in controversy among the parties."

The hybrid procedures in section 134 provide for oral argument on matters in controversy, preceded by discovery under the Commission's rules and the designation, following argument of only those factual issues that involve a genuine and substantial dispute, together with any remaining questions of law, to be resolved in an adjudicatory hearing. Actual adjudicatory hearings are to be held on only those issues found to meet the criteria of section 134 and set for hearing after oral argument.

The Commission's rules implementing section 134 of the NWPA are found in 10 CFR Part 2, Subpart K, "Hybrid Hearing Procedures for Expansion of Spent Fuel Storage Capacity at Civilian Nuclear Power Reactors" (published at 50 FR 41662 dated October 15, 1985). Under those rules, any party to the proceeding may invoke the hybrid hearing procedures by filing with the presiding officer a written request for oral argument under 10 CFR 2.1109. To be timely, the request must be filed within ten (10) days of an order granting a request for hearing or petition to intervene. The presiding officer must grant a timely request for oral argument. The presiding officer may grant an untimely request for oral argument only upon a showing of good cause by the requesting party for the failure to file on time and after providing the other parties an opportunity to respond to the untimely request. If the presiding officer grants a request for oral argument, any hearing held on the application must be conducted in accordance with the hybrid hearing procedures. In essence, those procedures limit the time available for discovery and require that an oral argument be held to determine whether any contentions must be resolved in an adjudicatory hearing. If no party to the proceeding timely requests oral argument, and if all untimely requests for oral argument are denied, then the usual procedures in 10 CFR Part 2, Subpart G apply.

For further details with respect to this action, see the application for amendment dated December 23, 1998, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Dated at Rockville, Maryland, this 7th day of January 1999.

For the Nuclear Regulatory Commission.

Scott Flanders,

Project Manager, Project Directorate II–3, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 99–758 Filed 1–12–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket Number 40–8102]

Exxon Coal and Minerals Company

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of receipt of Exxon Coal and Minerals Company's application for establishing alternate concentration limits in source material license SUA–1139 for the Highland Uranium Mill in Converse County, Wyoming; notice of opportunity for a hearing.

SUMMARY: Notice is hereby given that the U.S. Nuclear Regulatory Commission (NRC) has received, by letter dated December 18, 1998, an application from Exxon Coal and Minerals Company (ECMC) to establish Alternate Concentration Limits (ACLs) for nickel, radium (Ra 226+228), and natural uranium (UNAT); and amend accordingly Source Material License No. SUA–1139 for the Highland uranium mill.

FOR FURTHER INFORMATION CONTACT: Mohammad W. Haque, Uranium Recovery Branch, Division of Waste Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415–6640.

SUPPLEMENTARY INFORMATION: ECMC's application to amend Source Material License SUA–1139, which describes the proposed change and the reasons for the request, is being made available for public inspection at NRC's Public Document Room at 2120 L Street, N.W. (Lower Level), Washington, DC 20555.

The NRC hereby provides notice of an opportunity for a hearing on the license amendment under the provisions of 10 CFR Part 2, Subpart L, "Informal Hearing Procedures for Adjudications in Materials and Operator Licensing Proceedings." Pursuant to § 2.1205(a), any person whose interest may be affected by this proceeding may file a request for a hearing. In accordance with § 2.1205(c), a request for hearing must be filed within 30 days of the publication of this notice in the **Federal Register**. The request for a hearing must be filed with the Office of the Secretary, either:

(1) By delivery to the Docketing and Service Branch of the Office of the Secretary at One White Flint North, 11555 Rockville Pike, Rockville, MD 20852; or

(2) By mail or telegram addressed to the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Service Branch.

In accordance with 10 CFR 2.1205(e), each request for a hearing must also be served, by delivering it personally, or by mail, to:

(1) The applicant, Exxon Coal and Minerals Company, P.O. Box 1314, Houston, Texas 77251–1314, Attention: David Range; and

(2) The NRC staff, by delivery to the Executive Director for Operations, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, or by mail addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

In addition to meeting other applicable requirements of 10 CFR Part 2 of NRC's regulations, a request for a hearing filed by a person other than an applicant must describe in detail:

(1) The interest of the requestor in the proceeding;

(2) How that interest may be affected by the results of the proceeding, including the reasons why the requestor should be permitted a hearing, with particular reference to the factors set out in § 2.1205(g);

(3) The requestor's areas of concern about the licensing activity that is the subject matter of the proceeding; and

(4) The circumstances establishing that the request for a hearing is timely in accordance with § 2.1205(c).

