NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA: 2000-0170

RESPONSE NUMBER: 1

RESPONSE TYPE: ☑ FINAL ☐ PARTIAL

REQUESTER: Mr. Jim Warren

DATE: APR 10 2000

PART I. -- INFORMATION RELEASED

☐ No additional agency records subject to the request have been located.

☐ Requested records are available through another public distribution program. See Comments section.

☐ APPENDICES: Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.

☐ APPENDICES: Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.

☐ Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, NW, Washington, DC.

☑ APPENDICES A: Agency records subject to the request are enclosed.

☐ Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.

☐ We are continuing to process your request.

☑ See Comments.

PART I.A -- FEES

AMOUNT * $

* See comments for details

☐ You will be billed by NRC for the amount listed.

☐ You will receive a refund for the amount listed.

☐ None. Minimum fee threshold not met.

☑ Fees waived.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

☐ No agency records subject to the request have been located.

☐ Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.

☐ This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

You have requested that this Freedom of Information Act (FOIA) request be continuing in nature. For your information, a request made pursuant to the FOIA applies only to records in the possession of the agency on the date the request was received. You may, however, submit new FOIA requests at any time. This completes NRC's action on this request.

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Carol Ann Reed Carol Ann Reed

FOIA-2000-0170

APPENDIX A
RECORDS BEING RELEASED IN THEIR ENTIRETY
(If copyrighted identify with *)

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	Undated	Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools (278 pages)
2.	01/04/00	Minutes - Exhibits Supporting the Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely (73 pages)
3.	01/04/00	Minutes - Exhibits Supporting the Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely (200 pages)
4.	06/14/99	Letter to USNRC from D. B. Alexander; re: Response to NRC Request for Additional Information w/enclosures (152 pages)

Enclosure 7 to Serial: HNP-98-188

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

LICENSING REPORT FOR EXPANDING STORAGE CAPACITY IN HARRIS SPENT FUEL POOLS 'C' AND 'D' (NON-PROPRIETARY VERSION)

A/1

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053

Telephone (609) 797-0900
Fax (609) 797-0909

LICENSING REPORT
for
EXPANDING STORAGE CAPACITY
in
HARRIS SPENT FUEL POOLS C AND D

by

HOLTEC INTERNATIONAL
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

HOLTEC PROJECT NO. 70324
HOLTEC REPORT HI-971760
REPORT CATEGORY: A
REPORT CLASS: SAFETY RELATED
CLIENT CONTRACT NO. XTA7000024

REVIEW AND CERTIFICATION LOG FOR MULTIPLE AUTHORS

Sheet 1 of 2

REPORT NUMBER: 971760

PROJECT NUMBER: 70324

Document Portion	REVISION 0 Author	REVISION 0 Reviewer	REVISION 1 Author	REVISION 1 Reviewer	REVISION 2 Author	REVISION 2 Reviewer	REVISION 3 Author	REVISION 3 Reviewer
Chapter 1	S. Pellet SHP 1-22-98	[illegible] A.S. 1/23/98	S. Pellet SHP 4-15-98	[illegible] A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 2	S. Pellet SHP 1-22-98	[illegible] A.S. 1/23/98	S. Pellet SHP 4-15-98	[illegible] A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 3	S. Pellet SHP 1-22-98	[illegible] A.S. 1/23/98	S. Pellet SHP 4-15-98	[illegible] A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 4	[illegible] 1-22-98	[illegible] 1/23/98	[illegible] 4/15/98	[illegible] 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 5	[illegible] 1/22/98	[illegible] 1/23/98	[illegible]	[illegible]	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 6	S. Pellet SHP 1-22-98	[illegible] A.S. 1/23/98	S. Pellet SHP 4-15-98	[illegible] A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 7	S. Pellet SHP 1-22-98	[illegible] A.S. 1/23/98	S. Pellet SHP 4-15-98	[illegible] A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 8	S. Pellet SHP 1-22-98	[illegible] A.S. 1/23/98	S. Pellet SHP 4-15-98	[illegible] A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
QA APPROVAL	[illegible] 1-23-98		M. Phipps M. PHIPPS 4/15/98		M. Phipps M. PHIPPS 5/26/98			
PROJECT MANAGER'	[illegible] 1/23/98		Scott H. Pellet SHP 4-15-98		Scott H. Pellet SHP 5-26-98			

This document conforms to the requirements of the Design Specification and the applicable sections of the governing codes.

Note: Signatures and printed names are required in the review block.

- A revision of this document will be ordered by the Project Manager and carried out if any of its contents is materially affected during evolution of this project. The determination as to the need for revision will be made by the Project Manager with input from others, as deemed necessary by him.

' Must be Project Manager or his designee.

' Distribution
C: Client
M: Designated Manufacturer
F: Florida Office

THE REVISION CONTROL OF THIS DOCUMENT IS BY A "SUMMARY OF REVISIONS LOG" PLACED BEFORE THE TEXT OF THE REPORT. Form: RCL.02

REVIEW AND CERTIFICATION LOG FOR MULTIPLE AUTHORS

Sheet 2 of 2

REPORT NUMBER: 971760

PROJECT NUMBER: 70324

Document Portion	REVISION 0 Author	REVISION 0 Reviewer	REVISION 1 Author	REVISION 1 Reviewer	REVISION 2 Author	REVISION 2 Reviewer	REVISION 3 Author	REVISION 3 Reviewer
Chapter 9	S. Pellet SHP 1-22-98	Alan Solan A.S. 1/23/98	S. Pellet SHP 4-15-98	Alan Solan A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
~~Chapter 10~~	S. Pellet SHP 1-22-98	Alan Solan A.S. 1/23/98	S. Pellet SHP 4-15-98	Alan Solan A.S. 4/15/98	N/A	N/A		
Chapter 100	S. Pellet SHP 1-22-98	Alan Solan A.S. 1/23/98	S. Pellet SHP 4-15-98	Alan Solan A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
Chapter 101	S. Pellet SHP 1-22-98	Alan Solan A.S. 1/23/98	S. Pellet SHP 4-15-98	Alan Solan A.S. 4/15/98	S. Pellet SHP 5-26-98	C. [illegible] C.B. 5/26/98		
QA APPROVAL	V. Gupta V. GUPTA 1-23-98		M. Phipps M. PHIPPS 4/15/98		M. Phipps M. PHIPPS 5/26/98			
PROJECT MANAGER¹	[illegible] M.S. 1/23/98		Scott H. Pellet SHP 4-15-98		Scott H. Pellet SHP 5-26-98			

SHP 5-26-98

This document conforms to the requirements of the Design Specification and the applicable sections of the governing codes.

Note: Signatures and printed names are required in the review block.

- A revision of this document will be ordered by the Project Manager and carried out if any of its contents is materially affected during evolution of this project. The determination as to the need for revision will be made by the Project Manager with input from others, as deemed necessary by him.

- Must be Project Manager or his designee.

- Distribution C: Client
  M Designated Manufacturer
  F Florida Office

THE REVISION CONTROL OF THIS DOCUMENT IS BY A "SUMMARY OF REVISIONS LOG" PLACED BEFORE THE TEXT OF THE REPORT. Form: RCL.02

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Tables

Figures

1.0 INTRODUCTION

The Harris Nuclear Plant (HNP) is a single unit pressurized water reactor installation located in the extreme southwest corner of Wake County, North Carolina, and the southeast corner of Chatham County, North Carolina. The HNP installation is owned by the Carolina Power & Light Company (CP&L) and the North Carolina Eastern Municipal Power Agency (NCEMPA), located in Raleigh, North Carolina. CP&L has the overall responsibility to ensure that plant operations are performed without undue risk to the health and safety of the public. Table 1.1 contains key overview data for HNP's PWR Unit.

HNP was originally named Shearon Harris Nuclear Power Plant (SHNPP) and was initially designed as a four unit nuclear reactor site, of which only Unit 1 was completed. The Fuel Handling Building (FHB), however, was constructed to service all four Units as originally envisioned. During initial licensing, the possibility of transhipment from other Units was recognized and consequently the Spent Fuel Pools were licensed to store both PWR and BWR fuel. Transhipped fuel from the Robinson and Brunswick plants is already in stored in pools A and B.

The FHB is a long narrow structure intended to be sandwiched between the nuclear plants, in order to service all four Units. Each end of the building contains two large pools, with the South end pools (A and B) originally intended to service Units 1 and 4 and the North end pools (C and D) designed to service Units 2 and 3. The layout of the FHB and pools in relationship with Unit 1 is shown in Figure 1.1. The two pools in each end of the building were originally designated as the "New Fuel Pool" for the smaller of the two pools and the "Spent Fuel Pool" for the larger pool. These four pools have since been re-designated as pools A, B, C, and D, where pools A and D represent the smaller pools. All four pools are interconnected through "gated" passages and are capable of storing spent fuel.

Pools A and B, located at the South end of the building, 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 A has been, and will continue to be, used to store fresh (unburned) fuel, recently discharged Harris fuel and transshipped fuel. Pool B contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) PWR Region 1 style racks. Pool B also currently contains seventeen (11 x 11 cell) BWR racks, twelve of which have been supplied by Holtec International. 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. The combined pool A and B licensed storage capacity is 3669 assemblies.

Projected operation of the Harris Unit and transhipments from the Robinson and Brunswick Units will continue to demand incremental increases in spent fuel storage capacity. The Carolina Power & Light Company, HNP's principal owner and operator, has entered into a contract with Holtec International of Marlton, N.J. to design maximum density spent fuel storage racks for pools C and D. Under the proposed capacity expansion, fuel storage racks will be installed in campaign phases on an as needed basis. This process is consistent with the incremental capacity expansions already performed in pool B.

Pools C and D are unused and are located in the north end of the Harris Fuel Handling Building. Pool C will provide storage for both PWR and BWR fuel. This pool has nominal dimensions of 27 feet wide by 50 feet long and at maximum storage density can accommodate 927 PWR and 2763 BWR assemblies. Pool D will contain only PWR fuel and with nominal dimensions of 20 feet wide by 32 feet long can accommodate 1025 maximum density storage cells. Proposed storage configurations for pools C and D are provided in Figures 1.2 and 1.3, respectively.

The configuration shown in Figure 1.2 represents the mixture of PWR and BWR storage which will accommodate future storage needs based on the best information currently available. To provide the greatest flexibility in mixture of fuel types, the storage racks were sized to allow interchangeability. The dimensions of the 9x9 PWR storage rack are nearly identical to those of the 13x13 BWR rack. Therefore, configurations other than those shown in Figure 1.2 are possible by replacing one rack type by the other. The complete geometric fungibility between the 9x9 PWR and 13x13 BWR rack modules affords CP&L the latitude to alter the mix between PWR and BWR storage as the precise need for the two types of spent nuclear fuel storage become known. Interchanging of PWR and BWR modules would be performed after appropriate safety evaluations supported by reanalysis of the criticality, thermal-hydraulic, and structural analyses are successfully conducted to support such a substitution under Subpart 50.59.

The new Holtec racks are free-standing and self-supporting. The principal construction materials for the new racks are SA240-Type 304L stainless steel sheet and plate stock, and SA564-630 (precipitation hardened stainless steel) for the adjustable support spindles. The only non-stainless material utilized in the rack is the neutron absorber material which is a boron carbide and aluminum-composite sandwich available under the patented product name Boral™.

The new Holtec racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel (B&PV) Code. The material procurement, analysis, and fabrication of the rack modules conform to 10CFR50 Appendix B requirements.

The rack design and analysis methodologies employed in the Harris storage capacity expansion are a direct evolution of previous rerack license applications. This Licensing Report documents the design and analyses performed to demonstrate that the new Holtec racks meet all governing requirements of the applicable codes and standards, in particular, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", USNRC, 1978 and the 1979 Addendum thereto [1.0.1].

Sections 2 and 3 of this report provide an abstract of the design and material information on the new racks.

The criticality safety analysis requires that the neutron multiplication factor for the stored fuel array be bounded by the USNRC k_{eff} limit of 0.95 under assumptions of 95% probability and 95% confidence. The criticality safety analysis provided in Section 4 sets the requirements on the Boral panel length and the areal B-10 density for the new high density racks.

Thermal-hydraulic consideration requires that fuel cladding will not fail due to excessive thermal stress, and that the steady state pool bulk temperature will remain within the limits prescribed for the spent fuel pool to satisfy the pool structural strength, operational, and regulatory requirements. The thermal-hydraulic analyses carried out in support of this storage expansion effort are described in Section 5.

Demonstrations of seismic and structural adequacy are presented in Section 6.0. The analysis shows that the primary stresses in the rack module structure will remain below the allowable stresses of the ASME B&PV Code (Subsection NF) [1.0.2]. The structural qualification also includes analytical demonstration that the subcriticality of the stored fuel will be maintained under all postulated accident scenarios in the Harris Final Safety Analysis Report (FSAR). The structural consequences of these postulated accidents are evaluated and presented in Section 7 of this report.

Section 8 contains the structural analysis to demonstrate the adequacy of the spent fuel pool reinforced concrete structure. A synopsis of the geometry of the Harris reinforced concrete structure is also presented in Section 8.

The radiological considerations are documented in Section 9.0. Sections 10, and 11 discuss the salient considerations in the installation of the new racks, and a cost/benefit and environmental assessment to establish the prudence of CP&L's decision to exercise the wet storage expansion option, respectively.

All computer programs utilized to perform the analyses documented in this licensing report are benchmarked and verified. These programs have been utilized by Holtec International in numerous rerack applications over the past decade.

The analyses presented herein clearly demonstrate that the rack module arrays possess wide margins of safety in respect to all considerations of safety specified in the OT Position Paper, namely, nuclear subcriticality, thermal-hydraulic safety, seismic and structural adequacy, radiological compliance, and mechanical integrity.

1.1 References

[1.0.1] USNRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978, and Addendum dated January 18, 1979.

[1.0.2] ASME Boiler & Pressure Vessel Code, Section III, Subsection NF, and Appendices (1995).

Table 1.1 KEY HARRIS PLANT INFORMATION	
ITEM	DATA
Docket Number	50-400
Capacity, MWe	940
Applied to NRC	9-4-71
Construction Permit	1-27-78
Commercial Operation	1986
Present Capacity	<u>Cells</u>
Pool A	723
Pool B	2946
TOTAL	3669

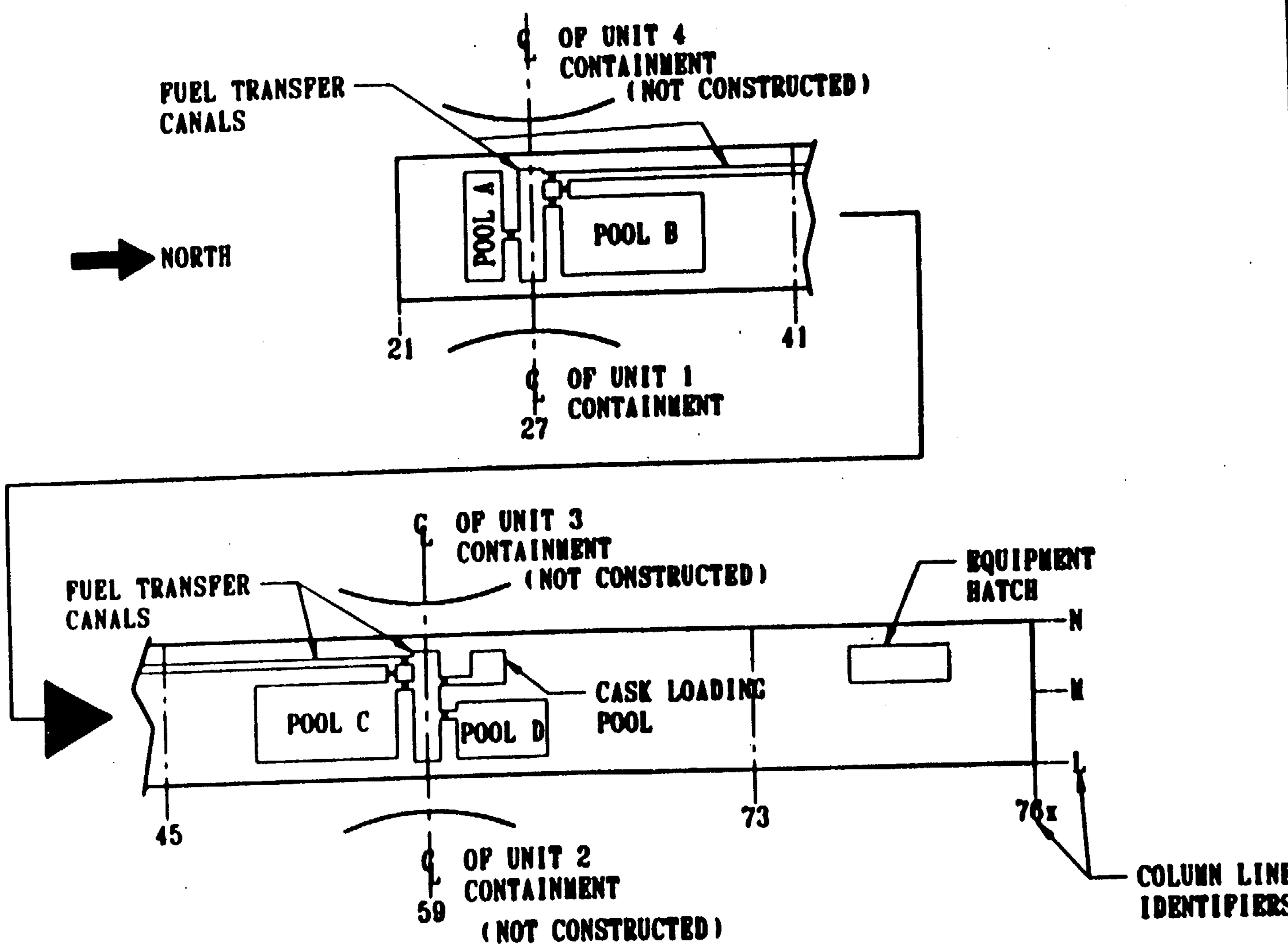


FIGURE 1.1; HARRIS FUEL HANDLING BUILDING PLAN LAYOUT

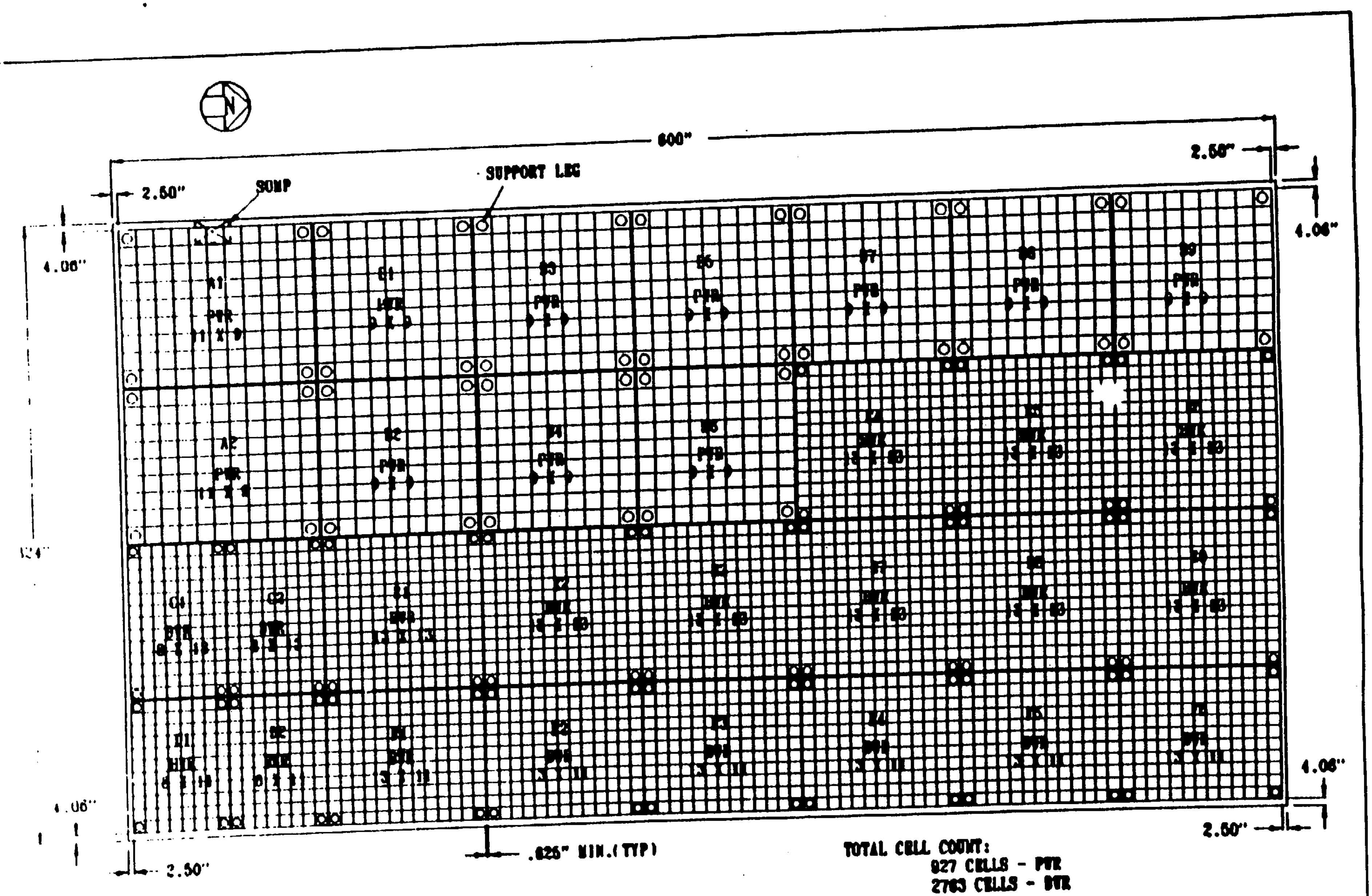


FIGURE 1.2; STORAGE CONFIGURATION FOR POOL C

HI-971760

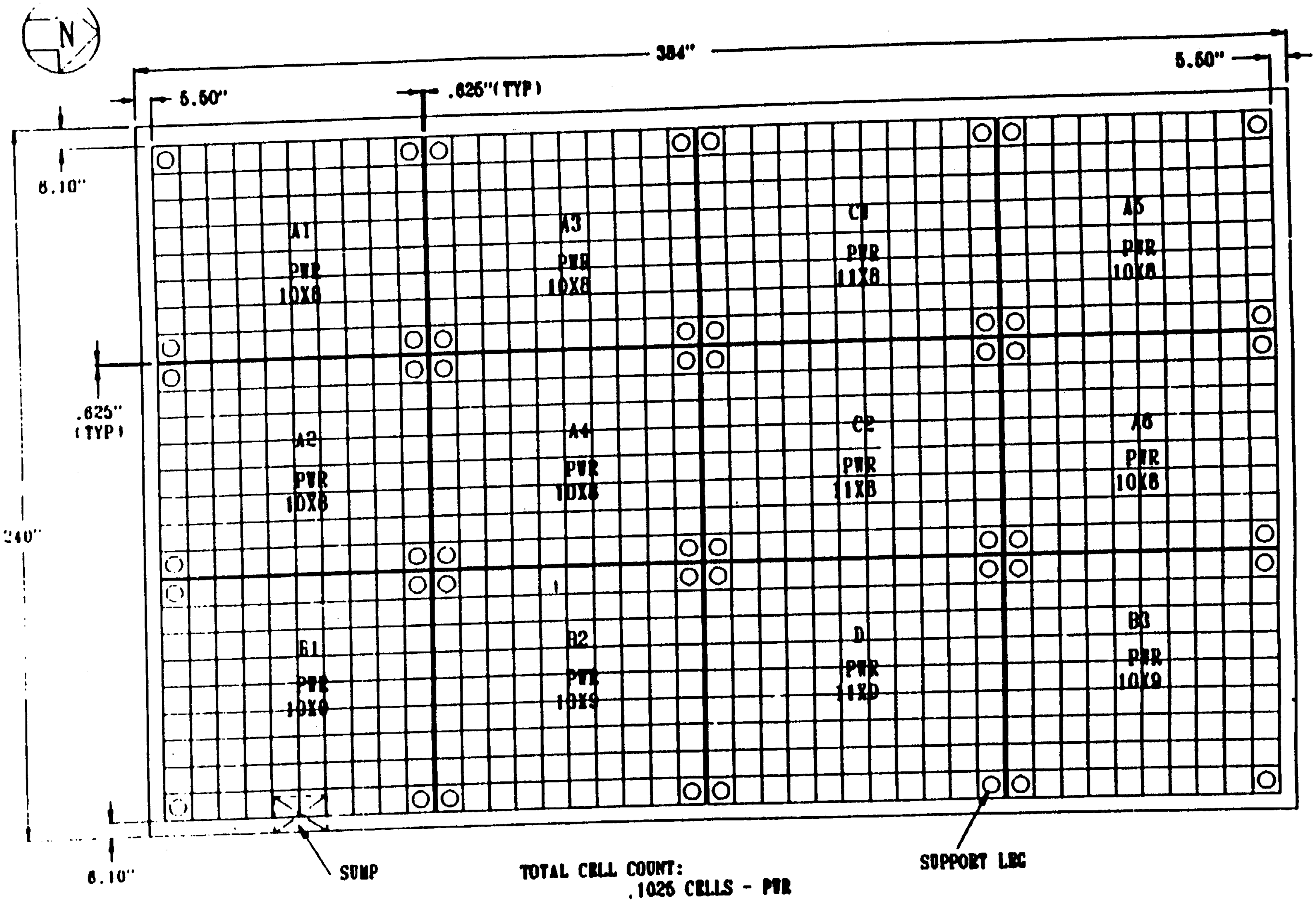


FIGURE 1.3; STORAGE CONFIGURATION FOR POOL D

HI-971760

2.0 OVERVIEW OF THE PROPOSED CAPACITY EXPANSION

2.1 Introduction

In its currently proposed fully implemented configuration, Pool C will contain eleven PWR racks and nineteen BWR racks. Pool D will contain twelve PWR racks. All storage racks arrays will consist of free-standing modules, made from Type 304L austenitic stainless steel containing prismatic storage cells interconnected through longitudinal welds. A panel of Boral cermet containing a high areal loading of the B-10 isotope provides appropriate neutron attenuation between adjacent storage cells. Figure 2.1.1 provides a schematic of the typical Region 2 storage module proposed for Harris. Data on the cross sectional dimensions, gross weight and cell count for each rack module in pools C and D are presented in Tables 2.1.1 and 2.1.2, respectively.

Each new rack module is supported by four legs which are remotely adjustable. Thus, the racks can be made vertical and the top of the racks can easily be made co-planar with each other. The rack module support legs are engineered to accommodate undulations in the fuel pool and cask pit floor flatness.

A bearing pad interposed between the rack pedestals and the pool liner serves to diffuse the dead load of the loaded racks into the reinforced concrete structure of the pool slab.

The overall design of the Harris racks is similar to those presently in service in the spent fuel pools at many other nuclear plants, among them Zion Nuclear Station of the Commonwealth Edison Company, Donald C. Cook of American Electric Power, and Connecticut Yankee of Northeast Utilities. Altogether, over 50 thousand storage cells of the Harris design have been provided by Holtec International to various nuclear plants around the world.

2.2 Summary of Principal Design Criteria

The key design criteria for the new Harris spent fuel racks are set forth in the classical USNRC memorandum entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 as modified by amendment dated January 18, 1979. The individual sections of this report expound on the specific design bases derived from the above-mentioned "OT Position Paper". Nevertheless, a brief summary of the design bases for the Harris racks are summarized in the following:

a. Disposition: All new rack modules are required to be free-standing.

b. Kinematic Stability: All free-standing modules must be kinematically stable (against tipping or overturning) if a seismic event (which is 150% of the postulated OBE or 110% of the postulated SSE) is imposed on any module.

c. Structural Compliance: All primary stresses in the rack modules must satisfy the limits postulated in Section III subsection NF of the 1995 ASME Boiler and Pressure Vessel Code.

d. Thermal-Hydraulic Compliance: The spatial average bulk pool temperature is required to remain under 137°F † in the wake of a normal refueling.

   In addition to the limitations on the bulk pool temperature, the local water temperature in the Harris pools must remain subcooled (i.e., below the boiling temperature coincident with local hydraulic pressure conditions).

† The 137°F limit is consistent with that currently in the Harris FSAR and procedures for pools A and B. CP&L is in the process of re-evaluating systems and components to allow for an increase the allowable bulk pool temperature.

2.6.2 Anatomy of the Harris PWR Rack Module

In addition to the composite box assembly, the baseplate and the support legs constitute the principal components of the Harris fuel rack modules. The following description provides details of all of the major rack components.

iii.

iv.

v.

The assembly of the rack modules is carried out by welding the composite boxes in a vertical fixture with the precision fabricated baseplate serving as the bottom positioner

(2) ASB 9-2 - Residual Decay Energy for Light-Water Reactors for Long-Term Cooling.

j. Standard Review Plan

(1) SRP 3.2.1 - Seismic Classification.

(2) SRP 3.2.2 - System Quality Group Classification.

(3) SRP 3.7.1 - Seismic Design Parameters.

(4) SRP 3.7.2 - Seismic System Analysis.

(5) SRP 3.7.3 - Seismic Subsystem Analysis.

(6) SRP 3.8.4 - Other Seismic Category I Structures (including Appendix D), Technical Position on Spent Fuel Rack.

(7) SRP 3.8.5 - Foundations for Seismic Category I Structures, Revision 1, 1981.

(8) SRP 9.1.2 - Spent Fuel Storage, Revision 3, 1981.

(9) SRP 9.1.3 - Spent Fuel Pool Cooling and Cleanup System.

(10) SRP 9.1.4 - Light Load Handling System.

(11) SRP 9.1.5 - Heavy Load Handling System.

(12) SRP 15.7.4 - Radiological Consequences of Fuel Handling Accidents.

k. AWS Standards

(1) AWS D1.1 - Structural Welding Code, Steel.

(2) AWS D1.3 - Structure Welding Code - Sheet Steel.

(3) AWS D9.1 - Welding of Sheet Metal.

(4) AWS A2.4 - Standard Symbols for Welding, Brazing and Nondestructive Examination.

(5) AWS A3.0 - Standard Welding Terms and Definitions.

(6) AWS A5.12 - Tungsten Arc-welding Electrodes.

(7) AWS QC1 - Standards and Guide for Qualification and Certification of Welding Inspectors.

2.4 Quality Assurance Program

The governing quality assurance requirements for design of the Harris spent fuel racks are enunciated in 10CFR50 Appendix B. The quality assurance program for design of the Harris racks is described in Holtec's Nuclear Quality Assurance Manual, which has been reviewed and approved by the Carolina Power & Light Company. This program is designed to provide a flexible but highly controlled system for the design, analysis and licensing of customized components in accordance with various codes, specifications, and regulatory requirements.

The manufacturing of the racks will be performed in accordance with the requirements setforth in 0CFR50 Appendix B.

2.5 Mechanical Design

The Harris rack modules are designed as cellular structures such that each fuel assembly has a prismatic square opening with conformal lateral support and a flat horizontal bearing surface. The basic characteristics of the Harris spent fuel racks are summarized in Table 2.5.1. The design of the PWR and BWR storage racks are very similar. The major differences are in the cell inside dimension and pitch, the baseplate flow holes, the support legs, and the poison width and length.

A central objective in the design of the new rack modules is to maximize their structural rigidity while minimizing their inertial mass. Accordingly, the Harris modules have been designed to simulate multi-flange beam structures. The multiple flanges are formed from the numerous cell walls in the rack cross-sectional array. These cells are connected through intermittent welds. The weld lengths, location, and size were chosen during the original design of this rack style/series to ensure adequate strength and to adjust the natural frequency of the rack modules to avoid resonance. In general, this effort has resulted in excellent detuning characteristics with respect to the applicable seismic events.

2.6 Rack Fabrication

This subsection presents an item-by-item description of the anatomy of the Harris rack modules in the context of the fabrication methodology. The object of this section is to provide a self-contained description of rack module construction for the Harris fuel pool to enable an independent appraisal of the adequacy of design.

2.6.1 Fabrication Objective

The requirements in manufacturing the high density storage racks for Harris may be stated in four interrelated points:

1. The rack modules are fabricated in such a manner that there is no weld splatter on the storage cell surfaces which would come in contact with the fuel assembly.

2. The storage locations are constructed so that redundant flow paths for the coolant are available.

3. The fabrication process involves operational sequences which permit immediate verification by the inspection staff.

4. The storage cells are connected to each other by austenitic stainless steel corner welds which leads to a honeycomb lattice construction. The extent of welding is selected to "detune" the racks from the stipulated seismic input motion.

2.6.3 Anatomy of the Harris BWR Rack Module

i.

ii.

iii.

iv.

v.

The assembly of the rack modules is carried out by welding the composite boxes in a vertical fixture with the precision fabricated baseplate serving as the bottom positioner.

An elevation view of three PWR and BWR storage cells is shown in Figures 2.6.2 and 2.6.3, respectively.

Table 2.1.1
GEOMETRIC AND PHYSICAL DATA FOR POOL C RACK MODULES†

Rack I.D. ††	Type	Number of Cells N-S	Number of Cells E-W	Number of Cells Per Module	Dimension (inches) N-S Direction	Dimension (inches) E-W Direction	Shipping Weight (lbs)	Submerged Weight (lbs)
A1	PWR							
A2	PWR							
B1	PWR							
B2	PWR							
B3	PWR							
B4	PWR							
B5	PWR							
B6	PWR							
B7	PWR							
B8	PWR							
B9	PWR							
C1	BWR							
C2	BWR							
D1	BWR							
D2	BWR							

† All dimensions are rounded off to the nearest 0.5 inch, and all weights are rounded off to the nearest 10 lbs.

†† See Figure 1.2 for pool configuration.

Table 2.1.1 (Cont'd.)

GEOMETRIC AND PHYSICAL DATA FOR POOL C RACK MODULES†

Rack I.D. ††	Type	Number of Cells		Number of Cells Per Module	Dimension (inches)		Shipping Weight (lbs)	Submerged Weight (lbs)
		N-S	E-W		N-S Direction	E-W Direction		
E1	BWR							
E2	BWR							
E3	BWR							
E4	BWR							
E5	BWR							
E6	BWR							
E7	BWR							
E8	BWR							
E9	BWR							
F1	BWR							
F2	BWR							
F3	BWR							
F4	BWR							
F5	BWR							
F6	BWR							

† All dimensions are rounded off to the nearest 0.5 inch, and all weights are rounded off to the nearest 10 lbs.

†† See Figure 1.2 for pool configuration.

Table 2.1.2

GEOMETRIC AND PHYSICAL DATA FOR POOL D RACK MODULES[†]

Rack I.D. ††	Type	Number of Cells		Number of Cells Per Module	Dimension (inches)		Shipping Weight (lbs)	Submerged Weight (lbs)
		N-S	E-W		N-S Direction	E-W Direction		
A1	PWR							
A2	PWR							
A3	PWR							
A4	PWR							
A5	PWR							
A6	PWR							
B1	PWR							
B2	PWR							
B3	PWR							
C1	PWR							
C2	PWR							
D	PWR							

† All dimensions are rounded off to the nearest 0.5 inch, and all weights are rounded off to the nearest 10 lbs.

†† See Figure 1.3 for pool configuration.

Table 2.5.1

MODULE DATA FOR HARRIS SPENT FUEL RACKS

Parameter	PWR	BWR
Storage cell inside dimension (nominal)		
Cell pitch (nominal)		
Storage cell height (above the baseplate)		
Baseplate hole size (away from pedestal)		
Baseplate thickness		
Support leg height (nominal)		
Support leg type		
Number of support pedestals		
Remote lifting and handling provisions		
Poison material		
Poison length		
Poison width		

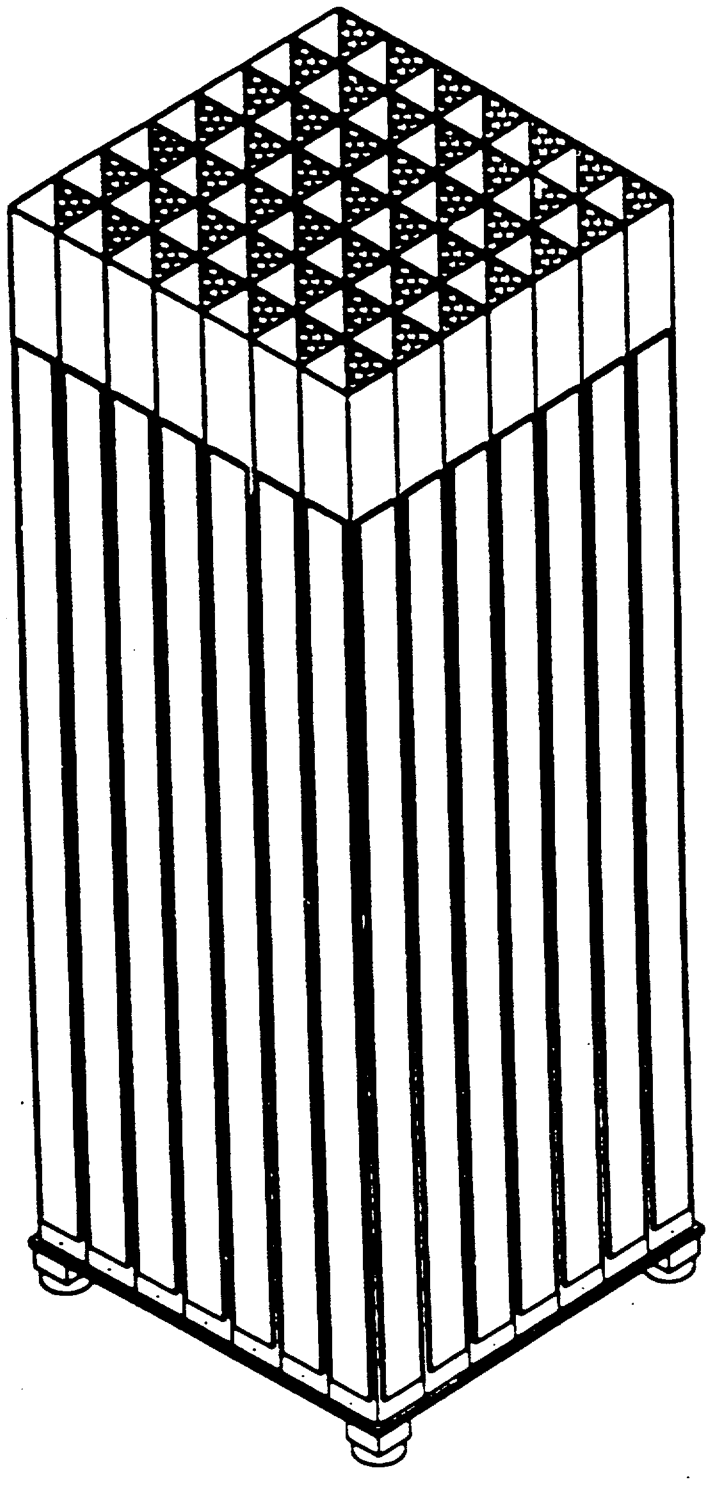


FIGURE 2.1.1; PICTORIAL VIEW OF TYPICAL HARRIS RACK STRUCTURE

HI - 971760

FIGURE 2.6.1; SEAM WELDED PRECISION FORMED CHANNELS

HOLTEC PROPRIETARY [illegible]

FIGURE 2.6.2; THREE PWR CELLS IN ELEVATION VIEW

Holtec Proprietary

[illegible]

FIGURE 2.6.3; THREE BWR CELLS IN ELEVATION VIEW

Holtec Proprietary

HI-971760

FIGURE 2.6.4; COMPOSITE BOX ASSEMBLY

FIGURE 2.6.5; TYPICAL ARRAY OF STORAGE CELLS
(NON-FLUX TRAP CONSTRUCTION)

FIGURE 2.6.6; SUPPORT PEDESTAL FOR PWR RACK

FIGURE 2.6.7; SUPPORT PEDESTAL FOR HOLTEC BWR RACKS

3.0 MATERIAL, HEAVY LOAD, AND CONSTRUCTION CONSIDERATIONS

3.1 Introduction

Safe storage of nuclear fuel in the Harris pools requires that the materials utilized in the rack fabrication be of proven durability and be compatible with the pool water environment. Likewise, all activities during the rack installations must comply with the provisions of NUREG-0612 to eliminate the potential of construction accidents. This section provides a synopsis of th considerations with regard to long-term service life and short-term construction safety.

3.2 Structural Materials

The following structural materials are utilized in the fabrication of the new spent fuel racks:

a. ASME SA240-304L for all sheet metal stock

b. Internally threaded support legs: ASME SA240-304L

c. Externally threaded support spindle: ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100°F)

d. Weld material - per the following ASME specification: SFA 5.9 ER308L

3.3 Poison Material (Neutron Absorber)

The racks employ Boral™, a patented product of AAR Manufacturing, as the neutron absorber material. Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a lightweight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal and chemical environment of a nuclear reactor or a spent fuel pool Boral has been shown [3.3.1] to be superior to alternative materials previously used as neutron absorbers in storage racks.

Boral has been the most widely used neutron absorbing material in fuel rack applications over the past 20 years. Its use in the spent fuel pools as the neutron absorbing material can be attributed to its proven performance (over 150 pool years of experience) and the following unique characteristics:

i. The content and placement of boron carbide provides a very high removal cross-section for thermal neutrons.

ii. Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.

iii. The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.

iv. The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.

v. Boral is stable, strong, durable, and corrosion resistant.

Boral will be manufactured by AAR Manufacturing under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants" As indicated in Tables 3.3.1 and 3.3.2, Boral has been licensed by the USNRC for use in numerous PWR and BWR spent fuel storage racks and has been extensively used in international nuclear installations

<u>Boral Material Characteristics</u>

Aluminum: Aluminum is a silvery-white, ductile metallic element that is the most abundant in the earth's crust. The 1100 alloy aluminum is used extensively in heat exchangers, pressure vessels and storage tanks, chemical equipment, reflectors, and sheet metal work

It has high resistance to corrosion in industrial and marine atmospheres Aluminum has atomic number of 13, atomic weight of 26.98, specific gravity of 2.69 and valence of 3 The physical, mechanical and chemical properties of the 1100 alloy aluminum are listed in Tables 3.3.3 and 3.3.4.

The excellent corrosion resistance of the 1100 alloy aluminum is provided by the protective oxide film that develops on its surface from exposure to the atmosphere or water. This film prevents the loss of metal from general corrosion or pitting corrosion.

Boron Carbide: The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The material conforms to the chemical composition and properties listed in Table 3.3.5.

References [3.3.2], [3.3.3], and [3.3.4] provide further discussion as to the suitability of these materials for use in spent fuel storage module applications.

3.4 Rack Material Compatibility with Coolant

All materials used in the construction of the Holtec racks have an established history of in-pool usage. Their physical, chemical and radiological compatibility with the pool environment is an established fact.

Austenitic stainless steel (304L) is perhaps the most widely used stainless alloy in nuclear power plants, since it provides both high strength and non-corrosive properties

3.5 Heavy Load Considerations for the Proposed Rack Installations

The Fuel Handling Building auxiliary crane will be used for installation of the new storage racks in pools C and D The Spent Fuel Cask Handling Crane (CHC) cannot be used for rack installation, since travel limitations prohibit its movement over the spent fuel pools · Storage capacity will be increased starting in the south end of pool C and proceeding north. This installation pattern will enable the storage racks to be manipulated without lifts over spent fuel.

The auxiliary crane is a single failure proof crane and is currently rated for 10 tons. A 20 ton hoist will be attached to the auxiliary crane hook to prevent submergence of the auxiliary crane hook. The auxiliary crane was used for installation of storage racks in pool B. Rigging and procedures for pools C and D rack installation will be similiar to those used previously.

The maximum lift weight during rack installation is determined by the following table.

Item	Weight (lbs)
Rack	15,700 (maximum)
Lift Rig	1,200
Rigging	500
20 ton hoist	1,420
Total Lift	18,820

The rack sizes were limited to ensure that the crane and lifting components remain single failure proof and it may be seen that the maximum lift of 18,820 lbs is below the auxiliary crane rating of 20,000 lbs. As a result, the auxiliary crane, which can travel over both pools C and D, is qualified to accept the anticipated load during the rack installation project

A remotely engagable lift rig, meeting NUREG-0612 [3 5 1] stress criteria, will be used to lift the new modules. The rig is designed for handling both PWR and BWR racks The new rack lift rig consists of independently loaded lift rods in a lift configuration which ensures that failure of one traction rod will not result in uncontrolled lowering of the load being carried by the rig (which complies with the duality feature called for in Section 5 1 6(3a) of NUREG 0612)

The rigs have the following attributes:

a. The traction rod is designed to prevent loss of its engagement with the rig in the locked position. Moreover, the locked configuration can be directly verified from above the pool water without the aid of an underwater camera.

b. The stress analysis of the rigs will be carried out using a finite element code, and the primary stress limits in ANSI 14.6-1978 [3.5.2] will be shown to be met by detailed analysis.

c. The rigs will be load tested with 300% of the maximum weight to be lifted. The test weight will be maintained in the air for 10 minutes. All critical weld joints will be liquid penetrant examined to establish the soundness of all critical joints.

Pursuant to the defense-in-depth approach of NUREG-0612, the following additional measures of safety will be undertaken for the racking operation.

i. The crane used in the project will be given a preventive maintenance checkup and inspection per the Harris maintenance procedures before the beginning of the racking operation.

ii. Safe load paths will be developed for moving the new racks in the Fuel Handling Building. The racks will not be carried directly over any fuel located in the pool.

iii. The rack upending and laying down will be carried out in an area which precludes any adverse interaction with safety related equipment.

iv. All crew members involved in the use of the lifting and upending equipment will be given training similar to that utilized in previous rack installation operations.

The rack installation activities will require Harris PNSC approval and will be conducted in accordance with written procedures which will be reviewed and approved by Carolina Power & Light.

The proposed heavy loads compliance will be in accordance with the objectives of the CP&L NRC-approved submittal to NUREG-0612. The guidelines of NUREG-0612 call for measures to "provide an adequate defense-in-depth for handling of heavy loads near spent fuel...". The NUREG-0612 guidelines cite four major causes of load handling accidents, namely

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The Harris racking program ensures maximum emphasis on mitigating the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas. A summary of the measures specifically planned to deal with the major causes is provided below.

Operator errors: As mentioned above, CP&L plans to provide comprehensive training to the installation crew. All training shall be in compliance with ANSI B30.2 [3.5.3].

Rigging failure: The lifting device designed for handling and installation of the new racks at Harris has redundancies in the lift legs and lift eyes such that there are four independent load members. Failure of any one load bearing member would not lead to uncontrolled lowering of the load. The rig complies with all provisions of ANSI 14.6 [3.5.2], including compliance with the primary stress criteria, load testing at 300% of maximum lift load, and dye examination of critical welds.

The Harris rig design is similar to the rigs used in the initial racking or the rerack of numerous other plants, such as Hope Creek, Millstone Unit 1, Indian Point Unit Two, Ulchin II, Laguna Verde, J.A. FitzPatrick and Three Mile Island Unit 1

Lack of adequate inspection: The designer of the racks will develop a set of QC hold points which will require inspections and approvals prior to proceeding. Additional hold points will be established for activities during the installatin process. These inspection points have been proven to significantly reduce any requirement for rework or instances of erroneous installation in numerous prior rerack projects

Inadequate procedures: CP&L is developing various operating procedures to address operations pertaining to the rack installation effort, including, but not limited to, mobilization, rack handling, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA compliance. Many of the procedures will be the same or revisions to those developed and currently in use for rack installations in pool B

The series of operating procedures planned for Harris rack installations are the successors of the procedures successfully implemented in previous projects.

Table 3.5.1 provides a synopsis of the requirements delineated in NUREG-0612, and their intended compliance.

3.6 References

[3.3.1] "Nuclear Engineering International," July 1997 issue, pp 20-23.

[3.3.2] "Spent Fuel Storage Module Corrosion Report," Brooks & Perkins Report 554, June 1, 1977.

[3.3.3] "Suitability of Brooks & Perkins Spent Fuel Storage Module for Use in PWR Storage Pools," Brooks & Perkins Report 578, July 7, 1978.

[3.3.4] "Boral Neutron Absorbing/Shielding Material - Product Performance Report," Brooks & Perkins Report 624, July 20, 1982.

[3.5.1] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

[3.5.2] ANSI N14.6-1978, Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds or more for Nuclear Materials," American National Standard Institute, Inc., 1978.

[3.5.3] ANSI/ASME B30.2, "Overhead and Gantry Cranes, (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," American Society of Mechanical Engineers, 1983.

[3.5.4] ANSI/ASME B30.20, "Below-the-Hook Lifting Devices," American Society of Mechanical Engineers, 1993.

[3.5.5] CMMA Specification 70, "Electrical Overhead Travelling Cranes," Crane Manufacturers Association of America, Inc., 1983.

[3.5.6] ANSI/ASME B30.20, "Below-the-Hook Lifting Devices," American Society of Mechanical Engineers, 1993

Table 3.3.1

BORAL EXPERIENCE LIST - PWRs

Plant	Utility	Docket No.	Mfg. Year
Maine Yankee	Maine Yankee Atomic Power	50-309	1977
Donald C. Cook	Indiana & Michigan Electric	50-315/316	1979
Sequoyah 1,2	Tennessee Valley Authority	50-327/328	1979
Salem 1,2	Public Service Electric & Gas	50-272/311	1980
Zion 1,2	Commonwealth Edison Co.	50-295/304	1980
Bellefonte 1, 2	Tennessee Valley Authority	50-438/439	1981
Yankee Rowe	Yankee Atomic Power	50-29	1964/1983
Indian Point 3	NY Power Authority	50-286	1987
Byron 1,2	Commonwealth Edison Co.	50-454/455	1988
Braidwood 1,2	Commonwealth Edison Co.	50-456/457	1988
Yankee Rowe	Yankee Atomic Power	50-29	1988
Three Mile Island I	GPU Nuclear	50-289	1990
Sequoyah (rerack)	Tennessee Valley Authority	50-327	1992
Donald C. Cook (rerack)	American Electric Power	50-315/316	1992
Beaver Valley Unit 1	Duquesne Light Company	50-334	1993
Fort Calhoun	Omaha Public Power District	50-285	1993
Zion 1 & 2 (rerack)	Commonwealth Edison Co.	50-295/304	1993
Salem Units 1 & 2 (rerack)	Public Gas and Electric Company	50-272/311	1995
Haddam Neck	Connecticut Yankee Atomic Power Company	50-213	1996
Gosgen	Kernkraftwerk Gosgen-Daniken AG (Switzerland)	--	1984
Koeberg 1,2	ESCOM (South Africa)	--	1985
Beznau 1,2	Nordostschweizerische Kraftwerke AG (Switzerland)	--	1985

Table 3.3.1 (Cont'd.)

BORAL EXPERIENCE LIST - PWRs

Plant	Utility	Docket No.	Mfg. Year
12 various Plants	Electricite de France (France)	–	1986
Ulchin Unit 1	Korea Electric Power Company (Korea)	–	1995
Ulchin Unit 2	Korea Electric Power Company (Korea)	–	1996
Kori-4	Korea Electric Power Company (Korea)	–	1996
Yonggwang 1,2	Korea Electric Power Company (Korea)	–	1996
Sizewell B	Nuclear Electric, plc (United Kingdom)	–	1997
Angra 1	Furnas Centrais-Electricas SA (Brazil)	–	1997

f. Radiological Compliance: The reracking of Harris must not lead to violation of the off-site dose limits, or adversely affect the area dose environment as set forth in the Harris FSAR. The radiological implications of the installation of the new racks also need to be ascertained and deemed to be acceptable.

g. Pool Structure: The ability of the reinforced concrete structure to satisfy the load combinations set forth in NUREG-0800, SRP 3.8.4 must be demonstrated.

h. Rack Cyclic Stress Fatigue: In addition to satisfying the primary stress criteria of Subsection NF, the alternating local stresses in the rack structure are evaluated to ensure that the "cumulative damage factor" due to at least ten SSE events does not exceed 1.0.

i. Liner Integrity: The integrity of the liner under cyclic in-plane loading during a seismic event must be demonstrated. A material fatigue evaluation is performed in accordance with ASME B&PV Code. The alternating local stresses in the liner are evaluated to ensure that the "cumulative damage factor" due to at least ten SSE events does not exceed 1.0.

j. Bearing Pads: The bearing pads must be sufficiently thick such that the pressure on the liner continues to satisfy the ACI limits during and after a design basis seismic event.

k. Accident Events: In the event of postulated drop events (uncontrolled lowering of a fuel assembly, for instance), it is necessary to demonstrate that the subcriticality of the rack structure is not compromised.

l. Construction Events: The field construction services required to be carried out for executing the reracking must be demonstrated to be within the "state of proven art".

The foregoing design bases are further articulated in Sections 4 through 9 of this licensing report.

2.3 Applicable Codes and Standards

The following codes, standards and practices are used as applicable for the design, construction, and assembly of the Harris fuel storage racks. Additional specific references related to detailed analyses are given in each section.

a. Design Codes

(1) AISC Manual of Steel Construction, 1970 Edition and later.

(2) ANSI N210-1976, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" (contains guidelines for fuel rack design).

(3) American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code Section III, 1986 Edition; ASME Section V, 1986 edition; ASME Section VIII, 1986 Edition; ASME Section IX, 1986 Edition; and ASME Section XI, 1986 Edition.

(4) ASNT-TC-1A June, 1984 American Society for Nondestructive Testing (Recommended Practice for Personnel Qualifications).

(5) American Concrete Institute Building Code Requirements for Reinforced Concrete (ACI318-63) and (ACI318-71).

(6) Code Requirements for Nuclear Safety Related Concrete Structures, ACI349-85/ACI349R-85, and ACI349.1R-80.

(7) ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities

(8) ASME NQA-2-1989, Quality Assurance Requirements for Nuclear Facility Applications.

(9) ANSI Y14.5M, Dimensioning and Tolerancing for Engineering Drawings and Related Documentation Practices

(10) ACI Detailing Manual - 1980

b. Material Codes - Standards of ASTM

(1) E165 - Standard Methods for Liquid Penetrant Inspection.

(2) A240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.

(3) A262 - Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.

(4) A276 - Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes.

(5) A479 - Steel Bars for Boilers & Pressure Vessels.

(6) ASTM A564, Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless and Heat-Resisting Steel Bars and Shapes.

(7) C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.

(8) A380 - Recommended Practice for Descaling, Cleaning and Marking Stainless Steel Parts and Equipment.

(9) C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.

(10) ASTM E3, Preparation of Metallographic Specimens.

(11) ASTM E190, Guided Bend Test for Ductility of Welds.

(12) American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section II-Parts A and C, 1995 Edition.

(13) NCA3800 - Metallic Material Manufacturer's and Material Supplier's Quality System Program.

c. Welding Codes: ASME Boiler and Pressure Vessel Code, Section IX - Welding and Brazing Qualifications, 1995 Edition.

d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling Requirements

(1) ANSI 45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants.

(2) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase).

(3) ANSI - N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants (Regulatory Guide 1.58).

(4) ANSI-N45.2.8, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Plants.

(5) ANSI - N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants.

(6) ANSI-N45.2.12, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants.

(7) ANSI N45.2.13 - Quality Assurance Requirements for Control of Procurement of Equipment Materials and Services for Nuclear Power Plants (Regulatory Guide 1.123).

(8) ANSI N45.2.15-18 - Hoisting, Rigging, and Transporting of Items For Nuclear Power Plants.

(9) ANSI N45.2.23 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Regulatory Guide 1.146).

(10) ASME Boiler and Pressure Vessel, Section V, Nondestructive Examination, 1995 Edition.

(11) ANSI - N16.9-75 Validation of Calculation Methods for Nuclear Criticality Safety.

e. Governing NRC Design Documents

(1) "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and the modifications to this document of January 18, 1979

(2) NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants", USNRC, Washington, D.C., July, 1980.

f. Other ANSI Standards (not listed in the preceding)

(1) ANSI/ANS 8.1 (N16.1) - Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

(2) ANSI/ANS 8.17, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.

(3) N45.2 - Quality Assurance Program Requirements for Nuclear Facilities - 1971.

(4) N45.2.9 - Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants - 1974.

(5) N45.2.10 - Quality Assurance Terms and Definitions -1973.

(6) ANSI/ANS 57.2 (N210) - Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.

(7) N14.6 - American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials.

(8) ANSI/ASME N626-3, Qualification and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code Section III, Div. 1, Certifying Activities.

(9) ANSI Y14.5M, Dimensioning and Tolerancing for Engineering Drawings and Related Documentation Practices.

g. Code-of-Federal Regulations

(1) 10CFR20 - Standards for Protection Against Radiation.

(2) 10CFR21 - Reporting of Defects and Non-compliance.

(3) 10CFR50 Appendix A - General Design Criteria for Nuclear Power Plants.

(4) 10CFR50 Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

(5) 10CFR61 - Licensing Requirements for Land Disposal of Radioactive Material.

(6) 10CFR71 - Packaging and Transportation of Radioactive Material.

h. Regulatory Guides

(1) RG 1.13 - Spent Fuel Storage Facility Design Basis (Revision 2 Proposed).

(2) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors.

(3) RG 1.28 - (ANSI N45.2) - Quality Assurance Program Requirements .

(4) RG 1.29 - Seismic Design Classification (Rev. 3).

(5) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Material.

(6) RG 1.38 - (ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants.

(7) RG 1.44 - Control of the Use of Sensitized Stainless Steel.

(8) RG 1.58 - (ANSI N45.2.6) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel.

(9) RG 1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants.

(10) RG 1.61 - Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0, 1973.

(11) RG 1.64 - (ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants.

(12) RG 1.71 - Welder Qualifications for Areas of Limited Accessibility

(13) RG 1.74 - (ANSI N45.2.10) Quality Assurance Terms and Definitions.

(14) RG 1.85 - Materials Code Case Acceptability - ASME Section 3, Div. 1.

(15) RG 1.88 - (ANSI N45.2.9) Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records.

(16) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis.

(17) RG 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.

(18) RG 1.123 - (ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.

(19) RG 1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Revision 1, 1978.

(20) RG 3.4 - Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities.

(21) RG 3.41 - Validation of Calculational Methods for Nuclear Criticality Safety, Revision 1, 1977.

(22) RG 8.8 - Information Relative to Ensuring that Occupational Radiation Exposure at Nuclear Power Plants will be as Low as Reasonably Achievable (ALARA).

(23) DG-8006, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants".

(24) IE Information Notice 83-29 - Fuel Binding Caused by Fuel Rack Deformation.

(25) RG 8.38 - Control of Access to High and Very High Radiation Areas in Nuclear Power Plants, June, 1993.

i. <u>Branch Technical Position</u>

(1) CPB 9.1-1 - Criticality in Fuel Storage Facilities.

