December 21, 2000

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: EXPANSION OF SPENT FUEL STORAGE CAPACITY (TAC NO. MA4432)

Dear Mr. Scarola:

The Nuclear Regulatory Commission has issued Amendment No. 103 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No. 1 (HNP), in response to your request dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000. This amendment supports a modification to HNP 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. Specifically, this amendment: 1) revises Technical Specification 5.6 to identify pressurized water reactor fuel burnup restrictions, boiling water reactor fuel 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; and 2) resolves an unreviewed safety question for additional heat load on the component cooling water (CCW) system.

In addition, your December 23, 1998, submittal included 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 SFPs C and D cooling and cleanup system piping. The staff has completed its review of your alternative plan, as documented in sections 3.7 and 3.8 of the enclosed Safety Evaluation. The staff has determined that the proposed alternative for the missing weld documentation for SFPs C and D piping would provide an acceptable level of quality and safety. Accordingly, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

A copy of the Notice of Issuance and Final Determination of No Significant Hazards Consideration, which is being sent to the Office of the Federal Register for publication, is also enclosed.

Sincerely,

/RA/

Richard J. Laufer, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 103 to NPF-63
2. Safety Evaluation
3. Notice

cc w/enclosures:

See next page

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cc w/enclosures:
See next page

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** no major changes to SE

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NAME	RLaufer	EDunnington	JWermiel*	KManoly **	MReinhart **	TEssig **	ESullivan **
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AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

PUBLIC
PDII Reading
OGC
G. Hill (2)
ACRS
B. Bonser, RII
H. Berkow
W. Beckner

cc: Harris Service List

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 103
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment 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;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(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. 103, 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.

Additionally, the license is amended to authorize revision of the Final Safety Analysis Report (FSAR) to reflect the change in the minimum specified component cooling water system flow to the residual heat removal system heat exchanger from 5600 gpm to 5200 gpm. The licensee shall make this update to the FSAR, as authorized by this amendment, in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 21, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 103

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
xvii	xvii
5-6	5-6
5-7	5-7
----	5-7a
----	5-7b

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

1.1 License Amendment Request

By letter dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000 (references 1 - 11), Carolina Power and Light Company (CP&L, the licensee), requested a change to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed amendment would support a modification to HNP 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. Specifically, the proposed action consists of: 1) a revision to TS 5.6 to identify pressurized water reactor (PWR) fuel burnup restrictions, boiling water reactor (BWR) fuel 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; 2) 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 component cooling water (CCW) and SFPs C and D cooling and cleanup system piping; and 3) an unreviewed safety question (USQ) for additional heat load on the CCW system.

The supplemental submittals dated March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, provided clarifying information that did not change the scope of the original *Federal Register* notice published on January 13, 1999 (64 FR 2237).

1.2 Atomic Safety and Licensing Board Hearing

In response to the January 13, 1999, *Federal Register* Notice, on February 12, 1999, the Board of Commissioners of Orange County North Carolina (BCOC) filed a Request for a Hearing and Petition to Intervene in the license amendment proceeding (ref. 23). Subsequently, on April 5, 1999, BCOC filed eight contentions (three technical and five environmental) (ref. 24). The NRC staff and the licensee provided responses to BCOC's contentions on May 5, 1999 (ref. 25, 26). The Atomic Safety and Licensing Board (ASLB) held a pre-hearing conference in Chapel Hill, North Carolina on May 13, 1999 (ref. 27).

On July 12, 1999, the ASLB issued its Ruling on Standing and Contentions (ref. 28). In its ruling, the ASLB stated that BCOC had standing and had submitted two admissible contentions. The two contentions related to (1) whether General Design Criterion (GDC) 62 allows the use of administrative controls to prevent criticality (TC-2); and (2) the adequacy of the licensee's proposed alternative plan for the cooling system piping (TC-3). On July 29, 1999, the ASLB granted CP&L's request to hold the hearing in accordance with the hybrid hearing procedures of 10 CFR Part 2, Subpart K (ref. 29).

On January 4, 2000, all parties filed written summaries (ref. 30, 31, 32) and on January 21, 2000, the ASLB heard oral arguments related to the two admitted contentions (ref. 33). On May 5, 2000, the ASLB issued a decision in favor of CP&L (ref. 34). In its decision, the ASLB concluded that "(1) there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing; and (2) contentions TC-2 and TC-3 are disposed of as being <u>resolved</u> in favor of CP&L."

On January 31, 2000, BCOC filed four late-filed environmental contentions (ref. 35), which challenged the adequacy of the staff's December 21, 1999, environmental assessment (ref. 36) related to CP&L's amendment request. On March 3, 2000, the NRC and CP&L responded to the late-filed contentions (ref. 37, 38), and on March 13, 2000, BCOC submitted its reply to the responses (ref. 39). On August 7, 2000, the ASLB issued its Ruling on Late-filed Environmental Contentions (ref. 44). In its ruling, the ASLB admitted one environmental contention (EC-6) regarding the probability of occurrence of BCOC's postulated accident scenario. On November 20, 2000, all parties filed written summaries (ref. 45, 46, 47) and on December 7, 2000, the ASLB heard oral arguments related to EC-6.

On May 22, 2000, BCOC filed a petition (ref. 40) for the Commission to review the ASLB's May 5, 2000, decision (ref. 34) in the Harris Nuclear Plant spent fuel storage case. BCOC stated in its petition that "The Commission should take review of clearly erroneous rulings in LBP-00-12 regarding criticality prevention and quality assurance issues." On June 6, 2000, the NRC and CP&L responded to BCOC's petition (ref. 41, 42). In a June 20, 2000, Memorandum and Order (ref. 43), the Commission dismissed BCOC's petition, without prejudice, on the ground that it was prematurely filed. The Commission stated that "After the Board ultimately rules on Orange County's environmental contentions and issues a final decision, Orange County may then resubmit to us its arguments that the Board erred in rejecting the merits of the two contentions concerning criticality prevention and quality assurance."

2.0 <u>BACKGROUND</u>

HNP was originally planned as a four nuclear unit site. 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 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 an SFP cooling and cleanup system to service SFPs A and B and a separate cooling and cleanup system to support SFPs C and D.

HNP Units 3 and 4 were canceled in late 1981, and HNP Unit 2 was canceled in late 1983. The FHB, all four SFPs (including liners), and the cooling and cleanup system to support SFPs A and B were completed. However, the construction on the SFP cooling and cleanup system for SFPs C and D was not completed.

The staff's Safety Evaluation Report, NUREG-1038, "Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," issued in November 1983 (ref. 12) (prior to the cancellation of HNP Unit 2), evaluates the design of all four pools SFPs and the associated cooling systems. 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, with each train consisting of a heat exchanger, strainer, and cooling pump. The cooling for the SFP heat exchangers was to be provided by the CCW system for the respective unit. The pools are designed to store both PWR and BWR fuel.

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 will be necessary to activate SFPs C and D and the associated cooling and cleanup systems. Activation of these two pools will provide storage capacity for all four CP&L nuclear units through the expiration of their current licenses.

3.0 EVALUATION

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

> a. A revision to TS 5.6 to identify PWR fuel burnup restrictions, BWR fuel 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.

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 fuel burnup and BWR fuel enrichment to ensure Keff less than or equal to 0.95. The proposed change will involve the addition of Region 2 (non-flux trap style) racks in SFPs C and D in incremental phases (campaigns), on an as-needed basis. In a fully implemented storage configuration, this modification will allow 927 PWR and 2763 BWR fuel assemblies in SFP C. Expansion of storage capacity in SFP D will occur once SFP C is substantially filled. SFP D can accommodate a maximum of 1025 PWR fuel assemblies.

Additionally, in its July 19, 2000, submittal, the licensee submitted updated TS pages to: (1) insert TS 5.6.1.a.2 (renumbered as TS 5.6.1.1.b), which had been inadvertently deleted from the pages submitted on December 23, 1998; (2) revise TS 5.3.1 to reflect the renumbering of TS 5.6.1.a.2; and (3) add Figure 5.6.1 to the TS Table of Contents.

> 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 SFPs C and D cooling and cleanup system piping.

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 American Society of Mechanical Engineers (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 provided an acceptable level of quality and safety.

c. A USQ for additional heat load on the CCW system.

As part of its preparation of the design package for the tie-in of the existing CCW system to provide cooling for SFPs C and D, CP&L prepared a 10 CFR 50.59 evaluation. The scope of the evaluation addressed 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. The evaluation considered a heat load of no more than 1.0 MBtu/hr (consistent with proposed TS 5.6) in SFPs C and D. 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 ultimate heat sink.

A reduction in the minimum specified residual heat removal (RHR) heat exchangers CCW 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 was prescribed by the new thermal-hydraulic analysis in order to maintain all thermal/hydraulic assumptions which are used in the HNP containment analysis. The licensee verified that the minimum specified ESW system flow of 8500 gpm to the CCW heat exchangers was within the capacity of the current system even considering the most limiting ESW system failure. Since the 5600 gpm RHR flow is discussed in NUREG-1038, the licensee determined that the reduction in flow to 5200 gpm was a reduction in an acceptance limit and, therefore, required NRC review.

The staff's evaluation of the licensee's proposed amendment follows.

3.1 Criticality

The licensee requested changes to TS 5.6, "Fuel Storage," to reflect the installation of additional spent fuel storage racks in pools C and D. Pools A and B have already been racked and are nearly full. Pool A contains six Region 1 type (6 x 10 cell) 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) PWR Region 1 type racks. Pool B also currently contains 17 (11 x 11 cell) BWR racks, 12 of which have been supplied by Holtec International (Holtec). Pool B is licensed to store one more (11 x 11 cell) Holtec BWR rack, which would increase the total pool storage capacity to 2946 assemblies. However, HNP is postponing installation of the last BWR rack in order to reserve the pool open area for fuel examination and repair. Therefore, the combined pool A and B storage capacity will remain as 768 PWR cells and 2057 BWR cells for a total of 2825 storage cell locations.

Under the proposed capacity expansion, storage racks would be added to the unused pools C and D on an as-needed basis. Pool C would provide storage for up to 927 PWR assemblies and 2763 BWR assemblies. Pool D would contain only PWR fuel assemblies with a maximum capacity of 1025 assemblies.

The proposed expansion would consist of the installation of maximum density spent fuel storage racks for pools C and D in phases on an as-needed basis. The new racks were designed by Holtec and are free-standing and self-supporting. The principal construction material is stainless steel. The only non-stainless material is the neutron absorber material, which is a boron carbide and aluminum composite sandwich commonly called boral. 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 11 cell), 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 and BWR storage rack styles as required. The racks in pool C will be installed as needed. 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.

The primary analysis of the reactivity effects of fuel storage in the HNP racks was performed with the CASMO-3 two-dimensional transport theory code. CASMO-3 was also used for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. The MCNP-4A Monte Carlo code was used to determine reactivity effects, to calculate the reactivity for fuel misloading outside the racks, and to determine the effect of having PWR and BWR racks adjacent to each other. MCNP-4A was also used for independent verification calculations against CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the HNP spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, these two independent methods of analysis (MCNP-4A and CASMO-3) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the HNP storage racks with a high degree of confidence.

The NRC acceptance criterion for subcriticality is that the effective multiplication factor (k-eff) in the spent fuel pool storage racks when fully flooded by unborated water shall be no greater than 0.95, including uncertainties at a 95 percent probability, 95 percent confidence level (95/95) under all conditions. The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Racks were fully loaded with the most reactive fuel authorized to be stored in the facility.

(2) Unborated pool water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures.

(3) Assumption of infinite array (no neutron leakage) of storage cells except for the assessment of peripheral effects and certain accident assessments.

(4) Neutron absorption in minor structural material is neglected (i.e., spacer grids are analytically replaced by water).

The staff concludes that appropriately conservative assumptions were made.

The Westinghouse 17x17 Standard, 17x17 Vantage 5, 15x15, and the Siemens 17x17 and 15x15 fuel assemblies were evaluated. The design basis fuel assembly used for the PWR rack criticality analyses was the Westinghouse 15 x 15 assembly with a maximum enrichment of 5.0 weight percent (w/o) U-235 since this was determined to have the highest reactivity at zero burnup as a function of burnup for an initial 5.0 w/o U-235 enrichment. For the nominal storage cell design, uncertainties due to manufacturing tolerances on boron loading, boral width, cell inner dimension, stainless steel thickness, and fuel density were included. In addition, a calculational uncertainty for burnup calculations and the effect of axial burnup distribution was included for burnup calculations. These uncertainties were appropriately determined at the 95/95 level, thus meeting the NRC acceptance criterion.

In order to store fuel with maximum initial enrichments up to 5.0 w/o U-235 in the pool C or D PWR racks, the concept of burnup reactivity equivalencing was used. This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for use in PWR fuel storage analyses. A series of reactivity calculations is performed to generate a set of enrichment versus burnup ordered pairs which yield an equivalent k-eff less than 0.95 for fuel stored in the HNP storage racks. The results are illustrated in TS Figure 5.6.1, which indicate that fuel initially enriched to 5.0 w/o U-235 must achieve a burnup of at least 41,447 MWD/MTU to be allowed storage in the PWR spent fuel racks in pools C and D.

The design basis fuel assemblies evaluated for the BWR rack criticality analyses were the General Electric GE-3, GE-4, GE-7, GE-8, GE-9, GE-10, and GE-13 BWR assemblies with a maximum planar average enrichment of 4.6 w/o U-235. The maximum planar average enrichment was assumed for all fuel rods in the assembly. The analyses included the same manufacturing tolerance uncertainties mentioned above for the PWR fuel. These uncertainties were appropriately determined at 95/95 level. In addition, the reactivity increase due to zirconium flow channel bulging was included as well as a reactivity uncertainty in the depletion calculations and an allowance for possible differences between fuel vendor calculations and those reported here.