The request must also set forth the specific aspect or aspects of the subject matter of the proceeding as to which petitioner wishes a hearing.

Dated at Rockville, Maryland, this 5th day of January 1999.

N. King Stablein,

Acting Chief, Uranium Recovery Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 99–756 Filed 1–12–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 72–09]

Public Service Company of Colorado, Fort St. Vrain Independent Spent Fuel Storage Installation; Exemption

I

Public Service Company of Colorado (PSCo, the licensee) holds Materials

Laufer Exhibit 7

[4310-84]

[Wyoming 62187]

WYOMING

Application

JANUARY 24, 1978.

Notice is hereby given that pursuant to Section 28 of the Mineral Leasing Act of 1920, as amended (30 U.S.C. 185), the Colorado Interstate Gas Co. of Colorado Springs, Colo. has filed an application for a right-of-way to construct an 8⅝ inch O.D. pipeline for the purpose of transporting natural gas across the following described public lands:

SIXTH PRINCIPAL MERIDIAN, WYOMING

T. 38 N., R. 90 W.,
Sec. 12, NW¼NW¼.

The pipeline will transport natural gas from wells located in the NW¼ of Section 7 and the NE¼ of Section 8, T. 38 N., R. 89 W., to an existing natural gas pipeline within NW¼ of Section 12, T. 38 N., R. 90 W., Fremont County, Wyo.

The purpose of this notice is to inform the public that the Bureau will be proceeding with consideration of whether the application should be approved, and, if so, under what terms and conditions.

Interested persons desiring to express their views should do so promptly. Persons submitting comments should include their name and address and send them to the District Manager, Bureau of Land Management, 1300 Third Street, P.O. Box 670, Rawlins, Wyo. 82301.

HAROLD G. STINCHCOMB,
Chief, Branch of Lands and Minerals Operations.

[FR Doc. 78-2862 Filed 2-1-78; 8:45 am]

[4310-84]

[Wyoming 62238]

WYOMING

Application

JANUARY 25, 1978.

Notice is hereby given that pursuant to Section 28 of the Mineral Leasing Act of 1920, as amended (30 U.S.C. 185), the Cities Service Gas Co. of Oklahoma City, Okla. filed an application for a right-of-way to construct a 4½ inch pipeline and install anodes for the purpose of transporting natural gas across the following described public lands:

SIXTH PRINCIPAL MERIDIAN, WYOMING

T. 21 N., R. 92 W.,
Sec. 2, S½S½;
Sec. 9, S½NE¼, SE¼NW¼, N½SW¼;
Sec. 10, N½NE¼, NE¼NW¼, S½NW¼.

The pipeline will transport natural gas from the Champlin 536 wellhead located in the SW¼ of section 1, T. 21 N., R. 92 W., in a southwesterly direction to connect with Cities Service Gas Co.'s proposed gathering line located in the NW¼SW¼ of section 9, T. 21 N., R. 92 W., Sweetwater County, Wyo.

The purpose of this notice is to inform the public that the Bureau will be proceeding with consideration of whether the application should be approved, and if so, under what terms and conditions.

Interested persons desiring to express their views should do so promptly. Persons submitting comments should include their name and address and send them to the District Manager, Bureau of Land Management, 1300 Third Street, P.O. Box 670, Rawlins, Wyo. 82301.

HAROLD G. STINCHCOMB,
Chief, Branch of Lands and Minerals Operations.

[FR Doc. 78-2863 Filed 2-1-78; 8:45 am]

[4310-31]

Geological Survey

UTAH

Known Geothermal Resources Area

Pursuant to the authority vested in the Secretary of the Interior by Sec. 21(a) of the Geothermal Steam Act of 1970 (84 Stat. 1566, 1572; 30 U.S.C. 1020), and delegations of authority in 220 Departmental Manual 4.1 H, Geological Survey Manual 220.2.3, and Conservation Division Supplement (Geological Survey Manual 220.2.1. G, the following described lands are hereby defined as a known geothermal resources area, effective April 1, 1977:

(44) UTAH

Meadow-Hatton Known Geothermal Resources Area

SALT LAKE MERIDIAN

T. 22 S., R. 6 W.,
Secs. 1, 14, and 22.
The area described aggregates 1,927 acres, more or less.

Dated: December 23, 1977.