Table 3.3.2

BORAL EXPERIENCE LIST - BWRs

Plant	Utility	Docket No.	Mfg. Year
Cooper	Nebraska Public Power	50-298	1979
J.A. FitzPatrick	NY Power Authority	50-333	1978
Duane Arnold	Iowa Electric Light & Power	50-331	1979
Browns Ferry 1,2,3	Tennessee Valley Authority	50-259/260/296	1980
Brunswick 1,2	Carolina Power & Light	50-324/325	1981
Clinton	Illinois Power	50-461/462	1981
Dresden 2,3	Commonwealth Edison Company	50-237/249	1981
E.I. Hatch 1,2	Georgia Power	50-321/366	1981
Hope Creek	Public Service Electric & Gas	50-354/355	1985
Humboldt Bay	Pacific Gas & Electric Company	50-133	1985
LaCrosse	Dairyland Power	50-409	1976
Limerick 1,2	Philadelphia Electric Company	50-352/353	1980
Monticello	Northern States Power	50-263	1978
Peachbottom 2,3	Philadelphia Electric	50-277/278	1980
Perry 1,2	Cleveland Electric Illuminating	50-440/441	1979
Pilgrim	Boston Edison Company	50-293	1978
Susquehanna 1,2	Pennsylvania Power & Light	50-387,388	1979
Vermont Yankee	Vermont Yankee Atomic Power	50-271	1978/1986
Hope Creek	Public Service Electric & Gas	50-354/355	1989
Shearon Harris Pool B	Carolina Power & Light	50-401	1991
Duane Arnold	Iowa Electric Light & Power	50-331	1993
Pilgrim	Boston Edison Company	50-293	1993
LaSalle 1	Commonwealth Edison Company	50-373	1992
Millstone Unit 1	Northeast Utilities	50-245	1989
James A. FitzPatrick	NY Power Authority	50-333	1990
Hope Creek	Public Service Electric & Gas Company	50-354	1991

Table 3.3.2 (Cont'd.)			
BORAL EXPERIENCE LIST - BWRs			
Plant	**Utility**	**Docket No.**	**Mfg. Year**
Duane Arnold Energy Center	Iowa Electric Power Company	50-331	1994
Limerick Units 1,2	PECO Energy	50-352/50-353	1994
Shearon Harris Pool 'B'	Carolina Power & Light Company	50-401	1996
Nine Mile Point Unit 1	Niagara Mohawk Power Corporation	50-220	1997
Chinshan 1,2	Taiwan Power Company (Taiwan)	--	1986
Kuosheng 1,2	Taiwan Power Company (Taiwan)	--	1991
Laguna Verde 1,2	Comision Federal de Electricidad (Mexico)	--	1991

Table 3.3.3

1100 ALLOY ALUMINUM PHYSICAL CHARACTERISTICS

Density	0.098 lb/in3
Melting Range	1190°F - 1215°F
Thermal Conductivity (77°F)	128 BTU/hr/ft^2/F/ft
Coefficient of Thermal Expansion (68°F - 212°F)	13.1 x 10^{-6} in/in-°F
Specific Heat (221°F)	0.22 BTU/lb/°F
Modulus of Elasticity	10 x 10^6 psi
Tensile Strength (75°F)	13,000 psi (annealed) 18,000 psi (as rolled)
Yield Strength (75°F)	5,000 psi (annealed 17,000 psi (as rolled)
Elongation (75°F)	35-45% (annealed) 9-20% (as rolled)
Hardness (Brinell)	23 (annealed) 32 (as rolled)
Annealing Temperature	650°F

Table 3.3.4

CHEMICAL COMPOSITION - ALUMINUM
(1100 ALLOY)

99.00% min.	Aluminum
1.00% max.	Silicone and Iron
0.05-0.20% max.	Copper
0.05% max.	Manganese
0.10% max.	Zinc
0.15% max.	Other

Table 3.3.5

CHEMICAL COMPOSITION AND PHYSICAL PROPERTIES OF BORON CARBIDE

CHEMICAL COMPOSITION (WEIGHT PERCENT)	
Total boron	70.0 min.
B^{10} isotopic content in natural boron	18.0
Boric oxide	3.0 max.
Iron	2.0 max.
Total boron plus total carbon	94.0 min.
PHYSICAL PROPERTIES	
Chemical formula	B_4C
Boron content (weight percent)	78.28%
Carbon content (weight percent)	21.72%
Crystal structure	rhombohedral
Density	0.0907 lb/in^3
Melting Point	4442°F
Boiling Point	6332°F

Table 3.5.1

HEAVY LOAD HANDLING COMPLIANCE MATRIX (NUREG-0612)

Criterion	Compliance
1. Are safe load paths defined for the movement of heavy loads to minimize the potential of impact, if dropped, on irradiated fuel?	Yes
2. Will procedures be developed to cover: identification of required equipment, inspection and acceptance criteria required before movement of load, steps and proper sequence for handling the load, defining the safe load paths, and special precautions?	Yes
3. Will crane operators be trained and qualified?	Yes
4. Will special lifting devices meet the guidelines of ANSI 14.6-1978?	Yes
5. Will non-custom lifting devices be installed and used in accordance with ANSI B30.20, latest edition?	Yes
6. Will the cranes be inspected and tested prior to use in rack installation?	Yes
7. Does the crane meet the intent of ANSI B30.2-1976 and CMMA-70?	Yes

.0 CRITICALITY SAFETY EVALUATION

4.1 Design Bases

The high density spent fuel PWR and BWR storage racks for Harris Pools C and D are designed in accordance with the applicable codes listed below. The rack design and fuel storage configuration acceptance criteria is to show that the effective neutron multiplication factor, k_{eff}, is equal to or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with un-borated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations including mechanical tolerances. All uncertainties are statistically combined, with uncertainties applied conservatively to calculate the final k_{eff} which must be shown to be less than 0.95 with a 95% probability at a 95% confidence level [4.1.1]. Reactivity effects of abnormal and accident conditions have also been evaluated to assure that under credible abnormal and accident conditions, the reactivity will not exceed the limiting design basis value.

Applicable codes, standards, and regulations or pertinent sections thereof, include the following:

- General Design Criteria 62, Prevention of Criticality in Fuel Storage and Handling.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3 - July 1981.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December 1981.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- ANSI/ANS-57.2-1983, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.

- ANSI N210-1976, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.

USNRC guidelines and the applicable ANSI standards specify that the maximum <u>effective</u> multiplication factor, k_{eff}, including uncertainties, shall be less than or equal to 0.95. The <u>infinite</u> multiplication factor, k_{inf}, is calculated for an infinite array, neglecting neutron losses due to leakage from the actual storage rack, and therefore is a higher and more conservative value. In the present criticality safety evaluation of the Harris storage racks, the design basis criterion was assumed to be a k_{inf} of less than 0.95, which is more conservative than the limit specified in the regulatory guidelines.

To ensure that the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made:

[redacted]

The PWR spent fuel storage racks are designed to accommodate any and all of the fuel assemblies listed in Table 4.3.1 with a maximum enrichment of 5 wt% ^{235}U. To assure the acceptability of the racks for storage of any and all of the above assembly types, the most reactive fuel assembly type was identified and used as the design basis fuel assembly. The Westinghouse 15x15 assembly was determined to have the highest reactivity at zero burnup and as a function of burnup for an initial 5 wt% ^{235}U enrichment and therefore was used as the design basis PWR fuel assembly.

The BWR spent fuel storage racks are designed to accommodate any and all of the fuel assemblies listed in Table 4.3.2 with a maximum planar average enrichments of 4.6 wt.% ^{235}U. Each fuel assembly type was analyzed independently to determine its acceptability in the rack. It is noted that individual fuel rods can have enrichments that are less than or greater than the maximum planar average enrichment.

4.2 Summary of Criticality Analyses

4.2.1 Normal Operating Conditions

4.2.1.1 PWR Fuel Results

The design basis PWR fuel assembly is a 15 x 15 Westinghouse fuel assembly containing UO_2 at a maximum initial enrichment of 5.0 wt% ^{235}U. All fuel assembly types listed in Table 4.3.1 were also evaluated and the Westinghouse 15x15 assembly was shown to exhibit the highest reactivity for the high density PWR storage racks at Harris.

The NRC guidelines specify that the limiting k_{eff} of 0.95 under normal storage conditions should be evaluated in the absence of soluble boron. Consequences of abnormal and accident conditions have also been evaluated assuming no soluble boron, where "abnormal" refers to conditions (such as higher water temperatures) which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions which are not expected to occur but nevertheless must be protected against.

The criticality analyses of the spent fuel storage pool are summarized in Table 4.2.1 for the design basis storage conditions. The maximum k_{eff} is 0.9450 (95% probability at the 95% confidence level) for the enrichment-burnup combinations shown in Figure 4.2.1. The calculated maximum reactivity includes burnup-dependent allowances for uncertainty in depletion calculations and for the axial distribution in burnup. Reactivity allowances for manufacturing tolerances and calculational uncertainties are also included. As cooling time increases in long-term storage, decay of Pu-241 and growth of Am-241 results in a significant decrease in reactivity, which will provide a continuously increasing subcriticality margin for the next 100 years.

The racks can safely accommodate fuel of various initial enrichments and discharge fuel burnups, provided the combination falls within the acceptable domain above the curve in Figure 4.2.1. For convenience, the minimum (limiting) burnup data for unrestricted storage can be described as a linear function of the initial enrichment (E, in weight percent ^{235}U) which conservatively encompasses the limiting burnup data. The equation for this curve is shown in Figure 4.2.1 and provided below.

For Unrestricted Storage of the following PWR fuel assemblies Westinghouse 17x17 Std Westinghouse 17x17 V5 Westinghouse 15x15 Siemens 17x17 Siemens 15x15 the enrichment must be less than or equal to 5 wt% ^{235}U and the burnup must satisfy the minimum burnup requirements Minimum Burnup in MWD/MTU = 12114*E-19123

The burnup criteria will be implemented by appropriate administrative procedures to ensure verified burnup as specified in the proposed Regulatory Guide 1.13, Revision 2, prior to fuel transfer into Spent Fuel Pools C or D.

4.2.1.2 BWR Fuel Results

All BWR fuel assembly types being considered were explicitly analyzed to determine the acceptability for storage in Spent Fuel Pool C. The maximum planar average enrichment was assumed for all rods in the assembly and no credit was taken for gadolinia which might be present.

The criticality safety was evaluated at the burnup corresponding to a k_{inf} of 1.32 in the Standard Cold Core Geometry (SCCG). SCCG is defined as an infinite array of fuel assemblies on a 6-inch lattice spacing at 20°C, without any control absorber or voids.

The maximum k_{eff} in the BWR storage rack was determined to be 0.9443 (95% probability at the 95% confidence level) including all known calculational and manufacturing uncertainties. [redacted] This allowance also encompasses any uncertainty in the burnup calculations.

The basic calculations supporting the criticality safety of the Harris fuel storage racks for the design basis fuel are summarized in Table 4.2.2. For the design basis fuel, the fuel storage rack satisfies the USNRC criterion of a maximum k_{eff} less than or equal to 0.95.

The acceptance criteria for storage of spent BWR fuel in Harris Pool C can be summarized in the following manner.

For Unrestricted Storage of the following BWR fuel assemblies GE 3, GE 4, GE 5, GE 6, GE 7, GE 8, GE 9, GE 10, GE 13 the maximum planar average enrichment must be less than or equal to 4.6 wt.% ^{235}U and the k_{inf} in standard cold core geometry must be less than or equal to 1.32

4.3 Input Parameters

4.3.1 Reference PWR Fuel Assembly and Storage Cell

The design basis PWR fuel assembly is a 15x15 array of fuel rods with 21 rods replaced by 21 control rod guide tubes. Table 4.3.1 summarizes the PWR fuel assembly design specifications for all fuel assemblies analyzed. Figure 4.3.1 shows the calculational model of the PWR spent fuel storage cell containing a 15x15 assembly.

The design basis for the Region 2 type storage cells is fuel of 5.0 wt.% ^{235}U maximum initial enrichment burned to 41,447 MWD/MTU [redacted]

4.3.2 Reference BWR Fuel Assembly and Storage Cell

The design basis BWR fuel assembly, used for uncertainty calculations, is a standard 8x8 array of BWR fuel rods containing UO_2 clad in Zircaloy (60 fuel rods with 4 water rods). Design parameters for all BWR fuel assemblies analyzed are summarized in Table 4.3.2. Figure 4.3.2 shows the calculational model of a BWR storage rack cell containing an 8x8 assembly.

[redacted]

[redacted]

4.4 Analytical Methodology

4.4.1 Reference Design Calculations

In the fuel rack analyses, the primary criticality analyses of the high density spent fuel storage racks were performed with a two-dimensional multigroup transport theory technique, using the CASMO-3 computer code [4.4.1 - 4.4.4]. Since CASMO-3 can not be directly compared to critical experiments, a calculational bias is not available for CASMO-3. Therefore, independent verification calculations were made with a Monte Carlo technique utilizing the MCNP-4A computer code [4.4.5]. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 ± 0.0011 for MCNP-4A, evaluated at the 95% probability, 95% confidence level. The MCNP-4A bias and uncertainty were included in the MCNP-4A to CASMO-3 comparison as discussed in Section 4.5.

CASMO-3 was also used for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. [redacted]

MCNP-4A 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 Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. [redacted]

4.4.2 Burnup Calculations and Uncertainties

CASMO-3 was used for burnup calculations during core operations. CASMO-3 has been extensively verified [4.4.4, 4.4.6] against Monte Carlo calculations, reactor operations, and heavy-element concentrations in irradiated fuel. In addition, Johansson [4.4.7] has obtained very good agreement in calculations of close-packed, high-plutonium-content, experimental configurations.

4.4.2.1 PWR Fuel Burnup Calculations

Since critical experiment data with spent fuel is not available for determining the uncertainty in burnup-dependent reactivity calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations. [redacted] Table 4.4.1 summarizes results of the burnup analyses to determine the allowances for uncertainties in burnup calculations. The reactivity allowances for uncertainties in burnup are listed for three different burnup ranges: less than 30,000 MWD/MTU, between 30,000 and 40,0000 MWD/MTU, and between 40,000 and 45,000 MWD/MTU. The appropriate uncertainty was used for each burnup range in determining the acceptable burnup versus enrichment combinations depicted in Figure 4.2.1. The allowance for uncertainty in burnup calculations is believed to be a conservative estimate, particularly in view of the substantial reactivity decrease with aged fuel as discussed in Section 4.4.4.

4.4.2.2 BWR Fuel Burnup Calculations and Comparison to Vendor Calculations

CASMO-3 was used to perform depletion calculations and to calculate the k_{inf} in the SCCG. As discussed, there are no depleted fuel critical experiments with which to benchmark CASMO-3's depletion calculations. Therefore a reactivity allowance for uncertainty in depletion is needed. [redacted]

[redacted] The allowance is used to also encompass any potential differences between the SCCG calculations performed here and the vendor calculations.

4.4.3 Effect of Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of lower reactivity worth due to neutron leakage. Consequently, it would be expected that over most of the burnup history, distributed burnup fuel assemblies would exhibit a slightly lower reactivity than that calculated for the average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

Generic analytic results of the axial burnup effect have been provided by Turner [4.4.8] based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnup, becoming positive at burnups greater than about 30,000 MWD/MTU. The trends observed [4.4.8] suggest the possibility of a small positive reactivity effect above 30,000 MWD/MTU increasing to slightly over 1% Δk at 40,000 MWD/MTU.

4.4.3.1 PWR Fuel Axial Burnup Distribution

Calculations for the Harris storage racks with PWR fuel of three different average burnups were made using an axial burnup distribution representative of spent PWR fuel[1]. At lower burnups, the

[1] The axial burnup distribution measured on spent fuel from the Surry plant was used as representative of PWR fuel.

reactivity increment is smaller as indicated in Table 4.4.1, being negative at 30,000 MWD/MTU and at lower burnups. No credit is taken for this negative reactivity effect at the lower burnups other than the suggestion of additional conservatism. Furthermore, the reactivity significantly decreases with time in storage (Section 4.4.4 below) providing a continuously increasing margin below the 0.95 limit.

The appropriate reactivity allowance for the effect of axial burnup distribution was used for each burnup range in determining the acceptable burnup versus enrichment values in Figure 4.2.1.

4.4.3.2 BWR Fuel Axial Burnup Distribution

The burnup at which k_{inf} in the SCCG reaches 1.32 is approximately 12,000 MWD/MTU. As discussed above and in [4.4.8] the effect of using the explicit axial burnup distribution as opposed to an average burnup distribution results in a negative effect on reactivity. Therefore, no reactivity allowance for axial burnup distribution is applied to the BWR fuel analysis.

4.4.4 Long Term Reactivity Changes

At reactor shutdown, the reactivity of the fuel initially decreases due to the growth of Xe-135. Subsequently, the Xenon decays and the reactivity increases to a maximum at several hundred hours when the Xenon is gone. Over the next 30 years, the reactivity continuously decreases due primarily to Pu-241 decay and Americium growth. At lower burnup, the reactivity decrease will be less pronounced since less Pu-241 would have been produced. No credit is taken for this long-term decrease in reactivity other than to indicate additional and increasing conservatism in the design criticality analysis.

4.5 PWR Storage Rack Criticality Analyses and Tolerance Variations

4.5.1 Nominal Design Case

The principal method of analysis for the racks was the CASMO-3 code, using the restart option in CASMO-3 to analytically transfer fuel of a specified burnup into the storage rack configuration at a reference temperature of 4°C (39°F). Calculations were made for fuel of several different initial enrichments and, at each enrichment, a limiting k_{inf} value was established which includes reactivity allowances for manufacturing tolerances, the uncertainty in the burnup analyses and for the effect of the axial burnup distribution on reactivity. The restart CASMO-3 calculations (cold, no-Xenon, rack geometry) were then interpolated to define the burnup value yielding the limiting k_{inf} value for each enrichment. A line was fitted to these converged burnup values and this line defines the boundary of the acceptable domain shown in Figure 4.2.1.

An independent MCNP-4A calculation was performed to verify the acceptability of the reference criticality analyses. Fuel of 5.0 wt% initial enrichment was analyzed by MCNP-4A and by CASMO-3. The results of this comparison are presented in Table 4.5.1. In comparing the MCNP values to the CASMO-3 values, the MCNP-4A calculational bias and calculational statistics were included. In addition, the MCNP-4A model correctly included the effect of axial neutron leakage which the CASMO-3 calculations conservatively neglect. Since the MCNP-4A model is at 20 °C and the CASMO-3 model is at 4 °C, a temperature correction had to be applied to the MCNP-4A result. The MCNP-4A result confirms that the reference CASMO-3 calculations are conservative.

4.5.2 Uncertainties Due to PWR Rack Manufacturing Tolerances

All reactivity allowances for manufacturing tolerances are summarized below and listed in Table 4.2.1. Since the tolerances are statistically independent, the allowances are statistically combined into a single reactivity allowance which was used in the final calculations (see Table 4.2.1).

[redacted]

4.5.2.2 Boral Width Tolerance

[redacted]

4.5.2.3 Tolerance in Cell Lattice Spacing and Cell Box Inner Dimension

Since the Region 2 style racks do not utilize a water gap between storage cells, the manufacturing tolerance on inner box dimension is identical to the tolerance on the storage cell lattice spacing. [redacted]

4.5.2.4 Stainless Steel Thickness Tolerance

[redacted]

4.5.2.5 Fuel Enrichment and Density Tolerances

[redacted]

4.6 BWR Storage Rack Criticality Analyses and Tolerance Variations

Nominal Design Case

The two-dimensional CASMO-3 code was used as the principal method of analysis for the Harris spent fuel pool BWR racks. CASMO-3 was used to perform depletion calculations on the fuel assembly and using the restart option in CASMO-3 the fuel of a specified burnup was analytically transferred into the storage rack at a reference temperature of 4°C (39°F). The same fuel of a specified burnup was also analytically transferred into the standard cold core geometry (SCCG) configuration which is an infinite lattice with 6 inch spacing at a temperature of 20°C without any burnable absorber or control blades and no voids. All Xenon which was present during the depletion calculations was removed during the restarts in the rack and SCCG. The reactivity effects of the natural uranium blanket normally located at the ends of the assemblies were conservatively neglected since an infinite fuel length was used.

All fuel assemblies specified were analyzed at the maximum enrichment specified. The maximum k_{eff} in the SCCG was specified as 1.32. Using the CASMO-3 results, the burnup corresponding to a k_{eff} in the SCCG of 1.32 was determined and the corresponding k_{eff} in the rack was determined. The reactivity adjustments were added to the rack k_{eff} to determine the maximum value and this was compared against the 0.95 k_{eff} limit. Based on this analysis, all specified fuel assemblies are acceptable for storage as stated in Section 4.2.1.2. Table 4.2.2 provides the final results of the BWR fuel assembly calculations.

An independent MCNP-4A calculation was used to verify the acceptability of the reference criticality analyses. Fuel of 4.6 wt% initial enrichment was analyzed by MCNP-4A and by CASMO-3. The results of this comparison are presented in Table 4.5.1. In comparing the MCNP values to the CASMO values, the MCNP-4A calculational bias and calculational statistics were included. In addition, the MCNP-4A model correctly included the effect of axial neutron leakage which the CASMO-3 calculations conservatively neglect. Since the MCNP-4A model is at 20 °C and the CASMO-3 model is at 4 °C, a temperature correction had to be applied to the MCNP-4A result. The MCNP-4A result confirm that the reference CASMO-3 calculations are conservative.

4.6.2 Uncertainties Due to Manufacturing Tolerances

The reactivity effects associated with manufacturing tolerances are discussed below and shown in Table 4.2.2. Since the tolerances are statistically independent, the allowances are statistically combined into a single reactivity allowance which was used in the final calculations (see Table 4.2.2).

4.6.2.1 Boron Loading Variation

[redacted]

4.6.2.2 Boral Width Tolerance Variation

[redacted]

4.6.2.3 Tolerance in Cell Lattice Pitch and Inner Box Dimension

Since the Region 2 style racks do not utilize a water gap between storage cells, the manufacturing tolerance on inner box dimension is identical to the tolerance on the storage cell lattice spacing. [redacted]

4.6.2.4 Stainless Steel Thickness Tolerances

[redacted]

4.6.2.5 Fuel Enrichment and Density Variation

The maximum planar average fuel enrichment was specified for each fuel assembly analyzed. Therefore, there is no reactivity allowance for variations in enrichment since the absolute maximum was used for all calculations.

The UO_2 density was specified for each fuel assembly analyzed. [redacted]

4.6.2.6 Zirconium Flow Channel

Elimination of the zirconium flow channel results in a small ([redacted]) decrease in reactivity. More significant is a positive reactivity effect resulting from potential bulging of the zirconium channel, which moves the channel wall outward toward the Boral absorber. It is conservatively assumed that the maximum bulging that could occur would result in the channel touching the cell walls. Since this would not occur over the entire length of the channel, the model assumed that the entire channel was enlarged so that the mid-point of the channel wall was placed equidistant between the nominal channel outer dimension and the cell wall. This results in an incremental reactivity of [redacted] as determined with MCNP-4A.

4.7 Abnormal and Accident Conditions

Strict administrative controls on the fuel transfer to Pools C and D will preclude fuel which is outside the range of the previously stated acceptance criteria from being brought into the spent fuel pool. Therefore, the only potential abnormal and accident conditions that exist are the isplacement of a fuel assembly outside the rack or the dropping of a fuel assembly on top of the rack. It is also possible to inadvertently place a BWR spent fuel assembly in the PWR rack.

4.7.1 Temperature and Water Density Effects

The spent fuel pool temperature coefficient of reactivity is negative. Using the minimum temperature of 4°C, therefore, assures that the true reactivity will always be lower than the calculated value regardless of the temperature. Temperature effects on reactivity have been calculated and the results are shown in Table 4.7.1. Introducing voids in the water internal to the storage cell (to simulate boiling) decreased reactivity, as shown in the table. Boiling at the submerged depth of the racks would occur at approximately 122°C.

4.7.2 Dropped Fuel Assembly

For a drop on top of the rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the fuel in the rack of more than 12 inches (which is considered infinite) , including an estimated allowance for deformation under seismic or accident conditions. At this separation distance, the effect on reactivity is insignificant.

It is also possible to vertically drop an assembly into a location occupied by another assembly. Such a vertical impact would at most cause a small compression of the stored assembly, reducing the water-to-fuel ratio and thereby reducing reactivity. In addition the distance between the active fuel regions of both assemblies will be more than sufficient to ensure no neutron interaction between the two assemblies.

Dropping an assembly into an unoccupied cell could result in a localized deformation of the baseplate of the rack. The resultant effect would be the lowering of a single fuel assembly by the amount of the deformation. This could potentially result in the active fuel height no longer being covered by the boral. The immediate eight surrounding fuel cells could also be affected.

However, the amount of deformation for these cells would be considerably less. The amount of localized deformation would not exceed three inches for a PWR assembly and would therefore be considerably less for the lighter BWR assembly. The criticality effect of this drop accident has been conservatively analyzed and it has been shown that this localized event (nine storage cells at most) has a negligible impact on reactivity.

4.7.3 <u>Lateral Rack Movement</u>

Lateral motion of the rack modules under seismic conditions could potentially alter the spacing between rack modules. Region 2 storage cells do not use a flux-trap and the reactivity is therefore insensitive to the spacing between modules. The spacing between modules is sufficiently large to preclude adverse interaction even with the maximum seismically-induced reduction in spacing.

4.7.4 <u>Abnormal Location of a PWR or BWR Fuel Assembly</u>

Strict administrative controls will prevent an unacceptable assembly, as determined by the acceptance criteria stated in Section 4.2, from being transferred to Harris Pools C and D. Therefore, the only potential mislocation of a fuel assembly is the mislocation of a fuel assembly of equal or lower reactivity to the design basis outside a PWR or BWR rack. Since the racks will have a Boral panel on the outside face (when the outside face is not against a wall) the reactivity effect of a misloaded fuel assembly outside the rack is negligible because of the neutron leakage that occurs from the rack itself. Therefore, the conservative infinite lattice calculations that were performed have k_{∞} values that are higher than any potential mislocation accidents.

Another mislocation event could occur with a BWR assembly This would be the inadvertent placement of a BWR assembly in the PWR racks Since, the BWR assembly is significantly smaller than a PWR assembly, the reactivity effect of placing a BWR assembly in the PWR rack is negligible The reverse scenario of misplacing a PWR assembly in the BWR rack is impossible because of the size of the PWR assembly.

4.7.5 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell and in the case of the BWR rack there are bottom fittings and spacers that mechanically restrict lateral movement of the fuel assemblies. Nevertheless, MCNP-4A calculations were made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicated that eccentric fuel positioning results in a decrease in reactivity ([redacted]). The highest reactivity, therefore, corresponds to the reference design with the fuel assemblies positioned in the center of the storage cells.

4.8 References

[4.1.1] M. G. Natrella, Experimental Statistics, National Bureau of Standards Handbook 91, August 1963.

[4.4.1] A. Ahlin, M. Edenius, H. Haggblom, "CASMO - A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik report (proprietary).

[4.4.2] A. Ahlin and M. Edenius, "CASMO - A Fast Transport Theory Depletion Code for LWR Analysis," *ANS Transactions*, Vol. 26, p. 604, 1977.

[4.4.3] M. Edenius, A. Ahlin, and B. H. Forssen, "CASMO-3 A Fuel Assembly Burnup Program, Users Manual", Studsvik/NFA-87/7, Studsvik Energitechnik AB, November 1986.

[4.4.4] M. Edenius and A. Ahlin, "CASMO-3: New Features, Benchmarking, and Advanced Applications," *Nuclear Science and Engineering*, 100, 342-351, (1988).

[4.4.5] J.F. Briesmeister, Editor, "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A," LA-12625, Los Alamos National Laboratory (1993).

[4.4.6] E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors (Lattice Physics)," YAEC-1232, Yankee Atomic Electric Co., December 1980.

[4.4.7] E. Johansson, "Reactor Physics Calculations on Close-Packed Pressurized Water Reactor Lattices," *Nuclear Technology*, Vol. 68, pp. 263-268, February 1985.

[4.4.8] S. F. Turner, "Uncertainty Analysis - Burnup Distributions", presented at the DOE SANDIA Technical Meeting on Fuel Burnup Credit, Special Session, ANS/ENS Conference, Washington, D.C., November 2, 1988.

Table 4.2.1

Summary of Criticality Safety Calculations for the PWR Fuel Racks

Fuel Assembly	Westinghouse 15x15
Enrichment	5%
Temperature	4°C
Burnup from Calculation (MWD/MTU)	41,352
Burnup from Curve (MWD/MTU)	41,447
CASMO-3 k_{inf}	0.9126
Uncertainties	
UO$_2$ density	
Inner box dimension	
Box wall thickness	
Boral width	
B-10 loading	
Burnup	
Total Uncertainty at 95%/95%	
Effect of Axial Burnup Distribution	
Maximum k_{inf}	**0.9450**
Regulatory Limit	**0.9500**

Notes:

1. Only the most reactive assembly is shown.
2. The total uncertainty is a statistical combination of the manufacturing uncertainties.

Table 4.2.2

Summary of Criticality Safety Calculations for the BWR Fuel Racks

Fuel Assembly		GE 3	GE 4	GE 7	GE 8	GE 9	GE 10	GE 13
Temperature		4°C	4°C	4°C	4°C	4°C	4°C	4°C
SCCG k_{inf}		1.32	1.32	1.32	1.32	1.32	1.32	1.32
Enrichment		4.6	4.6	4.6	4.6	4.6	4.6	4.6
CASMO-3 k_{inf}		0.9163	0.9140	0.9192	0.9214	0.9207	0.9201	0.9227
Uncertainties								
UO_2 density	[redacted]							
Inner box dimension								
Box wall thickness	[redacted]							
Boral width	[redacted]							
B-10 loading								
Total uncertainty at 95%/95%		[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Channel bulging Uncertainty for burnup and vendor comparison		[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Maximum k_{inf}		**0.9379**	**0.9356**	**0.9408**	**0.9430**	**0.9423**	**0.9417**	**0.9443**
Regulatory Limit		**0.9500**	**0.9500**	**0.9500**	**0.9500**	**0.9500**	**0.9500**	**0.9500**

Notes:

1. The total uncertainty is a statistical combination of the manufacturing uncertainties.
2. The GE 13 assembly has part length rods. Two CASMO-3 calculations were performed: one with all rods present and the other with only the full length rods present. The most reactive configuration was the second and the k_{inf} from this configuration is presented.
3. The GE 5 and GE 6 are identical to the GE 7 for the fuel parameters analyzed and therefore the GE 5 and GE 6 have a maximum k_{inf} equivalent to the GE 7.
4. The enrichment is the planar average enrichment.

Table 4.3.1

PWR Fuel Characteristics

Fuel Assembly	Westinghouse 17x17 Std	Westinghouse 17x17 V5	Westinghouse 15x15	Siemens 17x17	Siemens 15x15
NOTE: All Dimensions in inches					
Clad O.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Clad I.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Clad Material	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Pellet Diameter	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Stack Density	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Maximum Enrichment	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Active Fuel Length	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Number Fuel Rods	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fuel Rod Pitch	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Number of Thimbles	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Thimble O.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Thimble I.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

The highlighted data in the table above is the property of Westinghouse or Siemens and is proprietary information provided in confidence. Access to this information shall be limited to those individuals having a need for such access and shall not be disclosed or transmitted to any organization without the written permission of Westinghouse or Siemens, respectively.

Table 4.3.2

BWR Fuel Characteristics

Fuel Assembly	GE 3	GE 4	GE 7	GE 8	GE 9	GE 10	GE 13
NOTE: All dimensions in inches							
Clad O.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Clad I.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Clad Material	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Pellet Diameter	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Stack Density	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Maximum Enrichment	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
SCCG k_{eff}	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Active Fuel Length	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fuel Rod Array	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Number Fuel Rods	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fuel Rod Pitch	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Number of Water Rods	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Water Rod O.D.		[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Water Rod I.D.		[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Channel I.D.	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Channel Thickness	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

Notes:

1. The GE 13 assembly has 8 part length rods.
2. The GE 5 and GE 6 are identical to the GE 7 for the fuel parameters listed.
3. The enrichment is the maximum planar average enrichment.

The highlighted data in the table above is the property of GE and is proprietary information provided in confidence. Access to this information shall be limited to those individuals having a need for such access and shall not be disclosed or transmitted to any organization without the written permission of GE.

Table 4.4.1

Reactivity Allowance for Uncertainty in Burnup Calculations

and the Effect of Axial Burnup Distributions for PWR Fuel

		Δk	
Calculated Burnup (MWD/MTU)	Applicable Burnup Range (MWD/MTU)	Uncertainty in Burnup	Effect of Axial Burnup Distribution
45,000	40,000-45,000	[redacted]	[redacted]
40,000	30,000-40,000	[redacted]	[redacted]
30,000	< 30,000	[redacted]	[redacted]

Notes:

1. The uncertainty in burnup was calculated by taking 5% of the reactivity decrement from zero burnup to the calculated burnup using CASMO-3.
2. The effect of the axial burnup distribution was calculated using MCNP-4A by comparing results from two cases: the first had a uniform axial burnup and the second had a distributed axial burnup distribution represented by 10 axial zones.
3. The effect of the axial burnup distribution is negative at and below 30,000 MWD/MTU, therefore, conservatively no reactivity adjustment was made.

Table 4.5.1

Comparison of MCNP-4A and CASMO-3 Calculations

		PWR Rack	BWR Rack
Fuel Assembly		W 15x15	GE 8
Enrichment		5.0	4.6
Temperature		4°C	4°C
MCNP-4A k_{eff}		1.2004	0.9993
Uncertainties			
Calculational Statistics	[redacted]		
Bias Uncertainty	[redacted]		
Total Uncertainty at 95%/95%		[redacted]	[redacted]
Temperature correction from 20°C to 4°C		[redacted]	[redacted]
Bias		[redacted]	[redacted]
MCNP-4A Maximum k_{eff}		1.2056	1.0045
CASMO-3 k_{∞}		1.2076	1.0126

Notes:

1. The MCNP-4A calculation correctly includes the effect of axial neutron leakage.

Table 4.7.1

Reactivity Effects of Temperature and Void

	Incremental Reactivity Effect - Δk (relative to reference)	
Temperature	PWR Rack	BWR Rack
4°C (39°F)	reference	reference
20°C (68°F)	[redacted]	[redacted]
60°C (140°F)	[redacted]	[redacted]
120°C (248°F)	[redacted]	[redacted]
120°C with 10% void	[redacted]	[redacted]

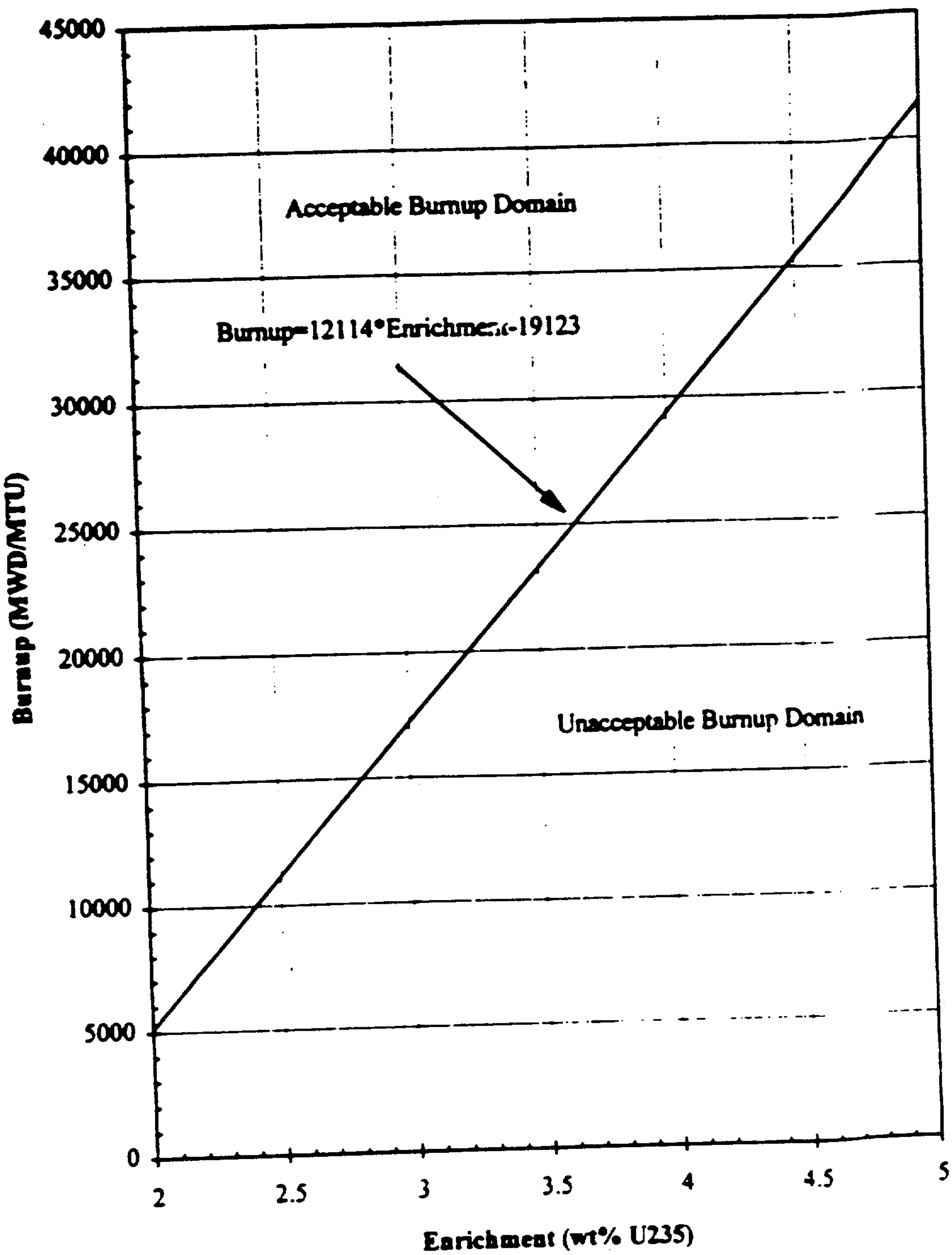
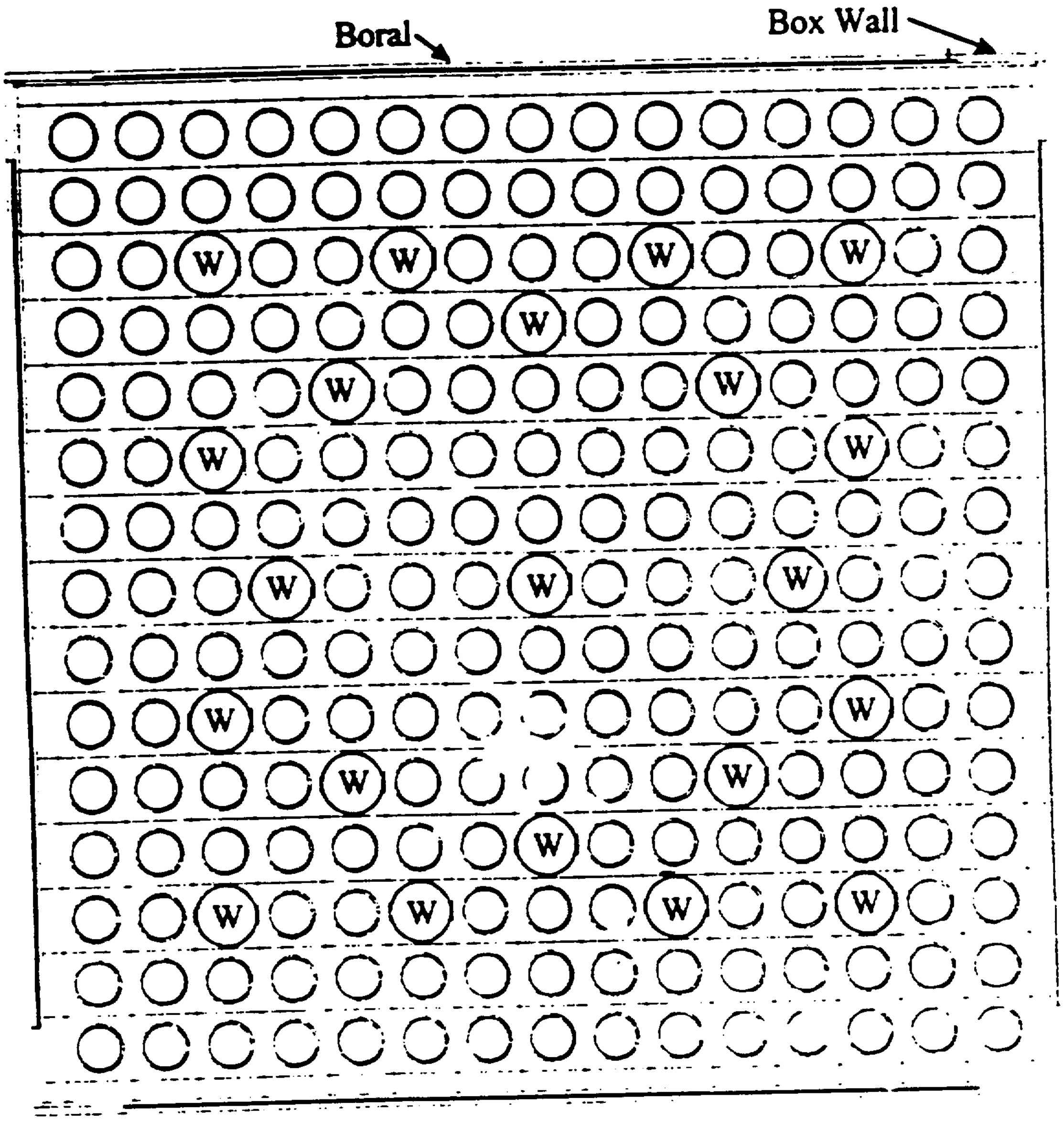
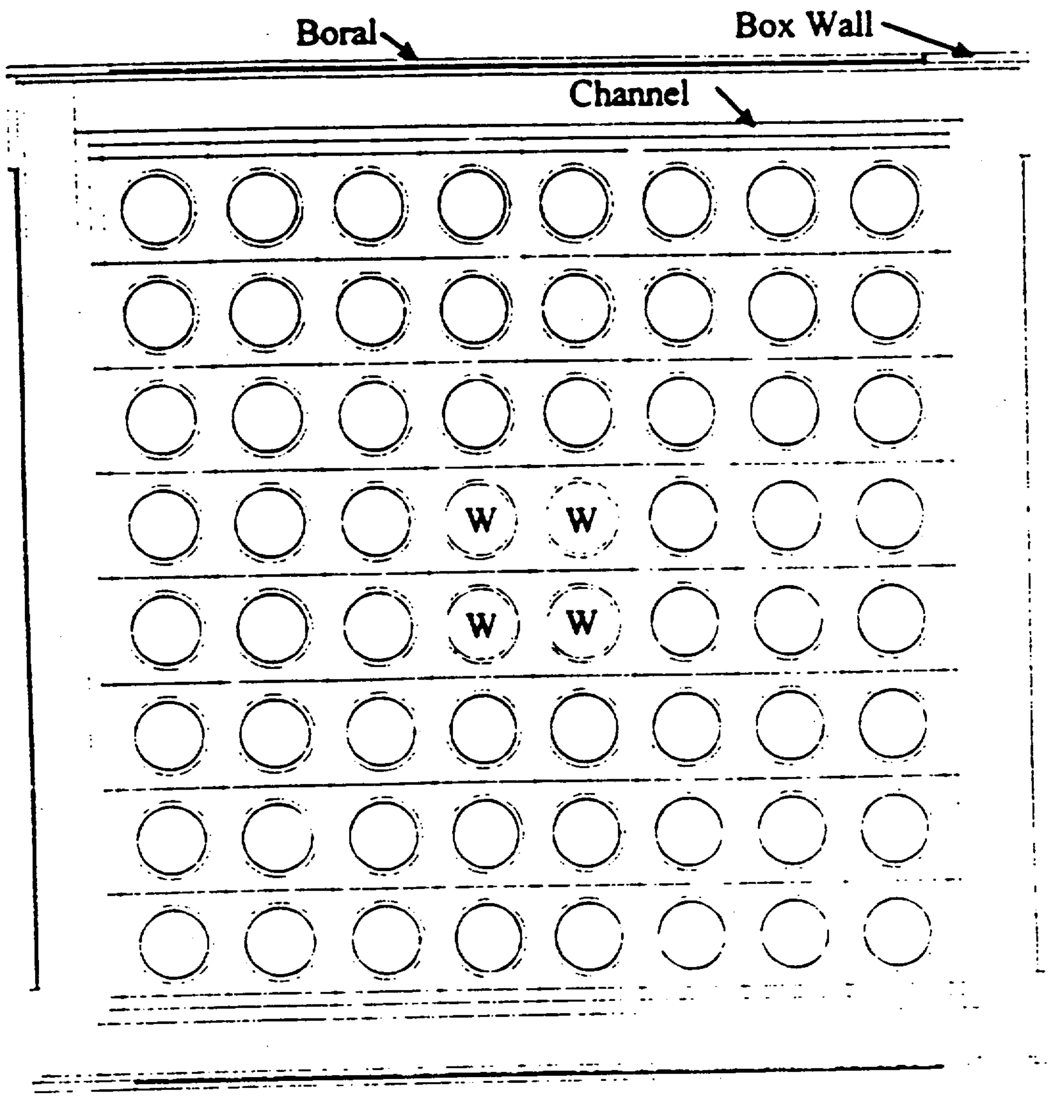


Figure 4.2.1: Burnup Versus Enrichment for PWR Fuel



W = guide tubes

Figure 4.3.1: This is a two dimensional representation of the calculational model used for the PWR storage rack analysis showing a Westinghouse 15x15 fuel design. This figure was drawn with the two dimensional plotter in MCNP-4A.



W = water rod

Figure 4.3.2: This is a two dimensional representation of the calculational model used for the BWR storage rack analysis showing a GE 8 fuel design. This figure was drawn with the two dimensional plotter in MCNP-4A.

5.0 THERMAL-HYDRAULIC CONSIDERATIONS

5.1 Introduction

This section provides a summary of the methods, models, analyses and numerical results to demonstrate the compliance of Harris Spent Fuel Pools C and D with the provisions of Section III of the USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications", (April 14, 1978) for a bounding configuration. Similar methods of thermal-hydraulic analysis have been used in other rerack licensing projects (see Table 5.1.1).

The thermal-hydraulic qualification analyses for the rack array may be broken down into the following categories:

(i) Evaluation of the long-term decay heat load, which is the cumulative spent fuel decay heat generation from all fuel assemblies stored in the C and D pools.

(ii) Evaluation of the steady-state bulk pool temperatures when forced cooling is available. The bulk pool temperatures are required to be maintained ≤ 137°F † under normal conditions with fuel pool cooling in operation.

(iii) Determination of the maximum pool local temperature at steady bulk pool temperatures.

(iv) Evaluation of the potential for flow bypass from pool inlet to outlet in the absence of a sparger line to the spent fuel pools racks.

(v) Evaluation of the "time-to-boil" if all forced heat rejection paths from the pool are lost.

† The 137°F limit is consistent with that currently in the Harris FSAR and procedures for pools A and B. CP&L is in the process of re-evaluating systems and components to allow for an increase in the allowable bulk pool temperature.

5.2 Decay Heat Load Calculations

This section presents a synopsis of the analysis methods employed, and final results. The decay heat load calculation is conservatively performed in accordance with the provisions of USNRC Branch Technical Position ASB9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling", Rev. 2, July, 1981.

The Pool C and D fuel rack configurations for proposed expansion are depicted in Figures 1.2 and 1.3. A total of 1,952 PWR cells and 2,763 BWR cells will be available in a bounding configuration to maximize fuel storage capacity.

To determine the limiting decay heat in the Harris spent fuel pools, a projected bounding decay period for fuel scenario is considered as shown in Table 5.2.1. The in-core irradiation time and limiting assembly specific power inputs are provided in Table 5.2.2. The C and D spent fuel pools (SFPs) are designated to store old fuel which has been cooled for at least 5 years. The fuel is envisaged to be transhipped from Brunswick and Robinson plants or shuffled from Harris' Pools A and/or B.

Since the decay heat load from the old assemblies varies very slowly as a function of time, the long-term decay heat in the bounding configuration is assumed to be constant. Based on the discharge scenario and fuel assemblies characteristics listed in Tables 5.2.1 and 5.2.2, the combined Pools C and D decay heat rates are determined and summarized in Table 5.2.3.

5.3 Bulk Pool Temperatures

The decay heat load to the two pools (C and D) will be removed by several passive and active heat rejection mechanisms, as listed below:

(a)
(b)
(c)
(d)
(e)
(f)

In the interest of conservatism, no credit is applied to removing heat by any of the mechanisms listed above from (a) to (e). Consequently, all of the decay heat generated in the C and D pools is considered to be removed by the forced flow of SFP cooling water circulating through a heat exchanger, which transfers heat to the CCW system. In a forced SFP cooling scenario, hot water from the pool is circulated by a pump through an exchanger cooled by the CCW system. The cooled SFP water is then directed back to the C and D pools. The decay heat load in the C and D pools is from old fuel discharges, which is relatively constant (i.e., steady heat load). Therefore, at equilibrium conditions, the total decay heat load to the pool is equal to the heat removed by the cooling system and a constant bulk temperature is maintained in the C and D pools.

The heat removal capacity of the SFP cooling system is principally characterized by two parameters, namely the water circulation flow rate and the fuel pool inlet water temperature. The bulk pool temperature of pools C and D is required to be maintained at or below 137°F *. The minimum SFP water flow rate required to comply with this bulk pool temperature criterion is thus a function of the fuel pool inlet water temperature. This requirement is graphically illustrated in Figure 5.3.1. A SFP cooling system design point, which is on the curve, satisfies the minimum cooling requirements. A design point above this curve exceeds the SFP cooling

---

* The 137°F limit is consistent with that currently in the Harris FSAR and procedures for pools A and B. CP&L is in the process of re-evaluating systems and components to allow for an increase in the allowable bulk pool temperature

requirements. Therefore, Figure 5.3.1 establishes the thermal-hydraulic design basis for SFP cooling system capacity and the final cooling system design shall comply with these flow vs. inlet temperature parameters.

5.4 Local Temperature Analysis

In this section, we present the methodology for calculating the local temperatures when forced cooling is available to the Pool C only. The results from evaluations performed with forced cooling in pool C only are conservative, since the pool cooling system will be connected to both pools and cooling water will be discharged to both pools. Therefore, these evaluations predict conservative local temperatures, especially in pool D.

Truncation of sparger lines has become a standard pool modification procedure in rerack campaigns in recent years. Over a dozen SFPs reracked in the past several years have removed sparger lines to enable a high density storage layout and thus maximize pool capacity. Absence of a sparger in the Harris C and D pools removes the mechanistic feed of cold water into the bottom plenum of the fuel racks. It is not apparent from heuristic reasoning alone that the cooled water delivered to the pool would not bypass the hot fuel racks and the stored spent fuel in the two pools and exit through the outlet piping. To demonstrate adequate cooling of fuel in the two areas, it is therefore necessary to rigorously quantify the velocity field in the pool created by the interaction of buoyancy driven flows and water ingress/egress. A CFD analysis for this demonstration is required. The objective of this study is to demonstrate that the principal thermal-hydraulic criteria of ensuring local subcooled conditions in the pool is met for the bounding fuel storage configuration. An outline of the CFD approach is described in the following.

Figure 5.4.1 depicts the fuel Pools C and D physical configuration in plan view. The two pools are connected by a transfer canal. Pool piping connections for introducing cooling water and discharge of heated water are shown for both pools. Currently, SFP cooling system design work is in progress to provide a forced cooling system which will provide suction and discharge to both pools. Thermal-hydraulic adequacy of the two pools shall be conservatively demonstrated by assuming that forced cooling is available to only Pool C. Adequate cooling of Pool D is enabled by a buoyancy-driven flow of relatively cooler bulk Pool C water to Pool D through the interconnecting transfer canal. Decay heat inputs to both pools are based on a bounding fuel storage configuration and spent fuel cooling times. The buoyancy-induced cooling of Pool D is demonstrated by performing a rigorous Computational Fluid Dynamics (CFD) analysis of the temperature and flow fields in the two pools. The CFD methodology is discussed in the next subsection. An additional assumption about the location of cooling inlet and outlet piping is included in the analysis to result in an extremely conservative thermal-hydraulic portrayal of the two interconnected pools. The pool cooling inlet and outlet piping connections are assumed to be located on the southeast end of the pool. Thus, forced cooling of the pool is in a diagonally opposite (i.e., farthest) corner from the northwest location of the connection from Pool C to the transfer canal. The forced cooling ingress and egress locations are in close proximity to each other and at the same elevation. The potential for flow bypass from inlet to outlet is conservative, since the modeled locations are closer than the actual relative positions.

There are several significant geometric and thermal-hydraulic features of the Harris SFPs which need to be considered for a rigorous CFD analysis. From a fluid flow modeling standpoint, there are two regions to be considered. One region is the bulk pool region where the classical Navier-Stokes equations are solved with turbulence effects included. The other region is the heat generating fuel assemblies located in the spent fuel racks located near the bottom of the SFP. In this region, water flow is directed vertically upwards due to buoyancy forces through relatively small flow channels formed by stored fuel assembly rod arrays in each rack cell.

[redacted] Bounding permeability and inertial resistance parameters for the rack cells loaded with PWR or BWR fuel is determined based on friction factor correlations for laminar flow conditions typically encountered due to low buoyancy induced velocities and small size of the flow channels. A large number of fuel assembly types have been analyzed for hydraulic flow resistance [5.4.1] determination. Table 5.4.1 provides flow resistance parameters which bound all PWR and BWR fuel assembly types which were analyzed in this study.

The pool geometry requires an adequate portrayal of large scale and small scale features, spatially distributed heat sources in the spent fuel racks and water inlet/outlet configuration. Relatively cooler bulk pool water normally flows down through the narrow fuel rack outline to pool wall liner clearance known as the downcomer. Near the bottom of the racks, the flow turns from a vertical to horizontal direction into the bottom plenum supplying cooling water to the rack cells. Heated water issuing out of the top of the racks mixes with the bulk pool water. An adequate modeling of these features on the CFD program involves meshing the large scale bulk pool region and small scale downcomer and bottom plenum regions with sufficient number of computational cells to capture the bulk and local features of the flow field.

(i)
(ii)
(iii)

5.4.1 Time-to-Boil

If all heat exchanger assisted forced pool cooling becomes unavailable, then the pool water will begin to rise in temperature and eventually will reach the normal bulk boiling temperature at 212°F. The time to reach the boiling point will be the shortest when the loss of forced cooling occurs at the point in time when the bulk pool temperature is at its maximum calculated value for a bounding fuel storage configuration. The calculation is conservatively performed for a bounding decay heat load to the pool, no credit for evaporation cooling and no credit for thermal inertia of racks. The amount of water holdup above the racks in the two pools is in excess of 48,000 ft^3 (2.9×10^6 lbs) of water. The maximum rate of temperature rise of bulk pools water at a bounding 15.63 million Btu/hr decay heat input (Table 5.2.3) is therefore less than 5.4°F/hr with no water makeup. If the initial temperature is conservatively assumed to be at a uniform maximum bulk average limit of 140°F †, then the time to reach normal boiling point of the bulk pool is in excess of 13 hours. This is a relatively long time period for operator action to start makeup water and re-initiate forced cooling to the pool.

5.5 CFD Analysis of C and D Fuel Pools

A summary of pools dimensional data used to generate a C nputational Fluid Dynamics (CFD) model of the two interconnected C and D pools is provided in Table 5.5.1. The CFD model provides a determination of the difference between the peak local and bulk pool temperatures. The local temperature corresponding to the maximum bulk pool temperature can then be determined by adding this local temperature rise to the bulk temperature limit. In the CFD model, a ***minimum*** bounding downcomer gap between racks outline to pool liner is applied as noted in Table 5.5.1. In this manner, the downcomer water flow path hydraulic resistance is ***maximized***. Consequently, the local rack cell temperature predictions shall be conservatively maximized. The background constant decay heat input to the pool is modeled as a uniform

† The assumption of an initial temperature of 140°F is conservative, since the bulk pool temperature is currently limited to 137°F.

volumetric heat source term in the active fuel region of the Pools C and D racks. The total heat generating volume is calculated to be 657 m^3. Thus, from the total decay heat input (Table 5.2.3), the volumetric heat source term is determined to be 6,956 W/m^3.

A plan view of the three-dimensional CFD model is presented in Figure 5.5.1. In this view, the two pools with an interconnected transfer canal is depicted. The water inlet/outlet connections are shown modeled in the top left end corner of the Pool C. The racks outline, modeled as a porous media, is depicted in blue color. A perspective view of the CFD model is presented in Figure 5.5.2. The bottom of the transfer canal, as shown in this figure, is at the same elevation as the top of the racks. The average background decay heat is applied to the model as a volumetric heat source term in the active fuel region of the fuel racks. The CFD model of the C and D pools is solved to obtain converged temperature and velocity profiles. The results obtained from the analysis are discussed next.

Peak local water temperature in the rack cells is shown as a contour plot in cross sectional plan view as shown in Figure 5.5.3. The plan view elevation is within the region of the racks above the active fuel region, but below the top of the racks.

An exchange of cold and hot water streams from the Pool D to Pool C is determined by the CFD solution with only pool C cooled by a forced cooling system. This exchange of cold and hot water between the two bulk pools is illustrated as a flow velocity vectors plot (Figure 5.5.4) in the pools' interconnecting channel. The peak local temperature is 6.8°F above the water temperature at the cooling system discharge from pool. Consequently, the peak local temperature corresponding to the maximum bulk pool temperature limit is obtained by adding this local temperature rise. Table 5.5.2 provides the bulk and local temperature summaries. The peak 143.8°F local temperature is below the local water boiling temperature by a large margin.

Figure 5.5.5 provides a flow velocity vectors plot in the pool cooling inlet/outlet piping region. The pool inlet piping is modeled to be 12 inches below the pool water level and the pool outlet piping suction is adjacent to the inlet piping discharge at the same elevation. From the velocity vectors plot, it is apparent that no bypass of incoming water to outlet is indicated for an extremely conservative configuration. In the actual pool piping arrangement for Pool C, the water inlet and outlet connections are widely separated. Consequently, it is concluded that any water bypass from inlet to outlet is *not* possible.

5.6 References

[5.4.1] Holtec Report HI-951325, "HI-STAR 100 System Thermal Design Package".

[5.4.2] "QA Documentation and Validation of the FLUENT Version 4.3 CFD Analysis Program", Holtec Report HI-961444.

[5.4.3] Batchelor, G.K., "An Introduction to Fluid Dynamics", Cambridge University Press, 1967.

[5.4.4] Hinze, J.O., "Turbulence", McGraw Hill Publishing Co., New York, NY, 1975.

[5.4.5] Launder, B.E., and Spalding, D.B., "Lectures in Mathematical Models of Turbulence", Academic Press, London, 1972.