The analysis for each BWR fuel assembly was performed at the maximum reactivity over burnup. At this point, the assembly was analytically transferred into the storage rack at a reference temperature of 4^{o}C and its k-eff in the rack geometry was determined. The same assembly was also analytically transferred into the HNP standard cold core geometry configuration, which is an infinite lattice with 6-inch spacing at a temperature of 20^{o}C without burnable absorber or control rods and no voids, and its k-inf in this cold core configuration was determined. All xenon which was present during the depletion calculations was removed for the rack and cold core analyses. The maximum calculated CASMO-3 reactivity for each BWR assembly resulted in a k-eff less than 0.95 for the proposed pool C storage racks. This meets the staff's criterion of k-eff no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

Based on these results, a BWR fuel assembly is acceptable for storage in the HNP storage racks in pool C if it has a k-inf in the standard cold core geometry, calculated at the maximum reactivity over burnup, of less than or equal to 1.32. This requirement will be incorporated into HNP TS 5.6.1.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events, such as the accidental insertion of an assembly outside and adjacent to the fuel storage rack, dropping an assembly on top of the rack, lateral rack movement during a seismic event, or the inadvertent placement of a fresh PWR assembly into a location restricted to a burned assembly as per TS Figure 5.6.1, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of soluble boron in the pool water based upon the double contingency principle which requires at least two unlikely, independent, concurrent events to occur before a nuclear criticality accident is possible. Therefore, since soluble boron is normally present in the SFP water, credit for soluble boron may be assumed in evaluating other accident conditions such as the misloading of fresh fuel. Plant procedure CRC-001 requires that the soluble boron concentration in the pool be maintained between 2000 and 2600 ppm and is confirmed by monthly surveillance measurements. The reduction in k-eff due to the boron more than offsets the reactivity addition caused by credible accidents. In fact, Holtec has determined that a soluble boron concentration of only 400 ppm would be sufficient to maintain k-eff less than 0.95 even if a fresh PWR assembly were inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1.

3.1.1 Summary

Based on the review described above, the staff finds that the criticality aspects of the proposed storage capacity expansion for HNP spent fuel pools C and D are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. Our review has also determined that the proposed changes to TS 5.6, “Fuel Storage,” are acceptable. In addition, the staff finds the proposed change to TS 5.3, to reflect the renumbering of TS 5.6.1.a.2 to TS 5.6.1.1.b, and the revision to the Table of Contents to reflect the addition of Figure 5.6.1, acceptable.

3.2 Plant Systems

This portion of the evaluation addresses the heat load limitations associated with the expanded capacity for pools C and D, and the resolution of the USQ regarding changes in CCW flow.

3.2.1 Systems Descriptions

HNP was originally planned as a four nuclear unit site. A single FHB was designed and constructed with four separate pools capable of storing spent fuel for all planned units. The two pools at the south end of the FHB were designated A and B, and were designed to store fuel for Units 1 and 4. Pool A, the smaller of the two pools, would store new and spent fuel, while pool B would store spent fuel. The two pools at the north end of the FHB are designated C and D and were designed to support Units 2 and 3. The multi-unit design of the FHB includes a spent fuel pool cooling and cleanup system for pools A and B, and a separate cooling and cleanup system for pools C and D.

Upon cancellation of Units 2, 3, and 4, the licensee decided not to complete the fuel pool cooling system for pools C and D (the cooling and cleanup system for SFPs C and D was approximately 80% complete when construction on the system was halted upon the cancellation of Unit 2). However, the FHB and SFPs A, B, C, and D, including the pool liners, were constructed and turned over to the operating staff as part of the construction and licensing of HNP, Unit 1. The licensee decided not to complete the cooling system for SFPs C and D until these pools were needed for spent fuel storage. The pools have been filled with coolant, but have not stored spent fuel assemblies since they were constructed.

SFP A contains six flux trap style PWR racks and three BWR racks for a total storage capacity of 723 fuel assemblies (FAs). SFP B contains 12 PWR racks and 17 BWR racks, and is licensed to hold one additional BWR rack for a total capacity of 2946 FAs. The licensee proposed the following fuel storage capacities for the four pools (note: the authorized capacity of SFPs A and B will not change):

Pool Designation	Capacity		
A	PWR FA:360	BWR FA: 363	Total: 723
B	PWR FA:768	BWR FA: 2178	Total: 2946
C	PWR FA:927	BWR FA: 2763	Total: 3690
D	PWR FA:1025	BWR FA: 0	Total 1025

Each fuel pool cooling and cleanup system (FPCCS), north and south, is designed with two 100% capacity cooling trains, and a cleanup loop to remove dissolved fission and corrosion products. Each cooling system is comprised of two shell and straight tube heat exchangers, two horizontal centrifugal pumps, a demineralizer, two filters, skimmers, fuel pool and refueling water purification pumps, isolation gates for each pool and transfer canal, and the requisite piping, valves, and system instrumentation. Electrical power for the FPCCS pumps can be aligned to independent emergency supplies. Although originally designed to be cooled by separate cooling water systems, the south and north FPCCS heat exchangers will be cooled by the Unit 1 CCW system. Each pool is outfitted with direct reading temperature and level instruments that provide operators with indication and alarm at local and remote (i.e., the control room) stations.

The CCW system serves as an intermediate closed cooling water system between the radioactive and potentially radioactive systems and the non-radioactive service water system. The FPCCS rejects its heat to the CCW system, which in turn rejects its heat via the service water system to the ultimate heat sink. In addition to the FPCCS, the CCW system provides cooling to various safety-related and non-safety-related heat loads supporting the operation of the reactor. Although the Unit 1 CCW system was originally designed to remove the heat rejected from the spent fuel stored in pools A and B, the system is being modified to remove decay heat from pools C and D as well. The CCW system contains two separate trains, each train containing a CCW heat exchanger. Three CCW pumps are shared by the two trains. During normal and accident operation, including refueling operations, only one CCW pump is required to be operated to remove the required heat loads from the plant. During plant cool down when heat removal demands on the CCW system are unusually high, two CCW pumps are operated.

3.2.2 Review

The focus of this review is to evaluate the licensee's plans to expand the storage capacity of fuel onsite and the effect the expanded capacity has on the heat removal capabilities of the FPCCSs to ensure they continue to meet staff guidelines on fuel storage. The staff organized the review into four sections:

- A review of the changes proposed to the FPCCS since the original design of the plant was accepted by the staff in NUREG-1038 (ref. 12).
- A review of the effects of the increased decay heat loads on the cooling water systems supporting the storage of spent fuel.
- A review of the heavy loads aspects of this amendment.
- A review of a USQ associated with a proposed reduction of CCW flow to the RHR heat exchangers under certain operating conditions.

The staff based its findings on information contained in the HNP Final Safety Analysis Report (FSAR, ref. 13), NUREG-1038 (ref. 12), and on information contained in letters from the licensee dated December 23, 1998 (ref. 1), and September 3, 1999 (ref. 7).

3.2.2.1 System Design Changes

The staff reviewed and accepted the design of the spent fuel storage system for HNP, Units 1 and 2, in NUREG-1038 (ref. 12). Although the system was never completed, the design of the system was reviewed by the staff in accordance with NUREG-0800, "Standard Review Plan" (SRP, ref. 14). The licensee's amendment dated December 23, 1998 (ref. 1) requested the activation of pools C and D, but also made fundamental changes in the design of the system, for example, changing the system that supplies cooling water to the FPCCS for SFPs C and D from Unit 2 CCW to Unit 1 CCW. As a result, the staff requested in a letter dated August 5, 1999, that the licensee address the differences between the system design that was accepted by the staff in NUREG-1038 and the "as-built" system. The licensee responded by letter dated September 3, 1999 (ref. 7).

In its September 3, 1999, response (ref. 7), the licensee provided a matrix which reconciles the differences between the "as-built" fuel storage system and the conclusions drawn by the staff in NUREG-1038 (ref. 12) concerning the original design of the spent fuel storage system. In general, most of the FPCCS supporting pools C and D was built to the design reviewed by the staff in NUREG-1038 (ref. 12). However, some portions of the system design underwent significant design changes. Those portions have been re-evaluated by the staff and the results are summarized in the following paragraphs.

The staff compared the conclusions drawn by the NRC in Section 9.1.2 of NUREG-1038 (ref. 12) about the original fuel storage system design to the changes to the fuel storage system proposed by the licensee in their December 23, 1998, amendment request (ref. 1). The staff performed this review to ensure the proposed changes did not impact on the staff's previous conclusions concerning the acceptability of design of the fuel storage facility.

In NUREG-1038 (ref. 12), the staff documented the acceptability of the spent fuel storage facility and the SFP cooling systems for both Units 1 and 2 in Sections 9.1.2 and 9.1.3. These sections frequently refer to both Unit 1 and Unit 2. Since Unit 2 was not completed, these references are inaccurate. However, the references are editorial in nature and do not affect the staff's previous conclusions about the acceptability of the fuel storage system.

For those portions of the system covered by Section 9.1.2 of NUREG-0800 (ref. 14), specifically, the FHB, the spent fuel storage racks, the SFP area ventilation system, and other portions of the fuel storage system described in Section 9.1.2 of NUREG-1038 (ref. 12), the staff concluded, based on our review, that the proposed changes do not impact the NRC's previous conclusions and are still acceptable in accordance with the guidance in NUREG-0800 (ref. 14), Section 9.1.2, and Regulatory Guide 1.13 (ref. 15).

The staff also compared the conclusions drawn by the NRC in Section 9.1.3 of NUREG-1038 (ref.12) concerning the original FPCCS design to the changes to the FPCCS proposed by the licensee in their December 23, 1998, amendment request (ref. 1). The staff performed this review to ensure the proposed changes did not impact on the staff's previous conclusions concerning the acceptability of design of the fuel pool cooling and cleanup system.

The licensee called out the following differences between the original FPCCS design accepted by the staff in NUREG-1038 (ref. 12) and the "as-built" system described in the proposed license amendment :

a. A single refueling water storage tank (RWST) to provide system makeup water to both FPCCSs versus an RWST for each cooling system.

b. Emergency makeup for pools C and D provided from the Unit 1 emergency service water (ESW) system, not the Unit 2 ESW system. Flanged connection described in NUREG-1038 (ref. 12) for ESW hookup from Unit 2 will not be installed in the FPCCS system supporting pools C and D. The Unit 1 ESW system is sized to accommodate the emergency fill requirements and can be cross-connected to all pools.

c. The current design limits the temperature of SFPs A and B to 137°F, assuming a single active failure, which is lower than the temperature stated in NUREG-1038 (ref. 12).

d. SFP chemistry limits are currently maintained consistent with guidelines established by the NSSS vendor, fuel manufacturer, and EPRI guidelines. NUREG-1038 (ref. 12) assumed a weekly sampling protocol.

e. NUREG-1038 (ref. 12) contains many references to Unit 2. Due to the cancellation of Unit 2, references to GDC 5, sharing of structures, systems and components, are no longer applicable.

Items c, d, and e, above were reviewed by the staff and found to be editorial in nature, or approved by the staff in previous licensing actions and part of the current system design basis, and are, therefore, acceptable.

For items a and b above, the staff previously accepted a design for the fuel storage system whereby separate RWSTs would be available to provide makeup water to pairs of SFPs (SFPs A and B from Unit 1 RWST, SFPs C and D from Unit 2 RWST). Similarly, the staff accepted a backup method of makeup to the fuel storage system from the Unit 1 and Unit 2 ESW systems through valved and flanged connections. Since Unit 2 was never constructed, the Unit 2 RWST and the Unit 2 ESW system are not available. Makeup from the RWST is used to compensate for coolant losses due to evaporation and cooling system leakage. The proposed change recognizes the Unit 1 RWST as the seismic Category 1 makeup water source for both FPCCSs, supplying makeup for all four SFPs. Similarly, the Unit 1 ESW system is available to provide a seismic Category 1 backup makeup water source through a cross-tie to all four fuel storage pools in the event of an emergency. The licensee has evaluated this configuration and determined that the Unit 1 RWST and the Unit 1 ESW system have sufficient capacity to supply makeup for all four pools. The staff reviewed the proposed changes to the seismic Category 1 makeup supplies for the FPCCS for pools C and D and finds that the Unit 1 RWST and the Unit 1 ESW system have sufficient capacity to provide makeup to the four fuel storage pools and are, therefore, acceptable.

In addition to the changes made to the systems that directly support SFPs C and D, the Unit 1 CCW system was also modified to account for the absence of the Unit 2 CCW system. Valves, piping and other components were added to the Unit 1 CCW system to provide heat removal capability for the SFPs C and D FPCCS heat exchangers. The staff's evaluation of the effects of adding an additional heat load to the Unit 1 CCW system is included in Section 3.2.2.3 of this Safety Evaluation.

3.2.2.2 <u>Changes in Decay Heat Load</u>

The licensee provided a summary of methods, models, analyses, and numerical results to demonstrate the compliance of the HNP spent fuel storage systems with the provisions of Section III of the USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications" (ref. 16). The licensee provided the following analyses as justification for the acceptability of the proposed changes to the spent fuel storage system:

- Evaluation of the long-term decay heat load in SFPs C and D.
- Evaluation of the steady-state bulk pool temperature with forced cooling available (fuel pool bulk temperature is limited to 137°F with the FPCCS in operation).
- Determination of the maximum pool local temperature.
- Evaluation of the potential for flow bypass from the pool inlet to the pool outlet with the sparger removed.
- Evaluation of the time-to-boil assuming all forced cooling is lost.