GEORGE H. HORN,
Conservation Manager, Central Region.

[FR Doc. 78-2864 Filed 2-1-78; 8:45 am]

[1410-03]

NATIONAL COMMISSION ON NEW TECHNOLOGICAL USES OF COPYRIGHTED WORKS

MEETING

The National Commission on New Technological Uses of Copyrighted Works (CONTU) will hold its twentieth meeting on Thursday, February 16 and Friday, February 17, 1978, in New York City.

The meetings, which will be held at the Time-Life Building, eighth floor anteroom, Rockefeller Center, 1271 Avenue of the Americas, will convene at 10 a.m. both days.

The Commission will discuss the reports of its Subcommittees. The meetings are open to the public.

ARTHUR J. LEVINE,
Executive Director, National Commission on New Technological Uses of Copyrighted Works.

[FR Doc. 78-2865 Filed 2-1-78; 8:45 am]

[7590-01]

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50-401, 50-402 and 50-403]

CAROLINA POWER & LIGHT CO.

Availability of Initial Decision of the Atomic Safety and Licensing Board for the Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4 and Issuance of Construction Permits

Pursuant to the National Environmental Policy Act of 1969 and the United States Nuclear Regulatory Commission's regulations in § 51.52(b)(3) of 10 CFR Part 51, notice is hereby given that an Initial Decision dated January 23, 1978, by the Atomic Safety and Licensing Board in the above captioned proceeding authorizing issuance of construction permits to the Carolina Power & Light Company for construction of the Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4 located about 20 miles southwest of Raleigh, N.C., in Wake and Chatham Counties, is available for inspection by the public in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. and in the Wake County Library, 104 Fayetteville Street, Raleigh, N.C. 27601.

The Initial Decision is subject to review by an Atomic Safety and Licensing Appeal Board prior to its becoming final. Any decision or action taken by an Atomic Safety and Licensing Appeal Board in connection with the Initial Decision may be reviewed by the Commission.

The Initial Decision is also being made available at the Office of Intergovernmental Relations, 116 West Jones Street, Raleigh, N.C. 27603 and at the Triangle J. Council of Governments, P.O. Box 12276, Research Triangle Park, Durham, N.C. 27709.

Based upon the record developed in the public hearing in the above captioned matter, the Initial Decision modified in certain respects the contents of the Revised Final Environmental Statement relating to the construction of the Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4, prepared by the Commission's Office of Nuclear Reactor Regulation. Pursuant to the provisions of § 51.52(b)(3) of 10 CFR Part 51, the Revised Final Environmental Statement is deemed modified to the extent that the findings and conclusions relating to environmental matters contained in the Initial Decision are different from those contained in the Revised Final Environmental Statement, dated March 1974. As required by § 51.52(b)(3) of 10 CFR Part 51, a copy of the Initial Decision, which modifies the Revised Final Environmental Statement, has been transmitted to the Council on Environmental Quality and distributed to the Environmental Protection Agency and other interested agencies and persons in accordance with § 51.26(c) of 10 CFR Part 51.

Pursuant to the above mentioned Initial Decision, the Commission has issued Construction Permits Nos. CPPR-158, CPPR-159, CPPR-160 and CPPR-161 to the Carolina Power & Light Company for construction of four (4) pressurized nuclear reactors, known as the Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4, each designed to operate at a core power level of 2775 megawatts thermal with a net electrical output of approximately 900 megawatts.

The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the construction permits. The application for the construction permits complies with the standards and requirements of the Act and the Commission's rules and regulations.

The Construction permits are effective as of their date of issuance. The earliest date for the completion of Unit 1 is June 1, 1983, the latest date for completion is June 1, 1984. The earliest date for the completion of Unit 2 is June 1, 1985, and the latest date for completion is June 1, 1986. The earliest date for the completion of Unit 3 is June 1, 1989, and the latest date for completion is June 1, 1990. The earliest date for the completion of Unit 4 is June 1, 1987, and the latest date for completion is June 1, 1988. Each permit shall expire on the latest date for completion of the facility.