Table 5.1.1

PARTIAL LISTING OF RERACK APPLICATIONS USING SIMILAR METHODS OF THERMAL-HYDRAULIC ANALYSIS

PLANT	DOCKET NO.
Enrico Fermi Unit 2	USNRC 50-341
Quad Cities 1 and 2	USNRC 50-254, 50-265
Rancho Seco	USNRC 50-312
Grand Gulf Unit 1	USNRC 50-416
Oyster Creek	USNRC 50-219
Pilgrim	USNRC 50-293
V.C. Summer	USNRC 50-395
Diablo Canyon Units 1 and 2	USNRC 50-275, 50-455
Byron Units 1 and 2	USNRC 50-454, 50-455
Braidwood Units 1 and 2	USNRC 50-456, 50-457
Vogtle Unit 2	USNRC 50-425
St. Lucie Unit 1	USNRC 50-425
Millstone Point Unit 1	USNRC 50-245
D.C. Cook Units 1 and 2	USNRC 50-315, 50-316
Indian Point Unit 2	USNRC 50-247
Three Mile Island Unit 1	USNRC 50-289
J.A. FitzPatrick	USNRC 50-333
Shearon Harris Unit 2	USNRC 50-401
Hope Creek	USNRC 50-354
Kuosheng Units 1 and 2	Taiwan Power Company
Chin Shan Units 1 and 2	Taiwan Power Company

Table 5.1.1 (continued) PARTIAL LISTING OF RERACK APPLICATIONS USING SIMILAR METHODS OF THERMAL-HYDRAULIC ANALYSIS	
PLANT	**DOCKET NO.**
Ulchin Unit 2	Korea Electric Power Corporation
Laguna Verde Units 1 and 2	Comision Federal de Electricidad
Zion Station Units 1 and 2	USNRC 50-295, 50-304
Sequoyah	USNRC 50-327, 50-328
La Salle Unit One	USNRC 50-373
Duane Arnold	USNRC 50-331
Fort Calhoun	USNRC 50-285
Nine Mile Point Unit One	USNRC 50-220
Beaver Valley Unit One	USNRC 50-334
Limerick Unit 2	USNRC 50-353
Ulchin Unit 1	Korea Electric Power Corporation

Table 5.2.1 DECAY PERIODS FOR A BOUNDING POOLS C AND D STORAGE CONFIGURATION			
PWR Fuel Assemblies		**BWR Fuel Assemblies**	
Number of Assys	Decay Period	Number of Assys	Decay Period
172	5 years	456	5 years
172	7 years	456	7 years
172	9 years	456	9 years
172	11 years	456	11 years
172	13 years	456	13 years
172	15 years	483	15 years
172	17 years		
172	19 years		
172	21 years		
172	23 years		
232	25 years		

Table 5.2.2 FUEL ASSEMBLIES INPUT DATA FOR DECAY HEAT EVALUATION	
Item	**Value**
PWR Assembly Irradiation Time	1,915 EFPD[†]
PWR Assembly Specific Power	19.11 MWt
BWR Assembly Irradiation Time	2,028 EFPD
BWR Assembly Specific Power	4.66 MWt

† Effective Full Power Days

Table 5.2.3

BOUNDING DECAY HEAT INPUT FROM STORED FUEL IN POOLS C AND D

Fuel Assemblies	Decay Heat Load (Million Btu/hr)
BWR Fuel Assemblies	4.47
PWR Fuel Assemblies	11.16
Total	15.63 (4.57 MW)

Table 5.4.1 BOUNDING FUEL ASSEMBLIES HYDRAULIC FLOW RESISTANCE PARAMETERS	
Parameter	**Value**
Permeability	10^{-6} m^2
Inertial Resistance Factor	95 m^{-1}

Table 5.5.1 POOLS C AND D DIMENSIONAL DATA	
Parameter	**Value**
Pool C: Length Width	597.88" 320.60"
Pool D: Length Width	383.36" 237.79"
Water Depth	38.5 ft
Pools-to-Transfer Canal Channel Width	24"
Bottom Plenum	6"
Pool C Downcomers North Wall South Wall East Wall West Wall	 1.44"* 1.44" 2.36" 2.36"
Pool D Downcomers North Wall South Wall East Wall West Wall	 5.15" 5.0" 5.0" 5.0"

* A minimum uniform downcomer gap equal to 1.44" applied to both pools for CFD analysis.

Table 5.5.2 BULK AND LOCAL TEMPERATURES SUMMARY	
Item	**Temperature (°F)** *
Local temperature rise above bulk	6.8
Bulk pool maximum temperature limit	137.0
Peak Local Temperature	143.8

* Local temperature values are conservatively computed based on neglecting forced cooling to pool D, as discussed at the beginning of Section 5.4

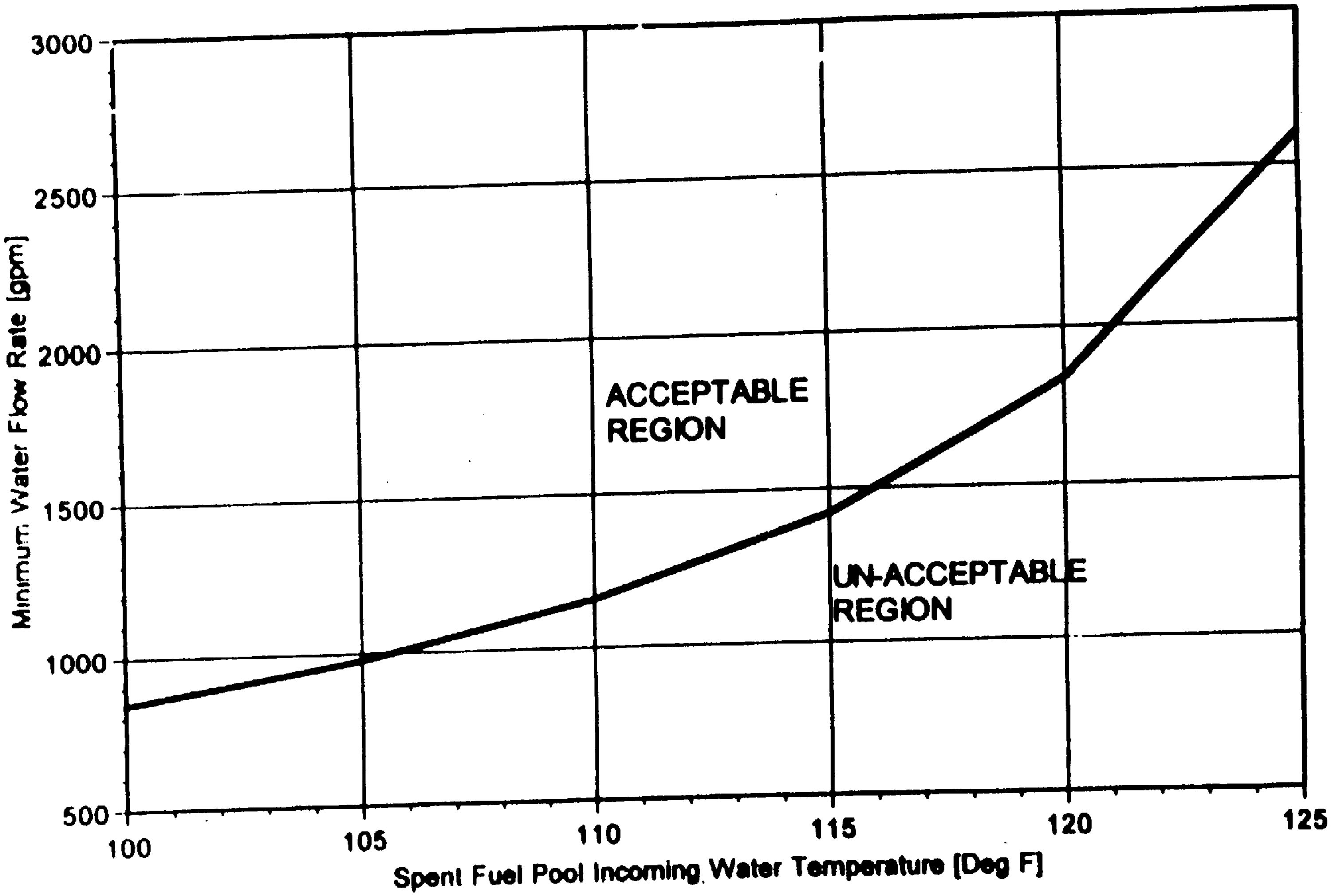


FIGURE 5.3.1: C AND D POOLS MINIMUM TOTAL COOLING SYSTEM REQUIREMENTS CURVE AT 137 Deg. F BULK POOL TEMPERATURE

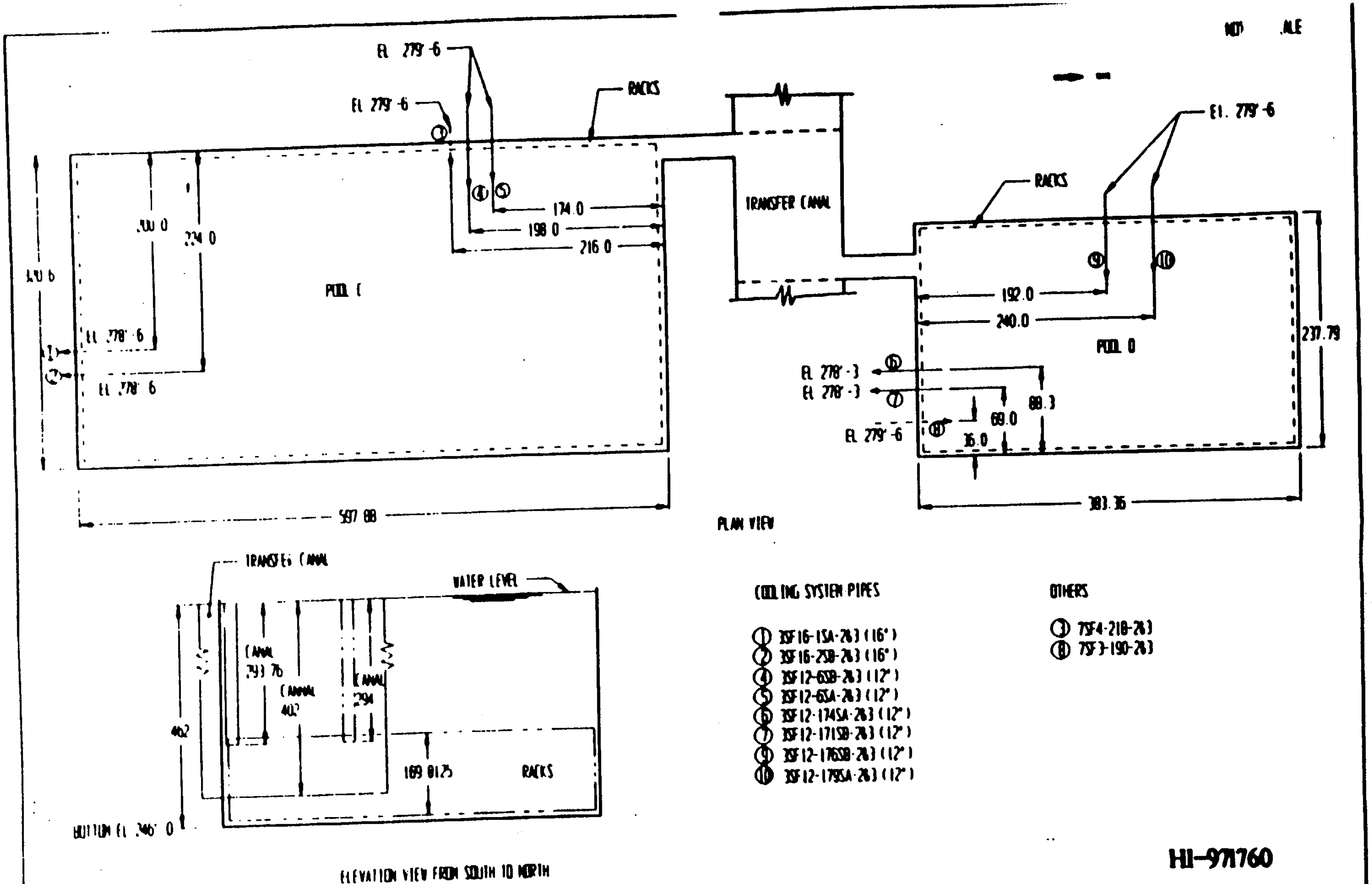


FIGURE 5.4.1: HARRIS C AND D POOLS PHYSICAL CONFIGURATION

FIGURE 5.5.1; PLAN VIEW OF THE HARRIS POOLS C AND D CFD MODEL

FIGURE 5.5.2; PERSPECTIVE VIEW OF THE HARRIS POOLS C AND D CFD MODEL

FIGURE 5.5.3; PEAK LOCAL WATER TEMPERATURE IN THE RACK CELLS

FIGURE 5.5.4: POOLS INTERCONNECTING CHANNEL FLOW VELOCITY VECTORS ELEVATION VIEW PLOT

FIGURE 5.5.5; POOL COOLING INLET/OUTLET PIPING REGION FLOW VELOCITY VECTORS PLOT

6.0 STRUCTURAL/SEISMIC CONSIDERATIONS

6.1 Introduction

This section considers the structural adequacy of the new maximum density spent fuel racks under all loadings postulated for normal, seismic, and accident conditions at Harris. The existing spent fuel storage racks are also examined for stability during the installation process. The analyzed storage rack configurations with the new racks in place are shown in Figures 1.2 and 1.3.

The analyses undertaken to confirm the structural integrity of the racks are performed in compliance with the USNRC Standard Review Plan [6.1.1] and the OT Position Paper [6.1.2]. For each of the analyses, an abstract of the methodology, modeling assumptions, key results, and summary of parametric evaluations are presented. Delineation of the relevant criteria are discussed in the text associated with each analysis.

6.2 Overview of Rack Structural Analysis Methodology

The response of a free-standing rack module to seismic inputs is highly nonlinear and involves a complex combination of motions (sliding, rocking, twisting, and turning), resulting in impacts and friction effects. Some of the unique attributes of the rack dynamic behavior include a large fraction of the total structural mass in a confined rattling motion, friction support of rack pedestals against lateral motion, and large fluid coupling effects due to deep submergence and independent motion of closely spaced adjacent structures.

Linear methods, such as modal analysis and response spectrum techniques, cannot accurately simulate the structural response of such a highly nonlinear structure to seismic excitation. An accurate simulation is obtained only by direct integration of the nonlinear equations of motion with the three pool slab acceleration time-histories applied as the forcing functions acting simultaneously.

Whole Pool Multi-Rack (WPMR) analysis is the vehicle utilized in this project to simulate the dynamic behavior of the complex storage rack structures. The following sections provide the basis for this selection and discussion on the development of the methodology.

6.2.1 Background of Analysis Methodology

Reliable assessment of the stress field and kinematic behavior of the rack modules calls for a conservative dynamic model incorporating all *key attributes* of the actual structure. This means that the model must feature the ability to execute the concurrent motion forms compatible with the free-standing installation of the modules.

The model must possess the capability to effect momentum transfers which occur due to rattling of fuel assemblies inside storage cells and the capability to simulate lift-off and subsequent impact of support pedestals with the pool liner (or bearing pad). The contribution of the water mass in the interstitial spaces around the rack modules and within the storage cells must be modeled in an accurate manner since erring in quantification of fluid coupling on either side of the actual value is no guarantee of conservatism.

The Coulomb friction coefficient at the pedestal-to-pool liner (or bearing pad) interface may lie in a rather wide range and a conservative value of friction cannot be prescribed *a priori*. In fact, a perusal of results of rack dynamic analyses in numerous dockets (Table 6.2.1) indicates that an upper bound value of the coefficient of friction often maximizes the computed rack displacements as well as the equivalent elastostatic stresses.

In short, there are a large number of parameters with potential influence on the rack kinematics. The comprehensive structural evaluation must deal with all of these without sacrificing conservatism.

The three-dimensional single rack dynamic model introduced by Holtec International in the Enrico Fermi Unit 2 rack project (ca. 1980) and used in some 50 rerack projects since that time (Table 6.2.1) addresses most of the above mentioned array of parameters. The details of this methodology are also published in the permanent literature [6.2.1]. Despite the versatility of the 3-D seismic model, the accuracy of the single rack simulations has been suspect due to one key element; namely, hydrodynamic participation of water around the racks. During dynamic rack motion, hydraulic energy is either drawn from or added to the moving rack, modifying its submerged motion in a significant manner. Therefore, the dynamics of one rack affects the motion of all others in the pool.

A dynamic simulation which treats only one rack, or a small grouping of racks, is intrinsically inadequate to predict the motion of rack modules with any quantifiable level of accuracy. Three-dimensional Whole Pool Multi-Rack analyses carried out on several previous plants demonstrate that single rack simulations under predict rack displacement during seismic responses [6.2.2].

Briefly, the 3-D rack model dynamic simulation, involving one or more spent fuel racks, handles the array of variables as follows:

Interface Coefficient of Friction: Parametric runs are made with upper bound and lower bound values of the coefficient of friction. The limiting values are based on experimental data which have been found to be bounded by the values 0.2 and 0.8. Simulations are also performed with the array of pedestals having randomly chosen coefficients of friction in a Gaussian distribution with a mean of 0.5 and lower and upper limits of 0.2 and 0.8, respectively. In the fuel rack simulations, the Coulomb friction interface between rack support pedestal and liner is simulated by piecewise linear (friction) elements. These elements function only when the pedestal is physically in contact with the pool liner.

Rack Beam Behavior: Rack elasticity, relative to the rack base, is included in the model by introducing linear springs to represent the elastic bending action, twisting, and extensions.

Impact Phenomena: Compression-only gap elements are used to provide for opening and closing of interfaces such as the pedestal-to-bearing pad interface, and the fuel assembly-to-cell wall interface. These interface gaps are modeled using nonlinear spring elements. The term "nonlinear spring" is a generic term used to denote the mathematical representation of the condition where a restoring force is not linearly proportional to displacement.

Fuel Loading Scenario: The fuel assemblies are conservatively assumed to rattle in unison which obviously exaggerates the contribution of impact against the cell wall.

Fluid Coupling: Holtec International extended Fritz's classical two-body fluid coupling model to multiple bodies and utilized it to perform the first two-dimensional multi-rack analysis (Diablo Canyon, ca. 1987). Subsequently, laboratory experiments were conducted to validate the multi-rack fluid coupling theory. This technology was incorporated in the computer code DYNARACK (a.k.a. MR216) [6.2.4] which handles simultaneous simulation of all racks in the pool as a Whole Pool Multi-Rack 3-D analysis. This development was first utilized in Chinshan, Oyster Creek, in earlier projects at the Harris plant [6.2.1, 6.2.3] and, subsequently, in numerous other rerack projects. The WPMR analyses have corroborated the accuracy of the single rack 3-D solutions in predicting the maximum structural stresses, and also serve to improve predictions of rack kinematics.

For closely spaced racks, demonstration of kinematic compliance is verified by including all modules in one comprehensive simulation using a WPMR model. In WPMR analysis, all rack modules are modeled simultaneously and the coupling effect due to this multi-body motion is included in the analysis. Due to the superiority of this technique in predicting the dynamic behavior of closely spaced submerged storage racks, the Whole Pool Multi-Rack analysis methodology is used for this project.

6.3 Description of Racks

The implementation of the storage capacity increase in pools C and D will be performed on an as needed basis through incremental phases (campaigns). Figures 6.3.1 and 6.3.2 identify the fully implemented configuration and also designates which racks will be included in each of the campaigns. The new high density storage racks are analyzed for the anticipated configurations at the completion of each of the installation campaigns. Evaluated configurations of the two pools are also handled separately, since the pools are physically separated by the surrounding concrete walls. The analyzed configurations considered are described as follows:

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

The materials utilized in fabrication of the rack components are identified in Table 6.3.1.

The cartesian coordinate system utilized within the rack dynamic model has the following nomenclature:

x = Horizontal ax.. along plant North
y = Horizontal axis along plant West
z = Vertical axis upward from the rack base

6.3.1 Fuel Weights

For the dynamic rack simulations, the dry PWR fuel weight is taken to be 1600 lbs and the dry BWR fuel weight is taken to be 680 lbs.

6.4 Synthetic Time-Histories

The synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) are generated in accordance with the provisions of SRP 3.7.1 [6.4.1]. In order to prepare an acceptable set of acceleration time-histories, Holtec International's proprietary code GENEQ [6.4.2] is utilized.

A preferred criterion for the synthetic time-histories in SRP 3.7.1 calls for both the response spectrum and the power spectral density corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions. The time-histories for the pools have been generated to satisfy this preferred (and more rigorous) criterion. The seismic files also satisfy the requirements of statistical independence mandated by SRP 3.7.1.

Figures 6.4.1 through 6.4.3 and 6.4.4 through 6.4.6 provide plots of the time-history accelerograms which were generated over a 20 second duration for OBE and SSE events, respectively.

Results of the correlation function of the three time-histories are given in Table 6.4.1. Absolute values of the correlation coefficients are shown to be less than 0.15, indicating that the desired statistical independence of the three data sets has been met.

6.5 WPMR Methodology

Recognizing that the analysis work effort must deal with both stress and displacement criteria, the sequence of model development and analysis steps that are undertaken are summarized in the following:

a.

b.

c.

6.5.1 Model Details for Spent Fuel Racks

The dynamic modeling of the rack structure is prepared with special consideration of all nonlinearities and parametric variations. Particulars of modeling details and assumptions for the Whole Pool Multi-Rack analysis of racks are given in the following:

6.5.1.1 Assumptions

a.

b.

c.

d.

e.

f.

g.

h.

i.

6.5.1.2 Element Details

Figure 6.5.1 shows a schematic of the dynamic model of a single rack. The schematic depicts many of the characteristics of the model including all of the degrees-of-freedom and some of the spring restraint elements.

Table 6.5.1 provides a complete listing of each of the 22 degrees-of-freedom for a rack model. Six translational and six rotational degrees-of-freedom (three of each type on each end) describe the motion of the rack structure. Rattling fuel mass motions (shown at nodes 1^*, 2^*, 3^*, 4^*, and 5^* in Figure 6.5.1) are described by ten horizontal translational degrees-of-freedom (two at each of the five fuel masses). The vertical fuel mass motion is assumed (and modeled) to be the same as that of the rack baseplate.

Figure 6.5.2 depicts the fuel to rack impact springs (used to develop potential impact loads between the fuel assembly mass and rack cell inner walls) in a schematic isometric. Only one of the five fuel masses is shown in this figure. Four compression only springs, acting in the horizontal direction, are provided at each fuel mass.

Figure 6.5.3 provides a 2-D schematic elevation of the storage rack model, discussed in more detail in Section 6.5.3. This view shows the vertical location of the five storage masses and some of the support pedestal spring members.

Figure 6.5.4 shows the modeling technique and degrees-of-freedom associated with rack elasticity. In each bending plane a shear and bending spring simulate elastic effects [6.5.4].

Linear elastic springs coupling rack vertical and torsional degrees-of-freedom are also included in the model.

Figure 6.5.5 depicts a single rack module with its surrounding impact springs (used to develop potential impact loads between racks or between rack and wall). Figures 6.5.6 through 6.5.13 show the rack numbering schemes used for the WPMR analyses of both pools. These figures also provide the numbering scheme for all of the rack periphery compression only gap elements.

6.5.2 Fluid Coupling Effect

In its simplest form, the so-called "fluid coupling effect" [6.5.2, 6.5.3] can be explained by considering the proximate motion of two bodies under water. If one body (mass m_1) vibrates adjacent to a second body (mass m_2), and both bodies are submerged in frictionless fluid, then Newton's equations of motion for the two bodies are:

$$(m_1 + M_{11})\, \ddot{X}_1 + M_{12}\, \ddot{X}_2 = \text{applied forces on mass } m_1 + O\,(X_1^2)$$

$$M_{21}\, \ddot{X}_1 + (m_2 + M_{22})\, \ddot{X}_2 = \text{applied forces on mass } m_2 + O\,(X_2^2)$$

$\ddot{X}_1$, and $\ddot{X}_2$ denote absolute accelerations of masses m_1 and m_2, respectively, and the notation $O(X^2)$ denotes nonlinear terms.

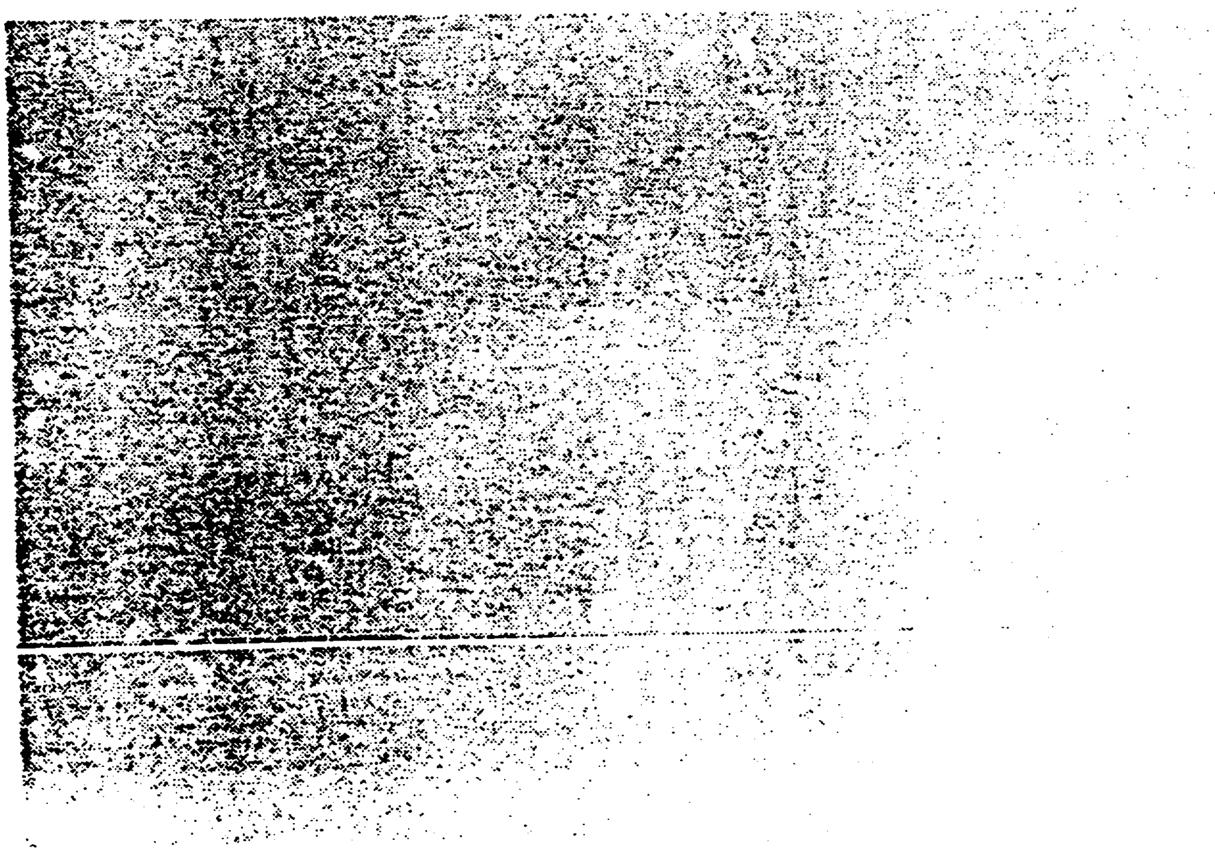
M_{11}, M_{12}, M_{21}, and M_{22} are fluid coupling coefficients which depend on body shape, relative disposition, etc. Fritz [6.5.3] gives data for M_{ij} for various body shapes and arrangements. The fluid adds mass to the body (M_{11} to mass m_1), and an inertial force proportional to acceleration of the adjacent body (mass m_2). Thus, acceleration of one body affects the force field on another. This force field is a function of inter-body gap, reaching large values for small gaps. Lateral motion of a fuel assembly inside a storage location encounters this effect.

For example, fluid coupling behavior will be experienced between nodes 2 and 2* in Figure 6.5.1. The rack analysis also contains inertial fluid coupling terms which model the effect of fluid in the gaps between adjacent racks.

Terms modeling the effects of fluid flowing between adjacent racks in a single rack analysis suffer from the inaccuracies described earlier. These terms are usually computed assuming that all racks adjacent to the rack being analyzed are vibrating in-phase or 180° out of phase. The WPMR analyses do not require any assumptions with regard to phase.

Rack-to-rack gap elements have initial gaps set to 100% of the physical gap between the racks or between outermost racks and the adjacent pool walls.

6.5.2.1 Multi-Body Fluid Coupling Phenomena



6.5.3 Stiffness Element Details

Table 6.5.2 lists all spring elements used in the 3-D, 22-DOF, rack model for Campaign I of pool D. This set of elements is chosen since it represents the smallest of the models and provides a sufficient example to describe spring element numbering of Campaign II of pool D and the larger pool C models, which are similar. Three element types are used in the rack models. Type 1 are linear elastic elements used to represent the beam-like behavior of the integrated rack cell matrix. Type 2 elements are the piece-wise linear friction springs used to develop the appropriate forces between the rack pedestals and the supporting bearing pads. Type 3 elements are non-linear gap elements which model gap closures and subsequent impact loadings (i.e., between fuel assemblies and the storage cell inner walls, and rack outer periphery spaces.

A detailed numbering scheme for the rack-to-rack and rack-to-wall gap elements for each of the pool models is provided in Figures 6.5.6 through 6.5.13.

If the simulation model is restricted to two dimensions (one horizontal motion plus one vertical motion, for example), for the purposes of model clarification only, then Figure 6.5.3 describes the configuration. This simpler model is used to elaborate on the various stiffness modeling elements.

Type 3 gap elements modeling impacts between fuel assemblies and racks have local stiffness K_I in Figure 6.5.3. In Table 6.5.2, for example, type 3 gap elements 5 through 8 act on the rattling fuel mass at the rack top. Support pedestal spring rates K_S are modeled by type 3 gap elements 1 through 4, as listed in Table 6.5.2. Local compliance of the concrete floor is included in K_s. The type 2 friction elements listed in Table 6.5.2 are shown in Figure 6.5.3 as K_f. The spring elements depicted in Figure 6.5.4 represent type 1 elements.

Friction at support/liner interface is modeled by the piecewise linear friction springs with suitably large stiffness K_f up to the limiting lateral load μN, where N is the current compression load at the interface between support and liner. At every time-step during transient analysis, the current value of N (either zero if the pedestal has lifted off the liner, or a compressive finite value) is computed.

The gap element K_s, modeling the effective compression stiffness of the structure in the vicinity of the support, includes stiffness of the pedestal. local stiffness of the underlying pool slab, and local stiffness of the rack cellular structure above the pedestal.

The previous discussion is limited to a 2-D model solely for simplicity. Actual analyses incorporate 3-D motions and include all stiffness elements listed in Table 6.5.2.

6.5.4 Coefficients of Friction

To eliminate the last significant element of uncertainty in rack dynamic analyses, multiple simulations are performed to adjust the friction coefficient ascribed to the support pedestal/pool bearing pad interface. These friction coefficients are chosen consistent with the two bounding extremes from Rabinowicz's data [6.5.1]. Simulations are also performed by imposing intermediate value friction coefficients developed by a random number generator with Gaussian normal distribution characteristics. The assigned values are then held constant during the entire simulation in order to obtain reproducible results.* Thus, in this manner, the WPMR ...sis results are brought closer to the realistic structural conditions.

6.5.5 Governing Equations of Motion

...ng the structural model discussed in the foregoing, equations of motion corresponding to each degree-of-freedom are obtained using Lagrange's Formulation [6.5.4]. The system kinetic energy includes contributions from solid structures and from trapped and surrounding fluid. The final system of equations obtained have the matrix form:

$$[M]\left[\frac{d^2q}{dt^2}\right] = [Q] + [G]$$

where:

[M] - total mass matrix (including structural and fluid mass contributions). The size of this matrix will be 22n x22n for a WPMR analysis (n = number of racks in the model)

---

* It is noted that MR216 has the capability to change the coefficient of friction at any pedestal at each instant of contact based on a random reading of the computer clock cycle. However, exercising this option would yield results that could not be reproduced. Therefore, the random choice ...coefficients is made only once per run

q - the nodal displacement vector relative to the pool slab displacement (the term with q indicates the second derivative with respect to time, i.e., acceleration)

[G] - a vector dependent on the given ground acceleration

[Q] - a vector dependent on the spring forces (linear and nonlinear) and the coupling between degrees-of-freedom

The above column vectors have length 22n. The equations can be rewritten as follows:

$$\left[\frac{d^2q}{dt^2}\right] = [M]^{-1}\ [Q] + [M]^{-1}\ [G]$$

This equation set is mass uncoupled, displacement coupled at each instant in time. The numerical solution uses a central difference scheme built into the proprietary computer program MR216 [6.2.4].

6.6 Structural Evaluation of Spent Fuel Rack Design

6.6.1 Kinematic and Stress Acceptance Criteria

There are two sets of criteria to be satisfied by the rack modules:

a. Kinematic Criteria

Per Reference [6.1.1], in order to be qualified as a physically stable structure it is necessary to demonstrate that an isolated rack in water would not overturn when an event of magnitude:

- 1.5 times the upset seismic loading condition is applied.
- 1.1 times the faulted seismic loading condition is applied.

b. Stress Limit Criteria

Stress limits must not be exceeded under the postulated load combinations provided herein.

6.6.2 Stress Limit Evaluations

The stress limits presented below apply to the rack structure and are derived from the ASME Code, Section III, Subsection NF [6.6.1]. Parameters and terminology are in accordance with the ASME Code. Material properties are obtained from the ASME Code, Section II, Part D [6.6.2], and are listed in Table 6.3.1.

(i) Normal and Upset Conditions (Level A or Level B)

a. Allowable stress in tension on a net section is:

$F_t = 0.6\ S_y$

Where, S_y = yield stress at temperature, and F_t is equivalent to primary membrane stress.

b. Allowable stress in shear on a net section is:

$F_v = .4\ S_y$

c. Allowable stress in compression on a net section

$$F_a = S_y \left(.47 - \frac{k\ \ell}{444\ r} \right)$$

kl/r for the main rack body is based on the full height and cross section of the honeycomb region and does not exceed 120 for all sections.

ℓ = unsupported length of component

k = length coefficient which gives influence of boundary conditions. The following values are appropriate for the described end conditions

= 1 (simple support both ends)
= 2 (cantilever beam)
= ½ (clamped at both ends)

r = radius of gyration of component

d. Maximum allowable bending stress at the outermost fiber of a net section, due to flexure about one plane of symmetry is:

$F_b = 0.60\ S_y$ (equivalent to primary bending)

e. Combined bending and compression on a net section satisfies:

$$\frac{f_a}{F_a} + \frac{C_{mx} f_{bx}}{D_x F_{bx}} + \frac{C_{my} f_{by}}{D_y F_{by}} < 1$$

where:

f_a = Direct compressive stress in the section
f_{bx} = Maximum bending stress along x-axis
f_{by} = Maximum bending stress along y-axis
C_{mx} = 0.85
C_{my} = 0.85
D_x = $1 - (f_a/F'_{ex})$
D_y = $1 - (f_a/F'_{ey})$
$F'_{ex,ey}$ = $(\pi^2 E)/(2.15\ (kl/r)^2_{x,y})$
E = Young's Modulus

and subscripts x,y reflect the particular bending plane.

f. Combined flexure and compression (or tension) on a net section:

$$\frac{f_a}{0.6S_y} + \frac{f_{bx}}{F_{bx}} + \frac{f_{by}}{F_{by}} < 1.0$$

The above requirements are to be met for both direct tension or compression.

g. Welds.

Allowable maximum shear stress on the net section of a weld is given by:

$$F_w = 0.3\ S_u$$

where S_u is the weld material ultimate strength at temperature. For fillet weld legs in contact with base metal, the shear stress on the gross section is limited to $0.4S_y$, where S_y is the base material yield strength at temperature.

(ii) <u>Level D Service Limits</u>

Section F-1334 (ASME Section III, Appendix F) [6.6.2], states that the limits for the Level D condition are the minimum of 1.2 (S_y/F_y) or ($0.7S_u/F_u$) times the corresponding limits for the Level A condition. S_u is ultimate tensile stress at the specified rack design temperature. Examination of material properties for 304L stainless demonstrates that 1.2 times the yield strength is less than the 0.7 times the ultimate strength.

Exceptions to the above general multiplier are the following:

a) Stresses in shear shall not exceed the lesser of $0.72S_y$ or $0.42S_u$. In the case of the Austenitic Stainless material used here, $0.72S_y$ governs.

b) Axial Compression Loads shall be limited to 2/3 of the calculated buckling load.

c) Combined Axial Compression and Bending - The equations for Level A conditions shall apply except that:

   F_a = 0.667 x Buckling Load/ Gross Section Area,

   and the terms F'_{ex} and F'_{ey} may be increased by the factor 1.65.

d) For welds, the Level D allowable maximum weld stress is not specified in Appendix F of the ASME Code. An appropriate limit for weld throat stress is conservatively set here as:

   F_w = ($0.3\ S_u$) x factor

   where:

   factor = (Level D shear stress limit)/(Level A shear stress limit)

6.6.3 <u>Dimensionless Stress Factors</u>

For convenience, the stress results are presented in dimensionless form. Dimensionless stress factors are defined as the ratio of the actual developed stress to the specified limiting value. The limiting value of each stress factor is 1.0, based on the allowable strengths for each level, for Levels A, B, and D (where $1.2S_y < .7S_u$).

The stress factors reported are:

R_1 = Ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression)

R_2 = Ratio of gross shear on a net section in the x-direction to its allowable value

R_3 = Ratio of maximum x-axis bending stress to its allowable value for the section

R_4 = Ratio of maximum y-axis bending stress to its allowable value for the section

R_5 = Combined flexure and compressive factor (as defined in the foregoing)

R_6 = Combined flexure and tension (or compression) factor (as defined in the foregoing)

R_7 = Ratio of gross shear on a net section in the y-direction to its allowable value

6.6.4 Loads and Loading Combinations for Spent Fuel Racks

The applicable loads and their combinations which must be considered in the seismic analysis of rack modules are excerpted from Refs. [6.1.2] and [6.6.3].

The load combinations considered are identified below:

Loading Combination	Service Level
$D + L$ $D + L + T_o$ $D + L + T_o + E$	Level A
$D + L + T_a + E$ $D + L + T_o + P_f$	Level B
$D + L + T_a + E'$ $D + L + T_o + F_d$	Level D The functional capability of the fuel racks must be demonstrated.

D = Dead weight-induced loads (including fuel assembly weight)

L = Live Load (not applicable for the fuel rack, since there are no moving objects in the rack load path)

P_f = Upward force on the racks caused by postulated stuck fuel assembly

F_d = Impact force from accidental drop of the heaviest load from the maximum possible height.

E = Operating Basis Earthquake (OBE)

E' = Safe Shutdown Earthquake (SSE)

T_o = Differential temperature induced loads (normal operating or shutdown condition based on the most critical transient or steady state condition)

T_a = Differential temperature induced loads (the highest temperature associated with the postulated abnormal design conditions)

T_a and T_o produce local thermal stresses. The worst thermal stress field in a fuel rack is obtained when an isolated storage location has a fuel assembly generating heat at maximum postulated rate and surrounding storage locations contain no fuel. Heated water makes unobstructed contact with the inside of the storage walls, thereby producing maximum possible temperature difference between adjacent cells. Secondary stresses produced are limited to the body of the rack; that is, support pedestals do not experience secondary (thermal) stresses.

6.7 Parametric Simulations

Whole Pool Multi-Rack (WPMR) simulations have been performed to investigate the structural integrity of each rack array. Pools C and D had separate runs performed for the SSE seismic event considering pools filled and partially filled with racks. The partially filled pools represent interim configurations subsequent to the installation campaigns identified for each pool in Figures 6.3.1 and 6.3.2. The configurations were considered with friction coefficients of 0.8, 0.2, and a guassian distribution with a mean of 0.5 (i.e., random coefficient of friction (COF) with upper and lower limits of 0.8 and 0.2). The SSE simulations were performed and conservatively compared against the allowables for OBE events. This process eliminated the need for performing OBE simulations to significantly reduce the number of runs needed. Due to the mild SSE earthquake postulated for Harris, this conservative evaluation technique yielded satisfactory design margins.

The overturning check simulations were performed to determine the behavior of the highest aspect (width/length) ratio racks under both the OBE and SSE events. The overturning check simulations considered a single rack (i.e., no dynamic fluid coupling to walls or other racks) half full with fuel all loaded along the long side of the rack.

The rack numbering schemes used to identify the racks in each simulation model are introduced in Figures 6.5.6 through 6.5.13 The circled rack numbers in the figures correspond to the rack numbers shown in the following tables

The following table presents a complete listing of the simulations discussed herein. Consideration of the parameters described above resulted in the following runs:

Run	Simulation Description	COF	Event
1	Pool C (Campaign I)	0.8	SSE
2	Pool C (Campaign I)	0.2	SSE
3	Pool C (Campaign I)	Random	SSE
4	Pool C (Campaign II)	0.8	SSE
5	Pool C (Campaign II)	0.2	SSE
6	Pool C (Campaign II)	Random	SSE
7	Pool C (Campaign III - Full)	0.8	SSE
8	Pool C (Campaign III - Full)	0.2	SSE
9	Pool C (Campaign III - Full)	Random	SSE
10	Pool D (Campaign I)	0.8	SSE
11	Pool D (Campaign I)	0.2	SSE
12	Pool D (Campaign I)	Random	SSE
13	Pool D (Campaign II - Full)	0.8	SSE
14	Pool D (Campaign II - Full)	0.2	SSE
15	Pool D (Campaign II - Full)	Random	SSE
16	Single Holtec Rack Overturning Check	0.8	OBE x 1.5
17	Single HoltecRack Overturning Check	0.8	SSE x 1.1

6.8 Time History Simulation Results

The results from the MR216 runs may be seen in the raw data output files. The MR216 output files archive all of the loads and displacements at key locations within each of the rack modules at every time step throughout the entire time history duration. However, due to the huge quantity of output data, a post-processor is used to scan for worst case conditions and develop the stress factors discussed in subsection 6.6.3.

Further reduction in this bulk of information is provided in this section by extracting the worst case values from the parameters of interest; namely displacements, support pedestal forces, impact loads, and stress factors. This section also summarizes other analyses performed to develop and evaluate structural member stresses which are not determined by the post processor.

6.8.1 Rack Displacements

A tabulated summary of the maximum displacement for each simulation is provided below with the location/direction terms defined as follows:

uxt = displacement of top corner of rack, relative to the slab, in the East-West direction for pool C racks and in the North-South direction for pool D rack modules.

uyt = displacement of top corner of rack, relative to the slab, in the North-South direction for pool C racks and in the East-West direction for pool D rack modules.

Simulations 16 and 17 were performed to evaluate the potential for overturning of a single Holtec rack isolated in the pool without any fluid coupling to adjacent racks or walls. This simulation was performed to account for the unlikely possibility of a seismic event occurring during the installation process.

The following maximum rack displacements (in inches) are obtained for each of the runs:

Pool	Event	Run	COF	Maximum Displacement (inches)	Location/ Direction	Rack
Pool C Campaign I	SSE	1	0.8	1.132	uxt	16
	SSE	2	0.2	0.631	uyt	5
	SSE	3	Random	0.878	uxt	16
Pool C Campaign II	SSE	4	0.8	1.494	uxt	28
	SSE	5	0.2	0.917	uxt	1
	SSE	6	Random	0.878	uxt	16
Pool C Campaign III	SSE	7	0.8	0.617	uyt	29
	SSE	8	0.2	0.740	uyt	1
	SSE	9	Random	0.684	uyt	3
Pool D Campaign I	SSE	10	0.8	0.520	uxt	2
	SSE	11	0.2	0.390	uyt	3
	SSE	12	Random	0.521	uxt	2
Pool D Campaign II	SSE	13	0.8	0.575	uyt	1
	SSE	14	0.2	0.595	uyt	1
	SSE	15	Random	0.576	uyt	1
Tipover: Single Holtec Rack	OBE	16	0.8	0.347	uyt	PWR
Tipover: Single Holtec Rack	SSE	17	0.8	1.054	uyt	PWR

The largest displacement of 1.494 occurs in run 4 for rack 28 in the X direction. Since this displacement maintains the centroid of the rack well within the boundaries represented by the support pedestals, there is no possibility of rack overturning (tipover).

6.8.2 Pedestal Vertical Forces

Pedestal number 1 for each rack is located in the +X, -Y corner of each rack. Numbering increases counterclockwise around the periphery of the rack. The following bounding vertical pedestal forces (in kips) are obtained for each run:

Pool	Event	Run	COF	..laximum Pedestal Load (kips)	Rack	Ped.
Pool C Campaign I	SSE	1	0.8	122	5	4
	SSE	2	0.2	115	5	4
	SSE	3	Random	123	5	1
Pool C Campaign II	SSE	4	0.8	153	5	2
	SSE	5	0.2	121	9	2
	SSE	6	Random	134	9	1
Pool C Campaign III	SSE	7	0.8	113	7	4
	SSE	8	0.2	110	9	1
	SSE	9	Random	122	26	1
Pool D Campaign I	SSE	10	0.8	118	5	4
	SSE	11	0.2	112	5	1
	SSE	12	Random	114	5	4
Pool D Campaign II	SSE	13	0.8	135	11	2
	SSE	14	0.2	116	11	4
	SSE	15	Random	130	11	4

As may be seen, the highest pedestal load is 153,000 lbs and occurs in run 4 for pedestal 2 of rack 5. Figure 6.8.1 provides a plot of the vertical force of this pedestal transmitted to the bearing pad over the entire duration of the SFP, 0.8 COF, SSE, campaign II simulation

6.8.3 Pedestal Friction Forces

The maximum (x or y direction) shear load (in kips) bounding all pedestals for each simulation are reported below and are obtained by inspection of the complete tabular data.

Pool	Event	Run	COF	Maximum Friction Load (kips)	Rack
Pool C Campaign I	SSE	1	0.8	46	11
	SSE	2	0.2	22.3	8
	SSE	3	Random	41.7	13
Pool C Campaign II	SSE	4	0.8	44.2	13
	SSE	5	0.2	22.2	9
	SSE	6	Random	40.9	11
Pool C Campaign III	SSE	7	0.8	43.4	3
	SSE	8	0.2	19.7	7
	SSE	9	Random	45.8	26
Pool D Campaign I	SSE	10	0.8	45 6	1
	SSE	11	0.2	19 7	2
	SSE	12	Random	34.4	1
Pool D Campaign II	SSE	13	0.8	42.3	11
	SSE	14	0.2	22.3	1
	SSE	15	Random	42.4	11

6.8.4 Rack Impact Loads

A freestanding rack, by definition, is a structure subject to potential impacts during a seismic event. Impacts arise from rattling of the fuel assemblies in the storage rack locations and, in some instances, from localized impacts between the racks, or between a peripheral rack and the pool wall. The following sections discuss the bounding values of these impact loads.

6.8.4.1 Rack to Rack Impacts

As is often the case with close rack spacing, some rack to rack impacts occur. The following instantaneous maximum impact forces and locations are identified for each of the simulations performed. Listings are only given for those simulations within which an impact occurred. The element numbering is identified in Figures 6.5.6 through 6.5.13.

Run	Impact Load (kips)	Element	Location	Run	Impact Load (kips)	Element	Location
1	3.0	494	Top	5	11.3	814	Bottom
1	8.1	503	Top	5	8.1	817	Top
1	8.1	504	Top	5	8.1	818	Top
1	8.1	583	Top	5	4.9	831	Bottom
1	8.1	584	Top	5	8.4	937	Bottom
2	8.1	493	Top	5	5.6	945	Bottom
2	8.1	494	Top	5	6.4	991	Bottom
3	6.7	493	Top	5	6.5	992	Bottom
3	8.1	494	Top	6	8.1	736	Top
3	8.1	503	Top	6	1.9	746	Top
3	8.1	504	Top	6	8.1	759	Top
3	3.0	539	Top	6	7.9	760	Top
3	2.1	540	Top	6	8.1	781	Top
3	8.1	583	Top	6	8.1	782	Top
3	8.1	584	Top	6	8.1	789	Top
3	8.1	599	Top	6	8.1	790	Top

Run	Impact Load (kips)	Element	Location	Run	Impact Load (kips)	Element	Location
3	8.1	600	Top	6	4.9	799	Top
4	5.3	736	Top	6	8.1	817	Top
4	8.1	759	Top	6	8.1	818	Top
4	8.1	760	Top	6	1.2	827	Top
4	8.1	781	Top	6	8.1	828	Top
4	8.1	782	Top	6	8.1	835	Top
4	8.1	799	Top	6	8.1	836	Top
4	8.1	800	Top	6	1.9	914	Top
4	8.1	817	Top	6	1.8	946	Bottom
4	8.1	818	Top	6	8.1	949	Top
4	8.1	827	Top	6	8.1	950	Top
4	8.1	828	Top	6	8.1	979	Top
4	8.1	835	Top	6	8.1	980	Top
4	8.1	836	Top	6	2.6	982	Bottom
4	8.1	907	Top	6	8.1	986	Top
4	8.1	908	Top	6	12.9	992	Bottom
4	8.1	913	Top	7	8.1	913	Top
4	8.1	914	Top	7	8.1	914	Top
4	8.1	979	Top	7	5.3	949	Top
4	8.1	980	Top	7	0.8	950	Top
5	6.7	736	Top	8	8.0	991	Bottom
5	16.7	743	Bottom	8	11.3	992	Bottom
5	7.7	744	Bottom	9	8.1	913	Top
5	10.7	756	Bottom	9	8.1	914	Top
5	4.5	777	Bottom	9	8.1	949	Top
5	22.1	778	Bottom	9	7.8	950	Top
5	16.1	813	Bottom				

6.8.4.2 Rack to Wall Impacts

Storage racks do not impact the pool walls under any simulation.

6.8.4.3 Fuel to Cell Wall Impact Loads

A review of the results from each simulation allows determination of the maximum instantaneous impact load between fuel assembly and fuel cell wall at any modeled impact site. The maximum values obtained are reported in the following table

Pool	Event	Run	COF	Maximum Fuel Impact Load (lbs)	Rack
Pool C Campaign I	SSE	1	0.8	532	2
	SSE	2	0.2	562	5
	SSE	3	Random	605	2
Pool C Campaign II	SSE	4	0.8	531	25
	SSE	5	0.2	548	9
	SSE	6	Random	535	22
Pool C Campaign III	SSE	7	0.8	525	17
	SSE	8	0.2	527	17
	SSE	9	Random	515	17
Pool D Campaign I	SSE	10	0.8	473	1
	SSE	11	0.2	591	4
	SSE	12	Random	473	1
Pool D Campaign II	SSE	13	0.8	472	12
	SSE	14	0.2	462	12
	SSE	15	Random	472	12

The maximum fuel to cell wall impact load is 605 pounds. Based on fuel manufacturer's data, loads of this magnitude will not damage the fuel assembly.

6.9 Rack Structural Evaluation

6.9.1 Rack Dimensionless Stress Factors for Level B and D Loadings

The vertical and shear forces at the bottom casting-pedestal interface are available as a function of time. The maximum values for the stress factors defined in Section 6.6.3 can be determined for every pedestal in the array of racks by scanning this data to select the limiting loads and performing calculations to determine member stresses. These two tasks are performed by a post-processor. With this information available, the structural integrity of the pedestal can be assessed and reported. The net section maximum (in time) bending moments and shear forces can also be determined at the bottom casting-rack cellular structure interface for each spent fuel rack in the pool. This allows the evaluation of the maximum stress in the limiting rack cell (box).

The tables presented in this section provide limiting stress factor results for male and female pedestals, and for the entire spent fuel rack cellular cross section just above the bottom casting. These locations are the most heavily loaded net sections in the structure so that satisfaction of the stress factor criteria at these locations ensures that the overall structural criteria set forth in Section 6.6.1 are met.

The tables below develop stress factors for all of the SSE (Level D) simulations based on the associated SSE allowables. However, as stated above the intent is to evaluate the stresses developed from the SSE loadings with the allowables associated with OBE (Level B). Since the OBE allowables are ½ of the SSE allowables, this comparison may be conservatively performed by reducing the acceptable stress ratio to 0.5. This is very conservative, since the actual OBE loads which should be compared against the OBE allowable would be much lower than the SSE loads herein.

6.9.1.1 Rack Cell Stress Factors

The rack cell dimensionless stress factors for each of the simulations are as follows:

Pool	Event	Run	COF	Maximum R6 Stress Factor	Rack
Pool C Campaign I	SSE	1	0.8	0.494	11
	SSE	2	0.2	0.289	9
	SSE	3	Random	0.384	9
Pool C Campaign II	SSE	4	0.8	0.454	6
	SSE	5	0.2	0.221	13
	SSE	6	Random	0.452	6
Pool C Campaign III	SSE	7	0.8	0.409	3
	SSE	8	0.2	0.266	3
	SSE	9	Random	0.432	24
Pool D Campaign I	SSE	10	0.8	0.230	1
	SSE	11	0.2	0.224	3
	SSE	12	Random	0.230	1
Pool D Campaign II	SSE	13	0.8	0.224	11
	SSE	14	0.2	0.227	3
	SSE	15	Random	0.232	2

The values for all other defined stress factors are also archived. As may be seen, all of the stress factors are well below 1.0. Therefore, the stresses developed during SSE conditions remain below the allowable SSE range and the rack modules are satisfactory to withstand the loadings. Note that stress factors for these SSE simulations are calculated based on SSE allowable strengths. However, since none of the stress factors exceed 0.5, the rack structures also adequately withstand the OBE conditions.

6.9.2 Pedestal Thread Shear Stress

The average shear stress in the thread engagement region is given below for the limiting pedestal in each simulation.

Pool	Event	Run	COF	Maximum Thread Shear Stress (psi)	Rack
Pool C Campaign I	SSE	1	0.8	4,682	11
	SSE	2	0.2	4,382	9
	SSE	3	Random	4,607	5
Pool C Campaign II	SSE	4	0.8	5,731	5
	SSE	5	0.2	4,532	9
	SSE	6	Random	5,019	9
Pool C Campaign III	SSE	7	0.8	4,232	7
	SSE	8	0.2	4,120	9
	SSE	9	Random	4,570	26
Pool D Campaign I	SSE	10	0.8	3,003	5
	SSE	11	0.2	2,850	5
	SSE	12	Random	2,901	5
Pool D Campaign II	SSE	13	0.8	3,435	11
	SSE	14	0.2	2,952	11
	SSE	15	Random	3,307	11

The ultimate strength of the female part of the pedestal is 66,200 psi. The yield stress for the female pedestal material is 21,300 psi, as shown in Table 6.3.1. The male pedestal material has much greater strength and is therefore not a controlling factor in the design. The allowable shear stress for Level B conditions is 0.4 times the yield stress which gives 8,520 psi. The allowable shear stress for Level D conditions is the lesser of: $0.72\ S_y$ = 15,336 psi or $0.42\ S_u$ = 27,804 psi. Therefore, the former criteria controls.

The largest thread shear stress computed by the post-processor is 5,731 psi. Since this value is below the allowable stresses for both OBE and DBE conditions, the thread shear stresses are within the acceptable range.

6.9.3 Local Stresses Due to Impacts

Impact loads at the pedestal base (discussed in subsection 6.8.2) produce stresses in the pedestal for which explicit stress limits are prescribed in the Code. The post-processor reports the stress factors in the pedestals which are developed, in part, from these impact stresses. The reported pedestal stress factors are included in the discussion above in Section 6.9.1.1 along with the rack cell stress factors. However, the post-processor does not develop stress factors for the localized areas of the cellular and baseplate regions of the racks which experience fuel to cell wall, rack to rack, and rack to wall impact loads. These impact loads produce stresses which attenuate rapidly away from the loaded region. This behavior is characteristic of secondary stresses.

Even though limits on secondary stresses are not prescribed in the Code for Class 3 NF structures, evaluations were made to ensure that the localized impacts do not lead to plastic deformations in the storage cells which affect the subcriticality of the stored fuel array.

a. Impact Loading Between Fuel Assembly and Cell Wall

Local cell wall integrity is conservatively estimated from peak impact loads. Plastic analysis is used to obtain the limiting impact load which would lead to gross permanent deformation. Table 6.9.1 indicates that the limiting impact load (of 3,238 lbf, including a safety factor of 2.0) is much greater than the highest calculated impact load value (of 605 lbf, see subsection 6.8.4.3) obtained from any of the rack analyses. Therefore, fuel impacts do not represent a significant concern with respect to fuel rack cell deformation.

b. Impacts Between Adjacent Racks

As may be seen from subsection 6.8.4.1, the bottom (baseplate) of the storage racks will impact each other at a few locations during seismic events. Since the loading is presented edge-on to the 3/4" baseplate membrane, the distributed stresses after local deformation will be negligible. The impact loading will be distributed over a large ar: a (a significant portion of the entire baseplate length of about 50.4 (minimum) inches by its 3/4 inch thickness). The resulting compressive stress from the highest impact load of 26,200 lbs distributed over 37 sq. inches is only 708 psi, which is negligible. Therefore, any deformation will not effect the configuration of the stored fuel.

Additional impacts will be experienced at the tops of some storage racks. These impacts will result in local yielding of the rack cell walls whenever the load exceeds 8,100 lbs. However, localized damage from all of these impacts occurs above the fuel active region. The fuel configuration and poison areas remain unaffected. Therefore, these impacts are acceptable.

6.9.4 Assessment of Rack Fatigue Margin

Deeply submerged high density spent fuel storage racks arrayed in close proximity to each other in a free-standing configuration behave primarily as a nonlinear cantilevered structure when subjected to 3-D seismic excitations. In addition to the pulsations in the vertical load at each pedestal, lateral friction forces at the pedestal/bearing pad-liner interface, which help prevent or mitigate lateral sliding of the rack, also exert a time-varying moment in the baseplate region of the rack. The friction-induced lateral forces act simultaneously in x and y directions with the requirement that their vectorial sum does not exceed μV, where μ is the limiting interface coefficient of friction and V is the concomitant vertical thrust on the liner (at the *given* time instant). As the vertical thrust at a pedestal location changes, so does the maximum friction force, F, that the interface can exert. In other words, the lateral force at the pedestal/liner interface, F, is given by

$$F \leq \mu N(\tau)$$

where N (vertical thrust) is the time-varying function of τ. F does not always equal μN; rather, μN is the maximum value it can attain at any time; the actual value, of course, is determined by the dynamic equilibrium of the rack structure.

In summary, the horizontal friction force at the pedestal/liner interface is a function of time; its magnitude and direction of action varies during the earthquake event.

The time-varying lateral (horizontal) and vertical forces on the extremities of the support pedestals produce stresses at the root of the pedestals in the manner of an end-loaded cantilever. The stress field in the cellular region of the rack is quite complex, with its maximum values located in the region closest to the pedestal. The maximum magnitude of the stresses depends on the severity of the pedestal end loads and on the geometry of the pedestal/rack baseplate region.

Alternating stresses in metals produce metal fatigue if the amplitude of the stress cycles is sufficiently large. In high density racks designed for sites with moderate to high postulated seismic action, the stress intensity amplitudes frequently reach values above the material endurance limit, leading to expenditure of the fatigue "usage" reserve in the material.

Because the locations of maximum stress (viz., the pedestal/rack baseplate junction) and the close placement of racks, a post-earthquake inspection of the high stressed regions in the racks is not feasible. Therefore, the racks must be engineered to withstand multiple earthquakes without reliance of nondestructive inspections for post-earthquake integrity assessment. The fatigue life evaluation of racks is an integral aspect of a sound design.

The time-history method of analysis, deployed in this report, provides the means to obtain a complete cycle history of the stress intensities in the highly stressed regions of the rack. Having determined the amplitude of the stress intensity cycles and their number, the cumulative damage factor, U, can be determined using the classical Miner's rule

$$U = \Sigma \frac{n_i}{N_i}$$

where n_i is the number of stress intensity cycles of amplitude σ_i, and N_i is the permissible number of cycles corresponding to σ_i from the ASME fatigue curve for the material of construction. U must be less than or equal to 1.0.

To evaluate the cumulative damage factor, a finite element model of a portion of the spent fuel rack in the vicinity of a support pedestal is constructed in sufficient detail to provide an accurate assessment of stress intensities. Figure 6.9.1 shows the essentials of the finite element model. The finite element solutions for unit pedestal loads in three orthogonal directions are combined to establish the maximum value of stress intensity as a function of the three unit pedestal loads. Using the archived results of the spent fuel rack dynamic analyses (pedestal load histories versus time), enables a time-history of stress intensity to be established at the most limiting location. This permits establishing a set of alternating stress intensity ranges versus cycles for several seismic events. Following ASME Code guidelines for computing U, it is found that U =0.464 due to the combined effect of 21 SSE events. This cumulative damage factor is below the ASME Code limit of 1.0 and therefore, fatigue failure is not expected. Selection of 21 SSE events represents a conservative evaluation compared to other previous fatigue assessments which were based on the damage resulting from 10 SSE events, as discussed in the Harris FSAR.