Holtec International, a contractor of the licensee, performed decay heat load calculations in accordance with USNRC Branch Technical Position ASB 9-2, “Residual Decay Energy for Light Water Reactors for Long Term Cooling.” (ref. 1, Enclosures 6 & 7). The calculations assumed that the spent fuel stored in SFPs C and D had cooled a minimum of 5 years before being placed in SFPs C or D. Holtec determined the bounding decay heat load in SFPs C and D based on fuel characteristics documented on Tables 5.2.1 and 5.2.2 of Enclosure 6 to the letter dated December 23, 1998 (ref. 1). Although the bounding calculations determined that the maximum decay heat load in SFPs C and D could reach 15.63 Mbtu/hr, the licensee has decided to limit the maximum decay heat load in SFPs C and D to 1 Mbtu/hr using administrative controls.

In an August 5, 1999, Request of Additional Information (RAI), the staff asked the licensee to provide an analysis showing that the maximum bulk temperature for SFPs C and D will not be exceeded assuming an increase in the decay heat load of 1 Mbtu/hr. In a letter dated September 3, 1999 (ref. 7), the licensee provided an analysis that shows the maximum bulk temperature of all four SFPs remains below the pool design temperature of 137°F under a variety of operational conditions, including those that conform to the guideline in NUREG-0800 (ref. 12) for partial and full core offloads. Where appropriate, the licensee assumed a single active failure and design temperatures (e.g., 95°F for the ESW system) in the systems providing cooling water to the FPCCS heat exchangers.

The staff reviewed the documentation and agrees with the licensee that there is sufficient thermal margin in the CCW and ESW systems to maintain the bulk fuel pool coolant temperature in all SFPs within their design limits assuming an additional decay heat load of 1 Mbtu/hr in SFPs C and D, and assuming a single active failure.

The licensee’s contractor, Holtec, also performed an analysis of the temperatures at various locations in the fuel pool to ensure localized boiling does not occur, especially in the fuel storage racks. Bounding assumptions for fuel storage location and cooling times were assumed, as well as for bulk coolant temperature and cooling flow to the SFPs. A computational fluid dynamics model was used to determine the difference between peak local and bulk coolant temperatures. The results indicate that peak local temperature in the pool will be 6.8°F higher than the maximum bulk coolant temperature of 137°F. Based on a review of the licensee’s methods and findings, the staff agrees that sufficient thermal margin exists to preclude localized boiling.

The licensee also provided the results of heat up calculations to determine the time-to-boil should a loss of all forced cooling occur. Section 5.4.1 of Enclosure 6 of the letter dated December 23, 1998 (ref. 1) discusses the results of time-to-boil calculations performed by Holtec. The results indicate that with a heat load of 15.63 Mbtu/hr in SFPs C and D, and an initial bulk coolant starting temperature of 140°F, more than 13 hours are available to take mitigating action. The staff considered this evaluation very conservative, given the heat load in SFPs C and D will be limited to 1 Mbtu/hr, and requested that the licensee evaluate the pool under its expected operating conditions. The licensee performed additional calculations that indicate several hundred hours are available to mitigate a total loss-of-cooling event in SFPs C and D assuming a 1 Mbtu/hr heat load limit. These calculations are documented in a letter dated September 3, 1999 (ref. 7).

The staff performed an independent heat-up evaluation to ensure the licensee's results were conservative. For added conservatism, the staff assumed the SFPs were isolated from each other when cooling was lost and that the entire decay heat load was located in a single pool. The staff's evaluation confirmed that more than 100 hours are available to identify and address a loss of all forced cooling event if the heat load were limited to SFP C, and more than 50 hours are available if the decay heat load were limited to SFP D.

Given the decay load in SFPs C and D will be limited to 1 Mbtu/hr, the staff agrees that sufficient time is available for plant operators to take mitigating actions prior to pool boiling.

3.2.2.3 Unreviewed Safety Question

The CCW system provides cooling to the RHR system heat exchangers, RHR pumps, the SFP heat exchangers, and other non-safety-related systems. Two RHR trains provide long-term cooling during the containment sump recirculation phase of a loss-of-coolant accident (LOCA) by circulating the reactor coolant from the containment sump, through the heat exchangers, and returning it to the reactor coolant system cold legs. Each RHR train is capable of removing up to 111.1 Mbtu/hr in the post-LOCA scenario. In the USQ thermal-hydraulic analysis, the licensee demonstrates that adequate excess thermal capacity existed in the CCW system to accommodate the additional heat loads of 1.0 MBTU/hr (which is a limitation specified in TS 5.6.3) from SFPs C and D during all normal and accident modes of system operation, i.e., the required RHR heat removal capability can be met with reduced CCW flow through the RHR heat exchanger due to the tie-in of the C and D FPCCS.

The USQ thermal-hydraulic calculations did not change any assumptions regarding maximum sump temperatures or RHR heat removal requirements under post-LOCA containment conditions. However, the licensee identified that fluid properties at the higher RHR temperatures associated with the post-LOCA scenario would result in an increase in the heat exchanger heat transfer coefficient values over the fixed value assumed in the existing analysis. The analyses used a "dynamic" RHR heat exchanger performance model in which the tube side inlet temperature is postulated to rise to 244.1°F during the initial phase of containment sump recirculation, rather than a fixed 139°F currently assumed. This increased tube side fluid temperature increases the overall RHR heat exchanger heat transfer coefficient (HTC) by approximately 10% due to the change in tube side fluid viscosity. Based on this increased heat exchanger HTC, the calculations showed that a minimum CCW system flow rate through the RHR heat exchanger of 4874 gpm at 120°F is required at the beginning of the sump recirculation phase. Assuming a 6% model uncertainty, the required CCW system flow to the RHR heat exchanger would be 5166 gpm, which is less than the 5600 gpm required by the existing analysis.

The licensee also provided, in response to a staff question (question 6, September 3, 1999, letter (ref. 7)), the results of analyses based on a time-dependent heat rejection load of the RHR heat exchanger, and the containment sump water temperature during a LOCA. The staff has performed an audit calculation of these results, and found that the analyses were conservatively based on a lower density and mass flow of the CCW volumetric flow rate of 4874 gpm. The staff concurs with the licensee's analysis conclusion that the required RHR heat removal capability can be met with the reduced CCW flow of approximately 5200 gpm.

3.2.2.4 Summary

The licensee proposed to modify Section 5.6.3, “Capacity,” of the TS to define the maximum capacity of the four SFPs. In addition, the licensee included a section to limit the total decay heat load in SFPs C and D to 1 Mbtu/hr. The licensee also identified a USQ and provided a justification why the changes to the design of the CCW system were acceptable. Information provided by the licensee in the amendment request dated December 23, 1998 (ref. 1), and letter dated September 3, 1999 (ref. 7), documented the licensee’s justification for requesting the staff’s approval of this amendment.

The staff has completed its review of the USQ justification and the decay heat load aspects of increasing capacity of the fuel storage system at HNP. Based on the above evaluation, the staff finds the licensee proposed changes acceptable.

3.3 Handling of Heavy Loads and Spent Fuel Assemblies

3.3.1 Background

NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either: (1) the potential for a load drop is extremely small, or (2) the potential hazards of load drops do not exceed acceptable limits. NUREG-0612 provides guidelines that are implemented in two phases. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads and provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, design, testing, inspection, and maintenance of cranes and lifting devices, and analyses of the impact of heavy load drops.

Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased handling system reliability, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drops and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, “Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612,” dated June 28, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety.

NRC Safety Evaluation Report related to the operation of HNP, Unit 1, NUREG-1038, Supplement No. 4, dated October 1986 (ref. 12) approved CP&L’s NUREG-0612, Phase I heavy loads program. In the proposed amendment, the licensee addresses heavy load issues, including the installation of spent fuel storage racks in SFPs C and D, fuel movement, movement of the gates that isolate the pools from the transfer canal, and movement of spent fuel dry storage casks. Considerations are given to the design and operation of the hoisting systems, safe load paths, procedures, crane operator training, and postulated load drop accidents and consequences on fuel and on the SFP.

3.3.2 Review

3.3.2.1 Hoisting Systems

The FHB auxiliary crane, which is rated at 10 tons, will be used to lift and move the new racks, the gates that isolate the pools from the transfer canal, and new FAs. The racks will be lifted up through the equipment hatch then transported along the safe load path to SFPs C and D. The same crane will be used to lower the racks into SFPs C and D. A 20-ton hoist will be suspended from the bridge of the FHB auxiliary crane and used in conjunction with the spent fuel rack lifting rig (special lifting device) to lift and move the racks into the pools. The use of the 20-ton hoist will allow the licensee to avoid contamination of the main hook during SFP rack movement in the pools. The 150-ton Spent Fuel Cask Handling Crane will be used to lift and move shipping casks containing offsite spent fuel from the railroad car through the equipment hatch to the cask loading pool.

In NUREG-1038, Supplement No. 4 (ref. 12), the NRC staff approved the load handling systems, including the FHB auxiliary crane and the spent fuel cask handling crane. The staff found that the load handling system provided adequate protection against heavy load drops and was consistent with NUREG-0612. However, the spent fuel storage rack lifting rig is specifically designed to lift the new rack modules; therefore, it was not addressed in NUREG-1038.

Both the FHB Auxiliary Crane and the Spent Fuel Cask Handling Crane are designed, fabricated, installed, inspected, tested, and operated in accordance with requirements of the Crane Manufacturers Association of America (CMAA) Specification No. 70, “Specifications for Electric Overhead Traveling Cranes,” and ANSI B30.2-1976, “Safety Standards for Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder).” The FHB auxiliary crane is single-failure-proof and although it has a rated capacity of 10 tons in the auxiliary hook, it can be used to handle items that weigh more than 10 tons but less than 12 tons provided that they are evaluated and administratively controlled. The single-failure-proof design of the crane enables the licensee to retain and hold the load in a stable and immobile safe position during an event. The crane is equipped with a means to safely move the load manually to a lay down area for emergency manual lowering of the load. Also, all the components in the load path of the crane hoist such as the hook, hoist rope, reeving, and braking mechanisms, either are redundant or have a large factor of safety. In addition, the crane is designed to maintain its structural integrity and hold the load under the dynamic loading conditions of a safe shutdown earthquake (SSE). The maximum lifted weight during rack installation includes the rack, lifting rig (special lifting device), rigging, and the 20-ton hoist for a total weight of 18,820 lbs.

The licensee states that the rack modules will be lifted using a remotely engaged spent fuel rack lifting rig that is specifically designed to lift both PWR and BWR spent fuel rack modules. The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612, Sections 5.1.6(1) and 5.1.6(3a), and the requirements in ANSI N14.6 (1978), “Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials.” Accordingly, in accordance with NUREG-0612, the lifting rig has twice the design safety factor with respect to the yield and ultimate strength (six (6) times and ten (10) times the combined concurrent static and dynamic loads for the yield and ultimate strength, respectively) of its material of construction. The lifting rig also is redundantly designed with four independently loaded lift rods that are configured such that failure of a single rod will not result in uncontrolled lowering of the rack. Therefore, the lift rods and lift points of the lifting rig are designed and tested as follows: (1) with a stress design factor of five times the lifted weight without exceeding the ultimate strength of the material; and (2) load tested to 300% of the maximum weight to be lifted, and hoisted and suspended for a minimum of 10 minutes. After load testing, examination of the critical weld joints using a liquid penetrant is performed.

The staff finds that the lifting capacity of the 10-ton single-failure-proof FHB auxiliary crane coupled with the capacity of the 20-ton hoist and the spent fuel rack lifting rig (special lifting device) will support the weight of the racks and the added rigging loads. In addition, the design, inspection and testing of the crane and lifting device will help to assure the licensee's safe handling of the racks with little to no risk of an accidental rack drop during rack installation.

3.3.2.2 Load Path

The new spent fuel storage racks will be installed in five sequential campaigns as follows:

	SFP C (date)	SFP D (date)
Campaign 1	14 (2000)	6 (2016)
Campaign 2	10 (2005)	6 (TBD)
Campaign 3	6 (2014)	

The new racks will be lifted to the FHB operating level through the equipment hatch, then moved along the safe load path that was previously identified as the path used for the spent fuel shipping cask. The racks are then moved over the fuel transfer canal from which they are moved over SFP C and D and lowered into position. The safe load path from the equipment hatch to the SFPs is clear of any safety-related equipment. The licensee stated that rack installation and fuel assembly storage will begin in the south end of SFP C and proceed north to SFP D. Therefore, lifts of the racks over spent fuel will be avoided.

As stated by the licensee, the new installed fuel storage racks will not significantly change the method of handling loads during normal plant operations because the same equipment (i.e., the spent fuel handling machine and tools) and procedures as those currently used in pools A and B will be used in pools C and D.

New and spent fuel shipping casks are lifted from the carrier through the equipment hatch up to the FHB operating level by the FHB auxiliary crane or the spent fuel cask handling crane. As stated in FSAR Section 9.1.4.3.2(b), spent fuel shipping casks are handled by the 150-ton cask handling crane. Therefore, the shipping casks containing offsite spent fuel will be lifted by the cask handling crane through the equipment hatch to the operating level then moved to the cask loading pool. Permanent mechanical stops on the cask handling crane limit any travel of the crane over the SFPs. This enables the licensee to avoid traversing or dropping the cask over spent fuel in the fuel pools.