In addition to the Initial Decision, copies of: (1) Construction Permit Nos. CPPR-158, CPPR-159, CPPR-160 and CPPR-161; (2) the report of the Advisory Committee on Reactor Safeguards dated August 19, 1977; (3) the Office of Nuclear Reactor Regulation's Safety Evaluation Report, dated December 22, 1972, and its four (4) supplements, dated April 27, 1973, May 6, 1974, July 30, 1977, and September 20, 1977, respectively; (4) the Preliminary Safety Analysis Report and amendments thereto; (5) the applicant's Revised Environmental Report dated March 16, 1972 and supplements thereto; (6) the Draft Environmental Statement, dated November 1972; and (7) the Revised Final Environmental Statement, dated March 1974, are available for public inspection at the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. and at the Wake County Library, 104 Fayeteville Street, Raleigh, N.C. 27601. Copies of the Initial Decision and the Final Environmental Statement are also available for public inspection at the above clearinghouses in North Carolina.

Single copies of the Initial Decision by the Atomic Safety and Licensing Board, the construction permits, the Safety Evaluation Report and supplements thereto, and the Revised Final Environemental Statement may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Md. this 27th day of January 1978.

For the Nuclear Regulatory Commission.

OLAN D. PARR,
Chief, Light Water Reactors Branch No. 3, Division of Project Management.

[FR Doc. 78-2831 Filed 2-1-78; 8:45 am]

[7590-01]

[Docket No. PRM 50-19]

CONNECTICUT CITIZEN ACTION GROUP, ET AL.

Denial of Petition for Rule Making With Regard To Stationing a Full-Time Federal Employee in the Reactor's Control Room

Louis J. Sirico, Jr., Esquire, filed with the Nuclear Regulatory Commission a petition for rulemaking dated January 21, 1977, on behalf of the Connecticut Citizen Action Group, the Public Interest Research Group, Free Environment, the Iowa Public Interest Research Group, Citizens United for Responsible Energy, Iowa Federation of Women's Clubs, and the Good News General Store Cooperative, Requesting the Commission to amend its regulations in 10 CFR Part 50, "Licensing of Production and Utilization Facilities".

The petitioners requested the Commission to amend 10 CFR Part 50 to require that:

1. Nuclear reactors be located below ground level;
2. Nuclear reactors be housed in sealed buildings in which permanent heavy vacuums are maintained; and
3. A full-time Federal employee, with full authority to shut down the plant in case of any operational abnormality, always be present in a reactor's control room.

The petitioners stated that the requested amendments would conform to reactor safety proposals discussed in the recent presidential campaign and that, as an interim measure, the proposal set out in the petition would significantly increase the protection offered to the public.

A notice of filing of petition for rulemaking was published in the FEDERAL REGISTER on March 10, 1977 (42 FR 13365). The comment period expired May 9, 1977. Nine interested parties have submitted comments regarding part 3 of the petition (full-time Federal employee in the control room). None supported the petition.

It was noted in the March 10, 1977, FEDERAL REGISTER notice that in November 1976, the Commission staff undertook an assessment of the feasibility of instituting a program of full-time inspection at operating reactors. It was also stated that this assessment which contains an analysis of various alternatives for inspecting operating reactors, including as one of the alternatives the placement of a full-time NRC employee in the control room of each nuclear powerplant, was expected to be completed by mid-1977. This study, which was broadened to also include the inspection program at reactor sites under construction and during the testing phase prior to commercial operation, has been completed. The study[1] concluded that the current inspection program can be improved by increasing NRC inspector presence on site and by increasing capabilities to perform independent verification.

After consideration of the results of this study of alternative inspection programs and their associated costs and benefits, the Commission decided to alter its inspection program. Based on this Commission decision and on the OMB-approved budget, an NRC resident inspector will be assigned to each site where there is an operating reactor and at selected construction sites. The NRC Regional Office will provide periodic technical support. There will be increased capability for independent verification of licensee actions.

The Commission has decided to deny the portion of the petiton for rulemaking concerning stationing a full-time Federal employee in the reactor's control room with full authority to shut down the plant in case of any operational abnormality on the grounds that the current inspection program, which will be improved by increasing onsite presence and by increasing capabilities to perform independent ver-

[1] Copies of the report entitled, "Study Report—NRC Inspection alternatives" are available at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C. 20555.