6.9.5 Weld Stresses

Weld locations subjected to significant seismic loading are at the bottom of the rack at the baseplate-to-cell connection, at the top of the pedestal support at the baseplate connection, and at cell-to-cell connections. Bounding values of resultant loads are used to qualify the connections.

a. Baseplate-to-Rack Cell Welds

Reference [6.6.1] (ASME Code Section III, Subsection NF) permits, for Level A or B conditions, an allowable weld stress $\tau = .3\ S_u = 19860$ psi. As stated in subsection 3.4.2 the allowable may be increased for Level D by the ratio (15336/8520) = 1.8, giving an allowable of 35,748 psi.

Weld dimensionless stress factors are produced through the use of a simple conversion (ratio) factor applied to the corresponding stress factor in the adjacent rack material. A 2.15 factor for PWR racks is based on the differences in material thickness and length versus weld throat dimension and length:

$$Ratio = \frac{0.075 * 8.875}{0.0625 * 0.7071 * 7} = 2.15165$$

Similarly, a 1.49 factor for BWR racks is developed as follows:

$$Ratio = \frac{0.075 * 6.135}{0.0625 * 0.7071 * 7} = 1.48736$$

The highest predicted weld stress for DBE is calculated from the highest R6 value (see subsection 6.9.1.1) as follows:

$$R6 * [(0.6)\ F_y] * Ratio =$$
$$0.494\ [(0.6)\ 21{,}300]*2.144 = 13{,}574\ psi$$

this value is less than the OBE allowable weld stress value, which is 19,860. Therefore, all weld stresses between the baseplate and cell wall base are acceptable.

b. Baseplate-to-Pedestal Welds

The weld between the baseplate and support pedestal are evaluated by development of a finite element model of the bearing pad/base plate interface and appropriate application of the maximum pedestal loads. The maximum weld stress was determined to be 10,194 psi, which is much less than the OBE allowable weld stress value of 19,860 psi. The results are also shown in Table 6.9.1.

c. Cell-to-Cell Welds

Cell-to-cell connections are made using a series of connecting welds along the cell height. Stresses in storage cell to cell welds develop due to fuel assembly impacts with the cell wall. These weld stresses are conservatively calculated by assuming that fuel assemblies in adjacent cells are moving out of phase with one another so that impact loads in two adjacent cells are in opposite directions; this tends to separate the two cells from each other at the weld.

Table 6.9.1 gives results for the maximum allowable load that can be transferred by these welds based on the available weld area. An upper bound on the load required to be transferred is also given in Table 6.9.1 and is much lower than the allowable load. This upper bound value is very conservatively obtained by applying the bounding rack to-fuel impact load from any simulation in two orthogonal directions simultaneously.

and multiplying the result by 2 to account for the simultaneous impact of two assemblies. An equilibrium analysis at the connection then yields the upper bound load to be transferred. It is seen from the results in Table 6.9.1 that the calculated load is well below the allowable.

6.9.6 Bearing Pad Analysis

To protect the pool slab from high localized dynamic loadings, bearing pads are placed between the pedestal base and the slab. Fuel rack pedestals impact on these bearing pads during a seismic event and pedestal loading is transferred to the liner. Bearing pad dimensions are set to ensure that the average pressure on the slab surface due to a static load plus a dynamic impact load does not exceed the American Concrete Institute, ACI-349 [6.9.1] limit on bearing pressures. Section 10.17 of [6.9.2] gives the design bearing strength as

$$f_b = \phi\ (.85\ f_c')\ \epsilon$$

where ϕ = .7 and f_c' is the specified concrete strength for the spent fuel pool. $\epsilon = 1$, except when the supporting surface is wider on all sides than the loaded area. In that case, $\epsilon = (A_2/A_1)^{.5}$, but not more than 2. A_1 is the actual loaded area, and A_2 is an area greater than A_1 and is defined in [6.9.2]. Using a value of $\epsilon > 1$ includes credit for the confining effect of the surrounding concrete. It is noted that this criteria is in conformance with the ultimate strength primary design methodology of the American Concrete Institute in use since 1971. For Harris, the concrete compressive strength is f_c' = 4,000 psi. The allowable bearing pressure is conservatively computed by taking $\epsilon = 1$ to account for lack of total concrete confinement in the leak chase region and a stress reduction factor of $\phi = 0.7$. Thus, the maximum allowable concrete bearing pressure is 2,380 psi.

The maximum vertical pedestal load is 153,000 lbs (SSE event). The bearing pad selected is 1.5" thick, austenitic stainless steel plate stock. The average pressure at the pad to liner interface is computed and compared against the above-mentioned limit. Calculations show that the average pressure at the slab/liner interface is 2,168 psi which is below the allowable value of 2,380 psi, providing a factor of safety of 1.1.

Therefore, the bearing pad design devised for the Harris pools C and D is deemed appropriate for the prescribed loadings.

6.9.7 Level A Evaluation

The Level A condition is not a governing condition for spent fuel racks since the general level of loading is far less than Level B loading. To illustrate this, the heaviest (fully loaded) spent fuel rack (which is an 11X9 PWR rack) is considered under the dead weight load. It is shown below that the maximum pedestal load is low and that further stress evaluations are unnecessary.

LEVEL A MAXIMUM PEDESTAL LOAD

Dry Weight of Largest PWR Holtec Rack	=	15,700 lbf †
Dry Weight of 99 PWR Fuel Assemblies	=	158,400 lbf
Total Dry Weight	=	174,100 lbf ††
Total Buoyant Weight (0.87 × Total Dry Weight)	=	151,467 lbf
Load per Pedestal	=	37,867 lbf

The stress allowables for the normal condition is the same as for the upset condition. An upset condition pedestal load may be conservatively (bounded on the low side) determined for the

† Conservative weight corresponding to the heaviest rack, which is a BWR storage rack. The heaviest PWR rack nominal weight is 15,620 lb.

†† This weight exceeds the weight of the heaviest fully loaded BWR rack, which is [15,700 lb + (13x13) x 680 lb] = 130,620 lb.

purpose of comparing with the load above by dividing the DBE pedestal load by a factor of 2.0. This would result in an OBE pedestal load of 153,000÷2=76,500, which is still much greater than the calculated Level A load. Since this load (and the corresponding stress throughout the rack members) is much greater than the 37,867 lb load calculated above, the Upset (OBE) condition controls over normal (Gravity) condition. Therefore, no further evaluation is necessary for Level A.

6.10 Hydrodynamic Loads on Pool Walls

The maximum hydrodynamic pressures (in psi) that develop between the fuel racks and the spent fuel pool walls will occur at those conditions and locations of greatest relative displacements. The greatest displacement was shown in Section 6.8.1 to be 1.494 inches, which occurs in rack 28 under simulation number 4. The maximum hydrodynamic pressure during this simulation was determined to be 19 psi. This hydrodynamic pressure was considered in the evaluation of the Fuel Handling Building and Pool structure.

6.11 Conclusions

- Time history simulations, including all non-linear impact and interface friction effects, have been applied to evaluate the structural margins in the Holtec spent fuel racks.

- The totality of simulations provide an extensive set of results for loads, stresses, and displacements, which taken together, demonstrate that the spent fuel racks meet the input specification and the governing Code requirements.

- Evaluation of structural margins have been performed for the array of racks in each pool with all racks loaded with fuel. The requirements of the specification and the governing Code documents are met for Level A, Level B, and Level D conditions.

- Based on all results presented in tabular form above the spent fuel racks are demonstrated to be acceptable for the service intended.

6.12 References

[6.1.1] USNRC NUREG-0800, Standard Review Plan, June 1987.

[6.1.2] (USNRC Office of Technology) "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978, and January 18, 1979 amendment thereto.

[6.2.1] Soler, A.I. and Singh, K.P., "Seismic Responses of Free Standing Fuel Rack Constructions to 3-D Motions", Nuclear Engineering and Design, Vol. 80, pp. 315-329 (1984).

[6.2.2] Soler, A.I. and Singh, K.P., "Some Results from Simultaneous Seismic Simulations of All Racks in a Fuel Pool", INNM Spent Fuel Management Seminar X, January, 1993.

[6.2.3] Singh, K.P. and Soler, A.I., "Seismic Qualification of Free Standing Nuclear Fuel Storage Racks - the Chin Shan Experience, Nuclear Engineering International, UK (March 1991).

[6.2.4] Holtec Proprietary Report HI-961465 - WPMR Analysis User Manual for Pre&Post Processors & Solver, August, 1997.

[6.4.1] USNRC Standard Review Plan, NUREG-0800 (Section 3.7.1, Rev. 2, 1989).

[6.4.2] Holtec Proprietary Report HI-89364 - Verification and User's Manual for Computer Code GENEQ, January, 1990.

[6.5.1] Rabinowicz, E., "Friction Coefficients of Water Lubricated Stainless Steels for a Spent Fuel Rack Facility," MIT, a report for Boston Edison Company, 1976.

[6.5.2] Singh, K.P. and Soler, A.I., "Dynamic Coupling in a Closely Spaced Two-Body System Vibrating in Liquid Medium: The Case of Fuel Racks," 3rd International Conference on Nuclear Power Safety, Keswick, England, May 1982.

[6.5.3] Fritz, R.J., "The Effects of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, Trans. of the ASME, February 1972, pp 167-172.

[6.5.4] Levy, S. and Wilkinson, J.P.D., "The Component Element Method in Dynamics with Application to Earthquake and Vehicle Engineering," McGraw Hill, 1976.

[6.5.5] Paul, B., "Fluid Coupling in Fuel Racks: Correlation of Theory and Experiment", (Proprietary), NUSCO/Holtec Report HI-88243.

[6.6.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NF, 1995 Edition.

[6.6.2] ASME Boiler & Pressure Vessel Code, Section II, Part D, 1995 Edition.

[6.6.3] USNRC Standard Review Plan, NUREG-0800 (Section 3.8.4, Rev. 2, 1989).

[6.9.1] ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Detroit, Michigan, 1985

[6.9.2] ACI 318-95, Building Code requirements for Structural Concrete," American Concrete Institute, Detroit, Michigan, 1995.

Table 6.2.1

PARTIAL LISTING OF FUEL RACK APPLICATIONS USING DYNARACK

PLANT	DOCKET NUMBER(s)	YEAR
Enrico Fermi Unit 2	USNRC 50-341	1980
Quad Cities 1 & 2	USNRC 50-254, 50-265	1981
Rancho Seco	USNRC 50-312	1982
Grand Gulf Unit 1	USNRC 50-416	1984
Oyster Creek	USNRC 50-219	1984
Pilgrim	USNRC 50-293	1985
V.C. Summer	USNRC 50-395	1984
Diablo Canyon Units 1 & 2	USNRC 50-275, 50-323	1986
Byron Units 1 & 2	USNRC 50-454, 50-455	1987
Braidwood Units 1 & 2	USNRC 50-456, 50-457	1987
Vogtle Unit 2	USNRC 50-425	1988
St. Lucie Unit 1	USNRC 50-335	1987
Millstone Point Unit 1	USNRC 50-245	1989
Chinshan	Taiwan Power	1988
D.C. Cook Units 1 & 2	USNRC 50-315, 50-316	1992
Indian Point Unit 2	USNRC 50-247	1990
Three Mile Island Unit 1	USNRC 50-289	1991
James A. FitzPatrick	USNRC 50-333	1990
Shearon Harris Unit 2	USNRC 50-401	1991
Hope Creek	USNRC 50-354	1990

Table 6.2.1

PARTIAL LISTING OF FUEL RACK APPLICATIONS USING DYNARACK

PLANT	DOCKET NUMBER(s)	YEAR
Kuosheng Units 1 & 2	Taiwan Power Company	1990
Ulchin Unit 2	Korea Electric Power Co.	1990
Laguna Verde Units 1 & 2	Comision Federal de Electricidad	1991
Zion Station Units 1 & 2	USNRC 50-295, 50-304	1992
Sequoyah	USNRC 50-327, 50-328	1992
LaSalle Unit 1	USNRC 50-373	1992
Duane Arnold Energy Center	USNRC 50-331	1992
Fort Calhoun	USNRC 50-285	1992
Nine Mile Point Unit 1	USNRC 50-220	1993
Beaver Valley Unit 1	USNRC 50-334	1992
Salem Units 1 & 2	USNRC 50-272, 50-311	1993
Limerick	USNRC 50-352, 50-353	1994
Ulchin Unit 1	KINS	1995
Yonggwang Units 1 & 2	KINS	1996
Kori-4	KINS	1996
Connecticut Yankee	USNRC 50-213	1996
Angra Unit 1	Brazil	1996
Sizewell B	United Kingdom	1996

Table 6.3.1 RACK MATERIAL DATA (200°F) (ASME - Section II, Part D)			
Material	**Young's Modulus** **E** **(psi)**	**Yield Strength** **S_y** **(psi)**	**Ultimate Strength** **Su** **(psi)**
SA240; 304L S.S.	27.6 x 10^6	21,300	66,200
SUPPORT MATERIAL DATA (200°F)			
SA240, Type 304L (upper part of support feet)	27.6 x 10^6	21,300	66,200
SA-564-630 (lower part of support feet; age hardened at 1100°F)	28.5 x 10^6	106,300	140,000

Table 6.4.1
TIME-HISTORY STATISTICAL CORRELATION RESULTS

OBE	
Data1 to Data2	0.0295
Data1 to Data3	0.0392
Data2 to Data3	0.0169
DBE	
Data1 to Data2	0.0183
Data1 to Data3	0.0588
Data2 to Data3	0.0299

Table 6.5.1
Degrees-of-freedom

LOCATION (Node)	DISPLACEMENT			ROTATION		

Table 6.5.2
(MR216) NUMBERING SYSTEM
FOR GAP ELEMENTS AND FRICTION ELEMENTS
IN THE POOL D, CAMPAIGN I MODEL

Table 6.5.2
(MR216) NUMBERING SYSTEM
FOR GAP ELEMENTS AND FRICTION ELEMENTS
IN THE POOL D, CAMPAIGN I MODEL

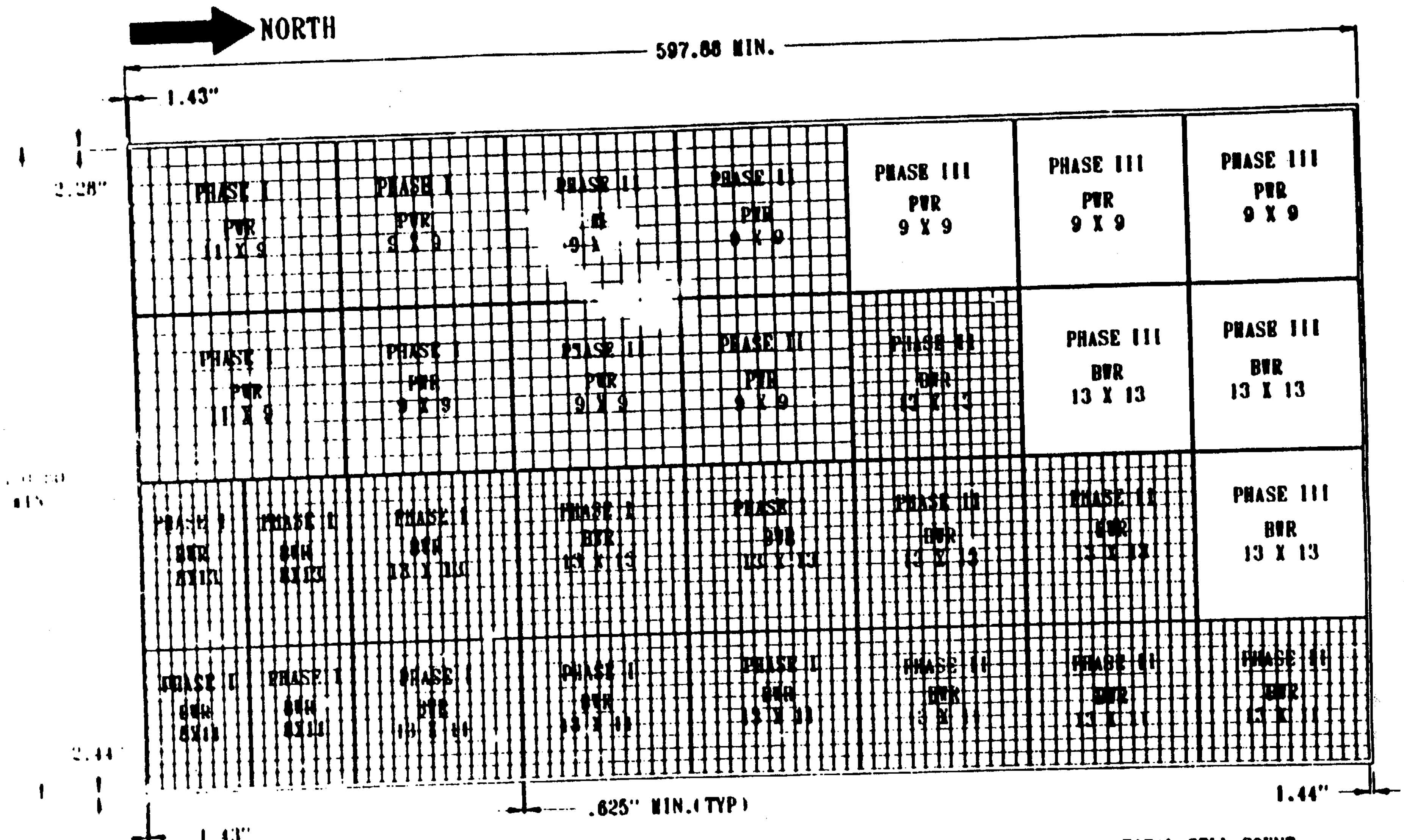
Table 6.9.1
COMPARISON OF BOUNDING CALCULATED LOADS/STRESSES
VS.
CODE ALLOWABLES
AT IMPACT AND WELD LOCATIONS

Item/Location	DBE Calculated	OBE Allowable
Fuel assembly/cell wall impact, lbf.	605 *	3,238 **
Rack/baseplate weld, psi	13,574	19,860
Female pedestal/baseplate weld, psi	10,194	19,860
Cell/cell welds, lbf.	1,711 ***	3,195

* See Section 6.8.4.3.

** Based on the limit load for a cell wall. The allowable load on the fuel assembly itself may be less than this value but is greater than 605 lbs.

*** Based on the fuel assembly to cell wall impact load simultaneously applied in two orthogonal directions.



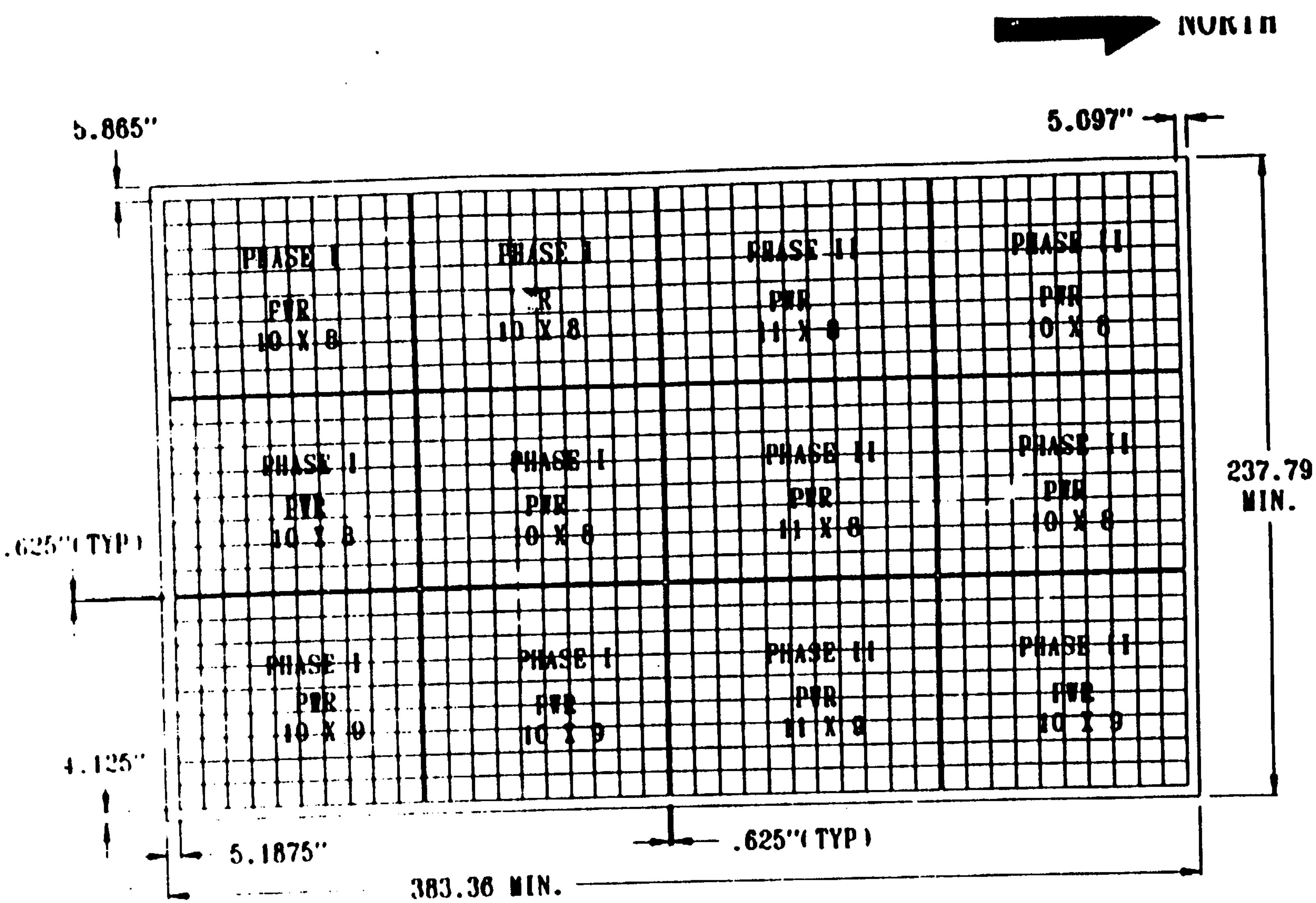
PHASE I CELL COUNT :
360 CELLS - PWR
1320 CELLS - BWR

PHASE II CELL COUNT :
324 CELLS - PWR
936 CELLS - BWR

PHASE III CELL COUNT :
243 CELLS - PWR
507 CELLS - BWR

TOTAL CELL COUNT :
927 CELLS - PWR
2763 CELLS - BWR

FIGURE 6.3.1; PHASED STORAGE CONFIGURATION FOR POOL C

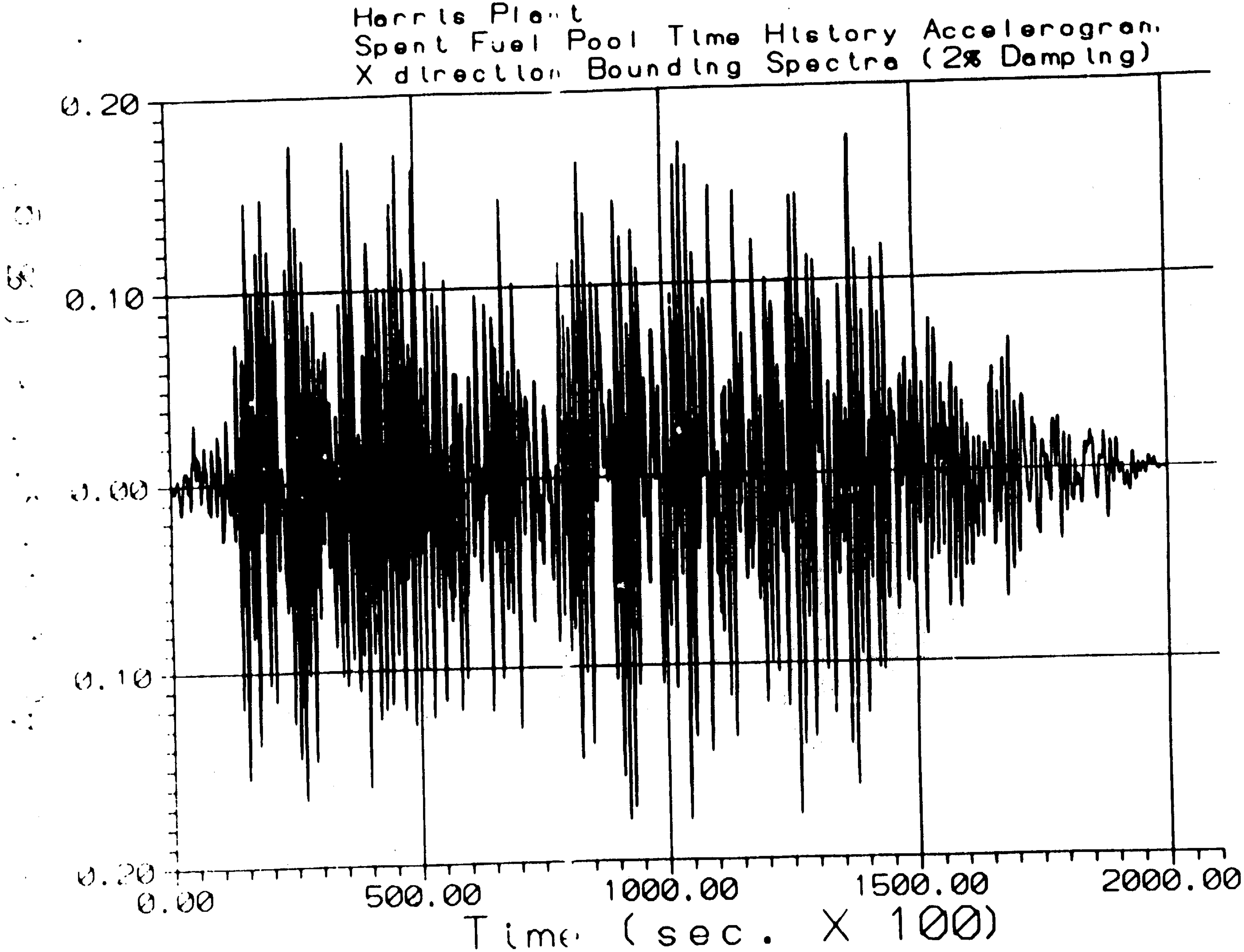


PHASE I CELL COUNT:
500 CELLS - PWR

PHASE II CELL COUNT:
525 CELLS - PWR

TOTAL CELL COUNT:
1025 CELLS - PWR

FIGURE 6.3.2; PHASED STORAGE CONFIGURATION FOR POOL D



HI-97176

Figure 6.4.1

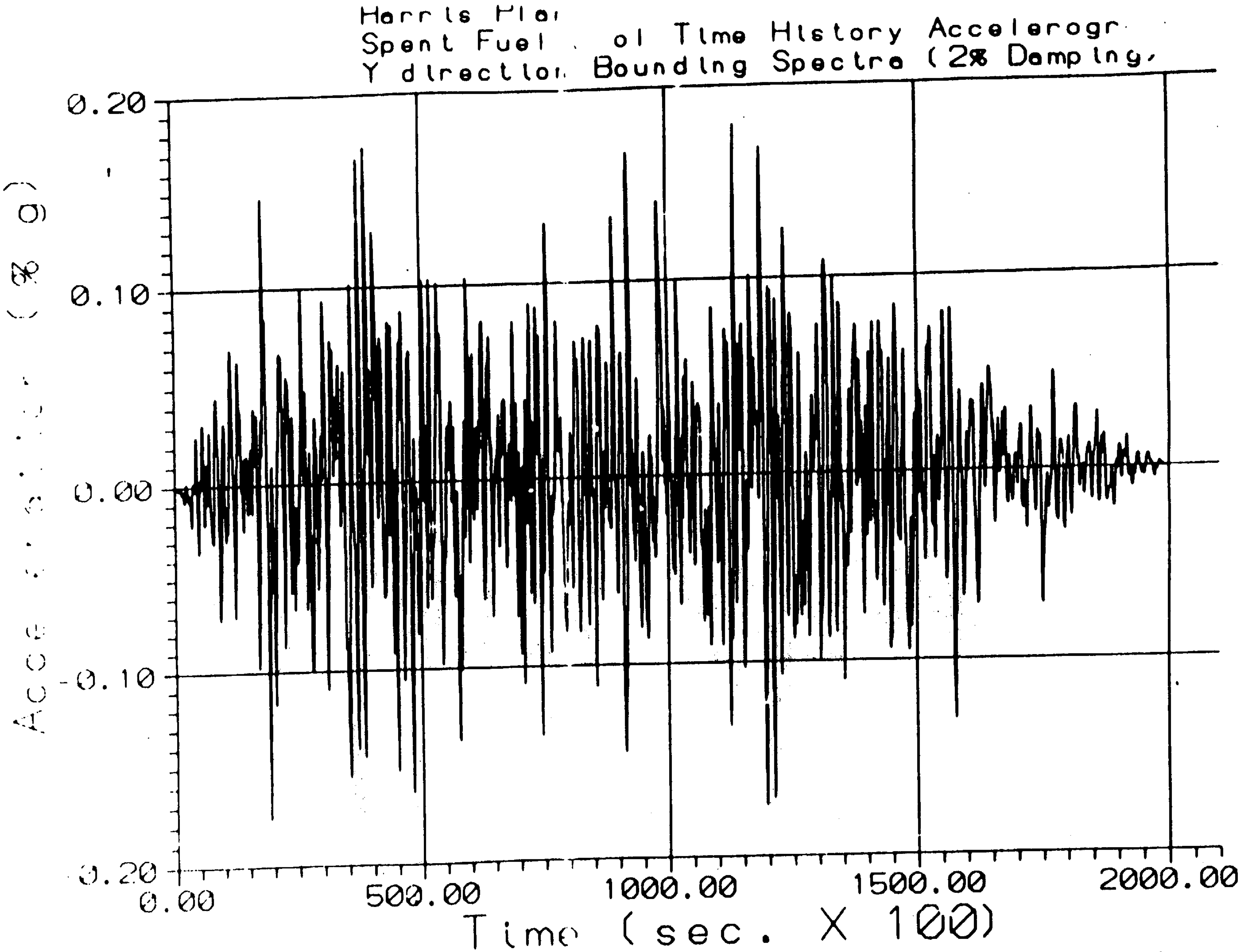

Harris Pla
Spent Fuel ol Time History Accelerogr
Y direction Bounding Spectra (2% Damping,
0.20
0.10
0.00
-0.10
-0.20
Acce (% g)
0.00
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)

Figure 6.4.2

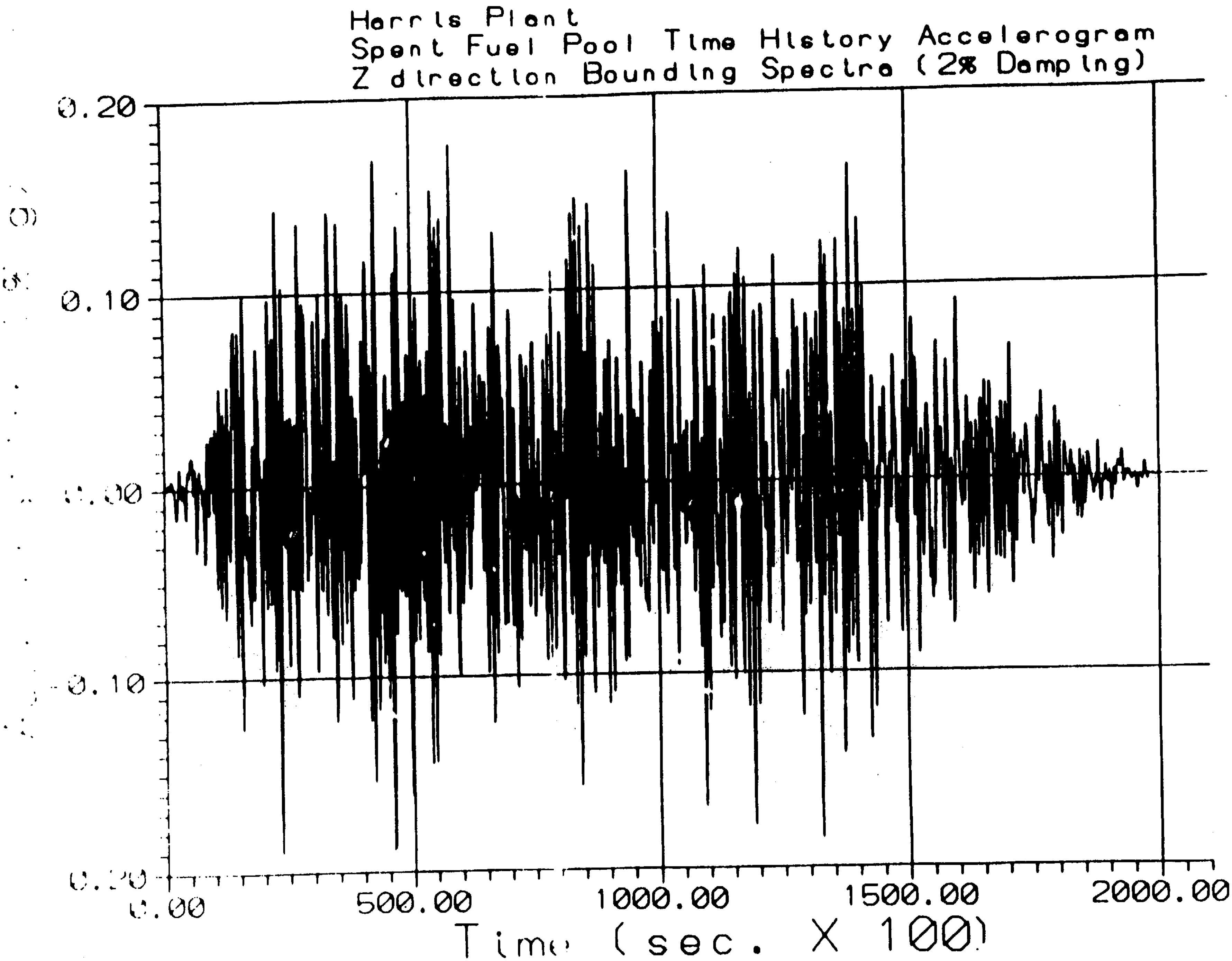

Harris Plant
Spent Fuel Pool Time History Accelerogram
Z direction Bounding Spectra (2% Damping)
0.20
0.10
-0.10
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)

Figure 6.4.3

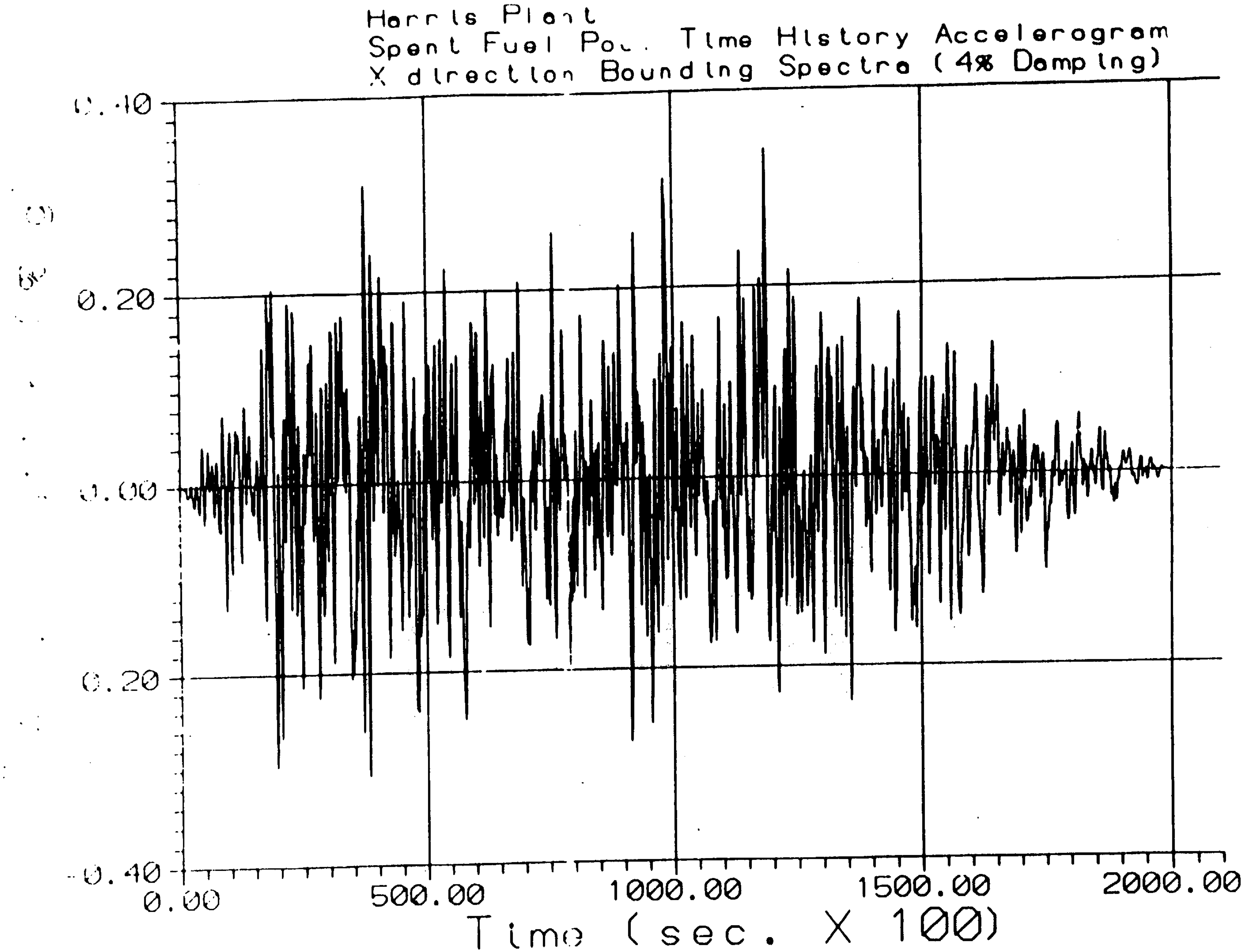
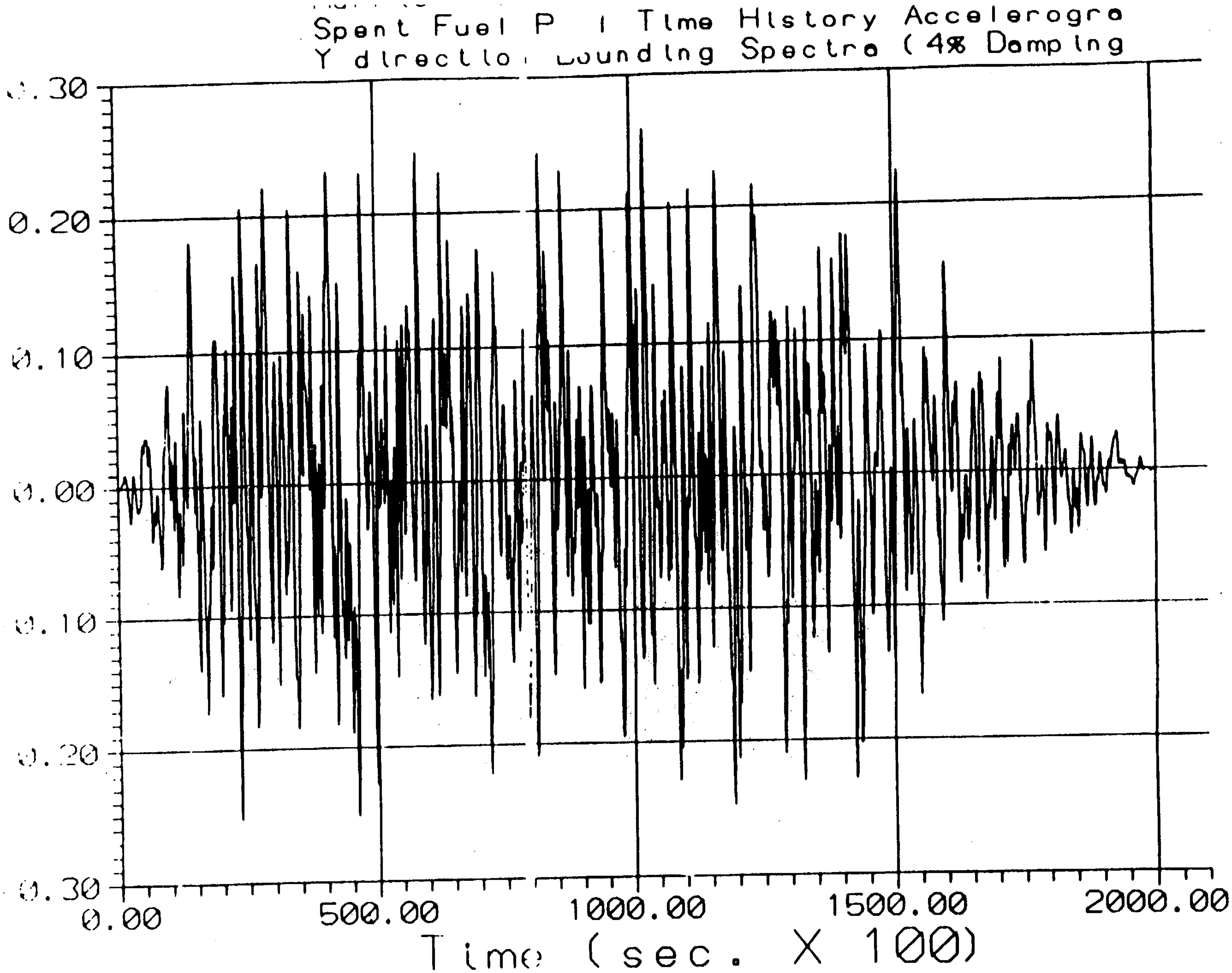


Figure 6.4.4

HI-971760



Spent Fuel P l Time History Accelerogra
Y directio Bounding Spectra (4% Damping
0.30
0.20
0.10
0.00
0.10
0.20
0.30
0.00
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)

Figure 6.4.5

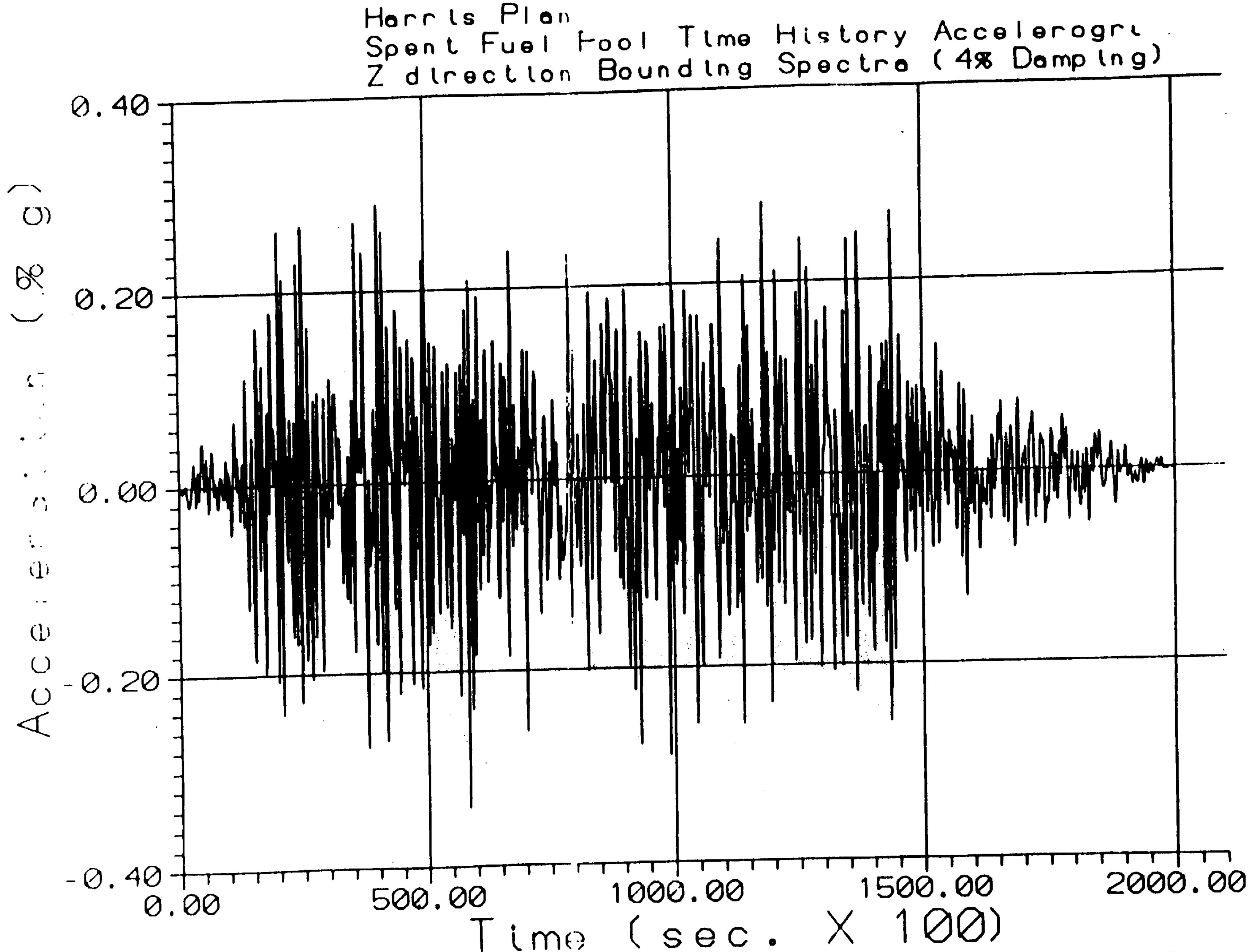

Harris Plan
Spent Fuel Fool Time History Accelerogr
Z direction Bounding Spectra (4% Damping)
0.40
0.20
0.00
-0.20
-0.40
0.00
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)

Figure 6.4.6

FIGURE 6.5.1: SCHEMATIC OF THE DYNAMIC MODEL FOR DYNARACK

REPORT # HI-971760

REPORT # HI-971760

FIGURE 6.5.3; TWO DIMENSIONAL VIEW OF THE SPRING-MASS SIMULATION

REPORT # HI-971760

Holtec Proprietary

FIGURE 6.5.4: RACK DEGREES-OF-FREEDOM FOR X-Z PLANE BENDING WITH SHEAR AND BENDING SPRING

REPORT # HI-971760

FIGURE 6.5.5; RACK-TO-RACK IMPACT SPRINGS

REPORT # HI-971760

Holtec Proprietary

HARRIS SPENT FUEL POOL C

FIGURE 6.5.6; RACK IMPACT SPRING NUMBERING SCHEME (BOTTOM)
CAMPAIGN I

HI-971760

HARRIS SPENT FUEL POOL C

FIGURE 6.5.7; RACK IMPACT SPRING NUMBERING SCHEME (TOP)
CAMPAIGN I

HI -971760

HARRIS SPENT FUEL POOL C

FIGURE 6.5.8; RACK IMPACT SPRING NUMBERING SCHEME (BOTTOM)
CAMPAIGNS II AND III

HARRIS SPENT FUEL POOL C

FIGURE 6.5.9; RACK IMPACT SPRING NUMBERING SCHEME (TOP)
CAMPAIGNS II AND III

HI -971760

HARRIS SPENT FUEL POOL D

FIGURE 6.5.10; RACK IMPACT SPRING NUMBERING SCHEME (BOTTOM)
CAMPAIGN I

Holtec Proprietary

HARRIS SPENT FUEL POOL D

FIGURE 6.5.11; RACK IMPACT SPRING NUMBERING SCHEME (TOP)
CAMPAIGN I

Holtec Proprietary

HI 971760

HARRIS SPENT FUEL POOL D

FIGURE 6.5.12; RACK IMPACT SPRING NUMBERING SCHEME (BOTTOM)
CAMPAIGN II

Holtec Proprietary

HARRIS SPENT FUEL POOL D

FIGURE 6.5.13; RACK IMPACT SPRING NUMBERING SCHEME (TOP)
CAMPAIGN II

Holtec Proprietary

Harris Pool C Run 4
Vertical Pedestal SSL Time History Loading
Rack 5, Pedestal 2

Figure 6.8.1

Figure 6.9.1: Rack Fatigue Analysis Model

7.0 FUEL HANDLING AND CONSTRUCTION ACCIDENTS

7.1 Introduction

The USNRC OT position paper [7.1] specifies that the design of the rack must ensure the functional integrity of the spent fuel racks under all credible drop events in the spent fuel pool. This section contains synopses of the analyses carried out to demonstrate the regulatory compliance of the proposed racks under postulated fuel assembly drop scenarios germane to HNP pools C and D.

In addition to the postulated fuel assembly free-fall scenarios, a gate drop accident event was also considered. In this case, the ability of the pool structure to avert primary structural damage (leading to rapid loss of water) needs to be demonstrated.

7.2 Description of Fuel Handling Accidents

Two categories of fuel assembly accidental drop events are considered. In the so-called "shallow drop" event, a fuel assembly, along with the portion of handling tool which is severable in the case of a single element failure, is assumed to drop vertically and hit the top of the rack. The "depth" of damage to the affected cell walls must be demonstrated to remain limited to the portion of the cell above the top of the "active fuel region", which is essentially the elevation of the top of the Boral neutron absorber. To meet this criterion, the plastic deformation of the rack cell wall should not extend more than 21.3 inches (downwards) from the top of a PWR rack. The distance separating the top of the rack from the Boral in the BWR racks is 13.75 inches. Therefore, to be conservative the smaller BWR dimension of 13.75 inches is selected as the maximum depth of damage of an object falling onto the tops of storage racks.

By observation, the drop of a PWR assembly onto a PWR rack is more limiting than any other combination of the two fuel types (PWR vs. BWR) with the two rack (PWR vs. BWR) types. This is obvious because of two reasons. The PWR assembly drop is a more severe case than the BWR assembly case, since the effect of the weight differences (approximately 1600 vs. 680 lbs, respectively) far exceeds the effect of the differences in the impact cross-section zone (about 8.4 vs. 5.5 inches, respectively). The PWR storage rack cell controls as an impact zone over the BWR cell because it is larger (8.4 vs. 6.06 inches, respectively) resulting in less capacity to withstand top of cell or baseplate impacts. (The nominal cell wall thicknesses of the two rack types is identical).

In order to utilize an upper bound of kinetic energy at impact, the impactor is assumed to weigh 2,100 lbs and the free-fall height is assumed to be 36 inches. The impactor weight corresponds to the heaviest fuel (plus handling tool) which will be handled in pools C and D.

It is readily apparent from the design of the rack modules described in Section 3, that the impact resistance of a rack at its periphery is less than its interior. Accordingly, the potential shallow drop scenario is postulated to occur at the periphery in the manner shown in Figure 7.2.1.

Finally, the fuel assembly assemblage is assumed to hit the rack in a manner to inflict maximum damage. The impact zone is chosen to minimize the cross sectional area which experiences the deformation. Placement of the impact at the corner would reduce the impact zone area, but actually increases the cross-sectional area experiencing deformation. Impact at the corner would involve the crushing of two cell walls under the dynamic impact. Therefore, impact on only one cell wall is chosen to simulate the worst case accident. Figure 7.2.2 depicts the impacted rack in plan view.

The second class of "fuel drop event" postulates that the impactor falls through an empty storage cell impacting the rack baseplate. This so-called "deep drop" scenario threatens the structural integrity of the "baseplate". If the baseplate is pierced, then the fuel assembly might damage the pool liner (which at 3/16" is rather thin) and create an abnormal condition of the enriched zone of fuel assembly outside the "poisoned" space of the fuel rack. To preclude damage to the pool liner, and to avoid the potential of an abnormal fuel storage configuration in the aftermath of a deep drop event, it is required that the baseplate remain unpierced and that the maximum lowering of the fuel assembly support surface is less than the distance from the bottom of the baseplate to the liner.

The deep drop event can be classified into two scenarios, namely, drop through cell located above a support leg (Figure 7.2.3), and drop in an interior cell away from the support pedestal (Figure 7.2.4).

In the former deep drop scenario (Figure 7.2.3), the baseplate is buttressed by the support pedestal and presents a hardened impact surface, resulting in a high impact load. The principal design objective is to ensure that the support pedestal does not pierce the lined, reinforced concrete pool slab.

The baseplate is not quite as stiff at cell locations away from the support pedestal (Figure 7.2.4). Baseplate severing and large deflection of the baseplate (such that the liner would be impacted) would constitute an unacceptable result.

7.3 Description of the Rack Drop Accident

The drop of a rack above spent fuel stored within in-place rack modules is precluded, since racks will not be lifted above spent fuel. The drop of a rack module during installation is also extremely remote, due to the defense-in-depth approach discussed in Sections 3.5 and

11.1. Despite the unlikelihood of this possibility, a rack dropping to the pool floor has been considered. To evaluate the consequences of an accidental, uncontrolled lowering of the heaviest rack module, a 13x13 BWR module conservatively considered with a submerged weight of 16140 lb (actual maximum nominal dry weight is only 15700 lb), from a height of 480 inches above the pool liner is considered (Figure 7.3.1). The objective of the analysis is to ensure that a rapid loss of pool water will not occur, leading to loss of shielding to the stored nuclear fuel.

7.4 Mathematical Model

In the first step of the solution process, the velocity of the dropped object (impactor) is computed for the condition of underwater free fall. Table 7.1 contains the results for the three drop events.

In the second step of the solution, an elasto-plastic finite element model of the impacted region on Holtec's computer Code PLASTIPACT (Los Alamos Laboratory's DYNA3D implemented on Holtec's QA system) is prepared. PLASTIPACT simulates the transient collision event with full consideration of plastic, large deformation, wave propagation, and elastic/plastic buckling modes. For conservatism, the impactor in all cases is assumed to be *rigid*. The physical properties of material types undergoing deformation in the postulated impact events are summarized in Table 7.2.

7.5 Fuel Drop Results

7.5.1 Shallow Drop Events

Figure 7.5.1 shows the finite element model utilized in the shallow drop impact analysis.

Dynamic analyses show that the top of the impacted region undergoes severe localized deformation. Figure 7.5.2 shows an isometric view of the post-impact geometry of the rack for the shallow drop scenario. The maximum depth of plastic deformation is limited to 11 inches, which is below the design limit of 13.75 inches. Figure 7.5.3 shows the plan view of the post-collision geometry. Approximately 10% of the cell opening in the impacted cell is blocked.

7.5.2 Deep Drop Events

The deep drop scenario depicted in Figure 7.5.4(b), wherein the impact region is located above the support pedestal, is found to produce a negligible deformation on the baseplate. The vertical force in the support pedestal remains below the loads generated during seismic events (see Section 6). Therefore, it is concluded that the pool liner will not be damaged.

The deep drop condition through an interior cell depicted in Figure 7.5.4(a) does produce some deformation of the baseplate and localized severing of the baseplate/cell wall welds (Figure 7.5.5). However, the fuel assembly support surface is lowered by a maximum of 2.89 inches, which is less than the minimum distance of 6 inches from the bottom of the baseplate to the liner. Therefore, the deformed baseplate will not strike the liner during this drop event and the pool liner will not be damaged. As stated in Subsection 4.7.2, criticality evaluations performed for this baseplate deformation have shown that the storage configuration remains acceptable.

7.6 Results of Other Drop Scenarios

Since the primary structural integrity of the slab is unimpaired subsequent to a rack drop to the pool floor liner, catastophic loss of pool water would not occur. Therefore, catastrophic failure of the pool structure or rapid loss of pool water will not occur.

No other credible in-pool drops have been identified. An object potentially carried over the pools is one of the 4,000 pound gates which isolate the pools. These gates are long rectangular metallic structures with a base area of 8 inches by 41 inches. During handling the gate is lifted using a single failure proof crane and double rigging. The rigging complies with the safety margin requirements of NUREG-0612. An accidental drop of the gate is not a credible event, because of the above mentioned defense-in-depth approach to the lifting of this heavy load. Additionally the gates are not located within the pools, but are installed inside of slots within adjacent transfer canals. Nevertheless, analyses were carried out for this accident scenario. A gate drop during handling from 40 feet above the pool liner was evaluated and it has been determined that a primary failure of the water retaining concrete structure will not occur. A gate drop during handling from 15 inches above the top of a PWR rack loaded with fuel was also evaluated. A schematic of the 3D finite element model is depicted in Figures 7.6.1 and 7.6.2. The gate is conservatively considered to strike only three rack storage cell walls, as shown in Figure 7.6.3. This impact zone is conservative, since the dimensions of the gate would span at least four cell walls. The gate is shown to penetrate the rack to a depth of less than 5 inches, as shown in Figures 7.6.4 and 7.6.5. Since this penetration remains above the tops of the stored fuel assemblies, no fuel damage occurs.

7.7 Closure

The fuel assembly and gate drop accident events evaluated for the HNP fuel pools were analyzed and found to produce localized damage well within the design limits for the racks. A construction accident event wherein the heaviest rack falls from a 40 foot height onto the pool floor was also considered. Analyses show that the pool structure will not suffer any primary structural damage. A similar conclusion is reached with regard to a gate drop event.