The staff finds that the load paths for movement of the spent fuel storage racks and the pattern for storing spent fuel in the racks does not involve any movement of the racks over spent fuel. Also, cask movements do not involve travel over fuel stored in the racks or over the SFP.

3.3.2.3 Analysis of Heavy Load Drop Accidents

The licensee analyzed postulated load drops of spent fuel assemblies, spent fuel storage racks, and the gates that isolate the pool from the transfer canal. Two drops of the fuel assembly using a bounding impact weight of 2100 lbs. (includes the heaviest fuel plus the handling tool) was considered: vertical drops on top of the racks, and vertical drops to the base plate of the racks. The FA drop on top of the spent fuel racks resulted in deformation of the racks to a depth of 11 inches below the top of the rack. However, the fuel in the racks would not be damaged. The FA drop onto the base plate of the racks resulted in damage to the baseplate but no damage to the SFP liner.

HNP FSAR (ref. 13), Appendix 9.1A, “Heavy Loads Analysis,” does not address the drop of a rack into the SFP. However, because a single-failure-proof crane is used, there is a reasonably low chance of a rack drop. Nonetheless, in its amendment request, the licensee considered a vertical drop of the heaviest rack (16140lbs.) from 40 feet above the SFP floor liner. The results indicated that some damage to the SFP liner and minor damage to the SFP concrete floor slab would occur. In telephone conferences on March 30 and April 4, 2000, the staff discussed additional information it needed on the results of the rack drop analyses. By letter dated April 14, 2000 (ref. 10), the licensee responded that a rack drop would pierce the SFP liner and result in leakage of SFP water. However, the plant’s design basis leakage detection system is designed and operated to detect, limit, and contain leakage from the SFP. Valves in the leakage detection system are normally closed and are only opened to check for and measure any leakage during the operator monthly rounds. Therefore, the closed valves will enable the system to limit any SFP leakage. In addition, SFP makeup can be made available from a number of sources to supplement any leakage from the SFP. Emergency makeup can be provided from the emergency service water system. Normal SFP makeup can be provided from the demineralized water system, RWST, the reactor coolant drain tank, and the reactor makeup water storage tank. Therefore, due to these capabilities, and because the structural integrity of the concrete slab remained unimpaired after the drop, the licensee concluded that neither catastrophic damage of the SFP structure nor rapid loss of pool water would occur.

The licensee did not analyze the potential for a rack drop on spent fuel assemblies or on safety-related equipment because: (1) the racks would not be moved directly over any fuel in the pool; and (2) upending and laying down the racks is planned to occur in an area that avoids any potential impact on safety-related equipment.

The drop of the gates (4,000 lbs. each) that isolate the pool from the transfer canal was analyzed. The gates will also be lifted using the single-failure-proof auxiliary crane and dual rigging that satisfies NUREG-0612 safety margins. The gate rigging will be load-tested to lift three times the weight of the rack and the other components of the lifting device without exceeding the minimum yield strength of the material. The gate rigging also will be capable of lifting five times that weight without exceeding the ultimate strength of the rig materials. The postulated gate drops were analyzed at 15 inches above loaded fuel racks and at 40 feet above the SFP liner. A gate drop would penetrate the racks to a depth of 5 inches with no impact on the stored fuel. It also would damage the pool liner; however, the SFP concrete slab would not fail. Therefore, fuel in the racks would not be affected if a gate drop occurred. If the SFP liner is breached, the leakage detection system and makeup capability as discussed above would be effective.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee stated that they will implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. Accordingly, the licensee plans to provide: (1) comprehensive training to the rack installation crew in accordance with ANSI B30.2; (2) redundantly designed and adequately tested lifting rigs in accordance with ANSI N14.6; (3) inspection and maintenance checks on the cranes, lifting devices, and racks themselves prior to and during the rack installation; and (4) specific procedures that cover the entire racking effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement. In addition, the licensee states that its rack installation pattern will enable the racks to be lifted and inserted without travel over spent fuel in the SFP during the rack installation operation.

Since both the spent fuel cask handling crane and the configuration of the FHB are designed to avoid any travel of the crane hook over the SFPs, the spent fuel shipping cask will not be moved over or have any opportunity to fall into the fuel pools. As a result, there is no need for a load drop analysis of the cask over the SFPs.

The staff accepts the licensee’s finding that, based on the load drop analyses, the integrity of the fuel and the SFP would be maintained if an FA or a spent fuel storage rack is dropped. The use of a single-failure-proof crane in conjunction with administrative procedures and controls that are focused on, but not limited to, the areas noted above would enable the licensee to maintain safety during the implementation of the proposed changes.

3.3.2.4 Summary

Based on the preceding discussions, the staff finds that the aforementioned considerations for the movement of heavy loads to support the proposed changes to TS 5.6 and the increase in the SFP storage capacity are acceptable. The licensee’s use of the 10-ton single-failure-proof FHB auxiliary crane, the 20-ton hoist, the spent fuel rack lifting rig, and administrative controls and procedures that are in accordance with NUREG-0612 and ANSI N14.6, will help to maintain safety during the installation of the new racks. The reliability of the crane coupled with the design, testing and inspection of the crane, the lifting rig and other lifting devices will enable the licensee to handle safely the racks and other heavy loads during the rack installation process. The postulated accident analyses involving a dropped spent fuel storage rack and the gate indicated that the SFP liner could be breached. However, during such a breach, the licensee could maintain the pool and its contents within the acceptable consequence limits set forth in NUREG-0612. In addition, the licensee’s use of administrative controls and procedures to improve the handling and control of heavy loads, including the racks, enhances the licensee’s capability to reduce the potential for load drops.

In addition, the licensee's cask handling operations will not occur over the SFPs. Therefore, movement of the cask will not impact the new racks nor the fuel configuration in the pools. Also, the licensee stated that there will be no significant changes in its method of handling new and spent fuel assemblies subsequent to the installation of the racks. Therefore, the staff believes that the existing fuel handling procedures as documented in the FSAR (ref. 13) will enable the licensee to continue to maintain safety during its fuel handling operations.

3.4 Structural Engineering

The primary purpose of this review is to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the SFP structures subject to the effects of the postulated loads (Appendix D of SRP Section 3.8.4)(ref. 14) and fuel handling accidents.

3.4.1 Storage Racks

CP&L has proposed to install 30 spent fuel storage racks in SFP C, and 12 racks in SFP D. The total storage capacity of the 30 racks is 3690 storage locations and the total storage capacity of the 12 racks is 1025 storage locations. All 42 storage racks are seismic Category I equipment and are required to remain functional during and after an SSE. CP&L, with its contractor Holtec, performed structural analyses of the racks for the requested license amendment.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the HNP spent fuel rack design under the combined effects of earthquakes and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor or walls of the SFP. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the program, were used to simulate the three dimensional (3-D) dynamic behavior of the rack and the stored FAs, including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Analyses of two models were performed: a 3-D single-rack (SR) model and a 3-D whole pool multi-rack (MR) model. For the 3-D MR analyses, all racks were considered to be fully loaded and partially loaded with three different coefficients of friction (μ=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the pool floor to investigate the fluid-structure interaction effects between the racks and the pool walls as well as those among the racks and to identify the worst-case response for rack movement and for rack member stresses. For the 3-D SR analyses, the rack was considered to be half loaded with a condition that all loaded fuels were placed at one side of the rack. The coefficient of friction of 0.8 between the rack pedestal and the pool floor was used to investigate the stability of the rack with respect to overturning.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) were generated from the design response spectra defined in the FSAR (ref. 13). CP&L demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in SRP Section 3.7.1 (ref. 14).

A total of 17 3-D SR and MR analyses were performed. The racks were subjected to the service, upset and faulted loading conditions (Level A, B and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 1.494 inches, indicating that there is adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the rack and the pool wall. However, the results show that there is impact potential between the racks. The staff compared the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NF. The stress results show that the induced impact forces under the SSE loading condition are small and all induced stresses in the racks are smaller than the corresponding allowable stresses specified in the ASME B&PV Code, indicating that the rack design is adequate.

CP&L also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal and cell-to-cell connections) under the dynamic loading conditions. CP&L demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

Based on (1) CP&L's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowables provided in the ASME B&PV Code, and (3) CP&L's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.4.2 Spent Fuel Storage Pool

CP&L analyzed the SFPs to demonstrate the adequacy of the structures under fully loaded fuel racks with all storage locations occupied by FAs. The fully loaded structures were subjected to the load combinations specified in the HNP FSAR (ref. 13).

The licensee’s July 23, 1999, submittal (ref. 6) shows the predicted factors of safety varying from 1.05 to 3.51 for shear force and bending moment of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the CP&L structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

3.4.3 Fuel Handling Accident

The following two refueling accident cases were evaluated by CP&L: (1) drop of an FA with its handling tool, which impacts the baseplate (deep drop scenario); and (2) drop of an FA with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of accident case (1) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the FA drop through the rack structure. The analysis results of accident drop case (2) show that damage will be restricted to a depth of 11.0 inches below the top of the rack, which is above the active fuel region. The staff reviewed CP&L's analysis results and concurs with its findings. These results are acceptable based on CP&L's structural integrity conclusions supported by the parametric studies.

3.4.4 Summary

Based on the review and evaluation of CP&L's submittals, the staff concludes that CP&L's structural analysis and design of the spent fuel rack modules is acceptable and that the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR (ref. 13) and applicable provisions of the SRP (ref. 14), and are, therefore, acceptable.

3.5 Materials Engineering

The new maximum storage rack arrays proposed for use in the SFP are manufactured by Holtec. The racks are free-standing and self-supporting. The racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME B&PV Code.

3.5.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include: (1) ASME SA240-304L for all sheet metal stock and internally threaded support legs; (2) ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100°F) for externally threaded support spindle; and (3) ASME specification SFA 5.9 ER308L for weld material.

These materials used in the Holtec racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the SFP environment. Therefore, they are acceptable for use in this application.

3.5.2 Poison Material

The Holtec racks employ Boral as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the SFP environments, where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed to vent the corrosion gases. The neutron-absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

3.5.3 Summary

Based on the above evaluation, the staff finds that the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec are compatible with the SFP environment at HNP. The type of degradation exhibited by the racks does not affect their neutron-absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

3.6 Radiological Assessment

3.6.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for placing SFPs C and D in service with respect to occupational radiation exposure. For this modification, the licensee plans to install region 2 (non-flux trap style) rack modules in pools C and D in incremental phases, on an as-needed basis. The licensee estimates that the collective dose associated with the proposed fuel rack installation is in the range of 2-3 person-rem.

All of the operations involved in racking will utilize detailed procedures prepared with full consideration of ALARA (as low as reasonably achievable) principles. The HNP racking project represents lower radiological risk due to the fact that the pools currently contain no spent fuel. The Licensee's Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP rack installation operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels, and dosimetry requirements. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment, including alarming dosimeters.

Since this license amendment does not involve the removal of any spent fuel racks, the licensee does not plan on using divers for this project. However, if it becomes necessary to utilize divers to remove any interferences that may impede the installation of the new spent fuel racks, the licensee will equip each diver with the appropriate monitoring equipment. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposure is maintained ALARA.

On the basis of our review of the licensee's proposal, the staff concludes that the expansion of HNP's spent fuel storage capacity can be performed in a manner that will ensure that doses to workers will be maintained ALARA. Therefore, the staff finds the licensee's proposal acceptable.

3.6.2 Solid Radioactive Waste

The necessity for pool filtration resin replacement is determined primarily by the requirement for water clarity, and the resin is normally expected to be changed about once a year. The licensee does not expect the resin change-out frequency of the SFP purification system to be permanently increased as a result of the expanded storage capacity. Overall, the licensee does not expect that the additional fuel storage made available by the increased storage capacity will result in a significant change on the generation of solid radioactive waste. The staff finds the licensee's conclusion to be acceptable.

3.6.3 Design Basis Accidents

In its application, the licensee stated that since the pertinent fuel parameters remain unchanged, the radiological dose consequences at the exclusion area boundary for the accidental drop of a fuel assembly in the SFP will not be increased from those previously calculated. Section 15.7.4.5 of the HNP FSAR (ref. 13) describes the worst-case fuel handling accident in the FHB for dose consequences. This case assumes a PWR assembly is dropped, initially strikes a stationary PWR spent fuel assembly, then falls onto an adjacent loaded BWR spent fuel rack. This scenario assumes 314 PWR spent fuel rods and 52 BWR spent fuel assemblies fail. FSAR Table 15.7.4-7 (ref. 13) gives the offsite dose consequences for the above bounding scenario. The addition of racks to the currently unused SFPs C and D would not change the bounding dose scenario. No change is being made to the handling of the spent fuel or the types of fuel stored in the HNP SFPs.

The staff agrees that the bounding scenario for the postulated fuel handling accident in the FHB does not change due to the addition of storage racks in SFPs C and D. Therefore, the inputs and assumptions for the dose consequences analysis do not change, and the current fuel handling accident dose assessments in the HNP FSAR (ref. 13) remain bounding. The staff agrees with the licensee’s assessment and finds their conclusion acceptable. The staff has determined that the addition of spent fuel racks to SFPs C and D is acceptable with regard to the radiological consequences of the postulated fuel handling accident.

3.7 10 CFR 50.55a Alternative Plan

As part of its December 23, 1998, application (ref. 1), the licensee proposed an alternative to the ASME B&PV Code (Code) documentation requirements for certain pipe welds in the cooling and cleanup water systems servicing SFPs C and D. The alternative was necessitated by an HNP determination that some documents required by the Code for welds were destroyed. The alternative consists of varying degrees of inspection coverage performed by a variety of examination methods, selective nondestructive testing (NDE), selective weld replacements, selected chemical analyses, and establishing that the missing documents had existed and had been found acceptable by the responsible parties prior to their destruction.

In a March 24, 1999, RAI, the NRC requested clarification of certain aspects of the licensee’s December 23, 1998, letter (ref. 1). The licensee responded to the RAI in a submittal dated April 30, 1999 (ref. 4). The staff sent the licensee a second RAI on September 20, 1999. The licensee responded in submittals dated October 15, 1999 (ref. 8), and October 29, 1999 (ref. 9). Additional information was gathered during an NRC staff on-site inspection of the SFPs on November 15 through 19, 1999 (ref. 17). A follow-up inspection was conducted from January 31 through February 4, 2000 (ref. 19).

3.7.1 Background

The licensee filed an application with the Atomic Energy Commission (AEC) in 1971 for licenses to construct and operate HNP Units 1, 2, 3, and 4. The AEC issued construction permits for each of the four units in January 1978. Construction proceeded until December 1981, when the licensee informed the NRC that Units 3 and 4 were being canceled, and requested that Units 1 and 2 be considered concurrently for operating licenses. NUREG-1038 (ref. 12) was issued in

November 1983 for Unit 1, and reflected ongoing construction for Unit 2. In December 1983, Unit 2 was canceled. Unit 1 was issued an operating license in January 1987.

The FHB was designed to have four SFPs. SFPs A & B were to support Units 1 and 4 and SFPs C & D were to support Units 2 and 3. These concrete pools are interconnected, which dictated that they be completed at least to the point of being watertight. For a watertight SFP, the reinforced concrete encasement, stainless steel pool liners, and closure of certain pipe openings had to be in place. By the time Units 2, 3, and 4 were canceled, the majority of mechanical piping and equipment associated with the operation of the pools were already installed. The four SFPs, the FPCCS for SFPs A & B, and the CCWS for SFPs A & B were completed and turned over as part of the construction and licensing of Unit 1. The construction of the FPCCS and CCWS for SFPs C & D was discontinued and their pipe stubs extending outside of the pools were capped. Some of the field weld documentation associated with the FPCCS and CCWS for SFPs C & D were inadvertently discarded during subsequent document control cleanup efforts.

10 CFR 50.55a(g)(4) states that ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the Code and addenda that become effective subsequent to the editions specified in paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

10 CFR 50.55a(a)(3) states that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.7.2 Requirements

The licensee filed an application with the AEC in 1971 for licenses to construct and operate HNP, Units 1, 2, 3, and 4. The construction code used to design, fabricate, erect, construct, and inspect the balance of plant piping requiring a Code stamp was the ASME Code, Section III, Division 1, 1971 Edition with Summer 1973 Addenda. Field fabrication and installation of Code items was performed in accordance with the ASME Code, Section III, Division 1, 1974 Edition with Winter of 1976 Addenda.

The specific paragraphs in the Code that pertain to missing documentation as discussed in this Safety Evaluation are listed in the licensee's letter dated April 30. 1999, Enclosure 7, "Matrix of Code Requirements vs. Missing Field Records" (ref. 4).

3.7.3 Licensee's Proposed Alternative

For the missing SFPs C & D construction documentation, the licensee's proposed alternative is to reinspect and reconstitute the records to the 1974 Edition with 1976 Winter Addenda of Section III of ASME Code. For the items that cannot be reinspected or reconstituted to Code, the proposed alternative is to perform chemical analyses/checks, visual examinations, and record reviews in order to establish an acceptable level of quality and safety.

The proposed alternative for specific paragraphs in the Code that pertain to missing documentation are listed in the licensee's letter dated April 30. 1999, Enclosure 7, "Matrix of Code Requirements vs. Missing Field Records" (ref. 4).

3.7.4 Basis for the Alternative

The licensee developed the proposed alternative based on the accessibility of the welds. For field welds that were accessible, the licensee relied mostly on re-inspections and testing. For inaccessible embedded field weld, the licensee relied mostly on non-Code methods of inspections, testing, and alternate documentation.

The weld filler material used for all accessible stainless steel field welds in the FPCCS piping were subjected to a limited chemical analysis with an "alloy analyzer." In addition, chip samples from three randomly selected field welds were subjected to complete chemical analyses. The chemical analyses demonstrate consistency and supported the validity of the chemical results from the alloy analyzer. Also, a limited number of these field welds were checked for ferrite number. The ferrite number checks provided supporting data of the acceptability of the stainless steel filler metal used in these applications.

All three CCWS carbon steel field welds with missing documentation were analyzed for material composition. The analyses verified the acceptability of filler metal for the application.

The as-built isometric drawings were reconstructed from data gathered during a detailed walk-down inspection of the current configuration.

The personnel used to complete and license SFPs A & B came from the same labor pool and used the same procedures as were used on the completed portions of SFPs C & D and their support facilities.

The NDE for the welds with missing documentation was recorded on the weld data record (WDR). WDRs are the only licensee records attesting to the Code acceptability of NDE. HNP recreated, to the extent possible, new WDRs for each field weld within the scope of the alternative. The recreation consisted of NDE reinspection of the accessible welds to the original construction Code. For field welds embedded in concrete, HNP visually examined them from inside the pipe. The examinations provided subjective information on the weld quality and, to the extent feasible, objective evidence of compliance with Code and procedural requirements.

The licensee used the existence of hydrostatic test records as evidence that WDRs were completed and contained the appropriate NDE records. The reinspections of accessible field welds with no rejections provided supporting evidence of acceptability of the construction process. The hydrostatic test records and reinspection records assure that missing WDRs had been satisfactorily completed with acceptable NDE results.

The initial construction program used the same pool of qualified craftsmen, quality control personnel, and engineers for all four SFPs. The construction program also used the same type of material, the same procedures, and the same program controls. The adequacy of these site programs and controls is validated by the acceptability of the operational SFPs A & B, including the acceptability of the construction documentation for SFPs A & B.

Beyond programmatic assurances, a large body of evidence was compiled attesting to the quality of construction of SFPs C & D. Vendor data packages, hydrostatic test records, quality control records, and other construction-era documentation were retrieved that validate compliance with site programs and procedures. An examination effort was completed in which Code-required external NDE of accessible welds were reexamined with no rejectable indications and filler metals were examined for chemical composition verifying material selection. These results provide direct evidence of the quality of accessible field welds, and by extension, the smaller group of welds that are embedded.

In the absence of any contradictory evidence, and the preponderance of supporting evidence, the licensee concludes that the construction that occurred during the original construction phase of SFPs C & D was performed to the appropriate level of quality and safety and in compliance with construction Code requirements. Therefore, the licensee concluded that the missing Code documentation does not infer a physical lack of quality in the field.

3.7.5 Evaluation and Discussion

In its letter dated April 30, 1999 (ref. 4), the licensee identified documentation deficiencies with 40 pipe welds and 12 hanger-to-pipe welds. Specifically, deficiencies were identified with 22 FPCCS welds that are accessible from the pipes' exterior, 3 CCWS welds that are accessible from the pipes' exterior, and another 15 FPCCS welds that are embedded in concrete and are accessible only from the pipes' interior. The 12 hanger-to-pipe welds are on the exterior surface of the FPCCS piping. Since then, the licensee has replaced three of the accessible FPCCS welds with fully documented Code welds (refs. 4, 8). The licensee provided the staff with a list of the SFPs C & D FPCCS and CCWS welds that had missing documentation and their proposed alternative. The submittals also identified the specific Code requirements that were not satisfied as a result of the missing documentation. In the following paragraphs, the staff evaluated the adequacy of the proposed alternative for the field welds against the objectives of the specific Code requirements associated with the missing documentation.

ND-2150 and ND-4122 require identification and control of pressure-retaining material. These requirements ensure that the appropriate materials are being used for the construction of the FPCCS and CCWS weld and pipe materials. The control of material is part of the licensee's Quality Assurance (QA) program. The licensee stated that a Code-compliant QA program was in existence during the fabrication of the welds with missing documentation. They used one construction program for the entire SFP facility of which SFPs A & B were completed. SFPs A & B are Code-compliant and the Code-required documentation exists. The same work procedures and work control procedures were used for the construction of all four SFPs (A, B, C, & D). After cancellation of Units 2, 3, and 4, the licensee inadvertently discarded field weld documentation for these 49 SFPs C & D field welds. In a letter dated April 30, 1999, the licensee provided specific procedural and program controls that traced the responsibility for verifying the completion of Code requirements to specific personnel. The procedural and program controls established QA oversight of each step in the construction process that required documentation.

The staff performed an on-site inspection that reviewed past QA procedures, the past construction program, current QA procedures, and current construction program (ref. 17). The purpose of the review of past QA procedures and the construction program was to establish that the documents that are now missing had been completed at the time of construction. The purpose of the review of current QA procedures and construction program was to verify Code compliance of documents recreated by means of re-inspections and the program being used to complete the construction of the SFPs C & D FPCCS and CCWS piping. The review encompassed the records used by the licensee as the bases for the alternative plan and included interviews with past and current Authorized Nuclear Inspectors (ANIs). (The ANI is an independent third party with the responsibility for verifying that construction is being performed according to Code requirements.) The review provided objective evidence of the acceptability of the past and current construction documents and past and current construction programs. The objective evidence consisted of construction and QA records signed off by the ANIs and re-inspections performed as part the alternative plan. Based on the above review of documentation and the on-site inspection, the staff concluded that the objectives of ND-2150 and ND-4122 were satisfied.

ND-4323 requires that only qualified procedures and qualified welders can be used for welding. The licensee stated that they used an ongoing construction program for the entire SFP facility. The same work procedures, work control procedures, welder pool, and welding procedures were used for construction of the four SFPs. During the on-site inspection, the staff reviewed welding records for embedded welds from SFPs A & B that were similar to welds in SFPs C & D. The review included WDRs, welder qualification records, weld quality control inspector records, NDE examiner qualification records, welding procedure specifications (WPS), and procedure qualification records (PQRs) for stainless steel welds. These SFPs A & B construction records were retrievable, legible, and complete. For SFPs C & D, the staff reviewed visual and liquid penetrant re-inspection records and the welder symbols retrieved from accessible welds. The records from the re-inspections and the selected original construction welder qualification records were retrievable and found to be in order. The staff's review provided objective evidence that a detailed quality program was in place and was followed during construction. Based on the data reviewed during the inspection and the existence of an acceptable QA program at the time of construction, the staff concluded that SFPs C & D welds were made by qualified personnel using qualified procedures in accordance with the objectives of ND-4323.

ND-4232.2(b) requires chemical analyses of filler metals or weld deposits be known. In a letter dated April 30, 1999 (ref. 4), the licensee provided information showing chemical checks for 21 accessible FPCCS welds (two of these welds were later replaced with Code welds) and 12 hanger-to-pipe FPCCS welds. The chemical checks were performed on the surface of the welds using an X-ray fluorescence analyzer. During the on-site inspection, the staff observed a performance demonstration of the X-ray fluorescence analyzer's ability to discriminate between different types of stainless steel and chemical extremes within a stainless steel type. The licensee's chemical checks verified that the welds were made from stainless steel filler metal. The licensee corroborated the chemical checks with chemical analyses performed on samples removed from three of the SFPs C & D FPCCS welds. During the inspection (ref. 17), the staff compared the chromium and nickel from the chemical analyses with filler metals used for similar welds in SFPs A & B, and determined that they were similar and within the ranges of Type 308 stainless steel. The chemical analyses provided objective verification that stainless steel filler metal was used for welding FPCCS piping. Based on the chemical checks, chemical analyses, comparison with SFPs A & B, and an acceptable QA program during construction, the staff concluded that the alternative for the FPCCS welds provided an acceptable level of quality and safety with respect to the chemical composition requirements of ND-2432.2(b).

In a letter dated October 29, 1999 (ref. 9), the licensee provided chemical analyses that were run on samples from the three SFPs C & D CCWS welds. During the on-site inspection, the licensee stated that they used SFA-5.1, E7018 filler metal for welding SFPs C & D CCWS piping. The staff compared the chemical analyses with the filler metal chemistry for SFA-5.1, E7018 and determined that they were similar. Based on the chemical analyses and the existence of a QA program during construction, the staff concluded that the alternative plan for the CCWS welds provided an acceptable level of quality and safety with respect to the chemical composition requirements of ND-2432.2(b).

ND-2433.2 requires that the filler metal for stainless steel have a ferrite number (FN) greater than 5. FN is an indication of resistance to cracking in the weld. In a letter dated October 29, 1999 (ref. 9), CP&L stated that they performed FN tests on 18 accessible FPCCS welds. The test results ranged from 4 to 9 (rounded off). This variation is in agreement with variations listed in paragraph A6.2 to SFA-5.4 of the 1995 Edition of Code which states that a specimen averaging 5.0 percent ferrite (based on data collected from participating laboratories) ranged in FN measurements from 3.5 to 8.0 percent. Paragraph A6.3 to SFA-5.4 of the 1995 Edition of Code states that ferrite variations from weld to weld must be expected due to slight changes in welding and measuring variables. Based on the information from SFA-5.4 and the above test results, the staff considers the FN measurements reasonable, and the measurements support the preceding conclusion that an acceptable austenitic stainless steel filler metal was used for FPCCS welds.