7.8 References

[7.1] "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

TABLE 7.1

IMPACT EVENT DATA

Case	Impactor Weight (lbs)	Impacter	Drop Height (inches)	Impact Velocity (inch/sec)
1. Shallow drop event	2,100	Fuel Assembly	36	152
2. Deep drop event	2,100	Fuel Assembly	205	353
3. Construction event	16,140	Rack Module	480	304

TABLE 7.2

MATERIAL DEFINITION

Material Name	Type	Density (pcf)	Elastic Modulus (psi)	Stress: First Yield (psi)	Stress: Failure (psi)	Strain: Elastic	Strain: Failure
Stainless steel	SA240-304L						
Stainless steel	SA240-304						
Stainless steel	SA564-630						
Concrete	4000 psi						

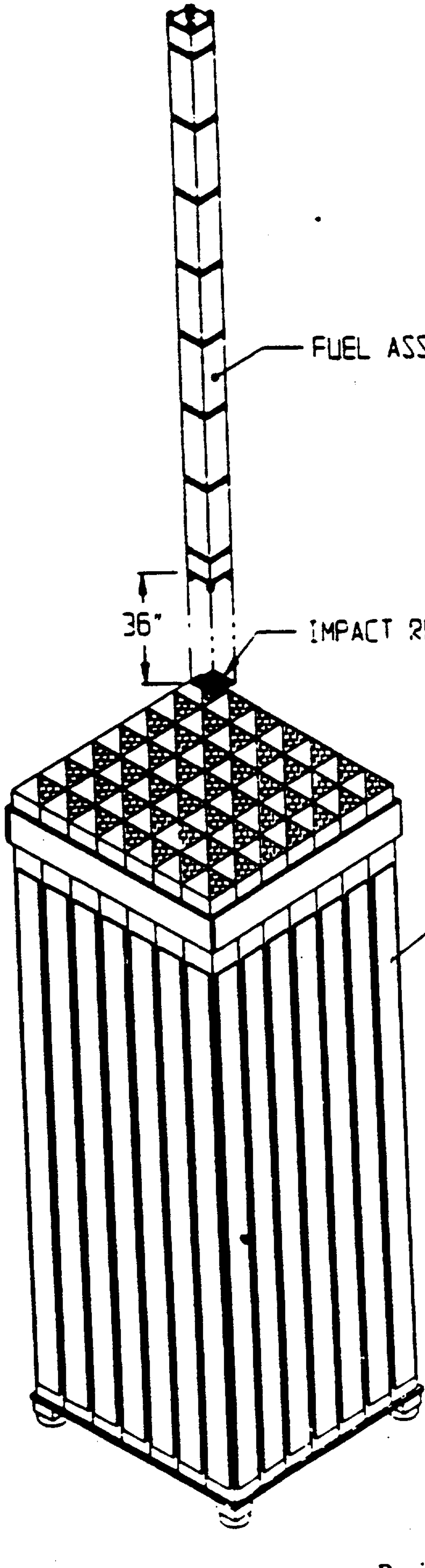


Figure 7.21: Shallow Drop on a Peripheral Cell

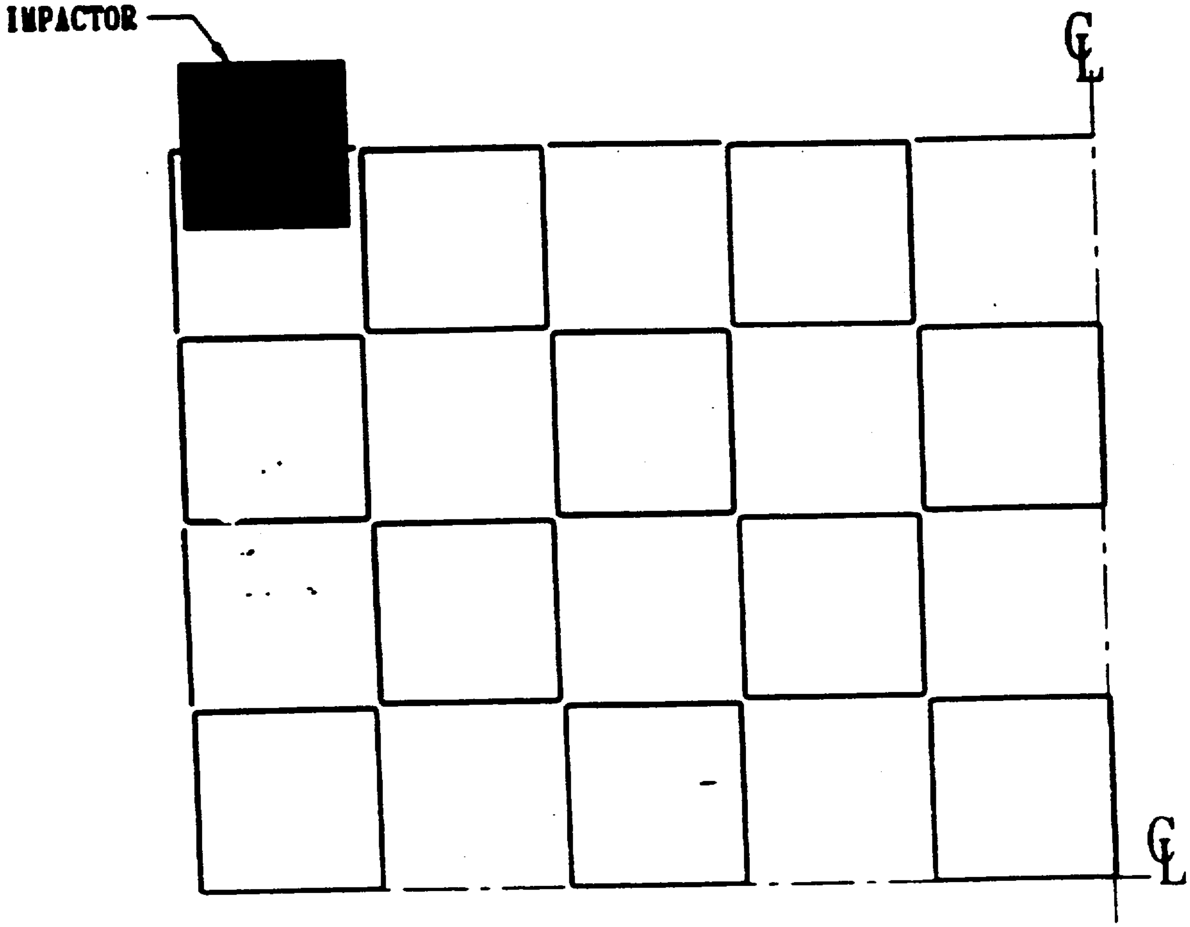


Figure 7.2.2: Plan View of Impactor and Impact Zone

(Shallow Drop Event)

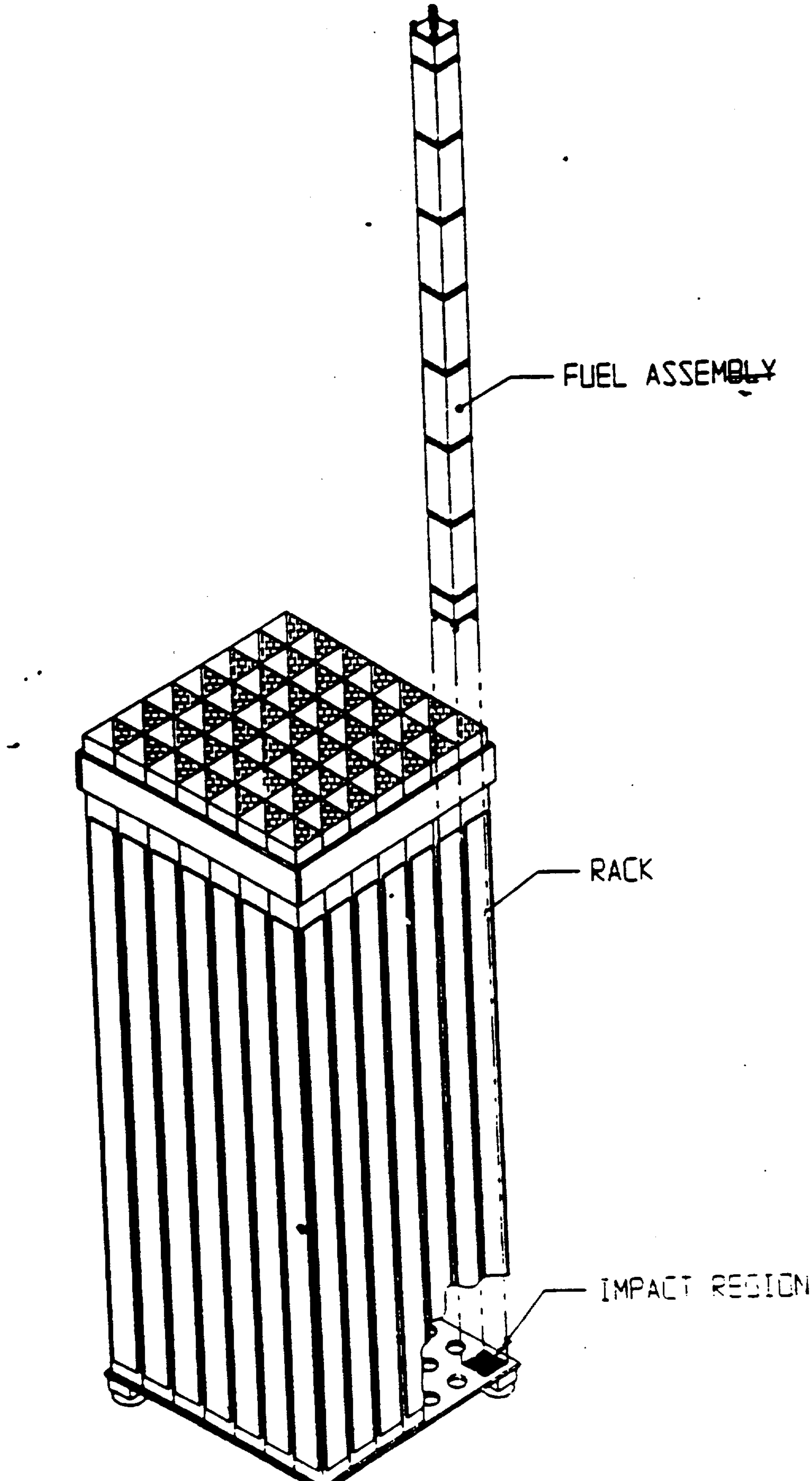


Figure 7.23: Deep Drop on a Support Leg Location

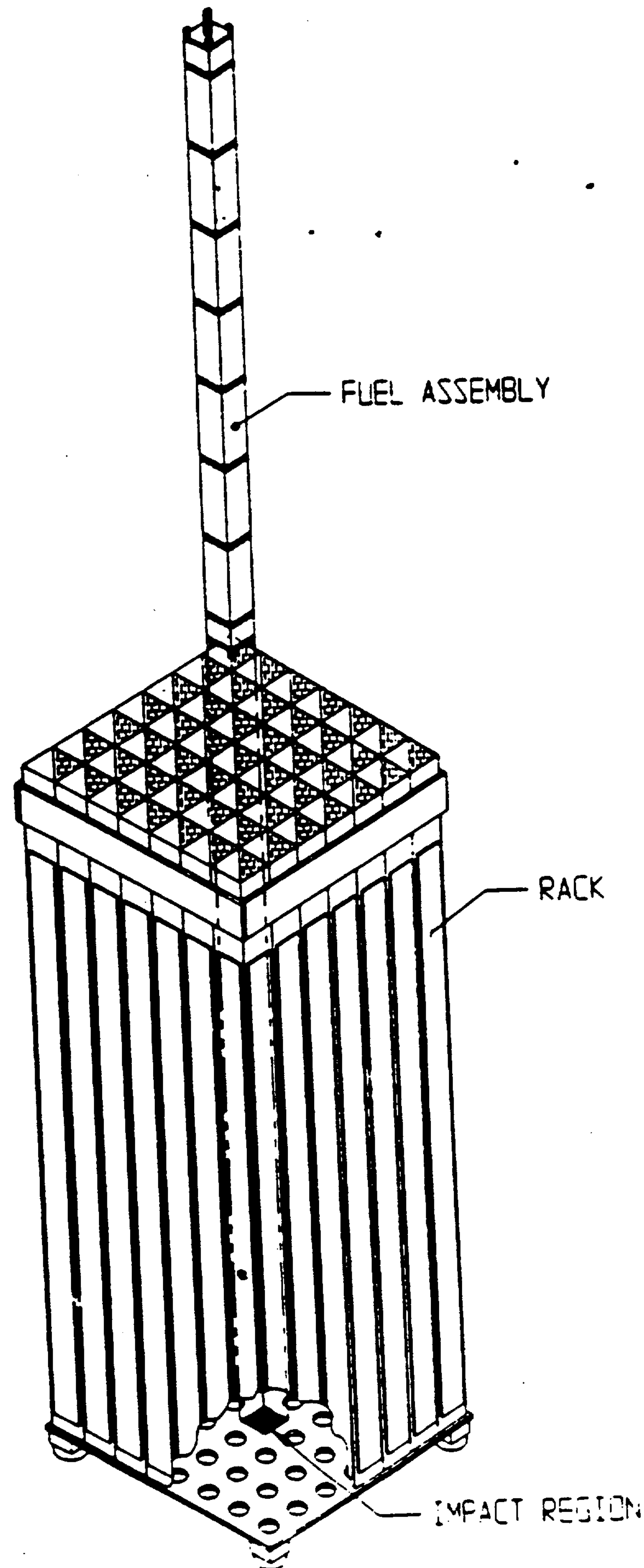


Figure 7.2.4: Deep Drop on a Center Cell Leg Location

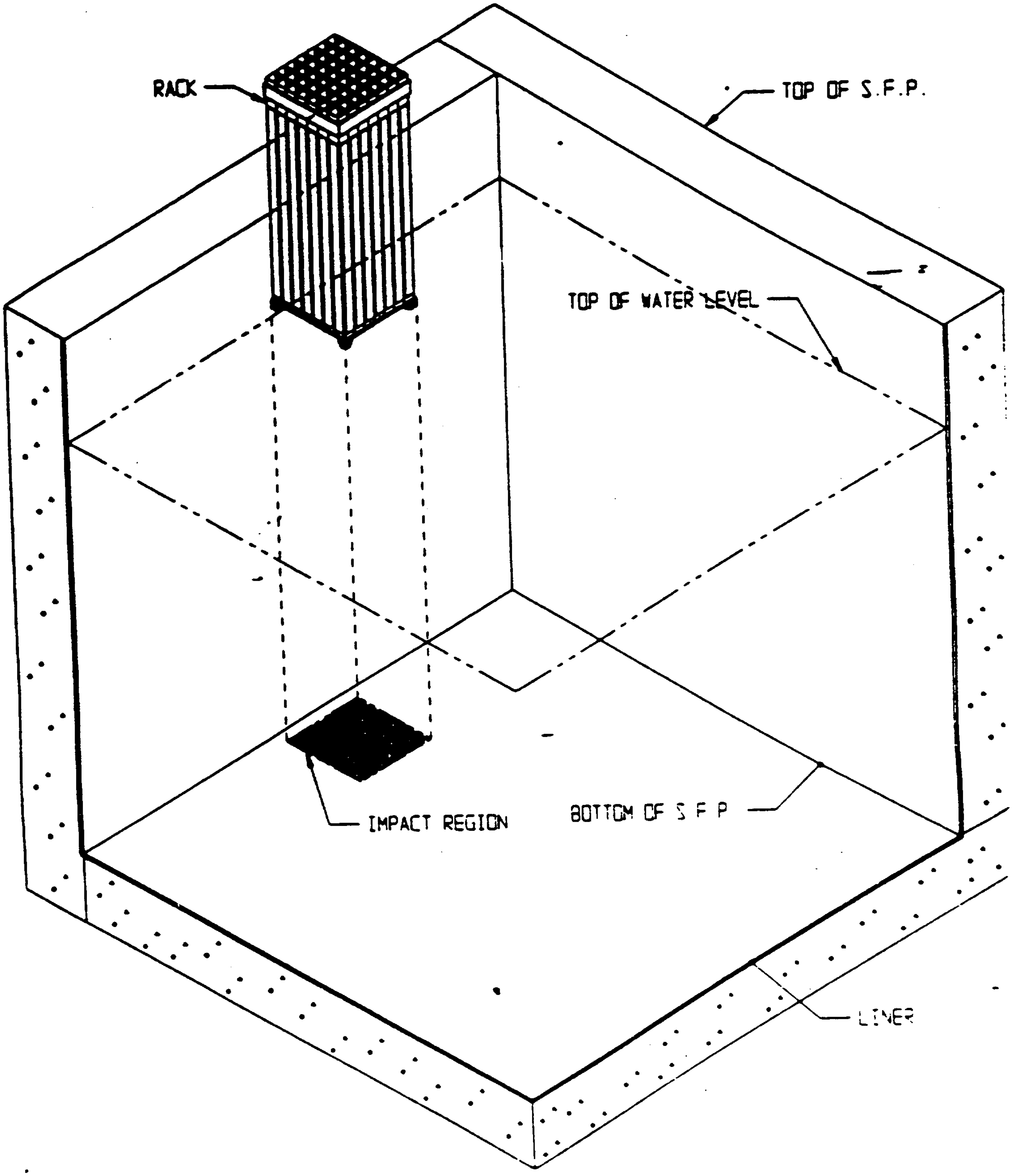


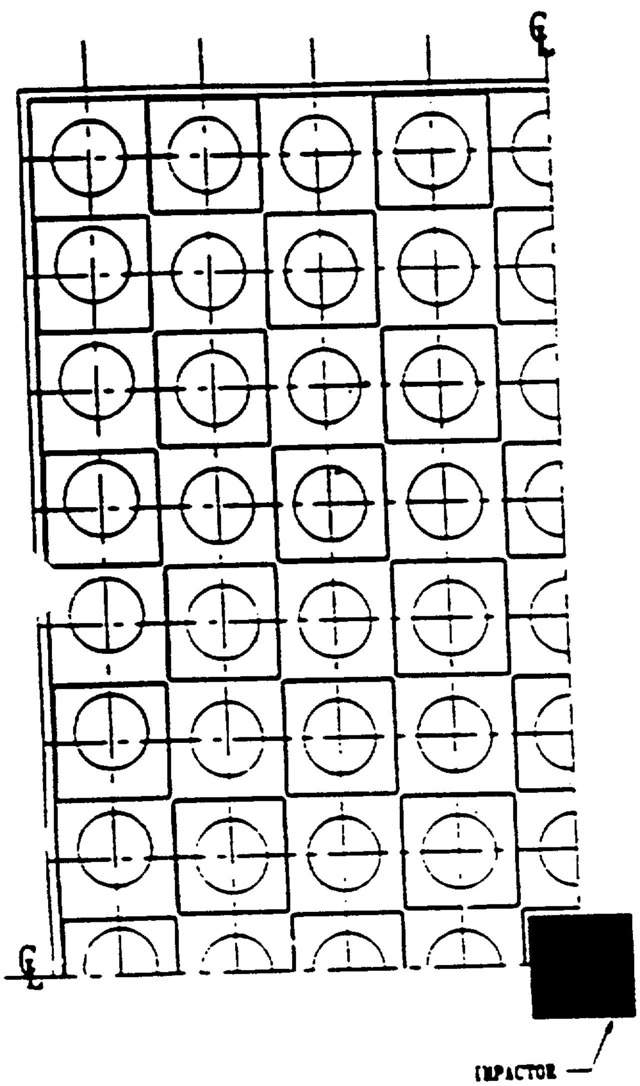
Figure 7.31: Heaviest Rack Drop

Figure 7.51: Shallow Drop: Finite Element Model Detail Impacted Region

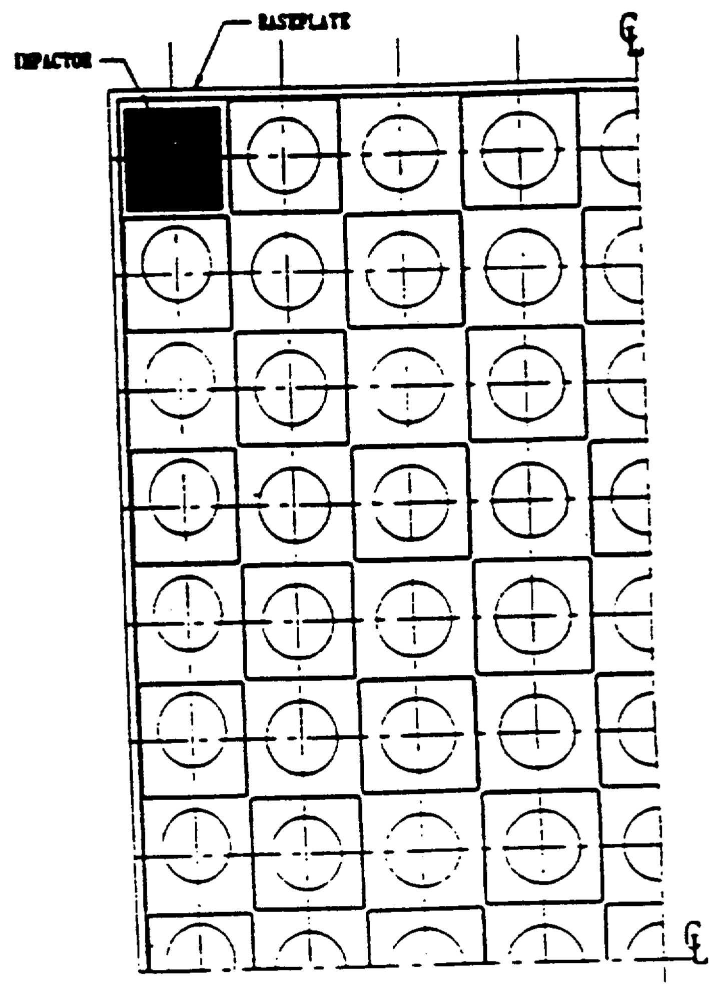
Figure 7.5.2: Maximum Cell Deformation for Shallow Drop on Exterior Cell

Figure 7.53: Shallow Drop: Maximum Cell Deformation

Impacted Region Plan



(a) *SCENARIO* SC1



(b) *SCENARIO* SC2

Figure 7.5.4: Plan View of Deep Drop Scenarios

Figure 7.5.5: Maximum Baseplate Deformation from Deep Drop Scenario

Figure 7.6.1: Gate Drop Finite-Element Model

Figure 7.6.2: Gate Drop Finite-Element Model. Detail of Impacted Region

Figure 7.6.3: Gate Drop Finite-Element Model

Detail of Impacted Region (Plan)

Figure 7.6.4: Gate Drop Maximum Deformation

Figure 7.6.5: Gate Drop Maximum Deformation

Impacted Region Plan

8.0 FUEL POOL STRUCTURE INTEGRITY CONSIDERATIONS

8.1 Introduction

The Harris Spent Fuel Pools (SFPs) C and D are safety related, seismic category I, reinforced concrete structures. Spent fuel is to be placed within storage racks located in both of these areas and they will be collectively referred to herein as the fuel pool structure. This section describes the analysis to demonstrate structural adequacy of the pool structure, as required by Section IV of the USNRC OT Position Paper [8.1.1].

The pool regions are analyzed using the finite element method. Results for individual load components are combined using factored load combinations mandated by SRP 3.8.4 [8.1.2] based on the "ultimate strength" design method of the American Concrete Institute (ACI 318) [8.1.3]. It is demonstrated that for the critical bounding factored load combinations, structural integrity is maintained when the pools are assumed to be fully loaded with spent fuel racks, as shown in Figures 1.2 and 1.3 with all storage locations occupied by fuel assemblies.

The regions examined in the SFPs are the floor slabs, and the highly loaded wall sections adjoining the slabs. Both moment and shear capabilities are checked for concrete structural integrity. Local punching and bearing integrity of the slab in the vicinity of a rack module support pedestal pad is evaluated. All structural capacity calculations are made using design formulas meeting the requirements of ACI 318.

8.2 Description of Pool Structures

The SFPs are located inside the Fuel Handling Building and are supported by a two way, reinforced concrete slab. The minimum thickness of the slab is 12.0 feet, including grout. The SFPs are separated by reinforced concrete walls and transfer canals.

Figure 1.1 shows the layout of the majority of the Fuel Handling Building. A plan of the building area of concern is shown in Figure 8.2.1, which shows the major structural dimensions of the pools. The floor liner plate of the SFPs are located at elevation 246.0 The spent fuel area operating floor is at elevation 286.0.

8.3 Definition of Loads

Pool structural loading involves the following discrete components:

8.3.1 Static Loading (Dead Loads and Live Loads)

1) Dead weight of pool structure includes the weight of the Fuel Handling Building concrete upper structure.

2) Maximum dead weight of rack modules and fuel assemblies in the fully implemented storage configuration, as shown in Figures 1.2 and 1.3.

3) Dead weight of a shipping cask including yoke of 250 kips.

4) The Cask Crane, Auxiliary Crane and Spent Fuel Handling Machine (Refueling Platform) are designed to move along the N-S direction. The dead weight and the rated lift weight of these cranes are considered as live load.

5) The hydrostatic water pressure.

8.3.2 Seismic Induced Loads

1) Vertical loads transmitted by the rack support pedestals to the slab during a SSE or OBE seismic event.

2) Hydrodynamic inertia loads due to the contained water mass and sloshing loads (considered in accordance with TID-7024 [8.3.1]) which arise during a seismic event.

3) Hydrodynamic pressures between racks and pool walls caused by rack motion in the pool during a seismic event.

4) Seismic inertia force of the walls and slab.

8.3.3 Thermal Loading

Thermal loading is defined by the temperature existing at the faces of the pool concrete walls and slabs. Two thermal loading conditions are evaluated: The normal operating temperature and the accident temperature.

8.4 Analysis Procedures

8.4.1 Finite Element Analysis Model

The finite element model encompasses the two SFPs, the Fuel Transfer Canal, the Cask Loading Pool, and adjacent transfer canals and building structure. The interaction with the rest of the Fuel Handling Building reinforced concrete, which is not included in the finite-element model, is simulated by imposing appropriate boundary conditions. The structural area of interest for the reracking project includes only two pools which are involved in the fuel storage capacity increase. However, by augmenting the area of interest, by considering in the constructed finite-element model and numerical investigation the additional areas described above, the perturbation induced by the boundary conditions on the stress field distribution for the area of interest is minimized. A finite element 3D view of the structural elements considered in the numerical investigation is shown in Figure 8.4.1.

The preprocessing capabilities of the STARDYNE computer code [8.4.1] are used to develop the 3-D finite-element model. The STARDYNE finite-element model contains 13,353 nodes, 3,564 solid type finite-elements, 7,991 plate type finite-elements and 24 hydro-dynamic masses. Figure 8.4.1 depicts an isometric view of the three-dimensional finite element model without the water and concentrated masses (racks, cask, etc.).

The dynamic behavior of the water mass contained in the SFPs and Transfer Canal during a seismic event is modeled according to the guidelines set in TID-7024.

8.4.2 Analysis Methodology

The structural region of concern, from column lines 43 to 73 and from line L to N, is isolated from the Fuel Handling Building. This region is numerically investigated using the finite element method. The pool walls and their supporting reinforced concrete slab are represented by a 3-D finite-element model.

The individual loads considered in the analysis are grouped in five categories: dead load (weight of the pool structure, dead weight of the rack modules and stored fuel, dead weight of the reinforced concrete Fuel Handling Building upper structure, the hydro-static pressure of the contained water), live loads (weights of the Cask Crane, Auxiliary Crane, and SFHM and their maximum suspended loads), thermal loads (the thermal gradient through the pool walls and slab for normal operating and accident conditions) and the seismic induced forces (structural seismic forces, interaction forces between the rack modules and the pool slab, seismic loads due to self-excitation of the pool structural elements and contained water, and seismic hydro-dynamic interaction forces between the rack modules and the pool walls for both OBE and SSE conditions). The dead and thermal loads are considered static acting loads, while the seismic induced loads are time-dependent.

Results for individual load cases are combined using the factored load combinations discussed below. The combined stress resultants are compared with the ultimate moments and shear capacities of all structural elements pertinent to the SFPs, which are calculated in accordance with the ACI 318 to develop the safety factors.

8.4.3 Load Combinations

The various individual load cases are combined in accordance with the NUREG-0800 Standard Review Plan [8.1.2] requirements with the intent to obtain the most critical stress fields for the investigated reinforced concrete structural elements.

For "Service Load Conditions" the following load combinations are:

- Load Combination No. 1 = 1.4* D + 1.7*L
- Load Combination No. 2 = 1.4* D + 1.7*L + 1.9*E
- Load Combination No. 3 = 1.4* D + 1.7*L - 1.9*E
- Load Combination No. 4 = 0.75* (1.4* D + 1.7*L + 1.9*E +1.7*To)
- Load Combination No. 5 = 0.75* (1.4* D + 1.7*L - 1.9*E +1.7*To)
- Load Combination No. 6 = 1.2*D + 1.9*E
- Load Combination No. 7 = 1.2*D - 1.9*E

For "Factored Load Conditions" the following load combinations are:

- Load Combination No. 8 = D + L + To + E'

- Load Combination No. 9 = D + L + To - E'

- Load Combination No. 10 = D + L + Ta + 1.25*E

- Load Combination No. 11 = D + L + Ta - 1.25*E

- Load Combination No. 12 = D + L + Ta + E'

- Load Combination No. 13 = D + L + Ta - E'

where:

D = dead loads;
L = live loads;
To = thermal load during normal operation;
Ta = thermal load under accident condition;
E = OBE earthquake induced loads;
E' = SSE earthquake induced loads.

8.5 Results of Analyses

The STARDYNE computer code is used to obtain the stress and displacement fields for the 1 individual load cases.

The STARDYNE postprocessing capability is employed to form the appropriate load combinations and to establish the limiting bending moments and shear forces in various sections of the pool structure. A total of 13 load combinations are computed. Section limit strength formulas for bending loading are computed using appropriate concrete and reinforcement strengths. For Harris, the concrete and reinforcement allowable strengths are:

concrete f_c'	= 4,000 psi
reinforcement f_y	= 60,000 psi

Table 8.5.1 shows results from potentially limiting load combinations for the bending and shear strength of the slab and walls. For each section, we define the limiting safety margins as the limited strength bending moment or shear force defined by ACI for that structural section divided by the calculated bending moment or shear force (from the finite element analyses). The major regions of the pool structure consist of the four concrete walls and floors delimiting each of the SFPs. Each area is searched independently for the maximum bending moments in different bending directions and for the maximum shear forces. Safety margins are determined from the calculated maximum bending moments and shear forces based on the local strengths. The procedures are repeated for all the potential limiting load combinations. Therefore, limiting safety margins are determined. Table 8.5.1 demonstrates that the limiting safety margins for all sections are above 1.0, as required.

8.6 Pool Liner

The pool liners are subject to in-plate strains due to movement of the rack support feet during the seismic event. Analyses are performed to establish that the liner will not tear or rupture under limiting loading conditions in the pool. These analyses are based on loadings imparted from the most highly loaded pedestal in the pool assumed to be positioned in the most unfavorable position. Bearing strength requirements are shown to be satisfied by conservatively analyzing the most highly loaded pedestal located in the worst configuration with respect to underlying leak chases.

8.7 Conclusions

Regions affected by loading the fuel pool completely with high density racks are examined for structural integrity under bending and shearing action. It is determined that adequate safety margins exist assuming that all racks are fully loaded with a bounding fuel weight and that the factored load combinations are checked against the appropriate structural design strengths. It is also shown that local loading on the liner does not compromise liner integrity under a postulated fatigue condition and that concrete bearing strength limits are not exceeded.

8.8 References

[8.1.1] OT Position for Review and Acceptance of Spent Fuel Handling Applications, by B.K. Grimes, USNRC, Washington, D.C., April 14, 1978.

[8.1.2] NUREG-0800, SRP-3.8.4, Rev. 1., July 1981.

[8.1.3] ACI 318-95 and ACI 318R-95, "Building Code Requirements for Structural Concrete and Commentary," American Concrete Institute, 1995.

[8.3.1] "Nuclear Reactors and Earthquakes, U.S. Department of Commerce, National Bureau of Standards, National Technical Information Service, Springfield, Virginia (TID 7024).

[8.4.1] STARDYNE User's Manual, Research Engineers, Inc., Rev. 4.4, July 1996.

[8.4.2] ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Detroit Michigan.

Table 8.5.1

BENDING AND SHEAR STRENGTH EVALUATION

Pool	Location	Limiting Safety Margin	Critical Flexure Load Combinations (see Section 8.4.3)	Limiting Safety Margin	Critical Shear Load Combinations (see Section 8.4.3)
C	North Wall	1.97	2	1.31	2
	South Wall	3.51	2	2.20	3
	East Wall	1.72	2	1.10	5
	West Wall	1.05	10	1.06	4
	Pool Floor Slab	2.1	2	2.71	2
D	North Wall	2.32	2	3.43	3
	South Wall	1.30	10	1.08	3, 7
	East Wall	1.48	2, 6	1.07	3, 7
	West Wall	1.05	4	1.06	2
	Pool Floor Slab	2.01	2	1.64	3

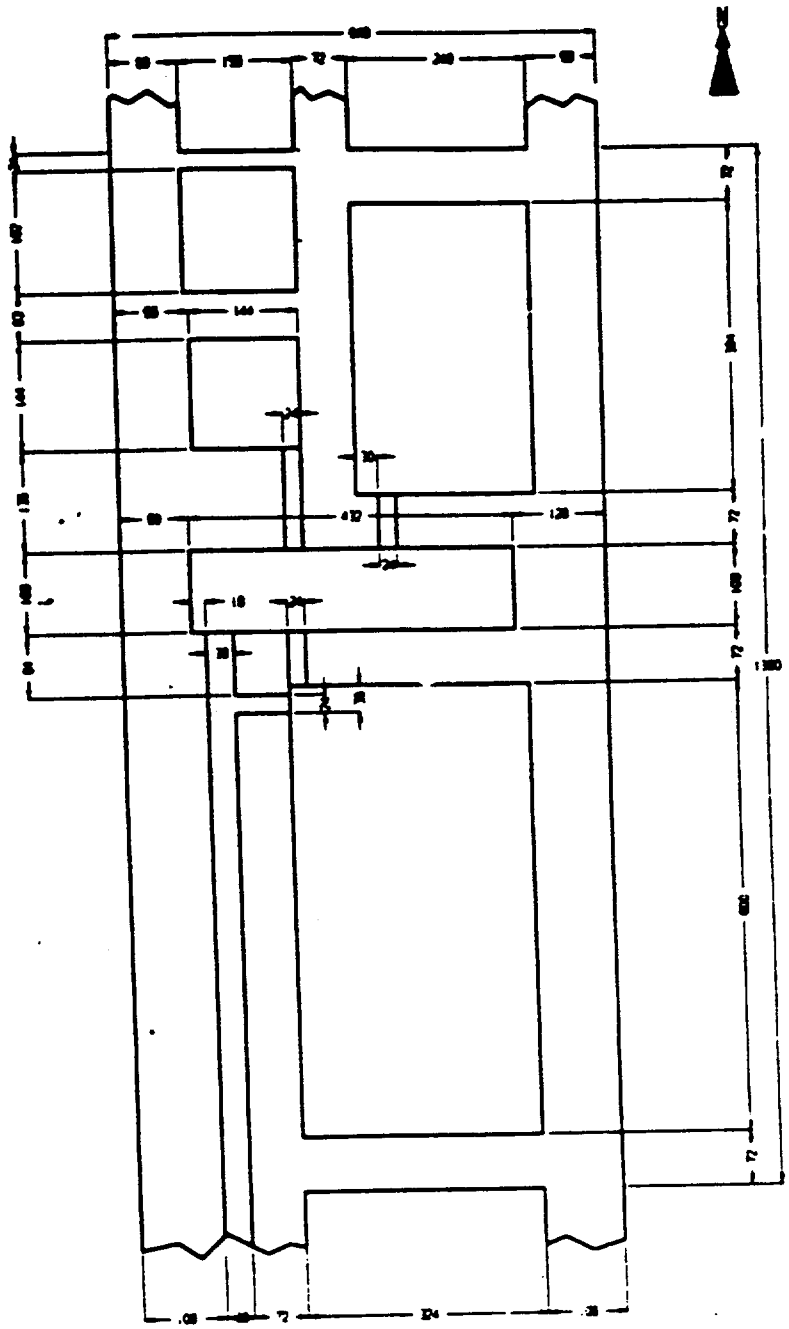


Figure 8.2.1: Pool Structure Dimensions

Figure 8.4.1; Fuel Handling Building

Finite Element Model

9.0 RADIOLOGICAL EVALUATION

9.1 Solid Radwaste

No significant increase in the volume of solid radioactive wastes is expected from opera[illegible] with the expanded storage capacity. 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. During racking operations, a small amount of additional resins may be generated by the pool cleanup system on a one-time basis.

9.2 Gaseous Releases

Gaseous releases from the fuel storage area are combined with other plant exhausts. Normally, the contribution from the fuel storage area is negligible compared to the other releases and no significant increases are expected as a result of the expanded storage capacity.

9.3 Personnel Doses

During normal operations, personnel working in the fuel storage area are exposed to radiation from the spent fuel pool. Operating experience has shown that area radiation dose rates originate primarily from radionuclides in the pool water. As expected, subsequent to the removal of transhipped fuel from the shipping casks, Harris has experienced increases in the pool water radionuclide concentrations due to sloughing of crud and other contaminants associated with fuel handling. Additionally, radionuclide concentration increases are also experienced subsequent to the discharge of fuel from the Harris Unit 1 reactor. These two conditions represent the the previously analyzed conditions for pool water radionuclide concentrations and will not be significantly changed by the capacity expansion of storing spent fuel in pools C and D. Therefore, no additional evaluations for pool water radionuclides are required for the proposed change.

Radiation dose rates in accessible areas around the SFPs will be determined for comparison with existing zone designations. Any changes required to the zone designations will be identified and included in an update to the Harris FSAR, if necessary.

Operating experience has also shown that there have been negligible concentrations of airborne radioactivity in the Spent Fuel Pool area. No increase in airborne radioactivity is expected as a result of the expanded storage capacity.

9.4 Anticipated Dose During Re-racking

All of the operations involved in racking will utilize detailed procedures prepared with full consideration of ALARA principles. Similar operations have been performed in a number of facilities in the past, and there is every reason to believe that racking can be safely and efficiently accomplished at Harris, with low radiation exposure to personnel. The Harris racking project represents lower radiological risks due to the fact that the pools currently contain no spent fuel.

[redacted]

The existing radiation protection program at Harris is adequate for the re-racking operations. Where there is a potential for significant airborne activity, continuous air monitors will be in operation. Personnel will wear protective clothing as required and, if necessary, respiratory protective equipment. Activities will be governed by a Radiation Work Permit, and personnel monitoring equipment will be issued to each individual. As a minimum, this will include thermoluminescent dosimeters (TLDs) and self-reading dosimeters. Additional personnel monitoring equipment (i.e., extremity TLDs or multiple TLDs) may be utilized as required.

Work, personnel traffic, and the movement of equipment will be monitored and controlled to minimize contamination and to assure that dose is maintained ALARA.

Table 9.4.1

PRELIMINARY ESTIMATE OF PERSON-REM DOSE DURING RACKING

Step	Number of Personnel	Hours	Estimated Person-Rem Dose
Clean and vacuum pool	[redacted]	[redacted]	[redacted]
Remove underwater appurtenances	[redacted]	[redacted]	[redacted]
Installation of new rack modules	[redacted]	[redacted]	[redacted]
Total Dose, person-rem	[redacted]	[redacted]	[redacted]

10.0 INSTALLATION

10.1 Introduction

The construction phase of the Harris Spent fuel pool rack installation will be executed by Carolina Power & Light. CP&L will also be responsible for specialized services, such as underwater diving and welding operations, if required. All construction work at Harris will be performed in compliance with NUREG-0612 (refer to Section 3.0), and site-specific procedures.

Crane and fuel bridge operators are to be adequately trained in the operation of load handling machines per the requirements of ANSI/ASME B30.2, latest revision, and the plant's specific training program.

The lifting devices designed for handling and installation of the new racks and removal of the old racks at Harris are remotely engageable. The lifting devices comply with the provisions of ANSI N14.6-1978 and NUREG-0612, including compliance with the primary stress criteria, load testing at a multiplier of maximum working load, and nondestructive examination of critical welds.

An intensive surveillance and inspection program shall be maintained throughout the rack installation phase of the project. A set of inspection and QC hold points will be implemented which have been proven to eliminate any incidence of rework or erroneous installation in numerous previous rack installation campaigns in Pools A and B.

Holtec International and CP&L have developed a complete set of operating procedures which cover the entire gamut of operations pertaining to the rack installation effort. Similar procedures have been utilized and successfully implemented by Holtec International on previous rack installation projects. These procedures assure that ALARA practices are followed and provide detailed requirements to assure equipment, personnel, and plant safety. The following is a list of procedures which will be available for use in implementing the rack installation phase of the project.

A. Installation/Handling Procedure:

This procedure provides direction for the handling/installation of the new high density modules. The procedure delineates the steps necessary to receive a new high density rack on site, and the proper method for unloading and uprighting the rack, staging the rack prior to installation, and installation of the rack. The procedure also provides for the installation of new rack bearing pads, adjustment of the new rack pedestals and performance of the as-built field survey.

B. Receipt Inspection Procedure:

This procedure delineates the steps necessary to perform a thorough receipt inspection of a new rack module after its arrival on site. The receipt inspection includes dimensional measurements, cleanliness inspection, visual weld examination, and verticality measurements.

C. Cleaning Procedure:

This procedure provides for the cleaning of a new rack module, if it is required, in order to meet the requirements of ANSI 45.2.1, Level C. Permissible cleaning agents, methods and limitations on materials to be employed are provided.

D. Pre-Installation Drag Test Procedure:

This procedure stipulates the requirements for performing a functional test on a new rack module prior to installation into Pools C or D. The procedure provides direction for inserting and withdrawing a "dummy" fuel assembly into designated cell locations, and establishes an acceptance criteria in terms of maximum kinetic drag force.

E. Post-Installation Drag Test Procedure:

This procedure stipulates the requirements for performing a functional test on a new rack module following installation into Pools C or D. The procedure will provide direction for inserting and withdrawing a "dummy" fuel assembly into designated cell locations, and establishes an acceptance criteria in terms of maximum kinetic drag force.

F. Underwater Diving Procedure:

Underwater diving operations may be required to assist in the positioning of new rack modules. This procedure describes the method for introducing a diver into Pools C or D, provides for radiological monitoring during the operation, and defines the egress of the diver from the fuel pool following work completion. Furthermore, this procedure requires strict compliance with OSHA Standard 29CFR-1910, Subpart T, and establishes contingencies in the event of an emergency.

G. ALARA Procedure:

Consistent with Holtec International's ALARA Program, this procedure provides details to minimize the total man-rem received during the rack installation project, by accounting for time, distance, and shielding. Additionally, a pre-job checklist is established in order to mitigate the potential for an overexposure.

H. Liner Inspection Procedure:

In the event that a visual inspection of any submerged portion of the Spent Fuel Pool liner is deemed necessary, this procedure describes the method to perform such an inspection using an underwater camera and describes the requirements for documenting any observations.

I. Leak Detection Procedure:

This procedure describes the method to test the Spent Fuel Pool liner for potential leakage using a vacuum box. This procedure may be applied to any suspect area of the pool liner.

J. Underwater Welding Procedure:

In the event of a positive leak test result, an underwater welding procedure will be implemented which will provide for the placement of a stainless steel repair patch over the area in question. The procedure contains appropriate qualification records documenting relevant variables, parameters, and limiting conditions. The weld procedure is qualified in accordance with AWS D3.6-93, Specification for Underwater Welding or may be qualified to an alternate code accepted by CP&L and Holtec International.

K. Job Site Storage Procedure:

This procedure establishes the requirements for safely storing a new rack module on-site, in the event that long term job-site storage is necessary. This procedure provides environmental restrictions, temperature limits, and packaging requirements.

10.2 Rack Arrangement

Pools C and D at Harris have been previously unused. The new rack arrangement has been prepared to maximize flexibility in the number and type (PWR vs. BWR) fuel assemblies stored. The new rack arrangement for Pool C consists of a mixture of free-standing PWR and BWR Holtec racks.

A breakdown of the number of racks and storage cells in the first campaign and completely filled configuration of Pool C is as follows:

	First Campaign		Filled Pool	
	Cells	Racks	Cells	Racks
PWR Cells	360	4	927	11
BWR Cells	1320	10	2763	19
Total	1680	14	3690	30

Pool D will store a maximum of 1025 PWR assemblies in 12 rack modules. Racks will be added to the pools on an as needed basis. A schematic plan view depicting the Spent Fuel Pools in the new maximum density configuration can be seen in Figure 1.1.

10.3 Pool Survey and Inspection

A pool inspection shall be performed to determine if any items attached to the liner wall or floor will interfere with the placement of the new racks or prevent usage of any cell locations subsequent to installation.

In the event that protrusions are found which would pose any interference to the installation process, it is anticipated that underwater diving operations and mechanical cutting methods would be employed to remove the protrusions.

10.4 Pool Cooling and Purification

10.4.1 Pool Cooling

The pool cooling system shall be operated in order to maintain the pool water temperature at an acceptable level. It is anticipated that specific activities, such as bearing pad elevation measurements, may require the temporary shutdown of the Spent Fuel Pool cooling system. At no time, however, will pool cooling be terminated in a manner or for a duration which would create a violation of the Harris Technical Specification or procedures.

Prior to any shutdown of the Spent Fuel Pool cooling system, the duration to raise the pool bulk temperature to 137°F will be determined. A margin temperature of 112°F is chosen such that the cooling system may be restarted prior to reaching this temperature. This will ensure that the pool bulk temperature will always remain below 137°F.

10.4.2 Purification

The existing Spent Fuel Pool filtration system shall be operational in order to maintain pool clarity. Additionally, an underwater vacuum system shall be used as necessary to supplement fuel pool purification. The vacuum system may be employed to remove extraneous debris, reduce general contamination levels prior to diving operations, and to assist in the restoration of pool clarity following any hydrolasing operations.

10.5 Installation of New Racks

The new high density racks shall be delivered in the horizontal position. A new rack module shall be removed from the shipping trailer using a suitably rated crane, while maintaining the horizontal configuration, and placed upon the upender and secured. Using two independent overhead hooks, or a single overhead hook and a spreader beam, the module shall be uprighted into vertical position.

The new rack lifting device shall be installed into the rack and each lift rod successively engaged. Thereafter, the rack shall be transported to a pre-levelled surface where the appropriate quality control receipt inspection shall be performed.

In preparing Pool C or D for the initial rack installation, the pool floor shall be inspected and any debris which may inhibit the installation of bearing pads will be removed. New rack bearing pads shall be positioned in preparation for the rack modules which are to be installed. Elevation measurements will then be performed in order to gage the amount of adjustment required, if any, for the new rack pedestals.

The new rack module shall be lifted with the Auxiliary Crane and transported along the safe load path. The rack pedestals shall be adjusted in accordance with the bearing pad elevation measurements in order to achieve module levelness after installation.

It is anticipated that the rack modules shall be lowered into the Pools C and D using the Cask Handling Crane. A hoist with sufficient capacity will be attached to the Auxiliary Crane for installation and removal activities in order to eliminate contamination of the main hook during lifting operations in the pools. The rack shall be carefully lowered onto its bearing pads. Movements along the pool floor shall not exceed six inches above the liner, except to allow for clearance over floor projections.

Elevation readings shall be taken to confirm that the module is level and as-built rack-to-rack and rack-to-wall offsets shall be recorded. The lifting device shall be disengaged and removed from the fuel pool under Radiation Protection direction.

10.6 Safety, Radiation Protection, and ALARA Methods

10.6.1 Safety

During the rack installation phase of the project, personnel safety is of paramount importance, outweighing all other concerns. All work shall be carried out in strict compliance with applicable approved procedures.

10.6.2 Radiation Protection

Radiation Protection shall provide necessary coverage in order to provide radiological protection and monitor dose rates. The Radiation Protection department shall prepare Radiation Work permits (RWPs) that will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels, and dosimetry requirements.

In addition, no activity within the radiologically controlled area shall be carried out without the knowledge and approval of Radiation Protection. Radiation Protection shall also monitor items removed from the pool or provide for the use of alarming dosimetry and supply direction for the proper storage of radioactive material.

10.5.3 ALARA

The key factors in maintaining project dose As Low As Reasonably Achievable (ALARA) are time, distance, and shielding. These factors are addressed by utilizing many mechanisms with respect to project planning and execution.

Time

Each member of the project team will be properly trained and will be provided appropriate education and understanding of critical evolutions. Additionally, daily pre-job briefings will be employed to acquaint each team member with the scope of work to be performed and the proper means of executing such tasks. Such pre-planning devices reduce worker time within the radiologically controlled area and, therefore, project dose.

Distance

Remote tooling such as lift fixtures, pneumatic grippers, a support levelling device and a lift rod disengagement device have been developed to execute numerous activities from the pool surface, where dose rates are relatively low. For those evolutions requiring diving operations, diver movements shall be restricted by an umbilical, which will assist in maintaining a safe distance from irradiated sources. By maximizing the distance between a radioactive sources and project personnel, project dose is reduced.

Shielding

During the course of the rack installation, primary shielding is provided by the water in the Spent Fuel Pool. The amount of water between an individual at the surface (or a diver in the pool) and an irradiated fuel assembly is an essential shield that reduces dose. Additionally, other shielding, may be employed to mitigate dose when work is performed around high dose rate sources.

10.7 Radwaste Material Control

Radioactive waste generated from the rack installation effort shall include vacuum filter bags, miscellaneous tooling, and protective clothing.

Vacuum filter bags shall be removed from the pool and stored as appropriate in a suitable container in order to maintain low dose rates.

Contaminated tooling shall be properly stored per Radiation Protection direction throughout the project. At project completion, an effort will be made to decontaminate tooling to the most practical extent possible.

11.0 ENVIRONMENTAL COST/BENEFIT ASSESSMENT

11.1 Introduction

Article V of the USNRC OT Position Paper [11.1] requires the submittal of a cost/benefit analysis for the chosen fuel storage capacity enhancement method. This section provides justification for selecting rack installation in Pools C and D as the most viable alternative.

11.2 Imperative for Increased Storage Capacity

The specific need to increase the limited existing storage capacity at the Harris facility is based on the continually increasing inventory in Pools A and B due to core offloads at Harris and transhipments from the Robinson and Brunswick plants, the prudent requirement to maintain full-core offload capability, and a lack of viable economic alternatives.

Based on the current number of stored assemblies and estimated discharge and transhipment rates, the Harris fuel pool is projected to lose the capacity to discharge one full core in 2001. This projected loss of storage capacity in the Harris pool would affect CP&L's ability to operate the reactors. CP&L does not have an existing or planned contractual arrangement for third party fuel storage or fuel reprocessing.

11.3 Appraisal of Alternative Options

CP&L has determined that rack installation at the Harris pools is by far the most viable option for increasing spent fuel storage capacity in comparison to other alternatives.

The key considerations in evaluating the alternative options are:

- Safety: minimize the number of fuel handling steps
- Economy: minimize total installed and O&M cost
- Security: protection from potential saboteurs, natural phenomena
- Non-intrusiveness: minimize required modification to existing systems
- Maturity: extent of industry experience with the technology
- ALARA: minimize cumulative dose due to handling of fuel

Rack installation was found by CP&L to be the most attractive option in respect to each of the foregoing criteria. An overview of the alternatives is provided in the following.

Rod Consolidation

Rod consolidation has been shown to be a potentially feasible technology. Rod consolidation involves disassembly of spent fuel, followed by the storage of the fuel rods from two assemblies into the volume of one and the disposal of the fuel assembly skeleton outside of the pool (this is considered a 2:1 compaction ratio). The rods are stored in a stainless steel can that has the outer dimensions of a fuel assembly. The can is stored in the spent fuel racks. The top of the can has an end fixture that matches up with the spent fuel handling tool. This permits moving the cans in an easy fashion.

Rod consolidation pilot project campaigns in the past have consisted of underwater tooling that is manipulated by an overhead crane and operated by a maintenance worker. This is a very slow and repetitive process.

The industry experience with rod consolidation has been mixed thus far. The principal 'antages of this technology are: the ability to modularize, compatibility with DOE waste management system, moderate cost, no need of additional land and no additional required surveillance. The disadvantages are: potential gap activity release due to rod breakage, potential for increased fuel cladding corrosion due to some of the protective oxide layer being scraped off, potential interference of the (prolonged) consolidation activity which might interfere with ongoing plant operation, and lack of sufficient industry experience.

On-Site Cask Storage

Dry cask storage is a method of storing spent nuclear fuel in a high capacity container. The cask provides radiation shielding and passive heat dissipation. Typical capacities for PWR fuel range from 21 to 37 assemblies that have been removed from the reactor for at least five years. The casks, once loaded, are then stored outdoors on a seismically qualified concrete pad. The pad will have to be located away from the secured boundary of the site because of site limitations. The orage location will be required to have a high level of security which includes frequent tours, reliable lighting, intruder detection, (E-field), and continuous visual monitoring.

The casks, as presently licensed, are limited to 20-year storage service life. Once the 20 years has expired the cask manufacturer or the utility must recertify the cask or the utility must remove the spent fuel from the container.

There are several plant modifications required to support cask use. Tap-ins must be made to the gaseous waste system and chilled water to support vacuum drying of the spent fuel and piping must be installed to return cask water back to the Spent Fuel Pools. A seismic concrete pad must be made to store the loaded casks. This pad must have a security fence, surveillance protection, a diesel generator for emergency power and video surveillance.

Finally, the cask facility must have equipment required to vacuum dry the cask, backfill it with helium, make leak checks, remachine the gasket surfaces if leaks persist, and assemble the cask on-site. For casks which have closure gaskets, the space between the inner and outer lid must be continuously monitored to check for inner seal failure.

Presently, no MPC cask has been licensed. Because of the continued uncertainty in the government's policy, the capital investment to develop a dry storage system is considered to be an inferior alternative for Harris at this time.

Modular Vault Dry Storage

Vault storage consists of storing spent fuel in shielded stainless steel cylinders in a horizontal configuration in a reinforced concrete vault. The concrete vault provides radiation shielding and missile protection. It must be designed to withstand the postulated seismic loadings for the site.

A transfer cask is needed to fetch the storage canisters from the fuel pool. The plant must provide for a decontamination bay to decontaminate the transfer cask, and connection to its gaseous waste system and chilled water systems. A collection and delivery system must be installed to return the pool water entrained in the canisters back to the fuel pool. Provisions for canister drying, helium injection, handling, and automatic welding are also necessary.

The storage area must be designed to have a high level of security similar to that of the nuclear plant itself. Due to the required space, the vault secured area must be located outside the secured perimeter. Consideration of safety and security requires it to have its own video surveillance system, intrusion detection, and an autonomous backup diesel generator power source.

Some other concerns relating to the vault storage system are: inherent eventual "repackaging" for shipment to the DOE repository, the responsibility to eventually decommission the new facility, large "footprint" (land consumption), potential fuel handling accidents, potential fuel/clad rupture due to high temperature and high cost.

At the present time, no MPC technology based vault system has yet been offered for licensing to the USNRC. Therefore, this option is considered to be unavailable at this time.

<u>Horizontal Silo Storage</u>

A variation of the horizontal vault storage technology is more aptly referred to as "horizontal silo" storage. This technology suffers from the same drawbacks which other dry cask technologies do, namely,

i. No fuel with cladding defects can be placed in the silo.

ii. Concern regarding long-term integrity of the fuel at elevated temperature.

iii. Potential for eventual repackaging at the site.

iv. Potential for fuel handling accidents.

v. Relatively high cumulative dose to personnel in effecting fuel transfer (compared to rack installation).

vi. Compatibility of reactor/fuel building handling crane with fuel transfer hardware.

vii. Potential incompatibility with DOE shipment for eventual off-site shipment.

viii. Potential for sabotage.

11.3.1 Alternative Option Summary

An estimate of relative costs in 1997 dollars for the aforementioned options is provided in the following:

Rack Installation:	$12 million
Horizontal Silo:	$35-45 million
Rod consolidation:	$25 million
Metal cask (MPC):	$68-100 million
Modular vault:	$56 million

The above estimates are consistent with estimates by EPRI and others [11.2, 11.3].

To summarize, there are no acceptable alternatives to increasing the on-site spent fuel storage capacity of Harris. First, there are no commercial independent spent fuel storage facilities operating in the U.S. Second, the adoption of the Nuclear Waste Policy Act (NWPA) created a de facto throw-away nuclear fuel cycle. Since the cost of spent fuel reprocessing is not offset by the salvage value of the residual uranium, reprocessing represents an added cost for the nuclear fuel cycle which already includes the NWPA Nuclear Waste Fund fees. In any event, there are no domestic reprocessing facilities. Third, at over $½ million per day replacement power cost, shutting down the Harris reactor is many times more expensive than simply installing racks in the existing Spent Fuel Pools.

11.4 Cost Estimate

The proposed construction contemplates installation of storage modules in Harris Pools C and D using free-standing, high density, poisoned spent fuel racks. The engineering and design is completed for rack installation in the pools. This rack installation project will provide sufficient pool storage capacity to maintain full-core offload capability until the end of the current plant license.

The total capital cost is estimated to be approximately $12 million as detailed below.

Engineering, design, project management:	$2 million
Rack fabrication:	$7 million
Rack installation:	$3 million

As described in the preceding section, many alternatives were considered prior to proceeding with rack installation, which is not the only technical option available to increase on-site storage capacity. Rack installation does, however, enjoy a definite cost advantage over other technologies.

11.5 Resource Commitment

The expansion of the Harris Spent Fuel Pool capacity is expected to require the following primary resources:

Stainless steel:	250 tons
Boral neutron absorber:	20 tons, of which 15 tons is Boron Carbide powder and 5 tons are aluminum.

The requirements for stainless steel and aluminum represent a small fraction of total world output of these metals (less than 0.001%). Although the fraction of world production of Boron Carbide required for the fabrication is somewhat higher than that of stainless steel or aluminum, it is unlikely that the commitment of Boron Carbide to this project will affect other alternatives. Experience has shown that the production of Boron Carbide is highly variable and depends upon need and can easily be expanded to accommodate worldwide needs.

11.6 Environmental Considerations

Prior to the proposed modification, Pools C and D were maintained full of water with levels consistent with those of Pools A and B. Although water was allowed to be exchanged between all four pools at various times, there was no heat load associated with Pools C and D. Therefore, the bulk pool temperatues in Pools C and D have always been maintained at or below the temperatures in Pools A and B. Due to the heat load arising from the spent fuel inventory, the pool cooling system will be connected to Pools C and D to provide adequate heat removal capabilities. The maximum normal bulk pool temperature will be realized when the capacity is maximized for Pools C and D, but will still be $\leq$ 137°F †.

Maintaining four pools (instead of the previous two pools) in the Fuel Handling Building with bulk pool temperatures $\leq$137°F † will result in an increase in the pool water evaporation rate. This pool water evaporation increase has been determined to increase the relative humidity of the Fuel Building atmosphere by less than 10%. This increase is within the capacity of both the normal and the ESF Ventilation Systems. The net result of the increased heat loss and water vapor emission to the environment is negligible.

---

† The 137°F limit is consistent with that currently in the Harris FSAR and procedures for pools A and B. CP&L is in the process of re-evaluating systems and components to allow for an increase the allowable bulk pool temperature.

11.7 References

[11.1] OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications, USNRC (April 1978).

[11.2] Electric Power Research Institute, Report No. NF-3580, May 1984.

[11.3] "Spent Fuel Storage Options: A Critical Appraisal", Power Generation Technology, Sterling Publishers, pp. 137-140, U.K. (November 1990).

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

10CFR50.55a ALTERNATIVE PLAN

10CFR50.55a ALTERNATIVE PLAN

I. Introduction

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

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

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

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

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

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

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

Construction Records Issue

The completed portion of the Unit 2 Fuel Pool Cooling and Cleanup System (FPCCS) and supporting facilities were constructed to the same codes and standards and using the same procedures and personnel as was Unit 1, which was fully completed and licensed. Appropriate records documenting field activities were generated at the time of construction as required by the construction codes and plant procedures, and maintained in storage under the control of the construction Quality Assurance (QA) program pending system completion and turnover. When construction on Unit 2 was halted, these records were transferred to temporary storage facilities maintained by the Harris Nuclear Plant Document Control. They were not microfilmed since they were associated with systems which were not fully completed and accepted under the site's N Certificate Program, and later were inadvertently discarded during a document control records cleanup effort.

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

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

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

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

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

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

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

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

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

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

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

Piping Pedigree Plan - Implementation

ASME Section III Piping:

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

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

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

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

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

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

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

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

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

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

B31.1 Piping:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

III. Alternative Plan for Continuance of Design and Construction

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

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

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

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

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

Enclosure 9 to Serial: HNP-98-188

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

UNREVIEWED SAFETY QUESTION ANALYSIS

CCW UNREVIEWED SAFETY QUESTION DISCUSSION

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

CURRENT SYSTEM CONFIGURATION

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

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

When the Emergency Core Cooling System is aligned to recirculate from the containment sump to the Reactor Coolant System, the CCW trains are separated from each other and from the non-essential header to maintain protection against a single passive failure and to provide sufficient flow to their respective RHR trains. In this alignment, each CCW train

[1] Controlled by revised Technical Specification 5.6

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

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

10CFR50.59 SAFETY EVALUATION OVERVIEW

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

UNREVIEWED SAFETY QUESTION DETERMINATION

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

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

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

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

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

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

SUMMARY

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

January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

EXHIBITS SUPPORTING THE SUMMARY OF FACTS, DATA, AND ARGUMENTS ON WHICH APPLICANT PROPOSES TO RELY AT THE SUBPART K ORAL ARGUMENT

VOLUME 1

EXHIBIT 1 (A)

4/7

EXHIBIT 1

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

AFFIDAVIT OF R. STEVEN EDWARDS

COUNTY OF WAKE	)	
	) ss:	
STATE OF NORTH CAROLINA	)	

I, Robert Steven Edwards, being sworn, do on oath depose and say:

1. I am a resident of the State of North Carolina. I am employed by Carolina Power & Light Company ("CP&L") and work at the Harris Nuclear Plant ("HNP" or "Harris Plant" or "Harris") in the Nuclear Engineering Department. Presently, I am the Supervisor, Spent Fuel Pool Project, and am responsible for commissioning and placing into service spent fuel pools C and D at the Harris Plant. My business address is 5413 Shearon Harris Road, New Hill, North Carolina 27562-0165.