ND-4322.1 requires that welder identification symbol be affixed near their welds. The welder symbol is used for tracing generic implications associated with defective welds. The generic implications are normally detected by nondestructive examinations during construction. The licensee demonstrated that welder symbols were affixed near the welds during construction by retrieving them from the accessible welds for inclusion on new WDRs. For the embedded welds, the welder symbols are not retrievable. As discussed above, the licensee constructed SFPs C & D under a QA program that was in compliance with the Code. Based on the licensee’s retrieval of the welder symbols from the accessible welds and their QA program requirements, the staff concluded that welder symbols were properly used. Because the welder identification symbol is a construction QA tool, the staff concludes the objectives of ND-4322.1 were satisfied.

ND-4231 specifies maximum alignment tolerances for pipe welds and the disposition of tack welds in the set-up of the weld joint. As discussed above, the licensee followed a QA program during construction. The QA program includes procedure CQC-19, “Welding Control,” Revision 0, with Exhibit 1, “Weld Inspection Checklist.” This checklist requires the welding QA/QC inspector to inspect the fit-up, alignment, and the removal of tack welds. To demonstrate that the alignment was in compliance with Code, the licensee performed a visual examination from the outside surface of all accessible welds and from the inside surface of the embedded welds with a camera. From these examinations, the licensee determined that the fit-up and alignment were acceptable. During the on-site inspection, the staff visually examined the outside surfaces of welds 2-CC-3-FW-207, 2-CC-3-FW-208, and 2-CC-3-FW-209, and determined that the fit-up and alignment looked satisfactory. The staff also looked at fit-up and alignment of the embedded welds recorded on the videotapes and determined that they looked satisfactory.

The staff did not detect evidence of the existence of any tack welds. Therefore, the tack welds were either consumed by the welding process or integrated into the weld. Based on the above discussion, the staff concluded that alignment, fit-up, and tack welds followed Code requirements. The missing documentation for alignment, fit-up, and tack welds have not affected weld quality. Therefore, the staff concludes the welds satisfy the objectives of ND-4231.

ND-4440 requires NDE of the welded surface. The NDE is normally performed on the exterior surface of the welds. As part of the alternative plan, the licensee proposed re-inspecting the exterior surfaces of the accessible welds and the interior surfaces of the embedded welds. A review by the staff of the re-inspection data generated by the licensee on the accessible welds showed that the welds were free of reportable flaws. In the absence of reportable flaws, the surfaces of the accessible welds were demonstrated to satisfy the NDE requirements of ND-4440.

As part of the alternative plan, the licensee examined all 15 embedded-field welds from inside the piping. The examination was performed using an enhanced remote visual technique. In a letter dated October 29, 1999 (ref. 9), the licensee described the visual examination as capable of assessing construction quality, pipe fit-up, pipe alignment, adequacy of a purge during welding, and fusion of the root pass. (An additional benefit of a visual examination is its ability to assess the entire piping system for corrosion and fouling that may have occurred during lay-up.) The licensee performed the visual examinations by sending a mobile video camera with focusing and magnifying capabilities to each embedded weld. The video camera sent images of the piping as it traveled to the welds and of each weld to a television monitor and video recorder.   A qualified level II visual examiner viewed the images and recorded any observations. The observations and videotapes were viewed by the level III visual examiner, the ANI, a contractor hired by the licensee (ref. 18), and the staff. Based on the reports generated by the individuals who viewed the observations and videotapes, the staff determined that there were no safety significant imperfections identified by the remote visual examinations. Certain minor imperfections were identified and, based on a detailed review of each, the staff concluded that they were innocuous. The staff concluded that the construction and the current condition of the embedded welds and piping are in accordance with the objectives of ND-4440 requirements.

ND-4452 and ND-4453 require that defects be removed, repaired, and examined. These paragraphs require that the examinations of surface repairs be performed using magnetic particle testing or liquid penetrant testing, and examinations of welded repairs be conducted using the original weld requirements. The licensee stated in their letter dated April 30, 1999 (ref. 4), that repair WDRs for defects identified during in-process welding may also be missing. The assurance that repair WDRs, if any, were completed is supported by the requirements in the licensee’s QA program, which stipulates that completed repair WDRs are verified by the QA inspectors and the ANI. The surface re-inspections performed by the licensee on all accessible welds and the visual inspection of the embedded welds showed no evidence of repairs. Any welds with undocumented repairs, therefore, were viewed from at least one surface. During its on-site inspection, the staff reviewed the licensee's surface inspection efforts and determined that the surface inspections were representative and in order. Based on the preceding discussion, the staff concluded that weld repairs, if any, were properly made and provided an acceptable level of quality.

From the evaluation and discussions above, the staff believes that the actions taken by the licensee to address the 46 SFPs C & D FPCCS welds with missing documentation, and 3 SFPs C & D CCWS welds with missing documentation provide an acceptable level of quality and safety with respect to the original Code construction requirements.

3.7.6 Summary

Based on the above evaluation, the staff concludes that the proposed alternative for the SFPs C & D FPCCS and CCWS welds with missing documentation provide an acceptable level of quality and safety. Pursuant to 10 CFR 50a(a)(3)(i), the staff authorizes the proposed alternative for the 46 SFPs C & D FPCCS welds with missing documentation and 3 SFPs C & D CCWS welds with missing documentation.

3.8 Quality Assurance

3.8.1 Introduction

The licensee's amendment request (ref. 1) included an alternative to certain ASME B&PV Code requirements. The licensee's Alternative Plan consists of two parts. The first part proposes an alternative to ASME documentation requirements, for which required records are missing. The second part proposes to achieve completion of the piping systems in accordance with an equivalent alternative to the original ASME Code requirements.

The formulation of an alternative to the requirements of the original ASME Code was dictated by the licensee's determination that some ASME Code construction documents were inadvertently destroyed. The alternative consists of inspections performed by a variety of examination methods, selective NDE, selective weld replacement, selective chemical analyses, and establishing with reasonable assurance that the missing documents had existed and had been found acceptable by the responsible parties prior to their destruction. In addition, the licensee proposes to complete construction of the ASME Code portion of the plant associated with SFPs C and D in accordance with the its approved 10 CFR Part 50, Appendix B QA program, as supplemented by requirements developed to conservatively reconcile differences from the ASME Code requirements used during original construction.

3.8.2 Background

Construction of HNP Units 1 and 2 was controlled under a single site-wide ASME QA program (ref. 4, Enclosure 5). All four SFPs and the SFPCCS and CCWS for pools A and B were turned over to the plant following completion. The four pools and cooling water systems are described in the HNP FSAR (ref. 13) and have been incorporated into the HNP Unit 1 operating license.

Construction of the cooling system for pools C and D, however, was discontinued following cancellation of HNP Unit 2. By the time HNP Unit 2 was canceled, the majority of mechanical equipment and piping associated with pools C and D was already installed.

Because construction of the cooling water and cleanup system for pools C and D was not complete, construction records were placed in temporary storage. As stated in the licensee's amendment request (ref.1, Enclosure 8), certain piping installation records were inadvertently destroyed. These records included installation verification data and field records for certain ASME Section III piping welds.

3.8.2.1 Regulatory Requirements

GDC 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

Regulatory Guide (RG) 1.26 (Ref. 20) is the principal document used in staff reviews for identifying, on a functional basis, quality standards for nuclear power plant components containing water, steam, or radioactive material. The HNP SFP piping for which certain required records have been lost are ASME Code, Section III, Class 3 components which are classified as "Quality Group C" in accordance with RG 1.26.

Regulatory requirements with respect to the application of the ASME Code for the design, fabrication, erection, construction, testing, and inspection of components for nuclear power plants are specified by §50.55a, "Codes and Standards." Section 50.55a requires that nuclear power plants meet the requirements of the ASME Code, Section III, Division 1 for Class 1, Class 2, and Class 3 components. Alternatives to these requirements may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. For proposed alternatives, the applicant must demonstrate that:

- The proposed alternatives would provide an acceptable level of quality and safety, or

- Compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.8.2.2 ASME Code Requirements Addressed by Alternative Plan

The licensee's submittal (ref. 1) provides a detailed description of the proposed alternatives to demonstrate compliance with ASME Code requirements for the cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i). The plan consists of two parts. The first part addresses the missing construction documentation associated with the piping installed during original plant construction and intended for the HNP SFP C and D cooling and cleanup system. The second part addresses completion of HNP SFP C and D under an alternative to an ASME N Certificate program.

Field fabrication and installation of the piping covered by the Alternative Plan was performed in accordance with the ASME Code Section III, 1974 Edition, Winter 1976 Addenda (the ASME

Code Edition and Addenda that were required by NRC regulations at the time of issuance of the HNP construction permit). Completion of the piping systems will also be performed in accordance with this edition of the ASME Code, as implemented by the licensee's Alternative Plan.

The ASME Code documentation requirements that cannot be satisfied as a result of the missing records are identified in Enclosure 7 of the licensee's April 30, 1999 RAI response, "Matrix of Code Requirements versus Missing Field Records" (ref. 4, Enclosure 7). The matrix identifies the specific section of the ASME Code, the deficiency, and method of reconciliation.

3.8.2.2.1 Alternative Plan for Missing Construction Records (Pedigree Piping Plan)

The plan addresses the missing construction documentation associated with the portion of the piping completed during original construction. To define the scope of the Alternative Plan, the boundaries of the piping system were determined through documented detailed field walkdowns, during which identification markings such as spoolpiece numbers, welder identification numbers and heat numbers were recorded. As-built isometric drawings were developed from the data collected from the walkdowns. Then, construction-era documents substantiating the quality of the piping were compiled. Where documentation was no longer available, reexamination and testing was performed. Finally, the compilation of original construction records and those generated by reexamination/ testing were reviewed against documentation required by the ASME construction Code. Where ASME Code-required documentation was missing, the body of evidence (e.g., alternate tests or inspections, construction procedures) was evaluated to determine if comparable quality and safety exists. Notably, the missing documentation includes piping isometric packages for approximately 40 of the nearly 200 large bore piping welds in the completed ASME Section III portions of the HNP FPCCS and CCWS for SFP C and D. The alternative plan addresses these 40 large bore field welds and 12 pipe hanger attachment welds.

3.8.2.2.2 Alternative Plan for Continuance of Design and Construction

The original construction of the HNP Nuclear Plant was subject to the full requirements of Section III of the ASME B&PV Code under the authorization of a single N Certificate program maintained by the licensee. The site ASME Section III QA program was discontinued shortly after completion and turnover of HNP Unit 1, and a corporate QA program meeting 10 CFR 50 Appendix B requirements was implemented as required to address plant operation, including implementation of ASME 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.

The licensee proposes to complete the design of this portion of the plant to meet applicable ASME Section III requirements, but to complete construction under the corporate QA program. Conflicts between the ASME QA program, used during original construction, and corporate QA program requirements will be conservatively dispositioned. The licensee has developed a set of supplemental QA requirements to augment the corporate QA program for completion of Code portions of the plant associated with SFPs C and D. These supplemental requirements were developed by review of the ASME Section III QA program used during original construction and are intended to conservatively reconcile differences between program requirements.

3.8.3 Evaluation

3.8.3.1 Alternative Plan for Missing Construction Records

This part of the Alternative Plan proposes to demonstrate that the originally installed equipment is acceptable for use. By walkdowns, compilation of original construction records, and by reexamination and testing, the licensee has determined that, except for certain field weld installation records, the piping systems are compliant with applicable ASME Code requirements. Because these records are not available, the cooling and cleanup system for HNP SFPs C and D cannot be demonstrated to have been completed in accordance with the applicable requirements of Section 50.55a.

The welds for which required documentation is missing are identified in the weld matrix submitted as Enclosure 3 to Reference 4. The data was recorded on WDRs, which contained information pertaining to weld attributes, including identification of the items being welded, specification of the weld procedure specification, welder identification, filler metal material identification, nondestructive examination requirements, and signatures attesting to satisfactory completion of activities associated with the welds. These signatures include that of the ANI, an independent third party representing the nuclear insurer, who verifies that construction activities are performed in compliance with ASME Code requirements.

The weld matrix identifies the 40 large bore pipe welds and 12 hanger attachment welds for which records are missing. Of the 40 piping welds, 37 are FPCCS welds (15 of which are embedded in concrete) and three are CCWS welds. All 52 welds were either reinspected or replaced (the 15 welds located on piping embedded in concrete were examined by remote camera). The staff evaluation and acceptance of weld reinspection and testing is addressed in Section 3.7 of this Safety Evaluation. In addition to reinspection and testing, the licensee compiled a substantial body of evidence to substantiate the acceptable quality of these welds. The staff reviewed much of this evidence (Ref. 1, 4, 9) as part of its evaluation, in addition to other plant records made available to NRC inspectors (Ref. 17, 19). The following summary is based on review of this material.

The NRC evaluation of the designs of HNP Unit 1 and Unit 2, including the four SFPs, is reported in NUREG-1038 (ref. 12). The four SFPs are located in the same building, and work was performed simultaneously on all four pools by a common pool of craft, quality control, and engineering resources. Only the cooling systems for pools C and D had not been turned over. The four pools were turned over to the operating organization after completion of construction, and are part of the HNP Unit 1 operating license. The pools and the cooling and cleanup system for SFP A and B are described in Chapter 9 of the FSAR for HNP Unit 1 (ref. 13). Pools A and B have been in operation since plant startup in 1987.