2. I was graduated from North Carolina State University in 1982 with a B.S. in Industrial Engineering. Since graduation, I have been employed by CP&L, first as

an Associate Engineer, then Engineer, at the Robinson Nuclear Plant, responsible for planning, scheduling and execution of outages and major projects. Beginning in 1986, I served in the Technical Support Unit at the Robinson Plant as a System Engineer – Mechanical Systems. Promoted to Senior Engineer in July 1988, I supervised a staff of contract engineers responsible for specific projects at the Robinson Plant. In June 1991, I assumed the position of Project Engineer – Mechanical Systems at the Robinson Plant and managed a staff of four system engineers and two component engineers responsible for the operation, performance, reliability and maintenance of various plant systems. In August 1992, I became the Director – Information Architecture (Nuclear) in CP&L's Corporate Management Services and served as the management-level liaison and project manager for nuclear-related information technology projects at CP&L's nuclear plants. In October 1994, I moved to the position of Director – Project Control in the Corporate Nuclear Business Operations Group. In that position, I facilitated the development of long-range planning at each CP&L nuclear plant and provided oversight and administration of project management and economic evaluation processes and activities. In July 1996, I moved to Corporate Nuclear Engineering and became Manager of Projects, responsible for scope, cost, schedule, and quality of various nuclear projects. In April 1998, I was assigned to Harris Plant Major Projects Section and became responsible for the spent fuel pools C and D activation projects, including the completion of the spent fuel pool cooling and cleanup system ("SFPCCS"), spent fuel storage rack design and installation, and related activities.

3. The purpose of this affidavit is to set forth facts and data on which CP&L relies in establishing that there is no genuine and substantial dispute of fact raised by Intervenor Board of Commissioners of Orange County in Technical Contentions 2 and 3 admitted in the above-captioned proceeding. First, I summarize the background of the license amendment request and the information submitted in support of the application. Second, I describe Harris Plant procedures, controls, physical conditions, physical constraints, and calculations that establish a single fuel assembly misplacement in HNP spent fuel pools C and D, involving a fuel element of the wrong burnup or enrichment, cannot cause criticality in the fuel pool. Third, I describe the basis for the 10 C.F.R. §50.55a Alternative Plan that provides assurance of acceptable quality and safety of the stainless steel piping that is part of the spent fuel pool cooling and cleanup system for spent fuel pools C and D -- notwithstanding the destruction of the weld data reports for the field welds in that piping. Fourth, I describe the measures set forth in the Equipment Commissioning Plan for spent fuel pools C and D to ensure that there has not been significant degradation of the components and piping in the SFPCCS that would affect their suitability for service. I provide the results of additional inspections and tests to confirm the acceptable condition of the SFPCCS piping embedded in concrete. Finally, I discuss the insignificant impact on Harris Plant operations and safety in the highly improbable event of a failure of a weld in the

embedded piping, and describe the counter-balancing hardship and unusual difficulty that would result if CP&L were required to commission spent fuel pools C and D without approval of the 50.55a Alternative Plan.

BACKGROUND

4. CP&L's application for a license amendment to place spent fuel pools C and D in service was submitted on December 23, 1998. The license amendment request includes nine enclosures with supporting information(Attachment A). As the project manager for the HNP spent fuel pool C and D activation projects, I was responsible for development of the factual information set forth in the license amendment request. The information in Attachment A is accurate to the best of my knowledge and belief. It has been updated by additional information that is also attached to this affidavit.

5. The license amendment request and the need to expand spent fuel storage at HNP results from the failure of the U.S. Department of Energy ("DOE") to begin taking delivery of spent fuel in 1998, as required by the contract between DOE and CP&L and by the Nuclear Waste Policy Act of 1982, as amended. CP&L requested that the license amendment to allow placement of spent fuel in spent fuel pools C and D be issued no later than December 31, 1999. CP&L plans to begin loading spent fuel in pool C starting in 2000. Delays would adversely impact CP&L's ability to maintain adequate spent fuel storage capacity and, with

the loss of full core discharge capability at one or more of CP&L's nuclear plants, could lead to a forced shutdown condition.

6. The NRC Staff reviewers requested additional information regarding the license amendment request by letters dated March 24, 1999, April 24, 1999, June 16, 1999, August 5, 1999, and September 20, 1999. CP&L responded to each request for additional information ("RAI") respectively on April 30, 1999, June 14, 1999, July 23, 1999, September 3, 1999, and October 29, 1999. CP&L also provided additional information to the NRC Staff on October 15, 1999 to supplement previous responses. Attachment B to this affidavit is CP&L's April 30, 1999 Response to RAI 1. Attachment C is CP&L's June 14, 1999 Response to RAI 2. Attachment D is CP&L's September 3, 1999 Response to RAI 4. Attachment E is CP&L's October 29, 1999 Response to RAI 5. Attachment F is CP&L's October 15, 1999 supplementary response to previous RAI's. (Not all of the enclosures to the RAI responses are attached; nor is the Response to RAI 3 (on seismic issues) attached.) As the project manager for the HNP spent fuel pool C and D activation projects, I was responsible for development of the factual information set forth in responses to NRC Staff RAIs. The information in Attachments B, C, D, E, and F is accurate to the best of my knowledge and belief.

7. The Harris Plant was originally planned as a four nuclear unit site (Harris 1, 2, 3 and 4). In order to accommodate four units, the Harris Fuel Handling Building was designed and constructed with four separate pools capable of storing spent fuel. Spent fuel pools A and B were originally intended to support Harris 1 and 4.

Spent fuel pools C and D were originally intended to support Harris 2 and 3. The original design included a spent fuel pool cooling and cleanup system ("SFPCCS") to service spent fuel pools A and B, and a separate SFPCCS to service spent fuel pools C and D. The purpose of the SFPCCS is to maintain water quality in the spent fuel pools, transfer canals, cask loading pool and the reactor cavity, and remove residual heat generated in the stored spent fuel. The SFPCCS consists of a cooling system and a cleanup system. The major system components are the fuel pool heat exchangers, fuel pool demineralizer, fuel pool cooling pumps, filters, skimmers, water purification pumps, valves, piping, fuel pool gates, strainers, instrumentation and system controls. Attachment G is the system description (SD-116) from Volume 6 of the Harris Plant Operating Manual and provides a more detailed description of the SFPCCS. Attachment H is a simplified schematic of the SFPCCS.

8. Harris 3 and 4 were canceled in late 1981. Harris 2 was canceled in late 1983. Spent fuel pools A, B, C and D and the SFPCCS for spent fuel pools A and B were completed as part of the Fuel Handling Building, are described in the HNP Final Safety Analysis Report, and are licensed as part of the HNP. In addition, HNP was licensed to accept spent fuel for storage from CP&L's other nuclear plants, H. B. Robinson Unit 2, and Brunswick Units 1 and 2. Beginning in 1989, spent fuel assemblies from Robinson and Brunswick have been regularly shipped to the Harris Plant and are stored in spent fuel pools A and B.

9. Construction on the SFPCCS for spent fuel pools C and D was discontinued after Harris 2 was canceled. By that time, all four spent fuel pools had been constructed, concrete had been poured, and the SFPCCS piping immediately outside and under the spent fuel pools was installed, welded in place and embedded in reinforced concrete. The SFPCCS for spent fuel pools A and B was completed and placed in service. Harris 1 began commercial operations in 1987. Sometime in late 1988 or 1989, before the first refueling of Harris 1 and discharge of spent fuel to the spent fuel pool, spent fuel pool A was filled with demineralized water. Boric acid was added to the water before spent fuel was discharged into the spent fuel pool. (A minimum 2000 parts per million ("ppm") boron concentration – boron absorbs neutrons -- is maintained in the spent fuel pool water to provide criticality control.) On or about the time spent fuel was first discharged from the Harris reactor, spent fuel pool B was filled with demineralized water with the same concentration of boric acid. Because spent fuel pools C and D are connected to spent fuel pools A and B by transfer canals, at some point in 1989 or later, spent fuel pools C and D were also filled with demineralized water with the same concentration of boric acid. This allows the gates in the transfer canal to be opened without a loss of water and precludes an inadvertent partial drain-down of spent fuel pools A and B to spent fuel pools C and D.

10. When Harris 2 was canceled in 1983, work was stopped on the SFPCCS for spent fuel pools C and D. The heat exchangers were subsequently filled with nitrogen

to inhibit any corrosion. The pump motors were stored in the warehouse for spares. The SFPCCS piping was "spared in place." Prior to filling spent fuel pools C and D with water, "plumbers plugs" were installed in the SFPCCS suction and discharge openings in the spent fuel pools and metal covers were installed at the uncompleted, open ends of the pipes (replacing wooden and/or sheet metal foreign material exclusion covers). The plumber's plugs were not leak-tight and eventually the sections of SFPCCS piping were filled with spent fuel pool water that leaked by. The SFPCCS piping was drained in 1995-1996, when drain valves were added to the accessible portions of the embedded lines. Thereafter, the lines refilled with water from the spent fuel pools leaking past the plumber's plugs. That water remained in the SFPCCS piping until drained for inspection earlier this year.

CRITICALITY CONTROL IN SPENT FUEL POOLS C AND D

11. As the project manager for the activation of spent fuel pools C and D, my work encompasses analytical design and engineering evaluations, management of the hands-on physical implementation of the modifications to the SFPCCS, and inspection and preparation of the spent fuel pools themselves. As a consequence of my extensive work at HNP and with the Harris spent fuel pools, I am familiar with the physical layout, operations, and operating procedures for the Harris Nuclear Plant, as they relate to the movement and storage of spent fuel and the operation of the spent fuel pools.

12. Each movement of an individual fuel assembly in the Harris spent fuel pools is a separate, independent action. Since there is only one set of fuel movement equipment that services the entire Fuel Handling Building, the Harris fuel movement equipment is physically able to move only one fuel assembly at a time. There is only one spent fuel bridge crane, one hoist, one upender, etc. Therefore, it is physically impossible to move more than one fuel assembly at a time. For that reason, Harris Nuclear Plant operating procedures are written to reference movement of only one spent fuel assembly at a time. See Harris operating procedures PLP-616 (Fuel Handling Operations), FHP-014 (Fuel and Insert Shuffle Sequence), FHP-020 (Refueling Operations), and FHP-024 (HNP Spent Fuel Handling Operations), included as Attachments I, J, K and L, respectively, for examples of implementing procedures which throughout discuss only one set of equipment and provide steps for movement of only one fuel assembly at a time. This same provision applies to all fuel movement operations in the Harris spent fuel pools.

13. There is no concurrent movement of more than one fuel assembly in the Harris spent fuel pools. Because of the physical constraints and procedural controls discussed above, two or more fuel assemblies cannot be moved concurrently in the Harris spent fuel pools. The maximum number of fuel assemblies that can be moved at any one time is one fuel assembly.

14. Misplacement of a single fuel assembly at HNP is highly unlikely for many reasons. The movement of fuel assemblies at the Harris Nuclear Plant is tightly

controlled by plant operating procedures to prevent challenging the spent fuel pool criticality control mechanisms. However, there are other reasons, wholly apart from criticality control, why the movement of fuel assemblies at Harris is tightly controlled.

15. NRC regulations require fissile material licensees, including CP&L at HNP, to tightly control and track the quantity and location of all fissile nuclear material in the licensee's possession. This process is referred to as materials control and accounting of special nuclear material, or "MC&A," and is controlled by 10 C.F.R. Part 74. CP&L rigorously tracks the location of all fuel assemblies for MC&A purposes through operating procedures and an electronic computer database referred to as the MAGIC database. A further description of the purpose and use of the MAGIC database is provided in CP&L Nuclear Fuels procedures NFP-NGGC-0021 (Corporate Special Nuclear Material Accountability Plan) and NFP-NGGC-0003 (Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask), which are included as Attachments M and N to this affidavit. In addition to the electronic database, paper records are kept of each fuel assembly movement which is made in the spent fuel pool, including the origination point from where the assembly started, and the destination point, typically a cell in the spent fuel pool racks, to which the fuel assembly is moved. These records are developed and retained pursuant to Harris operating procedure PLP-629 (Reactivity Management Program), Attachment 6 (Reactivity Management Controls, Spent Fuel Pool Activities) and FHP-014 (Fuel and Insert

Shuffle Sequence), which are included as Attachment O and J to this affidavit. These paper records provide the basis for entering factual information into the MAGIC electronic database.

16. The placement of fuel assemblies at Harris is also rigorously controlled for reactor operations purposes. Fuel assemblies are moved during every reactor core refueling exercise, which occurs approximately every 18 months at Harris. It is essential for reactor operations that the location of each fuel assembly be closely tracked and controlled in order to ensure that the reactor core is loaded with the correct fuel assemblies in the correct locations. Each restart reactor core is carefully designed and validated for compliance with the NRC's regulations regarding reactor safety. Knowing the location and characteristics of each assembly is essential to demonstrating compliance with the NRC's reactor operations regulations. The fuel assembly movement procedures and the MAGIC database that are used to control the location of fuel assemblies in the spent fuel pool are also used to control the location of fuel assemblies in the reactor core.

17. The MAGIC database is used to determine the location and the characteristics of each fuel assembly at the Harris Nuclear Plant. The location and characteristics of each assembly are required to be carefully tracked throughout the entire time a fuel assembly is at the Harris Nuclear Plant, from the time it arrives at the plant as either fresh fuel from the fuel vendor, or as spent fuel in a transportation cask from other CP&L plants, until it leaves the Harris site. The location and characteristics of each fuel assembly to be moved is determined in an independent

operation from the actual movement of the fuel assembly, by an individual who is different from the operators that actually move the fuel. The tracking of fuel assembly location and characteristics using the MAGIC database is discussed in CP&L Nuclear Fuels procedure NFP-NGGC-0021 (Corporate Special Nuclear Material Accountability Plan), which is included as Attachment M to this affidavit.

18. In addition to the rigorous control over the selection of a fuel assembly to move by the site reactor engineering staff, the actual physical movement of the fuel assembly has several additional features that further render a single fuel assembly misplacement highly unlikely. All fuel assembly physical movements are controlled by Harris operating procedures. The following steps summarize essential safeguards in the procedures against fuel assembly misplacement. Pursuant to procedure, fuel assemblies are moved only by licensed operators. These operators receive from the engineering staff a document identifying each fuel assembly to be moved, as well as the originating location, destination location, and fuel characteristics for each assembly. Before each fuel assembly is moved, the correct origination location is physically verified by two different operators through visual examination. The fuel assembly can only be moved after this redundant confirmation that the proper origination location is being accessed. The fuel assembly is then moved directly to its destination location. The destination location is also confirmed redundantly by two different operators through visual examination before the assembly can be inserted in that location.

The operators are not permitted to place the fuel assembly in any interim location. The fuel assembly is only permitted to be placed either in its origination location or its destination location, as identified on the Fuel Assembly and Insert Shuffle Data Sheet (Attachment 1 to FHP-014), Core Offload/Reload Fuel Transfer Data Sheet (Attachment 2 to FHP-014), or Cask to Storage Fuel Handling Data Sheet (Attachment 3 to FHP-014). The proper form to use is determined by the source location. In the event the assembly cannot be placed into its destination location, the fuel assembly is returned to its origination location. These procedures ensure that the misplacement of a single fuel assembly is highly unlikely. A typical fuel movement procedure for Harris Nuclear Plant that identifies these procedural safeguards is FHP-014 (Fuel and Insert Shuffle Sequence). A copy of this procedure is included as Attachment J to this affidavit. There is a separate fuel movement procedure for the fuel originating from each of the three CP&L plants from which fuel is stored in the Harris spent fuel pools – Harris Nuclear Plant, Robinson Nuclear Plant, and Brunswick Nuclear Plant. Each of these fuel movement procedures include the safeguards against fuel assembly misplacement discussed in this paragraph.

19. Misplacement of a single fuel assembly in the Harris spent fuel pools is highly unlikely because of the multiple procedures, independent operations, and independent operators involved in each fuel assembly movement. Based upon a review of Harris plant records, I have determined that no fuel assembly has ever been misplaced in the spent fuel pools at the Harris Nuclear Plant. The history of

fuel movement at Harris and the lack of a single fuel assembly misplacement ever occurring at Harris confirm, through practical experience, that a single misplacement at Harris is highly unlikely.

20. The misplacement of a single fresh fuel assembly in Harris pools C and D is even more unlikely because of the physical separation of pools C and D, relative to the Harris fresh fuel storage area. Fresh fuel at Harris is brought "dry" into the new fuel storage area and then stored in spent fuel pool A prior to moving it into the reactor. Spent fuel pool A is physically separated from pools C and D by approximately 300 feet of the fuel transfer canal. Fresh fuel is not passed through or adjacent to pools C and D as it is moved either from the Harris new fuel storage area to pool A, or from pool A to the reactor core.

21. The misplacement of multiple fuel assemblies in the Harris spent fuel pools is not credible. I am aware of no single failure in administrative or physical controls that could result in multiple fuel assembly misplacements in the Harris spent fuel pools. Because each fuel assembly movement is independent, and a single fuel assembly misplacement is highly unlikely, the likelihood of multiple fuel assembly misplacements is the product of several highly unlikely events, each in succession, and each of which is not detected. Based on Harris operations, operating procedures, and past record, the undetected misplacement of multiple fuel assemblies in the Harris spent fuel pools is not a credible event. The misplacement of multiple fresh fuel assemblies in Harris spent fuel pools C and D is even less credible because of the limited number of fresh fuel assemblies at the

Harris site at any one time. The Harris Nuclear Plant has, in practice, no more than about 52 fresh fuel assemblies on site at any one time. This is because only one-third of the reactor core is loaded with fresh fuel during the refueling operation. The entire Harris reactor core contains 157 fuel assemblies. One-third of the core, or about 52 fuel assemblies, is replaced with fresh fuel approximately every 18 months. New fuel is brought into the plant just prior to a refueling outage so the Harris plant only maintains inventory for one reloading at a given time.

22. Harris Nuclear Plant chemistry procedures require that at least 2000 ppm of soluble boron be maintained in the Harris spent fuel pools, including pools C and D, at all times. This is required pursuant to section M of Attachment 1.2 of Harris operating procedure CRC-001 (SHNPP Environmental and Chemistry Sampling and Analysis Program). A copy of this operating procedure is included as Attachment P to this affidavit. 2000 ppm of soluble boron is required to be maintained in the spent fuel pool water for criticality control of both the reactor core and the spent fuel pools. The water in the Harris Nuclear Plant reactor core also contains soluble boron. During refueling operations, the water in the reactor core and the water in the spent fuel pool are interconnected. The soluble boron level in the spent fuel pool water is maintained at 2000 ppm to ensure that the borated water in the reactor core is not inadvertently diluted with non-borated water. The soluble boron level in the spent fuel pool water also provides an

additional, redundant mechanism for criticality control in the Harris spent fuel pools.

23. Control of the spent fuel pools' soluble boron concentration and control of fuel assembly movements in the pools are completely separate and independent actions. Dilution of soluble boron and fuel assembly misplacement are entirely separate and independent hypothetical events.

24. In CP&L's RAI response serial HNP-99-094, dated June 14, 1999 (Attachment C), CP&L stated that analyses from Holtec International demonstrates that Harris spent fuel pools C and D would remain subcritical in the event of misplacement of a fresh fuel assembly if at least 400 ppm of soluble boron was in the spent fuel pool water. I am aware of no mechanism that could result in the dilution of the soluble boron level in the Harris spent fuel pool water from the required level of 2000 ppm down to 400 ppm.

25. In fact, loss of water from the spent fuel pools, which occurs in small quantities routinely through evaporation, results in the opposite effect – it <u>increases</u> the concentration of soluble boron in the spent fuel pool water. During evaporation, only the water is evaporated out of the pool, the soluble boron stays in the pool. In order for such a soluble boron dilution event to occur that would decrease the soluble boron concentration from 2000 ppm to 400 ppm, not only would eighty percent of the water volume in the spent fuel pools would have to be lost, an amount comprising in excess of a million gallons of water, this enormous quantity of water would <u>also</u> have to be inadvertently replaced with unborated water, in

violation of Harris operating procedures. I know of no credible mechanism to lose this quantity of water from the Harris spent fuel pools. Such a dilution event in the Harris spent fuel pools is not a credible event.

26. Supplemental analysis performed by Holtec International has demonstrated that Harris spent fuel pools C and D would, in fact, remain subcritical following the misplacement of a single fresh fuel assembly even if all of the soluble boron were removed from the spent fuel pools (i.e., 0 ppm). This analysis is included as Attachment B to Exhibit 3, the Affidavit of Everett L. Redmond II, Ph.D. This supplemental analysis, which demonstrates that Harris spent fuel pools C and D will remain subcritical even without soluble boron, renders further discussion of the likelihood of a boron dilution event moot.

BASIS FOR THE 50.55a ALTERNATIVE PLAN

27. The license amendment request includes a 10 C.F.R. §50.55a Alternative Plan (Attachment A, Enclosure 8) because CP&L had committed to construct the Harris SFPCCS piping to meet the requirements of Section III, Class 3 of the ASME Boiler and Pressure Vessel Code ("ASME Code"). Completing the SFPCCS strictly on the basis of the original HNP site construction program is not possible for two reasons: (1) CP&L discontinued its ASME N-Stamp certification program sometime after Harris Plant construction was complete, and (2) certain quality assurance ("QA") records associated with the field installation of SFPCCS piping were inadvertently destroyed along with a purging of the canceled Harris

Unit 2 records. The 50.55a Alternative Plan is intended to provide the basis for construction requirements for the completion of the SFPCCS and to provide reasonable assurance of an acceptable level of quality and safety of the SFPCCS in light of the missing documentation.

28. The first issue addressed in the 50.55a Alternative Plan is the supplemental QA requirements that have been developed and implemented at HNP to augment CP&L's Corporate QA Program (which meets the criteria in 10 C.F.R. Part 50, Appendix B) in order to address construction QA requirements that were part of the Harris ASME Code QA Program during construction. To the extent that it was completed, construction of the SFPCC for spent fuel pools C and D was accomplished in accordance with the ASME Code QA Program which CP&L maintained at that time for Harris Plant construction. Had Unit 2 not been canceled, the construction sequence for the SFPCCS piping would have eventually culminated in its receiving an "N-Stamp", an affirmation that it was constructed in accordance with Code requirements under an ASME approved N Certificate Program. However, since no partial turnover was conducted on the completed portion of the SFPCCS and CP&L's N Certificate program has been long since discontinued, it is not possible to now complete the construction of this system in a manner which could be stamped under Code rules. Similarly, since this construction effort does not fit the definition of repair/replacement activities in ASME Code, Section XI, it cannot be performed as such under the plant's Section XI program.

29. Given these conditions, CP&L decided to proceed with construction of the Harris SFPCCS on the basis of a construction program which retained the elements of the original program to the extent necessary to "provide an acceptable level of quality and safety," consistent with the criteria of 10 C.F.R. §50.55a(a)(3)(i). This was done by completing a detailed comparison of the requirements of the original ASME construction program with those of current plant programs, procedures and processes. Where the original construction program contained elements not adequately or not specifically addressed by current requirements, that item was evaluated for its implication on quality. Wherever necessary, supplemental rules were drafted to augment current plant programs, procedures and processes to meet the wording and / or intent of the original construction program. This set of supplemental rules was formalized into the "Supplemental QA Requirements for Construction", and incorporated into the design packages for completion of the Harris SFPCCS. For example, an Authorized Nuclear Inspector ("ANI") has been engaged through Hartford Steam Boiler Inspection and Insurance Company to provide third-party QA review and inspection during the completion of the SFPCCS, just as an ANI was on-site and provide such third-party oversight during construction. The ANI represents the nuclear insurer and, indirectly, pursuant to ASME Code requirements, the State of North Carolina in his review of work packages prior to field issuance, inspection activities, and final review and approval of QA documentation. A detailed comparison of the ASME Code QA Program and the CP&L Corporate QA Program, as supplemented, is described in

Attachment B, Enclosure 17. However, this part of the Alternative Plan is not a subject of Contention 3.

30. The second part of the Alternative Plan is to address the missing QA documentation for the SFPCCS piping. The Piping Pedigree Plan (as described in Attachment A, Enclosure 8) included a number of detailed reviews and inspections to document the quality of the as-found SFPCCS piping:

(a) The ASME Section III SFPCCS piping was walked down by CP&L engineering and Harris Nuclear Plant Quality Control personnel to compare the plant configuration with construction isometric drawings and ensure that all welds, both vendor and field constructed, were identified.

(b) All of the 44 vendor data packages associated with the ASME Section III portions of the SFPCCS were retrieved and reviewed to ensure that the requisite paperwork was in hand. Vendor records include records of material certification, welding activities, Nondestructive Examination ("NDE"), and hydrostatic testing. Vendor records were retained by the Harris Nuclear Plant Document Control Program and are available on microfilm. Thus, vendor piping and welds are not part of the 50.55a Alternative Plan relating to missing QA documentation. Of the nearly 200 existing large bore (12" and 16") ASME Section III SFPCCS piping welds, approximately 160 are vendor welds for which all required QA records exist.

(c) Other Construction Quality Control documentation was identified and reviewed. Hydrostatic test reports, Deficiency Disposition Reports, and Repair Weld Data Reports provided evidence that the missing QA documentation existed at the time of construction. The results of this review will be discussed in more detail below.

(d) Accessible SFPCCS piping was reinspected and NDE was performed to determine the quality of the welds. Weld Data Reports ("WDR") were generated to document the inspection results. The WDRs were inspected by the Harris QC personnel and the ANI. The inspections and NDE verified the acceptability of the accessible SFPCCS welds and the substitute WDRs address the QA documentation requirements.

(e) The inaccessible SFPCCS piping embedded in concrete includes 15 field welds for which WDRs are no longer available and an external reinspection and NDE is not possible. These 15 welds in six runs of piping were inspected, along with the piping, by remote video camera inspection. The results of the inspections are discussed in more detail below.

(f) All of the accessible SFPCCS piping field welds were subject to examination and/or testing to ascertain the composition of weld filler material. Generally, this was done using a nondestructive x-ray diffraction "alloy analyzer." In addition, chip samples were taken from three welds at random to support the validity of the alloy analyzer results. The results of this effort support that filler material alloy used in the field welds analyzed is consistent with that required by site specifications and welding procedures. See Attachment E. The composition of the filler material in the welds was not raised in Contention 3.

31. The results of the implementation of the 50.55a Alternative Plan demonstrate that the SFPCCS piping, as constructed, met ASME Code requirements applicable to Harris Plant construction and quality assurance and provides an acceptable level of quality and safety, for the following reasons:

(a) The SFPCCS for spent fuel pools C and D was constructed to the same exacting standards pursuant to the same ASME QA Program as was the SFPCCS for spent fuel pools A and B and the rest of Harris Plant. The requirements, processes and procedures which required rigorous inspections and documentation of the quality of the SFPCCS are described in more detail in the Affidavit of David L. Shockley (Exhibit 6) and in CP&L's ASME Quality Assurance Manual (Attachment A to Exhibit 6).

(b) Harris Unit 1 has operated the SFPCCS for spent fuel pools A and B successfully since startup. The installation of piping, welding and concrete placement was accomplished at all four spent fuel pools more or less concurrently, using the same pool of construction personnel, welders, supervisors, engineers, and QA inspectors, and the same group of ANIs. See the discussion of welding of piping during construction in the Affidavit of Charles H. Griffin (Exhibit 5) and the discussion of the QA inspections in the Affidavit of David L. Shockley (Exhibit 6) and the Affidavit of William T. Gilbert (Exhibit 7).

(c) Documentation for field welds joining these pipe spools was contained on Weld Data Reports ("WDRs"), which provided a record of all ASME Code required attributes pertinent to a given weld. Data such as joint and piece identification, filler material identification, weld procedure, welder identification and NDE requirements were all specified and documented on the WDR, and generally the WDR constituted the only permanent documentation for this information. Construction procedures required each WDR to be prepared by weld engineering personnel as part of work package preparation, and to be reviewed by both QA inspectors and the ANI prior to its release to the field. Subsequent to weld performance, each completed WDR would be reviewed again by QA inspectors and the ANI to verify that all requirements were met. WDRs were collected as part of piping isometric packages, which were compiled and stored pending system completion for N-Stamp review. Failure to complete the WDRs for the field welds in the embedded piping would have required a complete breakdown of the Welding Procedures and Processes and the QA Procedures and Processes. As attested to directly by Charles Griffin, David Shockley, and Tommy Gilbert, there was no such breakdown of the ASME Code welding program nor the ASME Code QA Program at the Harris Plant. See Exhibits 5, 6 and 7.

(d) Although direct QA documentation of attributes associated with the SFPCCS piping field welds no longer exists, a great deal of construction era information is available which conclusively supports that the WDRs did exist at the time of construction and were satisfactorily completed. The most direct QA documentation pertaining to this conclusion is found in hydrostatic test ("hydrotest") records for embedded spent fuel pool piping. Procedural requirements for conducting the hydrotest included a review by QA inspectors of all weld documentation associated with the piping being tested. Accordingly, the QA inspector performed a review of the WDR for each field weld within the test boundary, verifying that each WDR was completed, reviewed and approved, including the ANI's review. In addition, the hydrotest procedure required that each field weld be individually inspected for leakage while at test pressure, providing additional assurance as to the completion and quality of these welds. Hydrotest records are on hand for 13 of the 15 embedded field welds, and additional QC documents support the conclusion that the remaining two field welds were also hydrotested. This record set provides verification that WDRs did exist for each of the embedded field welds, that each WDR was fully completed, reviewed and accepted, and therefore, that these field welds were completed in full compliance with ASME Code construction requirements. Notably, several of the QA inspectors actually performing document reviews and hydrotest inspections associated with embedded SFPCCS piping are still employed by CP&L. These individuals readily attest that, to the extent indicated by their signature on the hydrotest records, they positively and personally confirm that the WDRs for eleven of the field welds within the test boundary did exist and were satisfactorily completed, and that each such weld was closely inspected as part of the hydrotest effort. See Affidavit of David L. Shockley (Exhibit 6, at ¶¶ 15, 16) and Affidavit of William T. Gilbert (Exhibit 7, at ¶ 10). They are also confident that the WDRs for the other four welds also were properly prepared and reviewed prior to the hydrotest and reviewed again prior to the concrete pour.

32. Our walk-downs and inspections of the SFPCCS piping and components and the retrospective review of construction procedures and processes, QA documentation, ASME Code QA Manual, and interviews with personnel who were part of the process provide reasonable assurance that the 15 SFPCCS field welds embedded in concrete were completed pursuant to applicable ASME Code requirements and that WDRs were prepared and reviewed by the QA inspectors and ANI. Former QA inspector David Shockley can attest to having personally reviewed 10 of the 15 WDRs, as indicated by his initials on hydrotest reports. (Exhibit 6, at ¶ 13). Former QA inspector Tommy Gilbert can attest to having personally reviewed 7 of the 15 WDRs (six of which were also reviewed by David Shockley). (Exhibit 7, at ¶ 10). Furthermore, the hydrotest reports confirm that 13 of the 15 SFPCCS field welds were visually inspected at test pressure by the QA inspectors. (Exhibit 6, at ¶ 10; Exhibit 7, at ¶ 7.) (Attachments S and T provide the hydrotest reports for the two field welds not included with Exhibit 6 or Exhibit 7.) The evidence is overwhelming that the SFPCCS piping and welds were properly installed and met ASME Code requirements at the time concrete was poured and the piping was embedded in concrete.

RESULTS OF INSPECTIONS OF SFPCCS PIPING

33. An Equipment Commissioning Plan was developed as part of the "Supplemental Quality Assurance Requirements for the Design Change Packages Associated with the Completion of the Units 2 & 3 Spent Fuel Pool Cooling System."

Attachment B, Enclosure 16, § 5.2. The Equipment Commissioning Plan prescribes a set of criteria to ensure that the components and equipment in the SFPCCS will meet the requirements of Appendix B to 10 C.F.R. Part 50 and is capable of performing their intended function in the completed design. The Equipment Commissioning Plan includes physical inspections and testing to verify that the lack of controlled storage conditions and regular maintenance has not caused any condition affecting quality, including damage from personnel, introduction of foreign material, scavenging of parts, corrosion, fouling, aging, or radiation exposure. Any identified deficiencies for Code items will be repaired in accordance with approved procedures pursuant to the ASME Code, Section XI, Repair and Replacement Program.

34. As part of the Equipment Commissioning Plan, a thorough test and inspection effort has also been completed to ascertain the condition of the embedded SFPCCS piping. The tests and inspections included testing of the water in the SFPCCS piping, a complete walk-down and visual inspection of all accessible piping, welds, components and equipment, re-inspection of all accessible welds, testing the weld filler material in the accessible welds, a visual inspection with a high-quality video camera of the segments of the embedded SFPCCS piping with field welds, and taking a sample and testing the composition of a deposit on one of the welds. Any indications observed during the visual inspection of the embedded SFPCCS piping were analyzed and dispositioned. An outside expert, Structural Integrity Analysis, Inc., provided an independent evaluation of the

structural integrity and suitability for service of the embedded SFPCCS piping. The Engineering Service Request, which provides the engineering evaluation of the tests and inspection of the embedded SFPCCS piping, is Attachment Q, which includes a copy of the data sheets for each indication recorded during the visual inspection (attachment 1), the Structural Integrity Associates Report (attachment 2), and a Technical Report prepared by the Dr. Ahmad Moccari at the CP&L Energy and Environmental Center (attachment 3). I was the supervisor responsible for the preparation of Attachment Q and the information contained therein is accurate to the best of my knowledge and belief.

Analysis of the Water in the SFPCCS Piping

35. The scope of this investigation included analysis of lay-up water in the embedded SFPCCS lines. The water, which has been sitting in these lines under extended lay-up conditions, was subject to chemical and microbiological analysis. This test effort determined that the water in these lines was of high purity (consistent with that in the spent fuel pools themselves). Nuisance bacteria capable of causing microbiologically induced corrosion ("MIC") were not detected and in general there were low levels of microbiological activity in the water samples for the SFPCCS piping. The results of this testing indicates a highly unlikely potential for chemically or microbiologically induced corrosion to have occurred during extended lay-up. The report of microbiological testing of the water in the SFPCCS piping is attached to the Affidavit of Dr. Ahmad Moccari. (Exhibit 8,

Attachment C). The results of the testing of the water are summarized in Attachment Q, §§ 3.4.3.1 - 3.4.3.2.

Results of Video-camera Inspection

36. All of the fifteen embedded field welds and associated SFPCCS piping runs were inspected using a high-resolution camera fitted to a pipe crawler. These inspections were conducted in accordance with Special Plant Procedure SPP-0312T, which provided specific acceptance criteria, as well as qualification requirements for the equipment and inspectors. The inspection included welds on six of the eight embedded cooling lines connected to spent fuel pools C and D. The remaining two lines have only approximately 6 feet of embedded pipe each, with no embedded shop or field welds. All of the lines inspected were 12" 304 stainless steel piping.

37. Per the acceptance criteria in SPP-0312T, welds which could be accepted without further evaluation must be completely free of the following indications:

Cracks
Lack of Fusion
Lack of Penetration
Oxidation
Undercut greater than 1/32"
Reinforcement ("Push Through") greater than 1/16"
Concavity (Suck Back") greater than 1/32"
Porosity greater than 1/16"
Inclusions

Generally, the inspection results were very good. It is noted that the welds in question were never subject to volumetric examination by Code, and were sufficiently far from the open end of the pipe at the time of welding that an internal visual examination would not have been performed. Some general discoloration of the weld and portions of the internal surfaces of piping was noted, as well as a number of minor surface indications. Each indication is described on a "Remote Visual Examination Data Sheet," which are included in Attachment Q, attachment 1. While none of the indications posed any threat to the structural integrity of the SFPCCS piping or its suitability for the intended purpose, CP&L performed additional inspections and sampling of surface deposits to identify the source of the deposit and determine if any appreciable corrosion had taken place. CP&L evaluated the impact on structural integrity of incomplete melting of consumable inserts, staining, linear indications, and deposits on the embedded field welds. The evaluations are included in Attachment Q.

Evaluation of Reddish-Brown Deposits

38. Visual inspection of the embedded piping found areas having a reddish-brown film adhering to the piping. This material is very similar in appearance to the iron oxide, which is introduced to the spent fuel pools by way of spent fuel transshipment from CP&L's other nuclear plants. This iron oxide neither results from, contributes to, or is otherwise associated with corrosion or degradation in the SFPCCS piping. Inspection of field weld FW-517 found three locations having a localized deposit of reddish-brown material at the field weld. Samples of this material were removed by fitting the head of the inspection camera with a arm and swab, and using pan and tilt manipulations to collect material directly from

the locations of interest. This material was subject to microbiological testing for the presence of bacteria associated with microbiologically influenced corrosion ("MIC"), as well as chemical analysis to determine its makeup. The results of this effort, also described in the Affidavit of Dr. Ahmad A. Moccari, a scientist specializing in corrosion studies and working for CP&L at its metallurgical laboratories, provided negative results relative to the presence of aggressive bacteria which are associated with MIC. Chemical analysis of this material confirms that it is primarily composed of iron oxide. (See Exhibit 8 and Attachment Q, attachment 3.)

Incompletely Consumed Inserts in the Root Pass of Field Welds

39. The typical field weld joint of the SFPCCS piping incorporated a consumable insert, with the ends of the pipe spools being prepped at the vendor facility for use with this configuration. The purpose of a welding consumable insert is to serve as a consumable retainer and filler metal during completion of a weld joint root pass (first welding pass). By design, the root pass of the weld would consume the insert while fusing both ends of the pipe together. A number of welds had locations where small portions of the insert could be discerned, indicating that it was not fully consumed by the root pass. Generally, these incidences of unconsumed insert were limited to several very small areas where a small portion of the insert could be discerned. The most significant indication of unconsumed insert was observed in field weld FW-516, which exists in the horizontal piping

on the supply line to the "D" SFP. This weld had several locations where a consumable insert had been utilized but was not fully melted by the root pass, including one area about 1.5" long where a continuous portion of the insert could be discerned. Notably, to the extent that could be discerned by closely reviewing multiple camera angles, inspection of these areas of unconsumed insert indicates that these pieces of insert material are completely fused around the edges.

40. Unconsumed inserts are typically the result of welder technique with this particular condition limited to the weld root pass. It is not an unusual condition. Unlike some welding flaws, such as hot cracking and piping porosity, which could possibly extend into subsequent weld layers, once the root pass is completed, subsequent weld passes are unaffected by an unconsumed insert condition. Unconsumed insert materials could typically be detected by visual observation of the pipe inside diameter surface (if accessible) or by conducting volumetric NDE examinations like radiography. However, consistent with ASME Code requirements, the final inspection requirements for these ASME Code Class 3 SFPCCS weld joints were a final visual exam and a liquid/dye penetrant examination of the weld joint outside diameter surface. Therefore the final inspections and NDE for these weld joints would not have detected indications such as these regions of unconsumed insert in the root pass, unless the weld inside diameter surface had been accessible for local visual observation during plant construction. See Affidavit of Charles H. Griffin (Exhibit 5, at ¶ 9).

41. The indications of unconsumed weld insert identified by camera inspection of the embedded field welds were evaluated and determined not to represent a challenge to piping integrity or otherwise affect its suitability for the intended service. The indications were determined to be relatively insignificant imperfections which are to some degree expected on field welds such as FW-516, which was only subject to surface examination and does not lend itself to internal visual examination. ASME Section III, Subsection ND design rules for vessels specifically recognize the potential for imperfections in welds which are not subject to volumetric examination, and provide compensation when necessary by a reduction in joint efficiency based on the type and extent of NDE performed. Although this consideration regarding joint efficiency does not directly apply to the embedded SFPCCS piping, it does demonstrate that the ASME acknowledges that minor imperfections will exist in welds of this nature which are not subject to fill volumetric examination. Based on these considerations and the additional discussion in the Report of Structural Integrity Associates, Inc., pertaining to structural integrity, the indications of incomplete fusion identified on these embedded field welds were deemed acceptable with no rework / repair. <u>See</u> Attachment Q, §3.4.2 and attachment 2.

Small Linear Indication in the Piping Base Metal

42. A small linear indication (approximately ½" long) was observed extending out of the seam weld on the pipe spool above field weld FW-515 and into the counter-bored region adjacent to this weld. This indication did not appear to originate in the field

weld itself, nor did it have the appearance of being corrosion related. The corrosion mechanisms which could possibly cause cracking in the Type 304 Stainless Steel spent fuel pool cooling lines are very unlikely due to a lack of the aggressive conditions (chemistry and temperature) which might initiate them. Further. the line is not exposed to cyclical loading or thermal variations, which might induce fatigue cracking.

43. At this point the specific cause for the linear indication in the seam weld adjacent to field weld FW-515 cannot be conclusively determined. What can be said is that an external visual and liquid penetrant examination was completed of this field weld after its construction, and that the indication of interest would have been identified if it extended to the exterior surface of the piping. Subsequently, this field weld was subjected to and successfully completed hydrostatic testing and additional close visual inspection prior to the concrete pour. These examinations and tests provide conclusive evidence that the crack is not through wall and will not result in leakage. Structural Integrity Associates was asked to provide an expert independent evaluation of the implications of the indication on the structural integrity of the piping. Their conclusion, based on critical flaw size analysis and consideration of the potential mechanisms for crack propagation, is that the indication does not pose any challenge to piping integrity, nor is there any reason to suspect that the indication might propagate beyond its existing condition. <u>See</u> Attachment Q, attachment 2 and Affidavit of George Licina (Exhibit 9).

Overall Condition of the SFPCCS Piping

44. The videotaped inspection allowed an assessment of the overall condition of the embedded SFPCCS piping. The videotapes were reviewed in their entirety by Dr. Ahmad Moccari and George Licina, both experts in materials and corrosion, by Charles Griffin, an expert in materials and welding, and by members of my engineering staff. I reviewed the videotapes in their entirety myself. The video camera was able to take high quality pictures of everything on the inside of the SFPCCS piping – longitudinal welds, circumferential welds, and the piping's inside surfaces. The camera work was very professional. The light clearly illuminated the surfaces examined. Areas of interest were inspected from a number of different angles as the camera moved back and forth over the same surface. I was able to inspect the piping and welds easily. It can readily be observed that the piping was without noticeable construction anomalies such as mismatch or other fit-up problems. There was no evidence of mishandling, such as dents or ovality, or of corrosion which might be evident of contamination or sensitization during handling and construction. Field welds and shop welds were all found to be in the expected location based upon isometric drawings and vendor manufacturing records. The camera inspection confirmed that the quality of construction was good, and provided no evidence to support that the piping was not in compliance with construction requirements. See Affidavit of Ahmad Moccari (Exhibit 8, ¶ 11).

45. The condition of the piping is not surprising because it is constructed of high-quality stainless steel, that is otherwise resistant to corrosion and cracking, and it has been maintained in a wet lay-up condition that is very benign. It has not been subject to extreme temperatures, pressure or other stresses. It would have been quite surprising to observe any degradation in the SFPCCS piping under these conditions. George Licina evaluated all of the possible causes of degradation in stainless steel piping and found that the conditions necessary for degradation of such piping absent from the conditions in the SFPCCS piping. He also noted that the SFPCCS piping was very conservatively designed for its intended operating conditions. The 0.375" wall thickness is approximately 30 times the minimum wall thickness for the actual service pressure; the stainless steel piping has a design rating of 150 psi and will have a maximum service pressure of about 25 psi. See Affidavit of George Licina (Exhibit 9).

46. A significant portion of the SFPCCS piping which connects to the spent fuel pools C and D is accessible, and subject to the same flooded conditions as the embedded piping. Importantly, these accessible portions are also the low points in this piping, and would be where any corrosion problems would be expected to evidence themselves. Since there has been no leakage or degradation identified with regard to this accessible SFPCCS piping, there was no reason to suspect degradation of the embedded SFPCCS piping. Further, since the active SFPCCS piping for spent fuel pools A and B had not shown any evidence of construction inadequacy, there was no reason to suspect any such problem might exist in the

inactive SFPCCS piping for spent fuel pools C and D. For these reasons, I had initially intended only to perform a camera inspection of a sample of the SFPCCS piping and welds as a confirmation of what was observed in the accessible piping, subject to the same construction procedures, inspections, and conditions and same wet lay-up conditions. We inspected all fifteen of the embedded welds and even pressure washed and re-inspected field weld FW-517 with reddish-brown deposits in order to be in a position to answer every question pertaining to the suitability of the SFPCCS piping for the intended purpose.

47. One of the issues raised by Mr. Lochbaum was the theoretical potential for contamination on the outside of the stainless steel SFPCCS piping that could somehow affect the integrity of the piping or welds. CP&L has considered the potential for external contamination and corrosion of embedded spent fuel pool piping, and concluded that degradation in this manner is not credible. This conclusion is based on the following:

    (a) During the time of construction, controls were in place to preclude contamination and sensitization of stainless steel piping, including SFPCCS piping. Based on our review, there are no recorded incidents of a through-wall failure in stainless steel piping attributed to external contamination at the Harris Nuclear Plant.

    (b) The embedded SFPCCS piping runs roughly through the center of 6-foot thick, heavily reinforced concrete walls. Inspection of the outside surfaces of these walls shows no indication of staining, spaulding, or other evidence of chemical attack. Moreover, none of the exposed portions of this piping or adjacent piping runs show any signs of chemical attack.

(c) At its closest incidence, the piping runs approximately 6 feet underneath the operating floor of the fuel handling building. This floor surface is open and visible, and like the wall surfaces shows no indications or evidence of chemical attack.

(d) Given the tooling and effort necessary to penetrate these concrete walls, it is not credible that a saboteur could have accessed embedded portions of SFPCCS piping without being detected. Moreover, given the considerable effort required and the relative lack of safety significance when compared to other portions of the plant, it is completely illogical that an attack would target this piping at any rate.

(e) It is not credible that the concrete itself contaminated the stainless steel piping. Concrete pour records provide documented evidence of the quality of the concrete used in this construction. Moreover, the ability of the natural alkalinity of concrete to produce a protective passivating film on steel surfaces is well documented. Indeed, many applications incorporate concrete lining specifically for that purpose.

48. The inspections show that the SFPCCS piping and welds embedded in concrete are in very good condition, show negligible degradation during the 17 years since construction (approximately 10 of which were in essentially wet lay-up), and have no credible source of contamination that could adversely affect the outside of the SFPCCS piping embedded in concrete. Furthermore, Structural Integrity Associates found that even if some corrosion or imperfections in welds or cracks in the piping did exist, it would have no effect on the structural integrity of the SFPCCS or on its suitability for service. Attachment Q, attachment 2, and Exhibit 9.

Public Health and Safety and the Environment Would Not be Affected by a Leak in the Embedded SFPCCS Piping

49. Finally, even in the highly improbably event that a weld were to fail or a pinhole leak occurred in the SFPCCS piping, there would be no impact on public health or safety, the environment, or plant operations, for the following reasons:

(a) The integrity of the six-foot thick reinforced concrete walls will not be challenged by the low pressures in the embedded spent fuel pool cooling lines.

(b) The most probable form of such a pinhole would have very small entrance and exit holes, consistent with a leakage rate measured in a few drops an hour at the very low pressures associated with this system open to atmosphere. The makeup rate required to compensate for this leakage would be considerably less than that associated with normal evaporation from the pools, and easily within the capabilities of the numerous makeup sources available to the spent fuel pools. Clearly, it is not credible that the occurrence of one or more such pinholes in this embedded piping would pose a liability to either spent fuel pool water level or the availability of spent fuel pool cooling.

(c) This embedded piping is located in interior walls of the Fuel Handling Building, and above open areas of 216' elevation of the Fuel Handling Building. Even assuming that a pinhole did exist in this piping, it would still not have a pathway to the outside environs without first migrating through several feet of concrete. Even then, it would only have traversed into one or more open areas in and adjoining the Fuel Handling Building. Once there, this moisture would simply evaporate on the outside surface of the wall, leaving an easily visible accumulation of boron crystals behind. Depending on the specific location, a variety of mitigation strategies could then be employed to monitor leakage and preclude its introduction into the soil or air outside of the Fuel Handling Building.

(d) In the worst case failure of the SFPCCS weld (the worst case being in the accessible piping where the path of the water would not be impeded by concrete), the level in the spent fuel pools cannot fall below the level of the suction and discharge openings in the pools. Thus, the spent fuel would remain covered with water. The leaking line could then be isolated. Furthermore, there is a fully redundant line for cooling and cleanup of each spent fuel pool. Indeed, calculations were performed which demonstrated that with only one set of SFPCCS lines in service to spent fuel pool C, adequate heat removal was still provided for spent fuel pool D, even with neither of the spent fuel pool D SFPCCS lines in service.

Failure to Approve the 50.55a Alternative Plan Would Impose Undue Hardship on CP&L Without Compensating Increase in Safety

50. Utilization of the embedded portions of the SFPCCS piping embedded in concrete is essential for operation of spent fuel pools C and D. Installing new piping to replace this embedded SFPCCS piping is not feasible. The cooling lines must enter the spent fuel pools at or about the elevation of the existing piping in order to ensure suction on the pools while precluding the potential for pool draindown. There are no existing alternate routes available to replace these embedded lines. The embedded piping was installed, inspected and tested prior to concrete pours for the spent fuel pools. In essence, the pools and the fuel handling building were built around the piping. Constructing new spent fuel pool piping runs would require extensive core bores through the steel reinforced concrete pool walls. As an example of the type of core bore required, the piping for the two D SFPCCS return lines runs under the full length of spent fuel pool C and the Unit 2/Unit 3

transfer canal for a distance of approximately 72.5 ft prior to a 37 ft vertical run in the wall that separates the D spent fuel pool and the transfer canal. Such a core bore through the over 100 feet of steel reinforced concrete is not technically feasible and would clearly constitute unusual difficulty and hardship in terms of effort, cost and time. It would be equally difficult and expensive to bore through the reinforced concrete to re-inspect the outside of the embedded welds. There would be no compensating increase in safety.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 30, 1999.

R Steve Edwards

R. Steven Edwards

Subscribed and sworn to before me
this 30 day of December 1999.

Darlene S. Yarborough

My Commission expires: 2-6-2000

DARLENE S. YARBOROUGH
NOTARY
PUBLIC
HARNETT COUNTY, NC

CP&L

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

James Scarola
Vice President
Harris Nuclear Plant

DEC 23 1998

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

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

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

Dear Sir or Madam:

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

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

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

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

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

Enclosure 5 provides the proposed Technical Specification pages.

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

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

PO

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

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

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

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

Sincerely,

James Scarola

RSE/KWS/kws

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

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

Lisa M. Randall
Notary (Seal)

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

My commission expires: 6-7-2003

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

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

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

BASIS FOR CHANGE REQUEST

BASIS FOR CHANGE REQUEST

Background:

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

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

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

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

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

Proposed Changes:

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

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

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

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

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

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

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

Basis for Change

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion:

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

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

10CFR50.92 EVALUATION
SIGNIFICANT HAZARDS CONSIDERATION

10CFR50.92 EVALUATION

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

Safety Assessment

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

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

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

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

The proposed cooling system modifications for Pools 'C' and 'D' have been designed to ensure that sufficient heat removal capability exists to maintain the temperature in the pools below the design limit. For the initial installation of racks into Pool 'C', the maximum heat load will be limited to 1.0 MBtu/hr consistent with revised Technical

Specification 5.6. In conjunction with the planned implementation of power uprate, additional analyses will be performed and any required system upgrades will be made to ensure the adequacy of the cooling system to dissipate the heat loads associated with the end of plant life. A comprehensive multi-system thermal-hydraulic analysis was performed in support of the plant design change package for the initial rack installation campaign. This analysis facilitates a reduction in this CCW flow requirement currently stated in the FSAR as being a basis for acceptance of postulated post-LOCA consequences. While the analysis methods are technically valid, this lower flowrate has been deemed to constitute an unreviewed safety question and requires NRC review and approval.

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

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

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

Significant Hazards Consideration Determination

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1. Material, mechanical and structural considerations

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

2. Nuclear criticality

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

3. Thermal-hydraulic and pool cooling

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

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

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

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

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

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

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

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

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

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

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

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

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

ENVIRONMENTAL EVALUATION

ENVIRONMENTAL EVALUATION

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

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

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

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

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

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

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

PAGE CHANGE INSTRUCTIONS

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

Enclosure 5 to Serial: HNP-98-188

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

TECHNICAL SPECIFICATION PAGES

5.6 FUEL STORAGE

CRITICALITY

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

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

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

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

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

DRAINAGE

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

CAPACITY

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

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

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

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

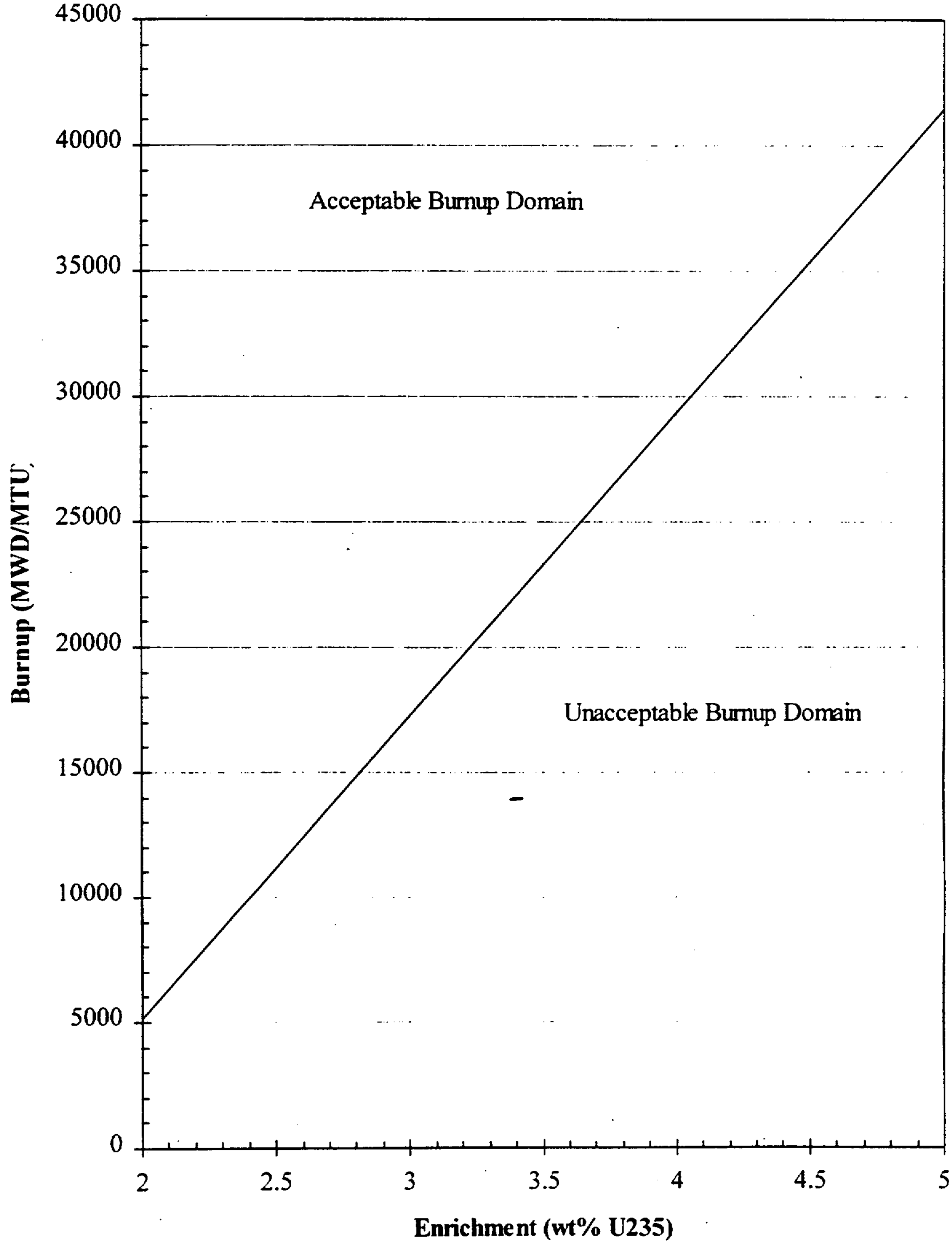


Figure 5.6.1: Burnup Versus Enrichment for PWR Fuel

BEFORE

5.6 FUEL STORAGE

CRITICALITY

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

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

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

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

DRAINAGE

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

CAPACITY

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

Enclosure 6 to Serial: HNP-98-188

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

LICENSING REPORT FOR EXPANDING STORAGE CAPACITY IN HARRIS SPENT FUEL POOLS 'C' AND 'D' (PROPRIETARY VERSION)

See Notebook
Containing Proprietary
Information

January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

EXHIBITS SUPPORTING THE SUMMARY OF FACTS, DATA, AND ARGUMENTS ON WHICH APPLICANT PROPOSES TO RELY AT THE SUBPART K ORAL ARGUMENT

VOLUME 2

EXHIBIT 1 (B – J)

A13

CP&L

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

SERIAL: HNP-99-069

APR 30 1999

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

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOL
COOLING AND CLEANUP SYSTEM PIPING

Dear Sir or Madam:

By letter dated March 24, 1999, the NRC requested additional information regarding the Harris Nuclear Plant (HNP) license amendment request to place spent fuel pools 'C' and 'D' in service. Enclosure 8 of the HNP license amendment request (ref. SERIAL: HNP-98-188, dated December 23, 1998) provided a detailed description of the proposed alternatives to demonstrate compliance with ASME B&PV Code requirements for spent fuel pool cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i). The NRC has determined that additional information is required to complete the review of the proposed alternative piping plan. Enclosed is the HNP response to the NRC request for additional information. The enclosed information is provided as a supplement to our December 23, 1998 submittal and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

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

Sincerely,

Donna B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosures

c:

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

bc: (w/o enclosures)

Mr. K. B. Altman
Mr. G. E. Attarian
Mr. R. H. Bazemore
Mr. S. R. Carr
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
Mr. W. J. Dorman (BNP)
Mr. R. S. Edwards
Mr. R. J. Field
Mr. K. N. Harris

Ms. L. N. Hartz
Mr. W J. Hindman
Mr. C. S. Hinnant
Mr. W. D. Johnson
Mr. G. J. Kline
Ms. W. C. Langston (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. J. H. O'Neill, Jr.
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File
Files: H-X-0511
H-X-0642

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOL COOLING AND CLEANUP SYSTEM PIPING

I. Existing Piping System

A. Detailed description of the proposed change:

Requested Item I.A.1

Provide isometric drawings (isometrics) showing all piping and piping systems within the scope of the proposed alternatives; i.e., for fuel pool cooling and cleanup system (FPCCS) and component cooling water system (CCWS) piping. Provide Isometric drawings to be used for continuance of design and construction without an N-Stamp.