Construction of HNP Units 1 and 2 was controlled under a single site-wide QA program. The ASME QA Program Manual (ref. 4, enclosure 5), effective during the construction period, was reviewed as part of this evaluation. The site-wide QA program and implementing quality procedures effective during the period when major welding activities were ongoing were also reviewed as part of a special inspection conducted to evaluate the Alternative Plan (Ref. 17). Based on this review, the staff concluded that the QA program provided adequate control of the welding process and that holdpoints were adequate to ensure the quality of welds before proceeding to a subsequent activity. These holdpoints included a detailed review of weld documentation to assure the welds had been completed in accordance with technical, Code, and regulatory requirements. For welds to be embedded in concrete, an additional holdpoint was provided to ensure that welds met all requirements prior to placement of concrete.

In addition to effective control of the welding process by the QA program, construction-era documents provide additional assurance that the subject welds were of acceptable quality. These records include hydrostatic test (hydrotest) records, concrete placement reports, and reports generated by QA personnel.

Hydrostatic tests were verified by QA personnel who verified that all tests, inspections, and documentation required by the Code were complete. The tests were also witnessed by an ANI. In addition to verifying that welds within the scope of a hydrotest met acceptance criteria, the hydrostatic test records also provide evidence that the welds were completed, inspected, and documented in accordance with the licensee's QA program. For the 15 embedded welds that cannot be reinspected directly, the staff reviewed hydrostatic test records for 13 of these welds. For the remaining two welds, the staff reviewed to its satisfaction corrective action documents identifying that these hydrostatic tests had been completed (ref. 17). These records provide evidence that the missing WDRs had been reviewed prior to performance of the hydrostatic tests.

Concrete placement records provide further evidence that the embedded welds were completed in compliance with Code requirements. Since embedding piping in concrete represented a point at which piping was no longer accessible for inspection or rework, procedural controls were established to ensure that all required work activities had been completed and that documentation was in order prior to authorizing concrete placement. For the 15 embedded welds, the staff reviewed and verified that the associated concrete placement reports were properly completed and signed.

Additional supporting documentation related to weld inspection activities is identified in the weld matrix. These documents were reviewed and found to support the conclusion that inspection of activities related to the subject welds was adequate (ref. 17).

In addition to the documents identified in the weld matrix, a sample of quality-related reports were reviewed to assess the effectiveness of QA program implementation. Based on review of these documents, it was concluded that inspection personnel actively monitored welding activities and processes for compliance with ASME and QA program requirements. Deficiencies were accurately reported, appropriately resolved, and corrective actions promptly taken. All corrective action documents reviewed were in compliance with the licensee's QA program and NRC requirements.

Finally, NRC inspection reports during the construction period from 1978 through 1983 were reviewed. This was the period during which the SFPs were under construction. The inspection reports document more than 50 separate inspections for this period for items related to the welding program and/or piping installation. Although several violations dealt with the general subject of welding, these violations were relatively minor and were resolved through the licensee's corrective action program.

3.8.3.2 Alternative Plan for Continuance of Design and Construction

This part of the Alternative Plan proposes to demonstrate that continuance of construction is technically acceptable and will ensure the requisite level of quality and safety in the completed systems.

The N Certificate program originally maintained by the licensee was discontinued after completion and turnover (to operations) of HNP Unit 1. The site is currently operating under an approved corporate QA program meeting 10 CFR Part 50 Appendix B requirements (ref. 4, Enclosure 14). (The corporate QA program is used in conjunction with site-specific QA requirements, documented in Chapter 17.3 of the HNP FSAR (ref. 13).) The corporate QA program includes elements of quality control (i.e., welder qualification, weld procedures, inspections, documentation, etc.) consistent with the original construction program. The HNP ASME Section XI program is implemented under the corporate QA program.

The licensee proposes to complete the design of the CCWS and FPCCS systems for pools C and D in accordance with the applicable ASME Section III requirements, but to use the corporate QA program and site procedures for those elements of quality assurance for which it is appropriate. A set of supplemental QA requirements has been developed to augment the corporate QA program for completion of the ASME Code portions of the plant associated with the C and D pools. These requirements were developed by a close comparison of the requirements in the approved ASME Section III Construction QA Program Manual as it existed at the time of construction.

The staff has conducted onsite inspections of the licensee's program for completing construction of pools C and D. The inspections included a review of the engineering documents prepared to complete construction, the construction and quality control program and procedures that control piping and pipe support installation, a walkdown inspection to examine completed work, the construction records documenting installation and inspection of the new piping and pipe supports, and the licensee's program for commissioning equipment for the C and D SFP. The inspectors found that the licensee has a comprehensive program to control and inspect piping installation and welding in accordance with Section III of the ASME Code and NRC requirements (ref. 17,19).

With respect to documentation requirements, Section 14 of the licensee's corporate QA program (ref. 4, Enclosure 14) establishes the requirements for accumulation, maintenance, and retention of QA records associated with HNP and establishes requirements for control of documents relative to activities affecting quality.

The licensee's commitments to NRC RGs are documented in Chapter 1.8 of the HNP FSAR (ref. 13). Notably for this evaluation, the licensee's QA program complies with RG 1.33 (ref. 21), and its QA recordkeeping program complies with RG 1.88 (ref. 22).

The licensee has submitted copies of the ASME Section III QA program, the corporate QA program, the supplemental requirements, and a comparison of the corporate QA and ASME QA programs, with comments on how these differences are reconciled by the supplemental QA requirements (ref. 4, Enclosures 5, 14, 16, 17). Based on review of these documents conducted as part of this evaluation, the staff has determined that the licensee's alternate approach to §50.55a applicable requirements provides an acceptable level of quality and safety.

3.8.4 Summary

Based on the above evaluation, the staff concludes that, pursuant to the provisions of 10 CFR 50a(a)(3)(i), the alternatives proposed by the licensee for continuance and completion of the piping systems associated with HNP SFPs C and D are acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if the operation of the facility, in accordance with the 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 provided its analysis of the issue of no significant hazards consideration in its December 23, 1998, amendment request (ref. 1). The staff reviewed the licensee's analysis and, based on its review, it appeared that the three standards of 10 CFR 50.92(c) were satisfied. Therefore, the NRC staff proposed to determine that the amendment request involves no significant hazards consideration, and published its proposed determination in the *Federal Register* for public comment on January 13, 1999 (64 FR 2237).

The staff has completed its evaluation of the licensee's proposed amendment as discussed in Section 3.0 above. Based on its evaluation, the staff has determined that the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in a margin of safety. The following staff evaluation in relation to the three standards of 10 CFR 50.92 supports the staff's final no significant hazards consideration determination.

First Standard:

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

The following postulated accidents and events involving spent fuel storage were identified and evaluated by the licensee. The staff likewise evaluated the same accidents and events.

1. a spent fuel assembly drop in a SFP
2. loss of SFP cooling flow
3. a seismic event
4. misloaded fuel assembly

The probability that any of the accidents in the above list can occur is not significantly increased by the proposed changes. The probabilities of a seismic event or loss of SFP cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misloadings are primarily influenced by procedures and equipment used for handling the fuel. Fuel handling is not a random event; it is strictly controlled using approved procedures, trained personnel, and specialized equipment. Using these methods, the probability of a fuel handling accident (fuel assembly drop or misloading) is minimized, and increasing the number of times it is done will not cause a significant increase in the probability of an accident.

In its submittal, the licensee re-evaluated the consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the SFP. The licensee found that the structural damage to the FHB, 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 staff reviewed the licensee's evaluation in Section 3.3 of this safety evaluation and accepts the licensee's finding. Similarly, the radiological dose at the exclusion area boundary will not be increased from those previously considered, since the pertinent fuel parameters remain unchanged. The staff reviewed the licensee's analysis as discussed in Section 3.6 of this safety evaluation. Based on its review, the staff concluded that bounding scenario for the postulated fuel handling accident in the FHB does not change due to the addition of storage racks in SFPs C and D. Therefore, the inputs and assumptions for the dose consequences do not change, and the current fuel handling accident dose assessments in the HNP FSAR (ref. 13) remain bounding. On this basis, the staff concluded that the consequences of this type of previously evaluated accident are not significantly increased by the proposed change.

The staff evaluated the consequences of a loss of SFP cooling event in Section 3.2 of this evaluation. On the basis of their review, the staff determined that sufficient time is available for plant operators to take mitigating actions to restore cooling prior to the pool boiling. In addition, sufficient makeup capability is available should boiling occur. Thus, the consequences of this type of accident are not significantly increased from previously evaluated loss of cooling events.

The staff evaluated the consequences of a design basis seismic event in Section 3.4 of this evaluation. On the basis of their review, the staff concluded that the licensee's structural analysis and design of the spent fuel rack modules is acceptable and that the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR (ref. 13) and applicable provisions of the SRP (ref. 14)., and are, therefore, acceptable. Thus, the consequences of a seismic event are not significantly increased from previously evaluated events.

The staff evaluated the consequences of fuel misloading and mislocation events in Section 3.1 of this evaluation. In their evaluation, the staff states that while most abnormal storage conditions will not result in an increase in the k-eff of the racks, its is possible to postulate events which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of soluble boron in the pool water based upon the double contingency principle which requires at least two unlikely, independent, concurrent events to occur before a nuclear criticality accident is possible. Therefore, since soluble boron is normally present in the SFP water, credit for soluble boron may be assumed in evaluating other accident conditions such as the misloading of fresh fuel. Plant procedure CRC-001 requires that the soluble boron concentration in the pool be maintained between 2000 and 2600 ppm and is confirmed by monthly surveillance measurements. The negative reactivity credited to the boron more than offsets the reactivity addition caused by credible accidents. In fact, Holtec has determined that a soluble boron concentration of only 400 ppm would be sufficient to maintain k-eff less than 0.95 even if a fresh PWR assembly were inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1. Thus, the consequences of fuel misloading and mislocation events are not significantly increased from previously evaluated events.

<u>Second Standard</u>:

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

As noted in various sections of this safety evaluation, the staff evaluated the proposed changes in accordance with appropriate NRC Regulatory Guides, SRP sections, and industry codes and standards. In addition, the staff has previously prepared several safety evaluations for rerack applications which are similar to this proposal. No unproven techniques or methodologies were used in the analysis and design of the racks to be used in SFPs C and D. No unproven technology will be used in the installation of the racks.

In its analysis, the licensee conservatively considered an accidental drop of a rack module during construction activity in SFPs C and D as an event which might represent a new or different kind of accident. However, this event had been considered previously by the licensee for rack installation in SFP B, which used the same equipment and procedures that will be used for installation activities in SFPs C and D.

The staff evaluated the handling of heavy loads and spent fuel assemblies in Section 3.3 of this evaluation. On the basis of their review, the staff determined that the licensee’s use of the 10-ton single-failure-proof FHB auxiliary crane, the 20-ton hoist, the spent fuel rack lifting rig, and administrative controls and procedures that are in accordance with NUREG-0612 and ANSI N14.6, will help to maintain safety during the installation of the new racks. The reliability of the crane coupled with the design, testing and inspection of the crane, the lifting rig and other lifting devices will enable the licensee to handle safely the racks and other heavy loads during the rack installation process. The postulated accident analyses involving a dropped spent fuel storage rack and SFP gate indicated that the SFP liner could be breached. However, during such a breach, the licensee could maintain the pool and its contents within the acceptable consequence limits set forth in NUREG-0612. In addition, the licensee’s use of administrative controls and procedures to improve the handling and control of heavy loads, including the racks, enhances the licensee’s capability to reduce the potential for load drops. In addition, the staff reviewed the licensee’s spent fuel shipping cask handling operations and determined that they will not occur over the SFPs. Therefore, movement of the cask will not impact the new racks nor the stored spent fuel configuration in the pools. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

Third Standard:

“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.

In evaluating a potential reduction in the margin of safety, the licensee addressed the safety issues related to the expanded pool storage capacity in the following areas. The staff likewise evaluated the same areas.

1. material, mechanical and structural considerations
2. nuclear criticality considerations
3. thermal-hydraulic and pool cooling considerations

The staff evaluated the material, mechanical and structural considerations of the proposed amendment in Sections 3.4 and 3.5 of this evaluation. Based on its evaluation the staff determined that the materials used in the fabrication of the spent fuel racks manufactured by Holtec are compatible with the SFP environment at HNP and that the type of degradation exhibited by the racks does not affect their neutron-absorbing capability. The staff concluded, therefore, that the materials used in the new spent fuel racks are acceptable. With respect to mechanical and structural considerations, the staff concluded that the licensee’s structural analysis and design of the spent fuel rack modules and the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR (ref. 13) and applicable provisions of the SRP (ref. 14), and are, therefore, acceptable. Thus there is no significant reduction in margin of safety related to the material, mechanical and structural considerations.

The staff evaluated the nuclear criticality aspects of the proposed amendment in Section 3.1 of this evaluation. The NRC acceptance criterion for subcriticality is that the effective multiplication factor (k-eff) in the spent fuel pool storage racks when fully flooded by unborated water shall be no greater than 0.95, including uncertainties at a 95 percent probability, 95 percent confidence level (95/95) under all conditions. On the basis of their review, the staff determined that the analysis methods used are acceptable and capable of predicting the reactivity of the HNP storage racks with a high degree of confidence. Therefore, the staff concluded that the criticality aspects of the proposed storage capacity expansion for HNP spent fuel pools C and D are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. Thus there is no significant reduction in margin of safety related to nuclear criticality considerations.

The staff evaluated the thermal-hydraulic and pool cooling aspects of the proposed amendment in Section 3.2 of this evaluation. On the basis or their review, the staff determined that there is sufficient thermal margin in the CCW and ESW systems to maintain the bulk fuel pool coolant temperature in all SFPs within their design limits assuming an additional decay heat load of 1 Mbtu/hr in SFPs C and D, and assuming a single active failure. In addition, given the decay load in SFPs C and D will be limited to 1 Mbtu/hr, the staff concluded that sufficient time is available for plant operators to take mitigating actions prior to pool boiling in the event of a loss of SFP cooling. Thus, there is no significant reduction in margin of safety related to thermal-hydraulic and pool cooling considerations.

On the basis of the above evaluation, the NRC has made a final determination that the proposed amendment does not involve a significant hazards consideration.

5.0 COMMENTS ON PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Several comments were received in response to the staff's January 13, 1999, proposed no significant hazards consideration determination (64 FR 2237). These comments and the staff's response are grouped into subject area categories and addressed below.

5.1 Risk/Accident Concerns

Comment - Several comments questioned how the probability or consequences of fuel handling accidents would not be significantly increased if the amount of fuel being moved and stored would be more than double what is currently licensed.

Response - As described in Section 4 above, the risk of a fuel handling accident is primarily influenced by procedures and equipment used for handling the fuel. Fuel handling is not a random event; it is strictly controlled using approved procedures, trained personnel, and specialized equipment. Using these methods, the probability of a fuel handling accident is minimized, and increasing the number of times it is done will not cause a significant increase in the probability of an accident.

As for the consequences of an accident, the fuel handling accident analysis for HNP was re-evaluated for the proposed changes. The resulting consequences are bounded by the existing analysis since the pertinent fuel parameters remain unchanged.

5.2 Cooling System Capacity

Comment - Several comments questioned the capacity of the existing HNP cooling systems to support the additional heat load of SFPs C and D.

Response - The staff's evaluation of the adequacy of the cooling system is discussed in section 3.2 above. Based on its review the staff concluded that there is sufficient thermal margin in the CCW and ESW systems to maintain the bulk fuel pool coolant temperature in all SFPs within their design limits, assuming an additional decay heat load of 1 Mbtu/hr in SFPs C and D, and assuming a single active failure.

5.3 Piping Concerns

Comment - Several comments questioned the adequacy of CP&L's alternative plan for demonstrating that the CCW and FPCCS piping to support SFPs C and D is capable of performing its design function.

Response - The licensee's alternative plan is discussed in detail in sections 3.7 and 3.8 above. Based on its review, the staff concluded that the proposed alternative for the SFPs C and D FPCCS and CCW system welds with missing documentation provides an acceptable level of quality and safety.

5.4 Transportation Concerns

Comment - A few comments questioned the safety of transporting spent fuel from CP&L's other two reactor sites (Brunswick and Robinson) for storage at HNP.

Response - The Operating License for HNP, which was issued in January 1987, authorized CP&L to receive and store fuel from Brunswick and Robinson at HNP. This amendment does not change that approval. CP&L can continue to ship fuel from its other sites using approved procedures as it has in the past.

5.5 Requests for Public Meeting/Hearing

Comment - Several comments requested that public hearings/meetings be held to discuss CP&L's proposed amendment.

Response - In response to the January 13, 1999, *Federal Register* Notice, BCOC filed a request for a hearing. As discussed previously, an ASLB panel was formed and BCOC was granted its hearing request. As part of the hearing process, a public pre-hearing conference was held in Chapel Hill on May 13, 1999, and Limited Appearance Statements were heard on December 7 and 8, 1999, in both Raleigh and Chapel Hill, North Carolina. The ASLB heard oral arguments on the admitted technical contentions on January 21, 2000, in Rockville, Maryland. The ASLB also heard oral arguments on the environmental contention on December 7, 2000, in Raleigh, North Carolina. The environmental contention was admitted by the ASLB on August 7, 2000, in response to BCOC's January 31, 2000, filing.

In addition, in response to the requests for a meeting, the staff held a public meeting in Raleigh, North Carolina, on February 28, 2000, to discuss its license amendment review process, and the scope and status of its review of CP&L's request. The majority of this meeting was used for public comments and questions. The staff issued a summary of this meeting on April 19, 2000.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on December 21, 1999 (64 FR 71514). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

Amendment Related

1. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Request for License Amendment Spent Fuel Storage," dated December 23, 1998. (Accession number 9812290056)

2. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Spent Fuel Storage Re-Designation of Proprietary Information," dated March 15, 1999. (Accession number 9903220064)

3. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Spent Fuel Storage - Page Additions - Holtec Report," dated April 5, 1999. (Accession number 9904130221)

4. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding the Alternative Plan for Spent Fuel Pools Cooling and Cleanup System Piping," dated April 30, 1999. (Accession number 9905050200).

5. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding the License Amendment Request to Place HNP Spent Fuel Pools C & D in Service," dated June 14, 1999. (Accession number 9906210117).

6. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding Amendment Request to Increase Fuel Storage Capacity by Placing Spent Fuel Pools C & D in Service," dated July 23, 1999. (Accession number 9907270169).

7. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, “Response to NRC Request for Additional Information Regarding Amendment Request to Increase Fuel Storage Capacity,” dated September 3, 1999. (Accession number 9909100158).

8. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, “Supplemental Information Regarding the License Amendment to Place HNP Spent Fuel Pools C & D in Service,” dated October 15, 1999. (Accession number 9910270013).

9. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, “Response to NRC Request for Additional Information Regarding the Alternative Plan for Spent Fuel Pools C & D Cooling and Cleanup System Piping,” dated October 29, 1999. (Accession number ML993160242).

10. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, “Response to NRC Request for Additional Information (RAI) Regarding Rack Installation Spent Fuel Pools C & D,” dated April 14, 2000. (Accession number ML003707409).

11. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, “Supplemental Changes to License Amendment Request - Spent Fuel Storage,” dated July 19, 2000. (Accession number ML003734906).

12. NUREG 1038, “Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2,” dated November 1983. Supplement 1 dated June 1984; Supplement 2 dated June 1985; Supplement 3 dated May 1986; Supplement 4 dated October 1986.

13. Final Safety Analysis Report - Shearon Harris Nuclear Power Plant.

14. NUREG 0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition.”

15. Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” Rev. 1, December 1975 (for comment); Rev. 2, December 1981 (for comment).

16. “OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,” dated April 14, 1978.

17. NRC inspection Report No.: 50-400/99-12, December 28, 1999. (Accession number ML003673416)

18. G. J. Licina, “Evaluation of Embedded Welds in Spent Fuel Piping at Harris Nuclear Plant,” Structural Integrity Associates, Inc., San Jose, CA, Report No. SIR-99-127, Rev. 0 and Rev. 2 dated December 1999. (Accession number ML003712432 (Rev. 0); ML003712470 (Rev. 2).

19. NRC inspection Report No.: 50-400/2000-05, February 16, 2000. (Accession number ML003685113)

20. Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants,” Revision 3, February 1976.

21. Regulatory Guide 1.33, Revision 2, “Quality Assurance Program Requirements (Operation), February 1978.

22. Regulatory Guide 1.88, Revision 2, “Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records,” October 1978.

<u>ASLB Hearing Related</u>

23. Orange County’s Request for Hearing and Petition to Intervene, February 12, 1999. (Accession number 9902250036)

24. Orange County’s Supplemental Petition to Intervene, April 5, 1999. (Accession number 9904080103)

25. NRC Staff’s Response to Orange County’s Supplemental Petition to Intervene, May 5, 1999. (Accession number 9905070018)

26. Applicant’s Answer to Petitioner Board of Commissioners of Orange County Contentions, May 5, 1999. (Accession number 9905100006)

27. Transcript of Pre-Hearing Conference RE: Carolina Power & Light Company, May 13, 1999. (Accession number 9905200044)

28. ASLBP No. 99-762-02-LA; Memorandum and Order (Ruling on Standing and Contentions), July 12, 1999. (Accession number 9907130054)

29. ASLBP No. 99-762-02-LA; Memorandum and Order (Granting Request to invoke 10 CFR Part 2, Subpart K Procedures and Establishing Schedule), July 29, 1999. (Accession number 9908020101)

30. Detailed Summary of Facts, Data, and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant, January 4, 2000. (Accession number ML003672688)

31. Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely at the Subpart K Oral Argument, January 4, 2000. (Accession number ML003672886)

32. NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff Proposes to Rely at Oral Arguments on Technical Contentions 2 and 3, January 4, 2000. (Accession number ML003673204)

33. Transcript of Oral Arguments RE: Carolina Power & Light Company, January 21, 2000. (Accession number ML003679424)

34. ASLBP No. 99-762-02-LA; Memorandum and Order (Ruling on Designation of Issues for an Evidentiary Hearing), May 5, 2000 (LBP-00-12). (Accession number ML003712091)

35. Orange County's Request for Admission of Late-Filed Environmental Contentions, January 31, 1999[2000]. (Accession number ML003680571)

36. Carolina Power & Light Company; Shearon Harris Nuclear Power Plant, Unit 1, Environmental Assessment and Finding of No Significant Impact; Federal Register, December 21, 1999 (64 FR 71514)

37. Applicant's Response to BCOC's Late-Filed Environmental Contentions, March 3, 2000. (Accession number ML003690965)

38. NRC Staff Response to Intervenor's Request for Admission of Late-Filed Environmental Contentions, March 3, 2000. (Accession number ML003690415)

39. Orange County's Reply to Applicant and Staff's Oppositions to Request for Admission of Late-Filed Environmental Contentions, March 13, 2000. (Accession number ML003694928)

40. Orange County's Petition for Review of LBP-00-12, May 22, 2000. (Accession number ML003719114)

41. NRC Staff Response to Orange County's Petition for Review of LBP-00-12, June 6, 2000. (Accession number ML003735783))

42. Applicant's Answer Opposing Commission Review of LBP-00-12, June 6, 2000. (Accession number ML003723761)

43. Commission Memorandum and Order (CLI-00-11) in response to BCOC's May 22, 2000, Petition, June 20, 2000. (Accession number ML003725065)

44. ASLBP No. 99-762-02-LA; Memorandum and Order (Ruling on Late-Filed Environmental Contentions), August 7, 2000 (LBP-00-19). (Accession number ML003738358)

45. Detailed Summary of Facts, Data, and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with respect to the need to prepare an Environmental Impact Statement to Address the Increased Risk of a Spent Fuel Pool Accident, November 20, 2000. (Accession number ML003772525)

46. Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely at the Subpart K Oral Argument Regarding Contention EC-6, November 20, 2000. (Accession number ML003771805)

47. NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff Proposes to Rely at Oral Arguments on Environmental Contention 6, November 20, 2000. (Accession number ML003771530)

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Date: December 21, 2000

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-400

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 103 to Facility Operating License No. NPF-63 issued to Carolina Power & Light Company (CP&L, the licensee), which revised the Technical Specifications (TS) for operation of the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), located in Wake and Chatham Counties, North Carolina. The amendment is effective as of the date of issuance.

The amendment modified the TS to support a modification to HNP 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. Specifically, the amendment consists of: 1) a revision to TS 5.6 to identify pressurized water reactor fuel burnup restrictions, boiling water reactor fuel 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; 2) 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 component cooling water (CCW) and SFPs C and D cooling and cleanup system piping; and 3) an unreviewed safety question for additional heat load on the CCW system.

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

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on January 13, 1999 (64 FR 2237). A request for a hearing was filed on February 12, 1999, by the Board of Commissioners of Orange County, North Carolina (BCOC).

On July 12, 1999, the Atomic Safety and Licensing Board (ASLB) ruled that BCOC had standing and had submitted two admissible contentions. The two contentions related to (1) whether General Design Criterion 62 allows the use of administrative controls to prevent criticality (TC-2); and (2) the adequacy of the licensee's proposed alternative plan for the cooling system piping (TC-3). On July 29, 1999, the ASLB granted CP&L's request to hold the hearing in accordance with the hybrid hearing procedures of 10 CFR Part 2, Subpart K. On January 4, 2000, all parties filed written summaries and on January 21, 2000, the ASLB heard oral arguments related to the two admitted contentions. On May 5, 2000, the ASLB issued a decision in favor of CP&L, stating that "(1) there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing; and (2) contentions TC-2 and TC-3 are disposed of as being resolved in favor of CP&L."

On January 31, 2000, BCOC filed four late-filed environmental contentions that challenged the adequacy of the staff's December 21, 1999, environmental assessment related to CP&L's amendment request. On March 3, 2000, the NRC and CP&L responded to the late-filed contentions, and on March 13, 2000, BCOC submitted its reply to the responses. On August 7, 2000, the ASLB issued its Ruling on Late-filed Environmental Contentions. In its ruling, the ASLB admitted one environmental contention (EC-6) regarding the probability of occurrence of BCOC's postulated accident scenario. On November 20, 2000, all parties filed written summaries and on December 7, 2000, the ASLB heard oral arguments related to EC-6.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding or completion of any required hearing, where it has determined that no significant hazards considerations are involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards considerations. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any hearing will be held after issuance.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (64 FR 71514).

For further details with respect to the action see (1) the application for amendment dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, (2) Amendment No. 103 to License No. NPF-63, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 21st day of December 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

advance of the holding or completion of any required hearing, where it has determined that no significant hazards considerations are involved.

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Dated at Rockville, Maryland, this 21st day of December 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

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Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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OFFICE	PM:PDII/S2	LA:PDII/S2	OGC	SC:PDII/S2	
NAME	RLaufer	Dunnington	SUttal *	RCorreia *	
DATE	11 / 20/00	11/20 /00	8 / 15 /00	11 / 16 /00	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

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