Response to Requested Item I.A.1

Copies of the original construction isometrics are provided in Enclosure 2 and have been marked up to show:

- installed piping (in scope of the Alternative Plan)
- embedded piping
- class boundaries, including safety vs. non-safety related
- location and identification of field welds

In addition, please note that these isometrics include the following information:

- material requirements for piping and fittings
- pipe spool numbers (traceable to vendor data packages)
- location of hanger attachment lug welds

These markups were based upon detailed field walk downs of the current system configuration. Documented verification of these details will be provided by the system turnover / certification process used to implement this activity (ref. responses to RAI items II.2 & 3). Piping outside of Code boundaries is identified on these isometrics only for the purpose of depicting continuity.

Requested Item I.A.2

Provide weld matrixes that list all the welds (each weld should be uniquely identified and traceable to I.A.1 above) within the scope of the alternatives.

Response to Requested Item I.A.2

A matrix is provided in Enclosure 3 for each of the field welds in the scope of the Code related piping discussed in I.A.1. For clarity, in-scope field welds are defined herein as that set of field welds which meet all of the following criteria:

(1) is installed in the ASME Section III Class 3 boundaries of the Component Cooling Water or Spent Fuel Pool Cooling Systems
(2) was installed during original plant construction,
(3) Code required field installation records are no longer available
(4) is consistent with the design of the system as it will be completed
(5) is in the "large-bore" piping on the main system flow path. Instrument lines, vents and drains, branch connections to other systems, etc., are not included.

Requested Item I.A.3

(i) In the matrixes or isometrics, identify the piping material (ASME / ASTM Specification), weld material (ASME / ASTM Specification), the existence of all required material documentation, and any specific missing documentation. (ii) Identify each missing document for each weld. (iii) Identify the method(s) used for reconciliation of each type of missing document. (e.g., missing Certified Material Test Report reconstructed with complete chemical analysis run on shavings taken from the material). (iv) For the sampling and testing methods used for reconciliation, identify references used for guidance. (i.e., NRC DG-1070, ASME, or EPRI). Explain any differences between the sampling / testing methods and the selected referenced guidance. (v) For chemical analysis, identify sample size and chemical analysis (mean and standard deviation for each element) for each analyzing technique.

Response to Requested Item I.A.3

(i) The weld matrix (Enclosure 3) includes a listing of weld material based on a review of applicable Weld Procedure Specifications (WPS) and Weld Data Reports (WDR) for comparable piping. Note that piping material requirements are included in the isometrics provided in response to requested item I.A.1. All Code piping in the scope of the Alternative Plan has been supplied by an NPT Stamp holder and vendor documentation for this material is on hand. This accounts for material certification for all of the piping within the scope of the Alternative Plan and the large majority of the welds in that piping. The outstanding material certification issue to be addressed herein is that associated with welding materials for a relatively small group of field installed welds on the large bore (12" and up) Code piping. During construction, filler metal traceability was accomplished by recording the material heat number on the WDR. The WDR was incorporated into the piping installation package, and typically became the only source of this information to be forwarded to document control. Since the WDRs for these field welds are not on hand, the traceability of filler metal cannot be established.

(ii) The WDR was used to provide the installation record for field welds. Generally, these reports are no longer on hand for the subject welds.

(iii) The WDR contained information pertaining to weld attributes, including identification of the items being welded, specification of the WPS to be used, welder identification, filler metal material identification, NDE requirements, and signature documentation (including that of the ANI) that all required attributes were satisfactorily performed and verified as complete. Reconciliation of missing information is presented in the weld matrix discussed in response to requested item I.B.4.

(iv) The sample size chosen for verifying filler metal composition of accessible (i.e., non-embedded) field welds is 100%. All of the accessible field welds (including welds for hanger lugs) in the large bore stainless steel Spent Fuel Pool Cooling System piping subject to the Alternative Plan have been evaluated for material composition using a Metorex X-Met Alloy Analyzer. Additionally, three of these stainless steel welds have been subject to laboratory analysis of chip samples to verify chemical composition. All three of the large bore carbon steel field welds in the CCW System subject to the Alternative Plan will be evaluated by laboratory analysis of chip samples since the alloy analyzer does not lend itself to reliable evaluation of this material. The use of these specific methods for determination of base metal is provided in the Corporate Welding Manual, Procedure NW-16. Chemical analysis was and will continue to be performed by a reputable and recognized laboratory (NSL Analytical Services, Inc of Cleveland, Ohio for completed analyses) to traceable standards. Since some blending of filler metal and base metal may have occurred with the field welds in question, the results of the filler metal analysis is being evaluated by CP&L's Materials Services Section - Metallurgy Unit (See Enclosure 4 for analysis of SFP field welds).

(v) Relative to physical sample size, Corporate Welding Manual Procedure NW-16 calls for the removal of about 5 grams of material for this type of analysis. The precise weight of the sample taken was not recorded, but was sufficient to facilitate the testing for which results are provided herein. Relative to the number of welds subject to chemical analysis, three of the field welds in the stainless steel Spent Fuel Pool Cooling piping were subject to composition analysis by both the alloy analyzer and chemical analysis of chip samples. Note that the purpose of subjecting these three welds to chemical analysis was not to provide inference to the entire population, but rather to demonstrate consistency with the alloy analyzer. Since the alloy analyzer does not lend itself to reliable composition analysis with carbon steels, all three CCW field welds will also be subject to laboratory analysis for material composition. The accuracy of the chemical analysis method for each element is listed in the laboratory test report. The laboratory analysis report from the three stainless steel samples already completed is included in Enclosure 4.

Requested Item 1.A.4

In the matrixes or on the isometrics, identify inaccessible non-embedded welds and embedded welds (all other welds should be accessible).

Response to Requested Item I.A.4

The isometrics are marked up to show which field welds are embedded and thereby inaccessible (Enclosure 2). All field welds which are not embedded are externally accessible.

Requested Item I.A.5

On the isometrics, indicate the specific location of each weld listed in I.A.2 and identify the boundaries of the systems that are considered safety related. Identify all non-safety related items that appear on the isometrics.

Response to Requested Item I.A.5

The isometrics are marked up accordingly (Enclosure 2).

Requested Item I.A.6

(i) Identify in the matrixes, or on the isometrics, the welds that will be or have been inspected or re-inspected that have Code documentation, welds that have been inspected that do not have Code documentation, and welds that will be or have been inspected or re-inspected not to Code. (ii) For the welds that will be or have been inspected or re-inspected but not to Code, describe the inspection technique, acceptance criteria, and documentation. (iii) Identify the edition and addenda of ASME Code that will be or has been used for the above inspections and re-inspections.

Response to Requested Item I.A.6

(i) Code documentation for welds performed by the piping vendor are included in the vendor data packages. As noted in the Alternative Plan (Enclosure 8 to HNP-98-188, dated 12/23/98), this accounts for approximately 160 of the roughly 200 welds in the large bore Spent Fuel Pool Cooling piping. Based on available evidence, all of the 40 piping field welds and the 12 hanger attachment pad welds were inspected to Code requirements, but generally do not have the Code required documentation available.

Documentation which is on hand for these field welds is listed on the matrix prepared in response to requested item I.A.2. (Enclosure 3).

(ii & iii) The accessible field welds within the scope of the Alternative Plan have been re-inspected using original surface examination criteria from ASME Section III, 1974 - winter 1976 Addenda, ND-5000. A portion of the inaccessible (embedded) field welds will be subjected to internal inspections using a high resolution, remotely operated video camera mounted on a pipe crawler. Details of these camera inspections, including inspection technique and acceptance criteria, are provided in response to requested items III.3 & III.4.

Requested Item I.A.7

Identify any non safety related items installed during the original construction which will be upgraded to safety related status by this amendment; e.g., will any of the non-safety-related ANSI B31.1 piping (Enclosure 8, page 7 of the submittal) be upgraded?

Response to Requested Item I.A.7

No such items installed during original construction will be upgraded for use in a Code application in support of this activity. No B31.1 piping will be upgraded for use in a Code or safety-related application. The turnover of piping and equipment within the scope of this activity will include a review of all Code items and documentation by the ANI to ensure that each item has the appropriate certification.

Requested Item I.A.8

Identify any commercial grade items requiring dedication installed during the original construction. For these items, is documentation of the dedication program available for review? Are the dedication packages for items available for review?

Response to Requested Item I.A.8

No commercial grade items were installed during the original construction which will now be used inside Code boundaries. The turnover of piping and equipment within the scope of this activity will include a review of all Code items and documentation by the ANI to ensure that each item has the appropriate certification.

Requested Item I.A.9

Identify any commercial grade items requiring dedication that will be used to complete construction.

Response to Requested Item I.A.9

No commercial grade items will be dedicated for use in a Code application by this activity. The turnover of piping and equipment within the scope of this activity will include a review of all Code items and documentation by the ANI to ensure that each item has the appropriate certification.

Requested Item I.A.10

(i) Was the piping system constructed in accordance with a 10CFR50 Appendix B Program? (ii) Is the construction Appendix B program documentation available for review? (iii) If construction was performed under a different program, identify the program. Is this program documentation available for review?

Response to Requested Item I.A.10

(i) The overall quality assurance program used by Carolina Power & Light Company for the design and construction of the Harris Nuclear Power Plant is described in the Shearon Harris PSAR. PSAR Section 1.8 states that "The Carolina Power & Light Company Quality Assurance Program for the engineering and construction of the Shearon Harris Nuclear Power Plant (SHNPP), which includes the quality assurance programs for both Ebasco and Westinghouse by reference, is structured with regard to safety-related equipment in accordance with the eighteen criteria of Appendix B to 10CFR50. In addition, the subject Program is structured in accordance with ANSI N45.2 and thereby Regulatory Guide 1.28 . . . ". The PSAR further states that the "Shearon Harris Nuclear Power Plant Quality Assurance Plan" was replaced by the "CP&L Corporate Quality Assurance Program" on April 1, 1974, and provides a cross reference on how the subject plan met the criteria of 10 CFR50 Appendix B.

(ii & iii) Certain aspects of Shearon Harris Nuclear Power Plant construction were subject to QA requirements beyond those outlined in the CP&L Corporate QA Manual. Since CP&L was not only the Owner, but also the constructor, installer, and a fabricator for Code items in the Shearon Harris Nuclear Power Plant, a separate QA Program was developed, reviewed, approved and implemented specifically to obtain the required ASME N, NA, and NPT Certificates of Authorization. ASME Code Section III, Subsection NA-4133.2 requires that an applicant for a Certificate of Authorization develop a QA program and implementing procedure specific to the proposed scope of work, and that "the applicant shall request the Society to review this procedure and Program prior to the issuance of a Certificate of Authorization." For construction of SHNPP, CP&L met this requirement by the formalization of its "ASME Quality Assurance Manual", intended to meet the criteria in Section III, Subsection NA-4100 of the

Code. All Code work by CP&L during the Construction of the Shearon Harris Nuclear Power Plant was performed to the requirements of this QA program manual. A copy of the ASME Quality Assurance Manual is provided in Enclosure 5.

Requested Item I.A.11

(i) Are the work control procedures and hold point sign-off documents from the original construction available for review? (ii) If these documents are required by Code, what documents are missing?

Response to Requested Item I.A.11

(i) Work control procedures and hold point sign-off documents from the construction era are available for review.

(ii) With the exception of the aforementioned WDRs and associated weld process control issues (including NDE) discussed in response to item I.B.4, CP&L has not identified any missing documents requiring consideration under the Alternative Plan.

Requested Item I.A.12

(i) Provide a list of qualified weld procedure specifications (WPS) used, and their procedure qualification records (PQRs). (ii) For welds missing welder identification, how will weld integrity be established.

Response to Requested Item I.A.12

(i) The welding procedures available for welding during the original construction of the piping in question were identified based on a review of available WPS in the welding manual at that time. A copy of these WPS and their PQRs are provided in Enclosure 6.

(ii) CP&L has located welder identification markings at each accessible field weld in the scope of the Alternative Plan. These Code required welder symbols can be traced back to the welder responsible for each such weld, and from there, qualification records on file can be used to establish that each welder was appropriately qualified.

These markings are not accessible on embedded welds. However, alternate QC records have been located which identify the welders for three of these fifteen welds, and numerous programmatic and procedural assurances existed to ensure that welds were made using qualified welders and weld procedures. For embedded welds, internal camera inspections (as described in response to RAI Items III.2, 3 & 4) will be used to augment programmatic and procedural assurances relative to the quality of these welds.

In addition, since the Spent Fuel Pool Cooling piping nozzles exit into the pools below the water level, the portions of the Spent Fuel Pool Cooling System piping attached to the spent fuel pools (including the embedded piping) are flooded as well. Beyond internal camera inspections, water chemistry in these legs of piping will be analyzed to ensure that Microbiologically Induced Corrosion or other corrosion mechanisms have not resulted in degradation of the integrity of field welds or piping.

B. Applicable Regulations for Welds and Piping Systems Within the Scope of the Proposed Alternatives

Requested Item I.B.1

1. Identify the edition and addenda of Code and any Code cases that were used for original construction of the welds and piping systems. If not the same for all the welds, identify the Code requirements for each weld or group of welds.

Response to Requested Item I.B.1

Piping was installed to ASME Section III, 1974 Edition, Winter 1976 Addenda. The PSAR and current FSAR provide the CP&L position on conformance to the requirements of Reg. Guides 1.84 and 1.85 relative to use of Code cases. A review of the N-5 Code Data Report associated with turnover of Unit 1 SFP piping identifies two Code cases used at some point in its construction; it is reasonable to assume that these same Code cases may have been used on the corresponding Unit 2 piping and equipment. These Code cases are:

N-240	"Hydrostatic Testing of Open Ended Piping, Section III, Division 1"
N-275	"Repair of Welds, Section III, Division 1"

Likewise, a review of the Unit 1 CCW N-5 Code Data Report shows these Code cases in association with its construction:

N-275	"Repair of Welds, Section III, Division 1"
N-224	"Use of ASTM A500 Gr. B and ASTM A501 Structural Tubing for Section III, Class 2, 3 and MC"
N-224-1	"Use of ASTM A500 Gr. B and ASTM A501 Structural Tubing for Section III, Class 2, 3 and MC"
N-282	"Nameplates for Valves, Section III, Division 1, Class 1, 2 and 3 Construction"
N-127	"Alternative Rules for Examination of Welds in Piping, Section III, Class 1 and 2 Construction"

Requested Item I.B.2

Identify the edition and addenda of Code and code cases that will be used to complete construction of the piping systems. Identify any exceptions to Code requirements and justifications for these exceptions.

Response to Requested Item I.B.2

Construction will be completed to ASME Section III, 1974 Ed, Winter 1976 Addenda. Code Case N-240 will be used to exempt formal requirements for hydro testing of the embedded piping connected to the atmospheric spent fuel pools due to the lack of accessibility. The need to invoke other specific Code cases has not been identified. Use of any such Code case would be consistent with CP&L's position regarding conformance with Reg. Guides 1.84 and 1.85. Relative to exceptions to Code requirements, CP&L does not take any such exceptions beyond those specifically identified and addressed by this Alternative Plan.

Requested Item I.B.3

Identify the edition and addenda of Code and code cases that were or will be used for repair and replacement of welds and piping.

Response to Requested Item I.B.3

No repair or replacement activities have been performed on the Code piping subject to the Alternative Plan. Future repair and replacement activities (after completion of construction and turnover) will be governed by the site Section XI Repair and Replacement program.

Requested Item I.B.4

Provide a matrix (See I.A.2) that identifies the specific paragraph in Code that is applicable to missing weld documents. Identify documentation deficiencies for each weld. Identify any exceptions to Code requirements. Provide alternatives and justifications for these exceptions.

Response to Requested Item I.B.4

A matrix has been provided in Enclosure 7 for Code requirements pertaining to missing weld documents. Additional information relative to specific welds is provided in Enclosure 3. Alternatives and justifications are identified in Enclosure 2 and discussed elsewhere in the Alternative Plan and this RAI response.

Requested Item I.B.5

Identify the ASME requirements, including administrative requirements, that were completed prior to stoppage of the original construction of the piping systems. Is documentation of these completed requirements available for review? What ASME data reports were filed and what were their filing dates?

Response to Requested Item I.B.5

None of the piping or equipment in question had completed the system certification process and received an N-Stamp. Generally, requirements which were met are consistent with the status of construction at the time work was halted. For instance, embedded piping had been installed, inspected and tested prior to pouring concrete, but accessible piping immediately adjacent was still under construction. The availability of records for the construction varies. Generally, records generated by site construction during the installation of the subject piping is not on hand. However, records generated as a result of QC oversight (NCRs, DDRs, audits, etc) are on hand and retrievable. Notably, hydro test records are also generally available for that portion of construction that proceeded to the extent of hydro testing, including embedded Spent Fuel Pool Cooling System piping. Hydro test documentation, including verification of weld documentation, is available for all but 2 of the 15 embedded field welds. The remaining 2 are included in the liner leak test boundary and would have been procedurally required to be verified as complete, but were not specifically included in the leak test as inspection items. (See Enclosure 3 for identification of records available, and Enclosure 8 for the hydro test records specifically discussed herein.) No partial data reports were filed on the subject piping systems. Manufacturer's Code data reports from NPT suppliers are available in document control for the subject piping, as are warehouse receipt inspection records. These records will be subject to review by the ANI as part of the system turnover process.

Requested Item I.B.6

Identify ASME survey inspections conducted prior to stoppage of the original construction of the piping systems. Provide documentation for representative internal / external audits conducted during the peak construction periods for the welds in question (1978 - 1979), particularly in the areas of work control, welding, material traceability and records.

Response to Requested Item I.B.6

There are no documented ASME survey inspections on hand specific to the construction of the piping systems in question. There were, of course, ASME surveys associated with CP&L obtaining and maintaining its N, NA and NPT Certificates of Authorization. This was originally accomplished by an interim letter of authorization in July, 1978 allowing CP&L to commence Code work. A follow up survey on the effectiveness of the program was conducted in July of the following year, with additional audits occurring in 1982 and 1985, in accordance with Code requirements.

Information pertaining to audits and inspections performed by parties other than the ASME is provided in response to requested item I.B.7, below. Also, note that the majority of construction for the welds in question occurred during the '81 - '83 time frame, as attested to by QC records and other documents associated with this construction.

Requested Item I.B.7

Identify third party inspections conducted prior to stoppage of the original construction of the piping systems. Provide a representative sample of documentation for these inspections.

Response to Requested Item I.B.7

A number of ANI inspections specifically associated with the construction of the Unit 2 & 3 SFP Cooling piping are documented in the form of QA surveillance records, hydro test records and other types of records which would have been subject to ANI review. Generally, the ANI inspection records which cannot be retrieved are those associated with WDRs and pipe spool packages. Records for which ANI inspections / reviews are documented are identified in Enclosure 3.

In addition, Corporate QA / QC, which operated independently of the site construction program, provided both quality inspections of work activities and audits on construction activities. Records for which QC inspections are documented are identified in Enclosure 3, and representative samples of QA audits of the construction program are provided in Enclosure 9. Finally, the NRC performed regular inspections of construction activities, with follow-up activities being initiated as needed for issues identified and tracked to satisfactory closure.

Requested Item I.B.8

With regard to piping system components / services performed by others, provide documented validations of these vendors services. Provide the documentation of the audits of the supplier of prefabricated piping.

Response to Requested Item I.B.8

A review has been conducted which identifies that Code data reports are on hand for pipe spools and components inside Code boundaries. The turnover process for completion and activation of this portion of the plant will include a review of these documents by the ANI. CP&L intends to replace any piping or equipment provided by an outside supplier for which appropriate Code records cannot be located. Audit records of the supplier of prefabricated piping and a representative sample of a piping vendor data package are included in Enclosure 10.

II. Completion of Piping System (General)

Requested Item II.1

(i) Identify the differences between HNP's proposed construction program to complete the SFP C and D and the original construction program under HNP's N certificate. (ii) How will these differences be reconciled?

Response to Requested Item II.1

(i) CP&L proposes to complete construction per the design requirements of the original construction Code. CP&L is requesting that exception be allowed under 10CFR50.55a.(a)(3)(i) to certain QA requirements generally found in Section III, Subsection NA and associated with having certificates of authorization for construction and installation of Code items, and to requirements regarding N-Stamping of the completed systems.

(ii) CP&L proposes to reconcile the differences between the original program and the program to be used for completion by providing comparable assurances, tests, inspections and reviews as needed to assure an acceptable level of quality and safety in accordance with 10CFR50.55a.(a)(3)(i). It is CP&L's intention to complete construction using the current Corporate Appendix B QA Program, augmented by supplemental QA requirements to ensure that the intent of Code requirements are adequately addressed. (See response to requested items III.14, 15 & 16).

Requested Item II.2

Will data packages be prepared?

Response to Requested Item II.2

Yes. CP&L is implementing a turnover plan which closely emulates that associated with the N-Stamping process, including preparation of Section III style data packages and third party (ANI) review.

Requested Item II.3

What third party verification is planned?

Response to Requested Item II.3

The Hartford Steam Boiler Insurance and Inspection Co. has been in discussions with CP&L throughout the development of the Alternative Plan. The role that Hartford will play in the certification / turnover process is very similar to that which would be followed in an N-stamping process. It is intended that the ANI will review work packages, participate in field inspections, participate in resolution of field discrepancies and non-conformances, and conduct a final review and certification process much like that done for the preparation of an N-5 data report for each affected system within Code boundaries. Details of this process are contained in a set of "Supplemental QA Requirements" developed for this activity (See response to III.14). A copy of the generic data report to be used for installation of Code items is provided in Enclosure 11.

III. Specific Comments on Submitted Information

Requested Item III.1

(i) What was the basis for selecting the four externally accessible field welds for internal examination? (ii) Identify these welds in the matrix provided in response to I.A.2 above.

Response to Requested Item III.1

(i) Field welds were generally used to join long sections of prefabricated piping, and so were (are) not typically accessible for internal examination with the naked eye. The four field welds in question join the strainer nozzles to the piping, and were identified by a field walk down as being those field welds which could be accessed without specialized pipe crawling / camera-equipment. One of these welds is only a few feet away from an open pipe end, lending itself well to visual examination with the assistance of an examination mirror. The other three field welds were subject to a more limited inspection by inserting a boroscope through nearby pressure taps. Note that a more detailed internal examination of these welds will be performed and formally documented when the strainers are disassembled, using the same internal inspection criteria as developed for the remote camera inspection discussed in III.2, 3, 4 & 5 below.
(ii) These welds are identified on the matrix (Enclosure 2) as 2SF-37-FW-441, 2SF-36-FW-449, 2-SF- 36-FW-450 & 2-SF-38-FW-451 .

Requested Item III.2

With reference to the "substantial portion of the embedded piping and field welds", identify these welds in the matrix provided in response I.A.2

Response to Requested Item III.2

These welds have been identified on Enclosure 3 as requested.

Requested Item III.3

Provide a summary of the inspection procedure used for remote inspection of embedded welds.

Response to Requested Item III.3

The procedure will use a pipe crawler mounted camera to perform a detailed inspection of the interior surfaces of embedded field welds. The procedure will include demonstration of camera resolution capability to at least 1/32" wire, and performance demonstration of inspector's ability to discern and disposition flaws of the nature which might be expected to be encountered. The inspection procedure will be developed and approved by a Level III inspector under the Corporate NDE Program. Inspections will be performed by an appropriately qualified Level II inspector.

Requested Item III.4

With reference to the remote inspection of the embedded welds, identify the critical characteristics that will be verified and the acceptance criteria to be used.

Response to Requested Item III.4

The inspection will specifically include examination of field welds for the following:

No cracks
No lack of Fusion (LOF)
No lack of Penetration (LOP)
No oxidation ("Sugaring")
No undercut greater than 1/32 inch
No reinforcement ("Push Through") greater than 1/16 inch
No Concavity ("Suck Back") greater than 1/32 inch
No porosity greater than 1/16 inch
No inclusions

Generalized inspections will be performed on the piping interior for indications of arc strikes, foreign material, high / low, mishandling indications, etc,. Any such indications shall be noted and characterized during the inspection and evaluated by Engineering if necessary.

In addition, since the Spent Fuel Pool Cooling piping nozzles exit into the pools below the water level, the portions of the Spent Fuel Pool Cooling System piping attached to the spent fuel pools (including the embedded piping) are flooded as well. The inspection procedure will also include criteria and instructions to conclusively ascertain if

Microbiologically Induced Corrosion or other corrosion mechanisms have resulted in degradation of this piping.

Data Recording - The following information will be recorded for each inspection:

1. The inspection will be recorded on videotape in a manner which will facilitate future review and evaluation.

2. Indication location (circumferential, side of weld, etc.), length, and depth (where applicable) shall be documented and recorded on tape.

References - The following references were used to establish this criteria:

ASME Section III, ND-4424 Winter 76 Addenda
ANSI B31.1 Paragraph 136.4.2, 1980 Edition
Corporate Welding Manual NGGM-PM-0003, NW-02, NW-06

Requested Item III.5

Provide results of remote inspection with any identified discrepancies

Response to Requested Item III.5

Camera inspections are currently planned for late May or early June of 1999. Results will be provided upon completion of this activity.

Requested Item III.6

Provide a completed weld data report, representative of those that were discarded. Identify the critical characteristics and explain how, in lieu of records, each will be validated.

Response to Requested Item III.6

A sample WDR is provided in Enclosure 12. Note that this is a WDR for one of the 15 embedded field welds, extracted from a DDR (Deficiency Disposition Report) in which a QA inspector questioned the identity of the adjacent pipe spool. Code required attributes recorded on the WDR are identified and reconciled in Enclosure 6.

Requested Item III.7

With reference to the procurement specification (SS-021, Purchasing Welding Materials for Permanent Plant Construction), did other specifications for other filler materials exist?

What assurances are provided that these other filler materials were not used for the embedded piping.

Response to Requested Item III.7

SS-021 is the site spec for procurement of filler material used in the SHNPP Construction Program and referenced in the Work Procedures which implemented this program. SS-021 is the specification for filler material specifically invoked by Code work procedures; no substitutes were identified or allowed. Research has not identified any other specification for this purpose in association with construction of SHNPP. Being a fairly new plant, CP&L still employs many of the weld engineers and craft personnel associated with the original construction effort. Numerous interviews of these personnel consistently provide the same conclusion; that filler material purchased by CP&L for use in Code work in construction of SHNPP was procured to this specification.

Requested Item III.8

Provide any updates / supplements to the Alternative Plan as they become available.

Response to Requested Item III.8

These will be provided as requested.

Requested Item III.9

With reference to the "large percentage of embedded field welds" that will be inspected, identify these welds on the matrix provided. Provide technical justification for not inspecting the remaining welds.

Response to Requested Item III.9

The matrix has been marked up as requested. The "large percentage of embedded field welds" referred to are those which CP&L has a high level of confidence can be accessed with available pipe crawling equipment based on a walk down with the vendor for pipe crawler / camera services. The enclosed weld matrix (Enclosure 3) specifically identifies the base scope of field welds which are targeted for inspection. Currently, 6 of the 15 embedded field welds are included, which notably includes both of the field welds for which hydro test records are not available.

Assurance of quality for any embedded field welds which are not subject to remote camera inspection is provided by conformance to the requirements of QA Program(s) and implementation procedures which existed at the time of construction along with the body of evidence which directly support adherence to those requirements. This evidence includes: uniform application of QA requirements for the entire site construction program, (including the completed and licensed Unit 1 facility), surveys, inspections, and audits verifying the effectiveness of QA program requirements, construction records which are on hand that attest to quality of construction, and re-performance of Code required inspections on accessible field welds in these same lines with no rejectable indications identified.

Requested Item III.10

(i) Explain what is meant by the statement that internal examination of the embedded welds provides a measure of quality assurance beyond Code requirements. (ii) What additional physical or material attributes will be verified?

Response to Requested Item III.10

(i) This statement is simply intended to identify that many of these welds would have been inaccessible for routine internal inspection at the time of construction (due to distance from an open pipe end), and since no Code requirements existed to do so, would not have been subject to an internal visual examination. Given this, internal camera inspections represent an activity above and beyond that which would have been required under the original construction program.
(ii) See response to requested items III.3 & 4.

Requested Item III.11

The submittal refers to opinions by Bechtel and Hartford concerning the benefits in accordance with an N certificate program. Are these opinions documented and available for review?

Response to Requested Item III.11

Hartford's endorsement of the Alternative Plan is provided in Enclosure 13. Note that this letter is authored by Dr. Richard E. Feigel, Vice President of Hartford Steam Boiler Inspection and Insurance Co. and Chairman of the ASME Council on Codes and Standards. Bechtel's endorsement of this plan is implicit in that they, as the design A/E, have fully reviewed and incorporated the Alternative Plan into the design change packages for this activity.

Requested Item III.12

Provide a copy of the site ASME Section III QA program used during original construction.

Response to Requested Item III.12

A copy of the ASME Section III QA Program manual is provided in Enclosure 5.

Requested Item III.13

(i) Provide a copy of the Corporate QA program that will be used to complete construction. (ii) (Provide a list of implementing quality control procedures for welder qualification, weld procedures, inspections, documentation, etc).

Response to Requested Item III.13

(i) A copy of the current Corporate QA Program Manual is provided in Enclosure 14. Note that this program manual is used with FSAR Section 17 to define the overall corporate QA program.
(ii) All welding will be accomplished in accordance with the Corporate Welding Manual, which conforms to the requirements of Section IX with regard to welder qualification, weld procedures and process control. NDE will be performed in accordance with the Corporate NDE Manual. The site Mechanical Modification Procedures (MMPs) are those procedures which will primarily be used to control work control processes. The list of MMPs most applicable to this activity and the index from the Corporate Welding and NDE Manuals are provided in Enclosure 15.

Requested Item III.14

Provide a copy of the supplemental quality assurance requirements developed to augment the Corporate QA Program, which was based on a review of the approved Construction QA Program at the time of construction versus the existing Corporate QA Program.

Response to Requested Item III.14

Supplemental QA Requirements are provided in Enclosure 16.

Requested Item III.15

Provide documentation of the referenced comparison of approved ASME Section III Construction QA Program Manual with the effective Corporate 10CFR50 Appendix B QA Program.

Response to Requested Item III.15

Documentation of the referenced comparison is provided in Enclosure 17.

Requested Item III.16

Provide documentation of the supplemental quality assurance requirements that have been developed specifically for the purpose of addressing differences between ASME Section III quality assurance requirements and the Corporate 10CFR50 Appendix B QA Program.

Response to Requested Item III.16

The ASME Section III QA Manual discussed in response to requested items III.14 and III.15 above is the document which was reviewed by the ASME and singularly credited for assuring compliance with Section III requirements in order to authorize CP&L to perform N, NA and NPT stamp activities. The overall corporate QA program may have shared procedures, facilities, etc. with this program, but was not directly relied upon to assure compliance with Section III during the construction effort. Given this, the Supplemental QA Requirements provided in response to requested item III.14 and the QA manual comparison provided in response to item requested item III.15 provide the documentation requested in this item as well.

Matrix of Construction Records Pertaining to Units 2 & 3 Spent Fuel Pool Cooling System

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-1-FW-1	Embedded	No, but assured by piping installation, hydro and concrete pour procedure requirements	No, but review of "Weld Documentation" contained in Hydro test record	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records; program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements.	Yes (ref. DDR 1347)	Hydro -QC & ANI (ref. DDR 1347)
2-SF-1-FW-2	Embedded	No, but assured by piping installation, hydro and concrete pour procedure requirements	No, but review of "Weld Documentation" contained in Hydro test record	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records; program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements.	Yes (ref. DDR 1347)	Hydro -QC & ANI (ref. DDR 1347)
2-SF-1-FW-3	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes B61, G97	Yes, by re-inspection.	No	
2-SF-1-FW-4	Embedded	Yes, on one side (ref. DDR-1347). Also assured by piping installation, hydro and concrete pour procedure requirements	No, but review of "Weld Documentation" contained in Hydro test record	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records; program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements.	Yes (ref. DDR 1347)	Hydro -QC & ANI (ref. DDR 1347)

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-1-FW-5	Embedded	Yes, WDR on hand (ref. DDR-1347) Also assured by piping installation, hydro and concrete pour procedure requirements	Yes, WDR on hand (ref. DDR-1347)	■ SFA 5.9/5.4 ■ Yes, WDR on hand in DDR-1347 ■ Not required	Yes, in weld documentation (ref. DDR-1347)	Yes, attested to in weld documentation.	Yes (ref. DDR 1347)	Hydro -QC & ANI (ref. DDR 1347) weld documentation - QC & ANI (ref. DDR 1347) DDR-1347 - QC & ANI
2-SF-1-FW-6	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D41	Yes, by re-inspection.	No	
2-SF-8-FW-65	Embedded	Yes, on one side by DDR-1387. Also assured by piping installation, concrete pour procedure requirements	No	■ SFA 5.9/5.4 ■ No ■ Assured by site specification SS-021, procedural requirements	No, but assured by weld program and procedural requirements	No, but assured by program and procedural requirements. Will be subject to internal camera inspection	No	liner leak test - QC
2-SF-8-FW-66	Embedded	Yes, on one side by DDR-1387 Also assured by piping installation, concrete pour procedure requirements	No	■ SFA 5.9/5.4 ■ No ■ Assured by site specification SS-021, procedural requirements	No, but assured by weld program and procedural requirements	No, but assured by program and procedural requirements. Will be subject to internal camera inspection	No	liner leak test - QC

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-70-FW-325	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes B7, D41	Yes by re-inspection	No, will be hydro-tested by Mod	
2-SF-72-FW-326	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21	Yes by re-inspection	No, will be hydro-tested by Mod	
2-SF-72-FW-327	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21, C97	Yes by re-inspection	No, will be hydro-tested by Mod	
2-SF-69-FW-328	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes B61	Yes by re-inspection	No, will be hydro-tested by Mod	
2-SF-69-FW-329	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes B61	Yes by re-inspection	No, will be hydro-tested by Mod	

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-71-FW-329	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer, chemical analysis	Yes B7	Yes by re-inspection	No, will be hydro tested by Mod	
2-SF-30-FW-381	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21, C97	Yes by re-inspection	No	
2-SF-148-FW-382	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21	Yes, by re-inspection.	No	
2-SF-149-FW-408	Embedded	Yes (ref. DDR-829). Also assured by piping installation, hydro and concrete pour procedure requirements	No, but have repair WDR on hand, also weld documentation review signoff in North New Fuel Pool Hydro Record,	■ SFA 5.9/5.4 ■ Partial, WDR on hand for repair weld. ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements.	B61 (NCR W-207) Also repair WDR on hand (DDR 829)	Yes, (full LP in DDR 829). Will be subject to internal camera inspection	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI Repair weld documentation - QC & ANI NDE rpt. - QC & ANI (ref. DDR-829) DDR-829 - QC & ANI

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-150-FW-412	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21	Yes, by re-inspection.	No	
2-SF-14-FW-424	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes B61	Yes by re-inspection	No, will be hydro tested by Mod	
2-SF-31-FW-426	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21, C97	Yes by re-inspection	No	
2-SF-35-FW-440	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21	Yes by re-inspection	No	NDE rpt. - QC & ANI (ref. NCR WP-016)
2-SF-37-FW-441	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E59, F60, E25, B47	Yes, by re-inspection. Will also be subject to direct internal examination when adjacent strainer is disassembled.	No, will be hydro tested by Mod	
2-SF-16-FW-447	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes B61	Yes by re-inspection	No, will be hydro tested by Mod	DDR-895 - QC & ANI

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt. Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-36-FW-448 * See Note	Yes *	*	*	*	*	*	*	*
2-SF-36-FW-449	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D41, B47	Yes, by re-inspection. Will also be subject to direct internal examination when adjacent strainer is disassembled.	No, will be hydro tested by Mod	
2-SF-36-FW-450	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer & chemical analysis	Yes B47	Yes, by re-inspection. Will also be subject to direct internal examination when adjacent strainer is disassembled.	No, will be hydro tested by Mod	
2-SF-38-FW-451	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer & chemical analysis	Yes G1, B7	Yes, by re-inspection. Will also be subject to direct internal examination when adjacent strainer is disassembled.	No, will be hydro tested by Mod	
2-SF-67-FW-452	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E21	Yes, by re-inspection.	No	
2-SF-68-FW-454	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes C97, D87	Yes, by re-inspection.	No	

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-143-FW-512	Embedded	Assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record,	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements.	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI
2-SF-143-FW-513	Embedded	Assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record,	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements.	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI
2-SF-143-FW-514	Embedded	Yes, ref. DDR-888 Also assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record,	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements Will be subject to internal camera inspection	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI DDR-888

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt. Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-144-FW-515	Embedded	Assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record,	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements. Will be subject to internal camera inspection	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI
2-SF-144-FW-516	Embedded	Yes, on one side, ref. DDR-869. Also assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record,	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements. Will be subject to internal camera inspection	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI DDR-921 - QC & ANI
2-SF-144-FW-517	Embedded	Yes - ref DDR-869. Also assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record.	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	Yes, partial UT & LP performed under DDR-869. Also, assured by weld documentation review in hydro records, program and procedural requirements. Will be subject to internal camera inspection	Yes, see North New Fuel Pool Hydro Test Record. Also ref. DDR-869.	Hydro -QC & ANI Repair weld documentation - QC & ANI (ref. DDR-869) DDR-869 - QC & ANI

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-159-FW-518	Embedded	Also assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	No, but assured by weld doc review in hydro records, program and procedural requirements	No, but assured by weld documentation review in hydro records, program and procedural requirements	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI
2-SF-159-FW-519	Embedded	Yes, on one side (ref. NCR-85-1318, Also assured by piping installation, hydro and concrete pour procedure requirements	No, but WDR review signoff in North New Fuel Pool Hydro Record	■ SFA 5.9/5.4 ■ No ■ Assured by weld doc review in hydro records, site specification SS-021, procedural requirements	Yes, C-20 (see NCR W-103)	No, but assured by weld documentation review in hydro records, program and procedural requirements	Yes, see North New Fuel Pool Hydro Test Record.	Hydro -QC & ANI NCR W-103 - QC
2-SF-71-FW-331 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D75, E50, D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-332 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E50, D69	Yes by re-inspection	N/A (Hanger attachment weld)	

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-71-FW-333 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-334 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D75, E50, D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-335 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-336 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-337 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes E50, D69	Yes by re-inspection	N/A (Hanger attachment weld)	

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt. Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-SF-71-FW-338 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-339 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D75, D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-340 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D75, E50, D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-341 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D75, E50, D69	Yes by re-inspection	N/A (Hanger attachment weld)	
2-SF-71-FW-342 (Hanger Attach. Weld)	Yes	Yes, by inspection	No	■ SFA 5.9/5.4 ■ No ■ by alloy analyzer	Yes D75, D69	Yes by re-inspection	N/A (Hanger attachment weld)	

Field Weld No.	Access?	Id of welded items ?	Weld documents available?	Specified Filler Material ID/ Documentation on hand/ Alt.Verification Method	Qual Record & Welder ID	NDE Records	Hydrotest Records	Inspections completed (ANI / QC)
2-CC-3-FW-207	Yes	No, but is accessible and will be visually verified.	No	■ SFA 5.18/5.1 ■ No ■ TBD by chemical analysis	Yes, K40	Yes by re-inspection	No, to be hydro tested by Mod	
2-CC-3-FW-208	Yes	No, but is accessible and will be visually verified	No	■ SFA 5.18/5.1 ■ No ■ TBD by chemical analysis	Yes, C11	Yes by re-inspection	No, to be hydro tested by Mod	
2-CC-3-FW-209	Yes	No, but is accessible and will be visually verified	No	■ SFA 5.18/5.1 ■ No ■ TBD by chemical analysis	Yes, B1	Yes by re-inspection	No, to be hydro tested by Mod	

Note * Field Weld 2-SF-FW-36-448 is a completed and stamped field weld, but will be cut out and replaced as it joins a section of piping which was affected by a pipe spool modification.

Enclosure 4 to Serial: HNP-99-069

Metallurgy Unit Report for
Spent Fuel Pool Weld Metal Composition Analysis

CAROLINA POWER & LIGHT COMPANY
MATERIALS SERVICES SECTION
METALLURGY SERVICES

TECHNICAL REPORT

To: Mr. Jeff Lane

Project Number: 98-125
Date: June 30, 1998

Investigators:
D. W. Brinkley, III

Reviewed by:
J. W. Wood

Distribution:
File/Metallurgy Services
Andy Bartrom/HNP
Emory Upchurch/HNP

Approved by: R. J. Block
Supervisor, Metallurgy Services

SUBJECT: Harris Nuclear Plant - Material Identification of Spent Fuel Piping Welds

Project Summary

The Unit 3/4 spent fuel piping field welds analyzed at the Harris Nuclear Plant with a Metorex X-Met 880 Alloy Analyzer were identified as being most similar in composition to either Type 304 stainless steel, Type 309 stainless steel, NIST 1154a SRM, or a combination of these reference materials. These results were confirmed by chemical analysis of chip samples from three different welds by an outside laboratory.

INTRODUCTION:

The objective of this investigation was to perform material identifications of field welds made on Unit 3/4 spent fuel piping at the Harris Nuclear Plant. It was reported that plant personnel wished to upgrade this system for possible future use. It was requested that the welds be analyzed nondestructively. Chips would be removed from two or three welds by Harris Nuclear Plant personnel for a more detailed chemical analysis.

FIELD EXAMINATION AND RESULTS:

The selected welds were identified by Mr. Andy Bartrom of the Harris Nuclear Plant Quality Control Unit. The welds had been prepared for a nondestructive evaluation by plant personnel. Field analysis of the welds was performed using a Metorex X-Met 880 Alloy Analyzer (Serial Number 69871) with a cadmium 109 isotope source (Serial Number 1256LY). The alloy analyzer was used in an identification mode and several standard reference materials had been entered into the alloy analyzer as references for comparison with the field welds. The reference materials are shown in Table 1. It should be noted that using this instrument in an identification mode, the unknown (or analyzed) material is compared to the reference materials loaded into the instrument during setup and calibration. If the unknown's composition exhibits very little difference to a known reference material, the unknown is identified as the reference material and as a "Good Match." If the unknown's composition is between those of the utilized reference materials, the unknown may be identified as either the nearest reference and "Possible Match" or a combination of the nearest references and "Possible Match." If the unknown's composition exhibits sufficient differences from the reference materials used, the instrument will respond as "No Good Match." Since the analyzed welds were reported to be a product of using a Type 308 stainless steel filler material to join Type 304 stainless steel piping, the resultant welds may exhibit a composition that is between these two materials due to dilution/mixing and, hence, a precise identification as either Type 304 stainless steel or Type 308 stainless steel may not be possible. Therefore, an identification as a reference (or the two nearest references) and as a "Possible Match" demonstrates the unknown's composition is similar to the references, but exhibits some variation due to dilution/mixing. In summary, all of the field welds were identified as being similar in composition to either the Type 304 stainless steel standard, the Type 309 stainless steel standard, the National Institute of Standards & Technology (NIST) standard reference material (SRM) 1154a, or a combination of two of the previous standards. The obtained results are summarized in Table 2.

Chip samples were obtained from three field welds by Harris Plant personnel. These samples were provided to NSL Analytical Services by Materials Dedication and Laboratory Services Unit personnel for chemical analysis using an expanded package for stainless steels. The obtained results are presented in Table 3 and as Attachment 1. These results were in agreement with those obtained by the alloy analyzer in that the majority of the welds were identified as being most similar in composition to either Type 309 stainless steel or the NIST 1154a SRM. The chemical analysis results showed the field welds to have carbon contents that were higher than the maximum limit specified for Type 304 stainless steel, but less than that specified for Type 309 stainless steel. The chromium content of all three welds was at the high end of the specified range for chromium in Type 304 stainless steel, but well under the lower limit of the specified range for chromium in Type 309 stainless steel. The nickel content of all three welds was in the middle of the range specified for Type 304 stainless steel and well under the lower limit for Type 309 stainless steel.

CONCLUSIONS:

The Unit 3/4 spent fuel piping field welds analyzed at the Harris Nuclear Plant with a Metorex X-Met 880 Alloy Analyzer were identified as being most similar in composition to either Type 304 stainless steel, Type 309 stainless steel, NIST 1154a SRM, or a combination of these reference materials. These results were confirmed by chemical analysis of chip samples from three different welds by an outside laboratory.

TABLE 1								
	Specified Elemental Composition, Weight Percent							
Identification	C	Cr	Ni	Mn	Si	P	S	Other
Standard Austenitic Stainless Steel Specifications								
Type 304 SS	0.08	18-20	8-10.5	2.00	1.00	0.045	0.03	...
Type 308 SS	0.08	19-21	10-12	2.00	1.00	0.045	0.03	...
Type 309 SS	0.20	22-24	12-15	2.00	1.00	0.045	0.03	...
Type 310 SS	0.25	24-26	19-22	2.00	1.00	0.045	0.03	...
Type 316 SS	0.08	16-18	10-14	2.00	1.00	0.045	0.03	2-3 Mo
Reference Material Compositions								
Type 304 SS	0.047	18.28	8.13	1.48	0.49	0.019	0.010	0.17 Mo
Type 309 SS	0.063	22.60	13.81	1.63	0.23	0.026	0.014	...
Type 310 SS	0.06	24.87	19.72	1.94	0.68	0.024	0.001	0.16 Mo
Type 316 SS	0.052	16.74	10.07	1.44	0.42	0.022	0.008	2.06 Mo
NIST 1154a	0.100	19.31	13.08	1.44	0.53	0.06	0.051	0.068 Mo
NIST 1155	0.046	18.45	12.18	1.63	0.502	0.018	0.020	2.38 Mo
NIST C1287	0.36	23.98	21.16	1.66	1.66	0.029	0.024	0.46 Mo

TABLE 2 Metorex X-Met 880 Alloy Analyzer Test Results				
Date	Specimen Identification	Alloy Identification	Match	Comments
5/12/98	Type 309 SRM	309 SS	Good	Instrument Check
5/12/98	Type 316 SRM	316 SS	Good	Instrument Check
5/12/98	NIST 1154a	NIST 1154a	Good	Instrument Check
5/12/98	2-SF-36-FW-450	304 SS	Possible	
5/12/98	2-SW-36-FW-449	NIST 1154a/309SS	Possible	
5/12/98	2-SF-38-FW-451	NIST 1154a	Possible	
5/12/98	2-SF-37-FW-441	NIST 1154a/309SS	Possible/Good	
5/12/98	2-SF-69-FW-328	NIST 1154a/309SS	Possible	
5/12/98	2-SF-70-FW-325	NIST 1154a/309SS	Good/Possible	
5/12/98	2-SF-69-FW-329	NIST 1154a	Possible	
5/12/98	NIST 1154a	NIST 1154a/309SS	Possible	Instrument Check
5/12/98	2-SF-14-FW-424	NIST 1154a/309SS	Possible/Good	
5/12/98	2-SF-71-FW-329	NIST 1154a	Possible	
5/12/98	2-SF-72-FW-327	NIST 1154a/309SS	Possible	
5/12/98	2-SF-16-FW-447	NIST 1154a	Possible	
5/12/98	2-SF-1-FW-6	NIST 1154a/309SS	Possible	
5/12/98	2-SF-1-FW-3	NIST 1154a/309SS	Possible	
5/12/98	2-SF-35-FW-440	NIST 1154a	Possible	
5/12/98	2-SF-68-FW-454	304SS	Good/Possible	
5/12/98	2-SF-31-FW-426	NIST 1154a/309SS	Possible	
5/12/98	NIST 1154a	NIST 1154a	Good	Instrument Check
5/12/98	2-SF-67-FW-452	NIST 1154a	Possible	
5/12/98	2-SF-72-FW-326	NIST 1154a	Possible	
5/12/98	2-SF-150-FW-412	NIST 1154a/304SS	Possible	
5/12/98	2-SF-148-FW-382	NIST 1154a/304SS	Possible	
5/12/98	2-SF-30-FW-381	NIST 1154a	Possible	
5/12/98	NIST 1154a	NIST 1154a	Good	Instrument Check
5/12/98	Type 309 SRM	309SS	Good	Instrument Check
5/12/98	Type 304 SRM	304SS	Possible	Instrument Check
5/13/98	Type 309 SRM	309SS	Good	Instrument Check
5/13/98	Type 310 SRM	310SS	Good	Instrument Check
5/13/98	NIST 1154a	NIST 1154a	Good	Instrument Check
5/13/98	2-SF-71-FW-341	304SS	Good	
5/13/98	2-SF-71-FW-335	NIST 1154a	Possible/Good	
5/13/98	2-SF-71-FW-336	304SS/NIST 1154a	Good/Possible	
5/13/98	2-SF-71-FW-342	304SS/NIST 1154a	Possible	
5/13/98	2-SF-71-FW-337	304SS/NIST 1154a	Possible	

TABLE 2 - Continued Metorex X-Met 880 Alloy Analyzer Test Results				
Date	Specimen Identification	Alloy Identification	Match	Comments
5/13/98	2-SF-71-FW-334	304SS/NIST 1154a	Possible/Good	
5/13/98	2-SF-71-FW-338	304SS	Good	
5/13/98	2-SF-71-FW-340	304SS	Good	
5/13/98	2-SF-71-FW-332	NIST 1154a/304SS	Possible/Good	
5/13/98	2-SF-71-FW-333	304SS/NIST 1154a	Possible/Good	
5/13/98	2-SF-71-FW-339	304SS/NIST 1154a	Possible/Good	
5/13/98	2-SF-71-FW-331	304SS/NIST 1154a	Good/Possible	
5/13/98	NIST 1154a	NIST 1154a	Good	Instrument Check
5/13/98	NIST 1155	316SS	Possible	Instrument Check
5/13/98	NIST C1287	310SS	Possible	Instrument Check
5/13/98	Type 309SS SRM	309SS/NIST 1154a	Good/Possible	Instrument Check

NOTE: The Metorex X-Met 880 was set up for analysis of the field welds using (1) a Type 304 stainless steel standard, (2) a Type 309 stainless steel standard, (3) a Type 310 stainless steel standard, (4) a Type 316 stainless steel standard, and (5) NIST 1154a standard reference material. NIST 1155 (Type 316 stainless steel) and NIST C1287 (Type 310 stainless steel) standard reference materials were used to check the instrument's response.

TABLE 3 NSL Analytical Services, Inc. Chemical Analysis Results					
Elemental Composition, Weight Percent					
	2-SF-36-FW-450	2-SF-38-FW-451	2-SF-71-FW-329	Type 304	Type 309
Carbon	0.13	0.10	0.064	0.08	0.20
Columbium	<0.05	<0.05	<0.05	...	...
Chromium	20.08	20.11	19.06	18-20	22-24
Copper	0.054	0.10	0.093	...	...
Manganese	1.46	1.39	0.79	2.00	2.00
Molybdenum	0.12	0.10	0.085	...	...
Nickel	9.30	9.24	9.63	8-10.5	12-15
Phosphorus	0.021	0.021	0.026	0.045	0.045
Sulfur	0.007	0.005	0.013	0.03	0.03
Silicon	0.37	0.39	0.25	1.00	1.00
Titanium	<0.01	0.011	<0.01	...	...

NOTE: The specified compositions for Type 304 and Type 309 stainless steels are provided for comparison and the single values represent maximum values.

ATTACHMENT 1

PROJECT NUMBER 98-125

NSL ANALYTICAL SERVICES, INC.
RMERLY NATIONAL SPECTROGRAPHIC LABORATORIES
QUALITY MATERIALS & ENVIRONMENTAL TESTING SINCE 1945

7650 HUB PARKWAY, CLEVELAND, OHIO 4412
PHONE: (216) 447-1550 FAX: (216) 447-071
(800) 497-6752
HTTP://www.nslanalytical.com

Carolina Power & Light Co
Harris Nuclear Plant
5413 Shearon Harris Rd.
New Hill, NC 27562
Attn: Gary Gray

Date: 5/22/98
Report No: 18630
P.O. No: 1L5577

Client Description: **Steel Chips**

NSL Lab No	Sample ID	Test	Results/Units
980011091	#2-SF-36-FW-450		
		C	0.13 %
		Cb	<0.05 %
		Cr	20.08 %
		Cu	0.054 %
		Mn	1.46 %
		Mo	0.12 %
		Ni	9.30 %
		P	0.021 %
		S	0.007 %
		Si	0.37 %
		Ti	<0.01 %
980011092	#2-SF-38-FW-451		
		C	0.10 %
		Cb	<0.05 %
		Cr	20.11 %
		Cu	0.10 %
		Mn	1.39 %
		Mo	0.10 %
		Ni	9.24 %
		P	0.021 %
		S	0.005 %
		Si	0.39 %
		Ti	0.011 %
980011093	#2-SF-71-FW-329		
		C	0.064 %
		Cb	<0.05 %
		Cr	19.06 %
		Cu	0.093 %
		Mn	0.79 %
		Mo	0.085 %
		Ni	9.63 %
		P	0.026 %
		S	0.013 %
		Si	0.25 %
		Ti	<0.01 %

Reporting Officers

H. Collins

Henry E. Collins, President
Steven M. Podolan, Vice President Technology

R01

NADCAP
Performance Review Institute
Accredited
MATERIAL TESTING LABORATORY

NSL ANALYTICAL SERVICES, INC.
FORMERLY NATIONAL SPECTROGRAPHIC LABORATORIES
QUALITY MATERIALS & ENVIRONMENTAL TESTING SINCE 1945

7650 HUB PARKWAY, CLEVELAND, OHIO 44125
PHONE: (216) 447-1550 FAX: (216) 447-0716
(800) 497-6752
HTTP://www.nslanalytical.com

Carolina Power & Light Co
Harris Nuclear Plant
5413 Shearon Harris Rd
New Hill, NC 27562

Date: 30 March, 1999
Report #: 18630
Lab #: 11091-11093
P.O.#: 1L5577
Page 1 of 1

Attn: Gary Gray

SUPPLEMENTAL REPORT- Traceability, Precision and Accuracy ADDED

Client Description: Steel Chips

NSL Lab No	Sample ID	Test	Results
980011091	#2-SF-36-FW-450	C	0.13%
		Cb	<0.05%
		Cr	20.08%
		Cu	0.054%
		Mn	1.46%
		Mo	0.12%
		Ni	9.30%
		P	0.021%
		S	0.007%
		Si	0.37%
		Ti	<0.01%
980011092	#2-SF-38-FW-451	C	0.10%
		Cb	<0.05%
		Cr	20.11%
		Cu	0.10%
		Mn	1.39%
		Mo	0.10%
		Ni	9.24%
		P	0.021%
		S	0.005%
		Si	0.39%
		Ti	0.011%
980011093	#2-SF-71-FW-329	C	0.064%
		Cb	<0.05%
		Cr	19.06%
		Cu	0.093%
		Mn	0.79%
		Mo	0.085%
		Ni	9.63%
		P	0.026%
		S	0.013%
		Si	0.25%
		Ti	<0.01%

Precision and Accuracy:
Cr +/-.20 Ni +/-.10 Cu +/- .03 P+/-.005 S+/- .002
Mn +/- .05 Si +/- .05 Mo +/- .03 C+/- .01

Traceability:
STD ARMI 2B

David M Kluk
David M. Kluk, Laboratory Manager

DK/mm

NADCAP
MATERIAL TESTING LABORATORY

The information and data in this report are rendered under the conditions ...

Matrix of Code Requirements vs. Missing Field Weld Records

Code Section	Code Requirement	Deficiency	Reconciliation
Section III, ND-2150 Section III, ND-4122	Requires identification and control of pressure-retaining materials	Identification of weld material for field welds was contained on WDRs which in most cases are no longer available. Likewise, records attesting to pipe spool id cannot be located for all of the embedded pipe spools. It is noted that numerous sections in the Code pertain to base metal and weld metal certification requirements. (ND-2121, 2130 & 2410, 2432, 2433, 4125)). There is no indication that CP&L did not conform to any of these requirements; rather, the deficiency is taken to be one of identification and traceability	For all accessible welds, material verification program will be undertaken to assure that correct material was used. For embedded SFP piping field welds, programmatic assurance is provided in that the procurement specification for welding materials during the time of construction (Site Specification No. 021)assured that all austenetic stainless steel welding material procured for Harris Plant construction was procured to Section III requirements. Construction procedure MP-03, "Welding Material Control" required that all filler material used for Code work at the Shearon Harris Nuclear Power Plant be purchased as specified by this procurement specification. Relative to embedded pipe spool id, in many cases this can be verified with alternate documentation from QA records. In those cases where this cannot be accomplished, program and procedure requirements provide additional assurance.
Section III, ND-4230	Requires that tack welds be removed or adequately prepared for incorporation into the final weld. Requires alignment of sections to be welded to specific criteria.	WDRs used to verify fit-up and alignment are generally not available.	All accessible field welds have been re-inspected using Code criteria with no gross fit-up deficiencies identified. A significant portion of embedded welds will be subject to internal camera inspection which would identify issues with fit-up and alignment.

Code Section	Code Requirement	Deficiency	Reconciliation
Section III, ND-4323	Requires that only those welding processes and welders qualified in accordance with Section IX be used.	Lack of documentation prevents verification of adherence to qualified processes and use of qualified welders.	Processes and programs at the time assure that the welding program was adhered to such that only qualified welders and processes were used. Construction Procedures MP-01(Qualification of WPS) & MP-02 (Qualification of Welders) required that all welders and welding procedures used for Power Plant construction be appropriately qualified. Construction Procedure MP-07, "General Welding Procedure for Stainless Steel Weldments", provided additional specific technical requirements beyond those found in the WPS. In addition, records associated with QA/QC oversight are available and provide assurance that issues were identified and resolved in accordance with QA program requirements. Finally, in most cases QC review of satisfactorily completed field welds is attested to by signature in hydro test records.
Section III, ND-4322.1	Requires identification of joint by application of welder id symbol	For embedded piping where WDRs are not available, lack of accessibility prevents verification of welder id	Program and procedural requirements would have required that the welder id be stamped at the weldment and included on the WDR (ref. Construction Procedure MP-05, "Stamping of Weldments")
Section III, ND-4440	Requires examinations of welds in accordance with ND-5000. For the welds in question, this would have resulted in either MT (for CCW Piping) or LP (SFP Piping), with acceptance standards per ND-5300.	Lack of documentation attesting to the satisfactory completion of required NDE.	All accessible Code field welds in scope of the Alternative Plan have been subject to visual examination, along with Code required external NDE (LP / MT) using original Code acceptance criteria. In addition, a large percentage of embedded field welds will be subject to internal camera inspection using documented inspection procedures and qualified inspectors. This examination program augments programmatic and procedural measures existing at the time of construction to assure that the necessary level of quality exists.
Section III, ND-4452 & 4453	Requires that defects be removed and repaired areas be examined.	Repair WDRs may not be available to document all inspection / repair activities	Records review finds that many Repair WDRs are on file as a result of QC oversight of the construction process. However, Repair WDRs which were identified "in process" may not be on hand. The same assurances which attest to the quality of completed field welds also apply to assure that defects were identified and removed per Code requirements.

Alternative Plan Letter of Endorsement from
The Hartford Steam Boiler Inspection and Insurance Co.

(1 Page)

Dr. Richard E. Feigel
Vice President

Post-it® Fax Note 7671	Date	# of pages ▸ /
To BARRY BOBO	From GENE FEIGEL	
Co./Dept.	Co.	
Phone #	Phone #	
Fax #	Fax #	

The Hartford Steam Boiler
Inspection and Insurance Co.
P.O. Box 5024
One State Street
Hartford CT 06102
(860) 722-5652
(860) 722-5530 (Fax)
rfeigel@ibm.net (Email)

RECEIVED
MAR. 8 1999.

March 8, 1999

Mr. Steve Edwards
Manager, SFP Activation Project
Carolina Power & Light Company
Harris Nuclear Plant
P.O. Box 165
New Hill NC 27562

Subject : 10CFR50.55a Alternative Plan
HH/99-001
HNP-98-188 a

Dear Mr. Edwards:

I have reviewed your letter to Mr. Bobo and the referenced attachments addressing various spent fuel pool piping systems. I have discussed the subject at length with Mr. Bobo, who is in responsible direct charge of Hartford Steam Boiler's (HSB) ASME Section III and XI inspection activities. Subject to detailed verification of completion of unfinished tasks and their compliance with commitments described, we believe that the plan proposed provides an acceptable alternative to code compliance in accordance with 10CFR50.55(a)(3). Our concurrence extends to both dispositioning issues related to the as-built condition of the systems and future activities under 'III. Alternative Plan for Continuance of Design and Construction.'

Our position is based principally on the following:

1. Site wide use of an integrated QA program at the site with evidence of adequacy provided by licensing of Unit 1.
2. Consistent reference by final acceptance documentation, e.g. hydrostatic test reports, of first tier inspection reports which establish review of records of welder qualification and similar code requirements.
3. Plan provisions to verify code compliance or establish technical equivalency, .e.g. deposited weld metal analysis.

Very truly yours,

RE Feigel

Richard E. Feigel, Ph.D.
Vice President, Engineering

Cc: B. Bobo R. Howard

LOGGED

Present Corporate (Appendix B) Quality Assurance Manual

CP&L

NGG PROGRAM MANUAL

Title: Quality Assurance Program Manual

Lead Department: PERFORMANCE EVALUATION & REGULATORY AFFAIRS

NGG Program Manual Number:	Revision Number:	Effective Date:
NGGM-PM-0007	Rev. 1	July 10, 1998

Revision 1:

Sections 19.0 and 20.0 were combined into Section 19.0 to provide more detailed requirements in establishing the Graded Approach to Quality for Software. The procedures implementing the requirements of these Sections will become effective August 18, 1998, after training has been presented on the implementing procedures and the changes to the QA Program Manual. Therefore the changes to this manual will also have an effective date of August 18, 1998. In addition, Section 3.4.2 was revised to correct an error in performing design verification, and Enclosure 1, CP&L Quality Assurance Program Policy, was added to ensure that the Quality Assurance Program Policy on the Intranet is appropriately controlled.

Approved By:

C S Hinnant
Senior Vice President and Chief Nuclear Officer

7-7-98
Date

TABLE OF CONTENTS

ACRONYMS

A/E - Architect-Engineer

ANI - AUTHORIZED NUCLEAR INSERVICE INSPECTOR for ASME Code compliance activities and items at the site

AIA - AUTHORIZED INSPECTION AGENCY for ASME Code compliance activities with which CP&L has contract for AUTHORIZED INSPECTOR coverage for each site

ANSI - American National Standards Institute, Inc.

ASME - American Society of Mechanical Engineers

ASTM - American Society of Testing & Material

BNP - Brunswick Nuclear Plant

BSEP - Brunswick Steam Electric Plant

BWR - Boiling Water Reactor

CMMS - Corporate Materials Management System

CP&L - Carolina Power & Light Company

10CFR50 - Title 10 (Atomic Energy), Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities."

ESR - Engineering Service Request

FSAR - Final Safety Analysis Report

HBRSEP - H. B. Robinson Steam Electric Plant, Unit 2

HNP - Harris Nuclear Plant

INPO - Institute of Nuclear Power Operations

M&TE - Measuring and test equipment

NAS - Nuclear Assessment Section

NIST - National Institute of Standards and Technology

NDE - Nondestructive examination

NED - Nuclear Engineering Department

ACRONYMS

NRC - Nuclear Regulatory Commission

N-Stamp - Official N-type symbol provided by the ASME and applied to plant items upon certification of compliance with applicable rules of the ASME Code

OESD - Operations & Environmental Support Department

PES - Performance Evaluation Support Unit

PO - Purchase order

PR - Purchase requisition

PWR - Pressurized Water Reactor

RFO - Released for operation

RNP - Robinson Nuclear Plant

SAR - Safety Analysis Report. The most recently updated collection of information pursuant to 10CFR50.34(b) and which the NRC uses to conclude that the facility may be operated without undue risk to the public health and safety, including, but not limited to , the following:

- UFSAR and FSAR (HNP), including its text, figures, drawings, and approved changes which have not yet been incorporated,
- Documents incorporated by reference including, but not limited to , the Emergency Plan, Security Plan, Operating License(s), Technical Specifications, and NRC Safety Evaluation Reports (SERs) (Documents merely listed as references are excluded), and
- Docketed correspondence related to 10CFR50.34.

SHNPP - Shearon Harris Nuclear Power Plant

SNM - Special Nuclear Material

SNT-TC-1A - Publications of the American Society for Nondestructive Testing which present recommended practices for qualifying and certifying personnel performing specific methods for nondestructive examination and evaluation of the examination results.

SSC - Structures, systems, and components

UFSAR - Updated Final Safety Analysis Report

DEFINITIONS

Certain terms are applied in the Carolina Power & Light Company (CP&L) Quality Assurance Program Manual (QAP Manual) with a special meaning or in a more restrictive sense than defined in a standard dictionary. The definitions listed are applicable to Nuclear Regulatory Commission (NRC) regulated activities and are generally used throughout the QAP Manual. All items which are defined in this section will appear as boldface type throughout this manual.

ACCEPTANCE CRITERIA: A limit or limits placed on the variation permitted in the characteristics of an item expressed in definitive engineering terms such as dimensional tolerances, chemical composition limits, density and size of defects, temperature ranges, time limits, operating parameters, and other similar characteristics. (ANSI N45.2.8)

ACTIVE SAFETY-RELATED INSTRUMENT: A permanently installed instrument that has been determined to be **safety-related** because it provides some required signal/output in the event of an accident.

ACTIVITIES AFFECTING QUALITY: Activities that affect or reasonably could affect the safety related functions of nuclear plant structures, systems, components, and parts. Activities included are design changes, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling and modifying.

ALGORITHM (COMPUTER): 1) A finite set of well-defined rules for the solution of a problem in a finite number of steps; for example, a complete specification of a sequence of arithmetic operations for evaluating sine *x* to a given precision. 2) Any sequence of operations for performing a specific task.

*ANNUALLY: Once per year, not to exceed 366 days.

APPROVED SUPPLIERS LIST (ASL): A listing of suppliers/contractors whose quality assurance programs have been evaluated to meet applicable requirements and found capable of supplying particular items or **services** to specified requirements.

BASIC COMPONENT: A structure, system, or component, or part thereof that affects its safety function necessary to assure the item is safety related. (10CFR21)

BENCHMARK (SOFTWARE): - See **qualification**

*BIANNUAL: Every 6 months, not to exceed 184 days.

*BIENNIAL: Every 2 years, not to exceed 732 days.

*BIMONTHLY: Every 2 months, not to exceed 62 days.

*BIWEEKLY: Every 2 weeks, not to exceed 14 days.

BUGS (SOFTWARE): Unexpected defects, faults, flaws, or imperfections.

CALIBRATION: Comparison of an item of **Measuring and Test Equipment** (M&TE) with a reference standard or with an item of **M&TE** of equal or closer tolerance to detect and quantify inaccuracies and to report or eliminate the inaccuracies.

COMMERCIAL GRADE: A structure, system, or component, or part thereof that affects its safety function, that was not designed and manufactured as a **basic component**. (10CFR21)

COMPLETED QUALITY ASSURANCE (QA) RECORD: A document becomes a QA record when the last reviewer or evaluator has completed his or her review or evaluation as prescribed in procedures. Completion includes being stamped, initialed, signed, or otherwise authenticated, and dated by authorized personnel. In the case of a record package (Engineering Service Request, Equipment Qualification, and so forth) made up of several individual documents, the package will be considered to be the document for the purpose of determining when the document is complete.

COMPUTER SOFTWARE: - Computer programs, procedures, and possibly associated documentation and data pertaining to the operation of a computer system. A sequence of instructions or actions implemented by procedure or **algorithm**, that may or may not be taken, suitable for processing by a computer.

CONDITION ADVERSE TO QUALITY (CATQ): See Section 12.0.

CONDITIONAL RELEASE: A document permitting limited work progression on nonconforming items.

CONSUMABLE/EXPENDABLE ITEM: Those designated items whose quality is necessary for the functional performance of **safety-related** structures, systems, and components and thus are subject to applicable provisions of 10CFR50, Appendix B. These designated items are purchased and controlled in accordance with plant procedures.

CONTRACT: The various documents which describe the scope of the contracted work and the conditions under which CP&L and the contractor have agreed to participate. The **contract** may include either the procurement of labor and/or **services** together with materials necessary in their performance.

DESIGN BASES: That information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen or controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted state-of-the-art practices for achieving functional goals or (2) requirements derived from analysis based on calculation and/or experiments of the effects of a postulated accident for which a structure, system, or component must meet its functional goals (refer to 10CFR50.2).

CP&L has provided the following clarification to the NRC. A system's Design Basis, as defined by CP&L, consists of:

- System and Component functional requirements (Reference 10CFR50.2),
- Regulatory Requirements and Commitments relative to system and component design (Reference 10CFR50, Appendix B, Criterion III),

- Original System and Component design codes and standards of record, unless clearly superseded by a Regulatory commitment to a later code or standard (Reference 10CFR50.2)

DESIGN CHANGE OPERABILITY: The installation of a completed design change such that the affected equipment is capable of performing its intended function, when sufficient acceptance testing has been completed to verify the changes will perform as specified by the design and to fulfill any testing requirements resulting from the change, and when sufficient documentation exists to support operation.

DESIGN DOCUMENTS: Specifications, calculations, drawings, and procedures derived from regulatory requirements and **design bases** that delineate item design, quality assurance and process requirements for use in procurement, fabrication, installation, examination, and testing; and analyses and reports that substantiate design characteristics or evaluate item performance.

DESIGN INPUTS: Those criteria, parameters, bases or other design requirements, updated to reflect all approved changes, upon which detailed final design is based.

DESIGN ORGANIZATION: An organization within CP&L or a contractor supporting CP&L assigned responsibility for development or revision and documentation of the design of a plant structure, system, equipment, or parts thereof.

DESIGN SPECIFICATIONS: The document describing the engineering and performance requirements which provide a basis for designing an item and/or the technical information necessary for purchasing an item.

DESIGN OUTPUT: Documents such as drawings, specifications, and other documents that define the technical requirements of Safety Systems and Components (SSC).

ENGINEERING EVALUATION: A documented assessment performed to disposition a concern, indeterminate condition, or other circumstance that provides a basis for the disposition and is reviewed and released as specified in procedures.

ENVIRONMENT (COMPUTER): The conditions under which a program is developed or run. This includes the type of processor, storage media, and other software-dependent hardware used, as well as the operating system used to run the program.

FIRE PROTECTION RELATED: Those fire protection systems and components that provide direct protection to **safety-related** items from fire or whose failure could prevent those fire protection systems and components from operating. Those components used for indication, backup, or information purposes are not considered fire protection related.

HOLDPOINT: A point beyond which work shall not proceed until mandatory verification, inspection, or approval is obtained from appropriate inspection/ verification organization(s).

INFORMATION MANAGEMENT SYSTEM: A data base or computing system containing information used to support a **safety-related** activity, i.e., Equipment Data Base System (EDBS).

MEASURING AND TEST EQUIPMENT (M&TE): Instrument, tools, gauges, fixtures, reference and transfer standards, and nondestructive test equipment which are used in the measurement, inspection, and monitoring of **safety-related** components, systems, and structures. (This includes [1] instrumentation permanently installed as required by the plant Technical Specifications, [2] instrumentation used to verify Technical Specifications but which are not specified in the Technical Specifications, and [3] **active safety-related instruments**. **M&TE** does not include rulers, tape measures, levels, and other such devices if normal commercial practices provide adequate accuracy, or installed or portable instruments used for preliminary or qualitative checks, where accuracy is not required, such as a circuit checking multimeter.)

*MONTHLY: Once per month, not to exceed 31 days.

PORTABLE MEASURING AND TEST EQUIPMENT (P-M&TE): **M&TE** items that are not permanently installed in the facility (e.g., test gauges, voltmeters, deadweight tester).

PURCHASE ORDER (PO): A formal agreement for procurement of items and those **services** allowed to be obtained without a **contract**.

QUALIFICATION: The process of demonstrating, through test methods, a given input for the software produces the expected output..

QUALITY CLASS B ITEMS: Nonsafety related, seismically designed items as discussed in Regulatory Guide 1.29, Regulatory Positions C2 and C4, and Category 2 instruments subject to Regulatory Guide 1.97, Revision 3.

QUALITY RELEASE: A document used by a vendor or CP&L to release item(s) for shipment from a vendor's facility.

QUALITY SOFTWARE/COMPUTING SYSTEMS: Computer software and/or computing systems used to support processes that have a direct or indirect affect on nuclear safety and/or operation.

*QUARTERLY: Every 3 months, not to exceed 92 days.

RECEIPT INSPECTION: Inspection activities performed by qualified personnel during the receiving of items to determine the conformance of those items to predetermined requirements.

REFERENCE CALIBRATION STANDARDS: Standards (e.g., primary, secondary, working, field, and shop where appropriate) used in a **calibration** program. These standards establish the basic accuracy limits for the program.

REPAIR: The process of restoring a nonconforming characteristic to a condition such that the capability of an item to function reliably and safely is unimpaired, even though that item still may not conform to the original requirement. (ANSI N45.2.10)

REWORK: The process by which a nonconforming item is made to conform to a prior specified requirement by completion, remachining, reassembling or other corrective means. (ANSI N45.2.10)

SAFETY-RELATED: A term applied to those plant features relied upon during or following a design basis event to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents which could result in off-site exposures comparable to the guideline exposures of NRC Regulation 10CFR100.

*SEMIANNUAL: Every 6 months, not to exceed 184 days.

*SEMIMONTHLY: Every 2 weeks, not to exceed 16 days.

*SEMIWEEKLY: Twice per week, not to exceed 4 days.

SERVICES: The performance by a Supplier of activities such as design, fabrication, inspection, non-destructive examination, repair, or installation. (ANSI N45.2.13)

SIGNIFICANT CONDITION ADVERSE TO QUALITY (SCATQ): See Section 12.0.

TRACEABILITY: The ability to trace the history, application, or location of an item or activity by means of recorded identification.

USE-AS-IS: A disposition which may be imposed for a nonconformance when it can be established that the discrepancy will result in no adverse conditions and that the item under consideration will continue to meet all engineering functional requirements including performance, maintainability, fit, and safety. (ANSI N45.2.10)

*WEEKLY: Every week, not to exceed 7 days.

*These frequency dependent terms are defined for application if not specifically defined in plant documents.

1.0 INTRODUCTION

1.1 CP&L QUALITY ASSURANCE (QA) PROGRAM - SCOPE

This manual amplifies the CP&L committed 10CFR50 Appendix B Quality Assurance Program (QAP) requirements described in (U)FSAR Section 17.3 and establishes measures for assuring that organizations performing **safety-related** activities perform their responsibilities in a manner which results in safe nuclear power production. This manual also establishes the QA programs for the non-safety related areas of RW-Q, FP-Q, and **Quality Class B**. Additional QA requirements imposed on individual plants by regulations and commitments shall be considered a part of the QAP. Other QA programs are established in this manual to comply with requirements, either required by regulators, or determined to assist the company implement structured programs beneficial to the operation of the nuclear plants.

The guidance provided in this manual is not all inclusive. It is intended to be used in conjunction with Sections 1.8 and 17.3 of the (U)FSARs to develop procedures that implement the CP&L Quality Assurance Program.

1.2 SCOPE OF APPLICATION

The measures described in this manual have been written to comply with the Quality Assurance requirements of certain regulatory documents identified in Sections 1.8 and 17.3 of the (U)FSARs. The applicable regulatory commitments are identified in each section.

The manual is arranged in functional sections to facilitate its use and includes additionally Appendix I which cross-references functional subjects with the applicable criteria of 10CFR50, Appendix B, and Appendix II which contains QA program regulatory guide references.

A list or system identifying items to which Sections 1.0 through 19.0 apply shall be maintained at each nuclear plant or work location. The responsibility for maintaining this list or system shall be identified in procedures or interface documents.

1.2.1 Sections 1.0 through 14.0--Scope of Application

For compliance with 10CFR50, Appendix B, and 10CFR72 the provisions of Sections l.0 through 14.0 shall be applied to activities associated with **safety-related** materials, equipment, and **services**.

1.2.2 Section 15.0--Scope of Application

This section identifies measures for compliance with the QAP requirements for fire protection systems, components, parts, and administrative programs.

1.2.3 Section 16.0--Scope of Application. (HNP Only)

This section identifies measures for compliance with the QAP requirements for radioactive waste systems, components, and administrative programs.

1.2.4 Section 17.0--Scope of Application

This section identifies measures for compliance with the QAP requirements for the IF-300 irradiated fuel shipping cask.

1.2.5 Section 18.0--Scope of Application

This section identifies measures for compliance with the QAP requirements for shipping "non LSA greater than Type A" packages.

1.2.6 Section 19.0--Scope of Application

This section identifies measures for compliance with the QAP requirements for **computer software** for **safety-related** applications.

1.2.7 Section 20.0--Scope of Application

This section has been deleted and the requirements incorporated into Section 19.0.

1.2.8 Section 21.0--Scope of Application (HNP only)

This section identifies QAP requirements for Class B items.

1.2.9 Section 22.0--Scope of Application (BNP and RNP only)

This section identifies QAP requirements for nonsafety related systems and equipment used to meet the Station Blackout Rule.

1.2.10 Section 23.0--Scope of Application

This section identifies requirements for the issuance of interpretations of the QAP by the Manager - Performance Evaluation and Regulatory Affairs (PERAS). Interpretations issued are included in this section.

2.0 ORGANIZATION AND RESPONSIBILITIES

2.1 SCOPE

This section sets forth the organizational structure and responsibilities for implementation of the Quality Assurance Program (QAP). While general managerial and supervisory responsibilities are delineated in this section, each organization performing activities described in this manual is responsible for assuring proper implementation of the applicable requirements for the activity being accomplished. Specific duties and responsibilities should be delineated in procedures and interface documents.

2.2 MANAGEMENT RESPONSIBILITIES

Ultimate responsibility for operation of the nuclear plants rests with the Senior Vice President, Nuclear Generation/Chief Nuclear Officer reporting to the Executive Vice President, Energy Supply who reports to the President/Chief Executive Officer.

Nuclear Generation - The Senior Vice President/Chief Nuclear Officer reports to the Executive Vice President, Energy Supply. This position is responsible for managing the company's nuclear plants and assuring they are in compliance with applicable regulations, codes, and other requirements. There are five departments in the Nuclear Generation Group: (a) the Brunswick Nuclear Plant Department, (b) the Harris Nuclear Plant Department, (c) the Robinson Nuclear Plant Department, (d) the Nuclear Engineering Department, and (e) the Operations and Environmental Support Department. Their responsibilities are summarized below:

2.2.1. The Brunswick Nuclear Plant Department - The Vice President, Brunswick Nuclear Plant Department reports to the Senior Vice President/Chief Nuclear Officer. This position is responsible for managing all aspects of: configuration control of the plant's design basis; services associated with the procurement, design, and modification installation; outage management; direct plant support functions; operation; and maintenance of the Brunswick Nuclear Plant. The department includes: (1) Director of Site Operations, (2) Plant General Manager, (3) Manager - Plant Support Services, (4) Manager - Regulatory Affairs, (5) Manager - Training, (6) Manager - Brunswick Engineering Support Services, (7) Manager - Nuclear Assessment, and (8) Manager - Environmental & Radiation Control.

2.2.2 The Harris Nuclear Plant Department - The Vice President, Harris Nuclear Plant Department reports to the Senior Vice President/Chief Nuclear Officer. This position is responsible for managing all aspects of: configuration control of the plants design basis; services associated with the procurement, design, and modification installation; outage management; direct plant support functions; operation; and maintenance of the Harris Nuclear Plant. The department includes: (1) Director of Site Operations, (2) Plant General Manager, (3) Manager - Plant Support Services, (4) Manager - Harris Engineering Support Services, (5) Manager - Training, (6) Manager - Regulatory Affairs, (7) Manager - Nuclear Assessment and (8) Manager - Environmental & Radiation Control.

2.2.3. The Robinson Nuclear Plant Department - The Vice President, Robinson Nuclear Plant Department reports to the Senior Vice President/Chief Nuclear Officer. This position is responsible for managing all aspects of: configuration control of the plant's design basis; services associated with the procurement, design, and modification installation; outage management; direct plant support functions; operation; and maintenance of the Robinson Nuclear Plant. The department includes: (1) Director of Site Operations, (2) Plant General Manager, (3) Manager - Plant Support Services, (4) Manager - Regulatory Affairs, (5) Manager - Training, (6) Manager - Robinson Engineering Support Services, (7) Manager - Nuclear Assessment and (8) Manager - Environmental & Radiation Control.

2.2.4. The Nuclear Engineering Department - The Vice President, Nuclear Engineering Department (NED) reports to the Senior Vice President/Chief Nuclear Officer. This position is responsible for complimenting the Plant Engineering Support Sections by providing an integrated technical, design control and configuration management function. The VP, NED is also responsible for engineering, procurement, and fabrication of nuclear fuel, probabilistic risk assessment (PRA) and spent fuel management services for the nuclear plants . Reporting to the Vice President - Nuclear Engineering Department are: (1) Manager - Nuclear Fuel Management and Safety Analysis, and (2) Chief Engineer.

2.2.5. The Operations and Environmental Support Department - The Director - Operations & Environmental Support Department reports to the Senior Vice President/Chief Nuclear Officer. This position is responsible for materials acquisition and administrative services for the Nuclear Generation Group; as well as providing analytical, chemistry, and metallurgy services; operations, maintenance, and configuration control of plant computing systems; environmental programs support; and radiological support for the company. The Department consists of: (1) Manager - Material Services, (2) Manager - Environmental Services, (3) Manager - Nuclear Information Technology, and (4) Manager - Business Planning & Budget Services.

The Manager - Performance Evaluation and Regulatory Affairs (PERAS) reports to the Senior Vice President/Chief Nuclear Officer. This position is responsible for generic licensing, independent oversight of the plant's Nuclear Assessment Sections, and maintenance of the Quality Assurance Program Manual.

The Manager - PERAS, as necessary, is responsible for updating this manual to maintain consistency with commitments, mandatory regulations, and codes. The Manager - PERAS shall assure a review of the status and adequacy of this manual is performed at least once a year by appropriate CP&L management and submit any recommended revisions to the Senior Vice President/Chief Nuclear Officer for approval. Revisions and distribution to the QAP Manual will be in accordance with NGGM-PM-0005, Development and Approval of Documents in the NGG Document Hierarchy.

The Senior Vice President - Administrative Services reports to the President/Chief Executive Officer. This position operates through the Vice President - Corporate Services to provide procurement activities and security access for each nuclear plant.

The three plant Nuclear Assessment Sections (NAS) independently monitor and assess the Company's nuclear programs on a continuing basis. The NAS performs assessments which incorporate the previous QA audits. These evaluations are primarily performance based with emphasis on quality of the end product.

Quarterly (approximately) a briefing of NAS activities, along with any potential issues and recommendations, shall be presented to the Senior Vice President/Chief Nuclear Officer. The Managers - NAS shall have access to the corporate management up to and including the Senior Vice President/Chief Nuclear Officer to resolve any quality or nuclear safety related concerns if the concerns cannot be resolved satisfactorily at a lower management level.

The Performance Evaluation Support Unit (PES) of PERAS is responsible to ensure that the results and effectiveness of the NAS organization and its processes in accomplishing its assigned objectives is regularly evaluated on a frequency not to exceed 24 months.

2.3 RESPONSIBILITY

The primary responsibility for quality performance, including the identification and effective correction of problems potentially affecting the safe and reliable operation of the Company's nuclear facilities, resides with the line organization. The term "line organization" used in this program refers to the production organization reporting to the Senior Vice President/Chief Nuclear Officer.

The managers of functions involving engineering, modification, maintenance, nuclear fuel, and operations shall assure that their personnel are adequately trained for their jobs and they have the experience and education required to carry out their assigned responsibilities. These managers shall ensure that adequate resources and procedures are available for correctly implementing the work activities to support the QA program.

Independent inspections are conducted in accordance with procedures to verify specific critical quality attributes. Individuals performing these inspections have access to necessary information to ensure that activities and equipment meet established acceptance criteria.

Procurement documents prepared in accordance with procedures require suppliers to operate in accordance with QA programs which are compatible with the applicable requirements of CP&L's QAP and procedures where their services are used in support of plant activities.

2.4 AUTHORITY

The QAP and procedures require that the authority and duties of persons and organizations performing activities affecting quality be clearly established and delineated in writing. In addition, the QAP requires that these individuals and organizations have sufficient authority and organizational freedom to:

1. Identify quality, nuclear safety, and performance problems.
2. Order unsatisfactory work to be stopped and control further processing, delivery, or installation of nonconforming material.

3. Initiate, recommend, or provide solutions for **conditions adverse to quality**.

4. Verify implementation of solutions.

3.0 OPERATING PLANT DESIGN ACTIVITY CONTROL

3.1 SCOPE

This section sets forth minimum requirements for control of design activities affecting systems, components, and structures. The major areas covered by this section are design, reviews and approvals, work execution, documentation, and design interface controls.

3.2 RESPONSIBILITIES

The responsibility for implementing this section is assigned to each nuclear plant department and the Nuclear Engineering Department. Responsibilities delegated to other Carolina Power & Light (CP&L) departments or contractors shall be documented through approved interface documents.

3.3 REGULATORY COMMITMENTS

This section used in conjunction with Regulatory Guides 1.64 and 1.33 and American National Standards Institute N45.2.11 and N18.7, as committed by Sections 1.8 and 17.3 of the (U)FSAR, establishes the requirements essential for compliance with the applicable portions of 10CFR50 Appendix B.

3.4 DESIGN PROCESS

The designated **design organization** shall have access to pertinent background information needed to fulfill its responsibility and shall have personnel with adequate understanding of the requirements and intent of the original plant design commensurate with the scope and complexity of the design activity to be performed.

Design activities carried out to develop final **design documents** or to support development of final **design documents** shall be accomplished in accordance with procedures of a type sufficient to ensure that design input requirements are correctly applied, the activity is documented in sufficient detail to permit verification, appropriate quality standards are identified, and the results of the activity are reviewed and approved. Design activities include such work as preparation of design input requirements, specifications, drawings, analyses, and procedures.

3.4.1 Design input requirements.

Applicable design input requirements shall be developed and documented. The **design inputs** shall be specified to a level of detail sufficient to allow translation into other **design documents** such as specifications, drawings, analyses, procedures, etc. Changes to design requirements during the design process shall be controlled to ensure such changes are factored into other ongoing design activities.

3.4.2 Design verification.

Sufficient design verification shall be performed by one or more methods to substantiate that the final design documents meet the appropriate design inputs. Verification activities shall be clearly documented, identifying the verifier and the results of the verification. Acceptable verification methods include but are not limited to:

3.4.2.1 Design reviews.

3.4.2.2 Alternate calculations.

3.4.2.3 Qualification testing using the most adverse specified design condition.

The design verification shall be performed by a competent individual or group of individuals, but shall not be performed by individuals who prepared the original design or the designer's immediate supervisor unless the immediate supervisor is the only one capable of verifying the design. Objective evidence documenting the completion of and satisfactory resolution of any concerns raised in the design verification shall be provided with the package prior to relying on the structure system or component to perform its function. A design verification of the completed design package shall be performed to verify the following:

3.4.2.3.1 Design interface between design disciplines is adequately established.

3.4.2.3.2 Sufficient **design documents** and procedures are included or referenced to allow implementation to be carried out in a planned and controlled manner.

3.4.2.3.3 Adequate provisions for in-process or post-installation examinations, inspections, and testing have been specified to assure quality of work and verification that the design performs as intended.

3.4.2.10.4 Adequate provisions have been provided to document installation and results of examinations, inspections, and testing within the package or documents referenced.

3.4.2.10.5 Consideration has been given to **design change operability**, reliability, maintainability, safety, and adherence to appropriate codes, standards, and regulatory requirements.

3.4.2.10.6 Appropriate design verification has been performed for applicable documents contained in the package.

3.4.2.10.7 Specified materials and processes are suitable for the intended application.

3.4.2.10.8 The design is technically adequate with respect to the **design bases**.

3.5 DESIGN CHANGE PACKAGE

For design changes that produce a physical modification to the plant, an approved design change package shall be issued. The design change package shall be prepared by the responsible **design organization** and shall include or reference **design documents** or procedures to provide for:

3.5.1 Installation of the physical change.

3.5.2 Identification of required inspections and **acceptance criteria.**

3.5.3 Identification of required testing and **acceptance criteria.**

3.5.4 Identification of specified materials for installation.

3.5.5 Identification of necessary revisions to existing **design documents** such as design basis documents, specifications, drawings, procedures, and manuals.

3.5.6 Identification of new **design documents.**

3.5.7 Identification of functional quality class and boundaries.

3.5.8 Control of design change package.

The content, revisions, format, reviews and approvals, issuance control, and interface with other processes affected by the design change shall be established in procedures. The procedures shall establish controls to assure that changes to Plant Operating Manual, **information management systems**, or other documents important to the configuration or to work execution are identified.

3.5.9 Design change package implementation.

Implementation including installation, examinations, inspections, and tests shall be performed in accordance with the procedures provided in the design change package or procedures referenced in the design change package. Deviations from the design change package, except where authorized in the package or referenced procedures, require an approved revision to the package prior to work execution.

3.6 SAFETY EVALUATION

Any proposed activity/change or discovered change to the facility or procedure or test or experiment as described in the applicable SAR whether permanent or temporary shall be evaluated for 10CFR50.59 applicability as described in procedures.

3.7 DESIGN CHANGE OPERABILITY AND CLOSEOUT

3.7.1 Prior to **design change operability**, verification of the work and a review of documentation shall be performed to assure work has been satisfactorily accomplished including examinations, inspections, and tests. Measures shall be established to document any exceptions identified by this review. These exceptions shall be either cleared prior to **design change operability** or exceptions shall be approved and tracked to ensure timely completion.

3.7.2 Training to familiarize plant personnel with the hardware, procedure changes, and Technical Specification changes resulting from the implementation of the design change package shall be conducted, as appropriate.

3.7.3 As part of a declaration of **design change operability**, measures shall be initiated to revise documents and **information management systems** identified per Section 3.5.8 in accordance with approved procedures. Methods shall be implemented to ensure that potential users of affected documents are notified of outstanding changes to documents and **information management systems.** The controlling procedures for design change packages shall provide a documented method for declaration of **design change operability.**

3.7.4 Final closeout of a design change package shall not be done until all exceptions and outstanding changes to documents and **information management systems** have been dispositioned.

3.8 TEMPORARY DESIGN CHANGES

3.8.1 Temporary design changes to the plant to support testing shall be accomplished in accordance with procedures. The procedure shall:

3.8.1.1 Control the installation of the change.

3.8.1.2 Require removal of the change upon test completion.

If the equipment or system affected is to remain in service during the change, the following additional requirements shall apply:

3.8.1.3 Design Verification

3.8.1.4 10CFR50.59 safety evaluation.

3.8.2 Temporary design changes including temporary repairs to the plant for reasons other than test or surveillance activities shall be controlled by procedures. The procedure may be for a specific change or a controlling process for a certain type of change. In either case, the document authorizing the change shall:

3.8.2.1 Provide instructions to implement the change.

3.8.2.2 Control removal of the change.

If the system is to remain in service during the change, the following additional requirements shall apply:

3.8.2.3 Perform a Design Verification

3.8.2.4 Perform a 10CFR50.59 safety evaluation.

3.8.2.5 Designate the responsible organization for control and removal of the change.

3.8.2.6 Provide appropriate notification and instruction, if needed, to operational personnel.

3.8.2.7 Identify drawing and procedure changes to be in effect during the time the temporary design change is in place.

3.8.2.8 Identify training requirements.

Such temporary changes shall be tracked to assure removal or permanent dispositioning within a specified time limit.

3.9 DISPOSITION OF DEVIATIONS BETWEEN DESIGN DOCUMENTS AND PLANT CONFIGURATION

When deviations are discovered between plant **design documents** and actual configuration, they shall be dispositioned in accordance with Sections 3.0, 11.0, or 12.0.

3.10 DESIGN INTERFACE CONTROL

Documented interface control over design activities assigned by each nuclear plant department to other CP&L organizations shall address the following:

3.10.1 Delegated areas of responsibility.

3.10.2 How the assistance is requested and the scope of work specified.

3.10.3 Methods of communication between the assisting organization and each nuclear plant department.

3.10.4 Applicable procedures which govern the conduct of design activities, design change authorizations, and work execution.

3.10.5 Handling and dispositioning of documentation and Quality Assurance Records generated during the performance of the requested activity.

The interface document shall be established prior to initiation of design. The interface may be in a document specific for a given assistance request, procedures concurred with by the affected departments, or a standardized interface agreement.

4.0 PROCUREMENT CONTROL

4.1 SCOPE

This section establishes requirements for controlling the activities and documents associated with procurement of items and services. It includes requirements for procurement document content and reviews, vendor selection and qualification, and surveillance after award.

4.2 RESPONSIBILITY

The responsibility for implementing this section is designated to the Carolina Power & Light (CP&L) organization having responsibility for the project or work function that determines the need for procurement. Delegation of tasks or functions related to this responsibility shall be accomplished through approved interface documents or procedures. The CP&L organization having primary responsibility shall retain the responsibility.

4.3 REGULATORY COMMITMENTS

This section used in conjunction with Regulatory Guides 1.33, 1.123, 1.144, and 1.146 and American National Standards Institute (ANSI) N45.2.12, N45.2.13, N45.2.23, and N18.7, as committed in Sections 1.8 and17.3 of the (U)FSAR, establishes the requirements for compliance with the associated portions of 10CFR50, Appendix B.

4.4 APPROVAL OF VENDORS

Where procurement documents require the vendor to implement a quality assurance (QA) program that complies with 10CFR50, Appendix B, the vendor's program shall be approved by CP&L before issuance of the purchase order (PO) or contract. Procurement from other nuclear plant facilities licensed for construction or operations by the NRC are exempt from this requirement. The Procurement, Dedication, and Vendor/Equipment Services Unit (PD&V/ES) of OESD shall maintain a list of approved vendors (ASL). The qualification of a vendor's QA program shall be based on an evaluation of the adequacy of the program compliance with the applicable requirements of 10CFR50, Appendix B, for the type of items or services supplied.

4.4.1 Vendor qualification shall be documented by one or more of the following methods:

4.4.1.1 The vendor's QA capabilities as determined by a direct survey/audit of the vendor's facilities and personnel and the implementation of the QA program.

4.4.1.2 Evaluation of the vendor's history of providing a product which performs satisfactorily in actual use. The following information should be considered:

4.4.1.2.1 Experience of users in identical or similar products of the same prospective vendor.

4.4.1.2.2 CP&L's records that have been accumulated in connection with previous procurement actions and product operation experience. Historical data should be representative of the vendor's current capability. If there has been no recent experience with the vendor, or the vendor is a new supplier, the prospective vendor shall be requested to submit information on a similar item or service for evidence of his current capabilities.

4.4.1.3 Evaluation of the vendor's current quality records supported by documented qualitative and quantitative information which can be objectively evaluated. This would include review and evaluation of the vendor's QA program manual and procedures, as appropriate, to ensure that the applicable requirements of 10CFR50, Appendix B, are appropriately applied and effectively implemented.

4.4.1.4 Verification that the vendor holds an active CERTIFICATE OF AUTHORIZATION from the American Society of Mechanical Engineers (ASME) to supply or manufacture item(s) described in the procurement document.

4.4.1.5 Evidence that material manufacturers or material suppliers hold an ASME Quality System Certificate (Materials).

4.4.2 Where procurement documents for **commercial grade** items take credit for the vendor's **commercial grade** quality assurance program for verifying any critical characteristics for acceptance, the vendor's program shall be approved by CP&L before issuance of the **purchase order**. Procurement from other nuclear plant facilities licensed for construction or operations by the NRC are exempt from this requirement. PD&V/ES shall maintain a list of approved **commercial grade** vendors. The qualification of a vendor's **commercial grade** QA program shall be based on an evaluation of the ability of the program and/or manufacturing processes to provide reasonable assurance that the critical characteristics for acceptance are verified for the items or **services** supplied.

4.4.2.1 Qualification of a vendor shall be documented by one or more of the following methods:

4.4.2.1.1 The vendor's QA and/or manufacturing process control capabilities as determined by a direct survey/audit of the vendor's facilities and personnel and the implementation of the QA program and/or manufacturing process controls.

NOTE: **Commercial grade** surveys/audit of distributors should not be employed alone unless the distributor has a **commercial grade** quality assurance program capable of verifying the applicable critical characteristics. Otherwise, a survey/audit of the distributor must also include a survey/audit of the original part manufacturer(s).

4.4.2.1.2 Evaluation of the vendor's history of providing a product which performs satisfactorily in actual use. The following information should be considered:

- Monitored performance of the item through user historical performance and evaluated results.
- Industry product tests.
- Manufacture of the item(s) to national codes and standards.
- Industry data bases on item performance.

The item performance record is required to be specific to the item and directly applicable to the item's critical characteristics and intended **safety-related** applications.

The vendor history method alone is not acceptable for dedication of **commercial grade** items and must be used in combination with supplier survey/audit, source inspection or special tests and inspection methods, provided the evaluation results are acceptable and the following are performed:

- **Receipt inspection** in accordance with a documented **receipt inspection** plan.
- Periodic revalidation of performance evaluation bases and results.

4.4.3 The evaluation of the adequacy of the vendor's QA program shall be performed by PD&V/ES and updated annually. Records supporting the vendor's listing on the ASL will be maintained as described in procedures.

4.5 ITEMS AND SERVICES PROCUREMENT BY PURCHASE ORDER

This subsection applies to the procurement by POs of items and services. Activities associated with the procurement process, including document preparation, reviews, approval, and changes to the PO, shall be controlled by procedures.

4.5.1 Initiation of purchase requisition (PR)

Procurement shall be initiated by preparation of a PR. The term "PR" shall apply to any document which initiates the process of procurement and is subject to the controls of this section. This may include manually processed or computer-generated documents. For computer-generated PRs, the technical and QA requirements applicable to the item being procured may be coded and recorded in a controlled data base. Text associated with technical and QA requirement codes shall be retrievable or included with the PR, as appropriate.

4.5.2 General requirements for PRs

PRs shall include a description of the item or service and delivery instructions. The quality class shall be specified on each PR.

4.5.3 Requirements for PRs, except **Commercial Grade**

This subsection applies to the procurement of items from a vendor who is required to implement an approved QA program that complies with the applicable requirements of 10CFR50, Appendix B, for the purpose of assuring quality and compliance with the order requirements.

4.5.3.1 Technical and documentation requirements

The PR shall specify:

4.5.3.1.1 The design technical requirements that adequately specify each item requirement imposed on the vendor. The requirements shall be established by:

4.5.3.1.1.1 Reference to applicable codes/standards, regulations, approved drawings, approved specifications, or other controlled documents including appropriate revision, editions, and addenda.

OR

4.5.3.1.1.2 Included in the PR based on requirements established in controlled engineering documents.

OR

4.5.3.1.1.3 A combination of both the above.

4.5.3.1.2 Fabrication requirements and controls essential to the item's final quality, as appropriate.

4.5.3.1.3 Required vendor inspection and tests, as appropriate.

4.5.3.1.4 Vendor shelf-life limitations specified by the vendor, if applicable, unless they are to be established by CP&L.

4.5.3.1.5 Packaging and shipping requirements, as appropriate.

4.5.3.1.6 Documentation submittal requirements including schedule for submittal and any limitations on work progression related to their review, if appropriate.

4.5.3.1.7 As applicable, record retention requirements by the vendor including type of records and retention time if the vendor is to retain custody of final QA Records.

4.5.3.2 QA requirements

The PR shall require the vendor to:

4.5.3.2.1 Implement a documented QA program that complies with the applicable requirements of 10CFR50, Appendix B, and is approved by CP&L.

4.5.3.2.2 Allow CP&L and other parties authorized by CP&L right of access to the vendor's facilities and QA Records for source inspection and QA audits.

4.5.3.2.3 Incorporate appropriate QA program requirements in subtier procurement documents.

4.5.3.2.4 Notify CP&L of nonconformances to the order requirements which consist of one or more of the following. Documented vendor notification shall include the vendor's recommended disposition and technical justification.

4.5.3.2.4.1 Technical or material requirement is violated.

4.5.3.2.4.2 Vendor documents approved by CP&L are violated.

4.5.3.2.4.3 Nonconformance(s) cannot be corrected by continuation of the original manufacturing process or by **rework.**

4.5.3.2.4.4 The item does not conform to the original requirements even though the item can be restored to a condition such that the capability of the item to function is unimpaired.

4.5.3.3 Waiver of requirement for vendor-approved QA program.

The requirement for a vendor to have a CP&L-approved QA program may be waived under the following circumstances:

4.5.3.3.1 For procurement of replacement or spare parts where the original design, fabrication inspection, and test requirement are adequate; the vendor is the original equipment manufacturer; and the original specification or order did not require the vendor's QA program to be approved.

4.5.3.3.2 For procurement of items and **services** where CP&L is substituting its QAP, in whole or in part, in place of the vendor's normal controls. Such circumstances will require the requisition, as a minimum, to reference the documents and methods (e.g., **Engineering evaluations**, ESRs, modifications, etc.) which will be used to invoke CP&L's QAP on the applicable activities such that appropriate 10CFR50, Appendix B, controls are assured.

4.5.3.3.3 Special procurement in accordance with Section 4.10.

4.5.3.3.4 When there has been no recent experience with the supplier, the prospective supplier shall be evaluated by reviewing appropriate procedures, instructions, and specifications on a similar item (or service) for evidence of current capabilities.

The requisition shall include, as appropriate, that the vendor have approved procedures for the specific work being performed, that personnel be qualified to perform the specific activity, that **calibration** of instrumentation shall be traceable to nationally recognized standards, and that the activity being performed shall be evaluated by CP&L, or its agent, at the supplier facility. These activities shall be in accordance with Section 4.8.

4.5.4 Determination of **Commercial Grade** Items.

An evaluation shall be performed in accordance with approved procedures to determine the applicability of using **commercial grade** items for the intended **safety-related** application(s) and should include:

4.5.4.1 A confirmation that the item meets the criteria as defined in the definition section.

4.5.4.2 Identification of the critical characteristics of the item to be verified. Critical characteristics are identifiable and measurable attributes/variables of the item which, once verified, provide reasonable assurance that the item received is the item specified on the PR.

4.5.4.3 Identification of methods to be employed for verification of critical characteristics including **acceptance criteria**.

4.5.4.4 Identification of technical and QA requirements sufficient to assure the product requirements are clearly specified to vendor.

4.5.5 Requirements for **Commercial Grade** PRs.

Establishment of technical requirements, quality requirements, documentation requirements, dedication methodologies, etc., for **commercial grade** items shall be performed in accordance with approved procedures.

4.5.6 **Measuring and test equipment (M&TE) calibration** service PRs

A PR for **M&TE calibration services** shall include the following requirements:

4.5.6.1 Description of the **calibration** service being requested including calibration ranges; accuracy and repeatability requirements, where appropriate; and any restrictions on service, if warranted.

4.5.6.2 Traceability of **calibrations** to a nationally recognized standard. Where no nationally recognized standard exists, the method and standard used in performing the **calibration** shall be documented by the vendor.

4.5.6.3 Written notification to purchaser when equipment is found out of **calibration**, including the amount of out of **calibration**.

4.5.6.4 Specific documentation to be submitted.

4.5.7 PR review and approval

4.5.7.1 PRs shall be reviewed prior to release for purchase by qualified individuals knowledgeable in technical and QA requirement considerations to assure that the PRs are adequate for the intended item. The responsible organization(s) designated to perform the review(s) shall be established in procedures.

This review shall determine, as appropriate, that:

4.5.7.1.1 The PR has been prepared in accordance with procedures.

4.5.7.1.2 Adequate technical and quality requirements are specified.

4.5.7.1.3 References and attachments are appropriate for the intended item.

4.5.7.1.4 Adequate QA documentation requirements have been specified.

4.5.7.2 After satisfactory completion of the review(s), the reviewer(s) shall document concurrence in a manner specified in procedures. The PR is considered approved within the context of this QAP upon satisfactory completion of the review(s).

4.5.8 PR changes

Once the initial PR has been reviewed by the reviewer(s), any changes to the technical and quality requirements including references or attachments of the PR shall be reviewed for adequacy. The review shall be equivalent to that performed on the initial PR and performed by the organization(s) assigned this responsibility in approved procedures.

4.5.9 Request for quotation/proposal

4.5.9.1 When required or requested, a request for quotation (RFQ) shall be prepared from an approved PR and issued to selected bidders. Items on a PR may be regrouped in the RFQ; however, each item and its requirements shall remain unchanged in transcription from the PR to the RFQ.

4.5.9.2 Quotations received with exceptions to the technical or quality requirements of the PR shall be evaluated by the department initiating the PR. Changes to the technical or quality requirements resulting from acceptance of vendor exceptions shall be translated into a change to the PR per Section 4.5.8 prior to issuance of a PO to the selected vendor.

4.5.10 POs

4.5.10.1 POs shall be prepared from an approved PR. When required by the PR, the vendor's QA program shall be approved by CP&L prior to issuance of the PO.

4.5.10.2 POs may be issued to agents or distributors of a vendor. In such cases, the PO shall include the name and location of the approved vendor. The agent or distributor does not have to appear on the **Approved Suppliers List (ASL)** provided that the items are shipped directly from the approved vendor.

4.5.10.3 PR items may be regrouped in the PO to facilitate procurement; however, each item and its requirements shall remain unchanged in the transcription from the PR to the PO. Validation of the accuracy of the PO against the approved PR will be performed in accordance with approved procedures.

4.5.10.4 Any exceptions to the PO received from the vendor involving the technical or quality requirements of the order shall be forwarded to the appropriate materials acquisition organization for evaluation. Any exceptions granted to technical and quality requirements shall be translated into a change to the initial PO as per Section 4.5.8.

4.6 PROCUREMENT BY CONTRACT

This subsection applies to the procurement by contract of items and services. Activities associated with the contract process including document preparation, review, approval, and changes to the contract document shall be controlled by procedure.

4.6.1 Contract Requisition

A contract will be generated from an approved contract requisition. The requisition shall be reviewed, prior to release, by qualified individuals knowledgeable in technical and QA requirement considerations to assure the requisition is adequate for the intended scope of work.

The review shall verify, as appropriate, that:

4.6.1.1 The requisition has been prepared in accordance with procedures.

4.6.1.2 Adequate technical and QA requirements are specified.

4.6.1.3 References and attachments are appropriate for the intended work scope.

4.6.1.4 When required, the vendor's QA program has been approved by CP&L for the scope of work. The vendor's QA program does not have to be approved prior to issuance of a RFQ.

4.6.1.5 Appropriate QA documentation requirements have been specified.

4.6.1.6 The record of reviews shall be retained as a QA record.

4.6.2 General requirements

Each **contract** shall specify:

4.6.2.1 Vendor's name and address.

4.6.2.2 Location where the work will be performed.

4.6.2.3 Scope of work requested.

4.6.2.4 QA requirements.

4.6.2.5 Special conditions the vendor shall comply with to be able to perform the work.

4.6.2.6 Documentation submittal requirements.

4.6.2.7 Material requirements, if applicable.

4.6.3 CP&L designated representative

The organization requesting a **contract** shall identify a CP&L designated representative, and the individual shall be specified by name in the **contract**. Changes in the assigned CP&L designated representative should be communicated to the vendor in writing. This individual will function as the **contract** administrator and, as such, shall be a person knowledgeable of the:

4.6.3.1 Work scope requested.

4.6.3.2 Technical and quality requirements of the work.

4.6.3.3 Responsibilities of a designated representative in monitoring the vendor, handling changes in work scope, and processing any documentation resulting from the service.

4.6.4 Technical requirements

The work scope of the **contract** shall clearly specify technical requirements which govern the work and establish an interface process for transmittal of requirements not identified in the work scope. Consideration shall be given to the following areas when developing the **contract** work scope description:

4.6.4.1 Applicable codes, standards, regulations, etc.

4.6.4.2 Methods of interface between vendor and CP&L for transmittal of **design inputs** and outputs, documents for review and approval, and other applicable design information.

4.6.4.3 Applicable specifications, drawings, or documents which shall be invoked.

4.6.4.4 Submittals required for CP&L review and approval with any limitations on work progression related to their approval by CP&L.

4.6.4.5 Reference to existing interface documents between vendor and CP&L if the document will be used to govern the interface.

4.6.4.6 For **M&TE calibration services**, the requirements of Section 4.5.6 shall apply.

4.6.5 QA requirements

4.6.5.1 The **contract** shall identify whether the work will be performed under the controls of the vendor's QA program and resultant procedures or under the control of CP&L's QAP with work performed to CP&L procedures. For work performed under CP&L's program, the CP&L designated representative will be responsible for identifying applicable procedures and making available a copy of these procedures and the QAP to the vendor or vendor's personnel.

4.6.5.2 For vendors implementing their QA program the **contract** shall required the vendor to:

4.6.5.2.1 Implement a documented QA program that complies with the applicable requirements of 10CFR50, Appendix B, and is approved by CP&L prior to the initiation of any work.

4.6.5.2.2 Allow CP&L and/or other parties authorized by CP&L right of access to the vendor's facilities, work areas, and records for the purposes of audits, surveillances, and inspections.

4.6.5.2.3 Incorporate appropriate QA requirements of the **contract** in subtier procurement documents.

4.6.5.2.4 Notify CP&L's designated representative of any nonconformances to the **contract** or any CP&L-approved document that results in one or more of the following. Documented vendor notification shall include the vendor's disposition and technical justification.

4.6.5.2.4.1 Technical or material requirement is violated.

4.6.5.2.4.2 Vendor documents approved by CP&L are violated.

4.6.5.2.4.3 Nonconformance(s) that cannot be corrected by continuation of the original manufacturing process or by **rework**.

4.6.5.2.4.4 The item or work product does not conform to the specified requirements even though the item or work product can be restored to a condition such that the capability of the item or work product is unimpaired.

4.6.5.3 Documentation submittal requirements to CP&L shall be specified or referenced. If any QA Records are to be retained by the vendor for CP&L, the **contract** shall specify the records to be retained, retention period for each record, and appropriate storage requirements.

4.6.6 **Contract** reviews

The **contract** shall be reviewed prior to release to assure the **contract** requisition requirements have been incorporated in accordance with approved procedures.

4.6.7 Contract/contract requisitions (CR) changes

Once the contract/CR has been reviewed by the reviewer(s), any changes to the technical and QA requirements including the scope, references, and attachments shall be reviewed for adequacy. The review shall be equivalent to that performed on the initial contract/CR and performed by the organization(s) assigned this responsibility in approved procedures.

4.6.8 Request for quotation/proposal

4.6.8.1 When required or requested, a RFQ shall be prepared and issued to selected bidders. The RFQ shall be reviewed for compliance with the contract requisition prior to issuance.

4.6.8.2 Quotations received with exceptions to the technical or quality requirements of the RFQ shall be evaluated by the organization responsible for the work. Changes to the technical or quality requirements resulting from acceptance of vendor exceptions shall be reflected in the final contract with final reviews performed per Section 4.6.7.

4.6.8.3 If the vendor's QA program is required to be approved by CP&L, qualification of the vendor's program shall be approved per Section 4.4 prior to issuance of the formal contract.

4.6.9 Approved contract

A copy of the approved safety-related, FP-Q, RW-Q, 10CFR71-Q, 10CFR72-Q, or Q Class B contract including attachments shall be provided to the Procurement, Dedication and Vendor/Equipment Services Unit and the CP&L designated representative, except for contracts controlled by the Nuclear Fuel Management & Safety Analysis Section of NED.

Copies of all safety-related contracts shall be retained as QA Records.

4.7 DISPOSITION OF VENDOR NONCONFORMANCES

Nonconformances reported by a vendor and the recommended disposition shall be evaluated by the responsible individual/group within the initiating department of the procurement documents. Approval of the vendor's disposition or CP&L-selected alternate disposition shall be provided to the vendor in writing. A copy of the nonconformance report and CP&L's disposition approval shall be retained as QA Records.

4.8 VERIFICATION OF VENDOR ACTIVITIES

4.8.1 Verification activities shall be a function of relative importance, complexity, and quantity of the item or service being procured and the vendor's past quality performance.

4.8.2 Verification activities shall be documented and executed by a surveillance or audit plan for an awarded **PO** or **contract**. These plans shall include the following provisions as required:

4.8.2.1 For in-process and final source surveillance of vendor's product or activities at the vendor's facilities or facilities of subtier vendors. The source surveillance shall be documented in a report and a copy of all reports sent to the appropriate plant's materials acquisition organization.

4.8.2.2 For surveillance or audits, as necessary, to assure that vendor planning and execution of work at the work location is controlled in accordance with the procurement document requirements.

4.9 MATERIAL UPGRADING

Material may be upgraded providing the item complies with the specification or applicable requirements for the intended application. Upgrades shall be performed by documented **engineering evaluations** or in accordance with criteria established in a procedure. The results of the evaluation of intended application requirements to the actual attributes of the item to be upgraded shall be documented and auditable. The upgrade evaluation shall assess the adequacy of QA data. The upgrade process shall address the need for **receipt inspection** based on the circumstances of the situation. Upon approval, the upgrade evaluation, along with the relevant vendor documentation and **receipt inspection** package, shall form the equivalent to a **PO** for item **traceability**.

4.10 SPECIAL PROCUREMENT

Items and **services** may be procured from other nuclear plant facilities licensed for construction or operations by the NRC that are not on CP&L's ASL so long as the technical and quality attributes of the item or service comply with the necessary requirements of specifications or intended application. The methods to be used for procurement in such special cases shall be set forth in procedures and shall provide adequate controls to assure technical and quality requirements are met.

4.11 VENDOR AUDITS

Audits, including preaward and periodic audits, are performed at the facilities of contractors providing material, parts, components, and **services** to CP&L. Preaward audits are performed when alternate methods of qualification are not sufficient to support the initial qualification of contractors. Periodic audits are performed when the results of periodic evaluations and the status and nuclear safety importance of items and **services** indicate that an audit is required. Audits shall be planned, conducted, and reported in accordance with procedures.

4.11.1 Personnel performing audits shall be appropriately trained, indoctrinated, and qualified to plan, conduct, and report audits.

4.11.2 Personnel qualified as lead auditors shall be responsible for: leading audits; audit notification, audit agenda preparation; auditor assignments, checklist preparation, auditor orientation, leading and scheduling preaudit and postaudit meetings, audit report preparation, and audit follow-up action.

4.11.3 Audits shall be conducted using checklists as guidelines. The checklists shall be prepared to cover the scope of the contractor's QA program for the desired items and **services**.

4.11.4 Audit planning shall include a review of previous audit reports of contractors. Unresolved **conditions adverse to quality** from previous audits shall be documented on the checklist.

4.11.5 Audits are performed to evaluate contractors' abilities to comply with the QA requirements of CP&L's procurement documents. Audits include an evaluation of QA practices, procedures, and instructions; assessment of contractors' QA programs implementation; review of work activities and processes; and review of quality-related documents and records.

4.11.6 Audit reports will include any identified findings, concerns, comments and, when appropriate, recommended corrective action. Audit reports will be maintained as QA Records.

4.11.7 Audit reports will be distributed to the appropriate CP&L and contractor management.

4.11.8 The contractor shall be requested to respond to the **conditions adverse to quality**. The request shall indicate the period of time required to respond to the **conditions adverse to quality**.

4.11.9 The lead auditor is responsible for the evaluation of corrective action. The results of this evaluation shall be documented. Follow-up audits will be planned and conducted to verify implementation of corrective action when necessary.

4.11.10 A system shall be maintained which indicates the status of nonconformances identified during audits.

5.0 MATERIAL AND EQUIPMENT CONTROL

5.1 SCOPE

This section establishes the requirements for controlling items which by definition include material, parts, and components. It includes requirements for the verification of identification, inspection status, handling, and storage of items. Material and equipment control assures that items used or installed in nuclear plants comply with the QAP, regulatory requirements, applicable technical requirement, and codes and standards.

5.2 REGULATORY COMMITMENTS

This section used in conjunction with Regulatory Guides 1.33, 1.38, and 1.146, and American National Standards Institute N45.2.2, N45.2.23 and N18.7, as committed in Sections 1.8 and 17.3 of the (U)FSAR, establishes the requirements for compliance with the associated portions of 10CFR50 Appendix B.

5.3 MATERIAL ACCEPTANCE

5.3.1 Personnel responsible for receiving material shipments shall verify that items do not exhibit shipping damage.

5.3.2 The determination of the need for, methods to be used, and **acceptance criteria** for material acceptance shall be determined prior to acceptance.

5.3.3 For acceptance by **receipt inspection**, the **acceptance criteria** shall be determined in accordance with guidance set forth in procedures. The following shall be considered in establishing the **receipt inspection** requirements and **acceptance criteria**:

5.3.3.1 Identification, marking, and labeling

5.3.3.2 Packaging requirements

5.3.3.3 Cleanliness

5.3.3.4 Physical attributes and electrical characteristics

5.3.3.5 Special inspections

5.3.3.6 Special environmental conditions (such as inert gas atmospheres, specific moisture content, and temperature levels)

5.3.3.7 Statistical sampling methods that may be used for **receipt inspection** of groups of similar items

5.3.3.8 Documentation required and the review requirements (such as legibility and completeness)

5.3.4 Results of the receipt inspection shall be documented. If the items and supporting documentation are found to be acceptable, the items shall be identified as acceptable. Acceptable items shall be released for storage or installation. Acceptable items shall indicate information that will provide **traceability** to procurement documents.

5.3.5 When necessary, source inspection shall be performed to verify that contractors have performed manufacturing, testing, and inspecting of items in accordance with the requirements of procurement documents. A **quality release** shall be prepared which authorizes contractors to ship items that are acceptable. Personnel responsible for **receipt inspection** shall verify that source-inspected items are marked, labeled, and traceable to documentation packages and that documentation packages include records required by procurement documents, as a minimum. Any inspections or tests required per Section 5.3.3 which are not performed during source inspection shall be performed by appropriately qualified personnel upon receipt of the item(s) by CP&L.

5.3.6 Measures shall be taken to assure that items, including those subdivided, are properly identified from the time of receipt to the point of installation. Identification markings shall be applied in a manner that will not affect the function of the item.

5.3.7 The required identification and status markings shall be retained with the items or records traceable to the items. The identification of each item shall be included in the record of assembly or installation. For uninstalled items in work areas, status indicators such as markings, tags, or notations on work control documents shall be applied to show the latest status.

5.3.8 When items or required documentation for the items do not conform to requirements, the items shall be identified as nonconforming. Nonconforming items will be identified and controlled until proper disposition is made.

5.3.9 A receipt inspection documentation package shall be prepared and will include or reference for **traceability** the procurement documents, receipt inspection report, special inspection reports, certifications, plant-generated documents, and contractor-furnished documents. The documentation package shall be retained as QA Records.

5.4 CONDITIONAL RELEASE OF NONCONFORMING ITEMS

5.4.1 A **conditional release** may be initiated to permit progression of work involving a nonconforming item awaiting resolution. The request shall contain the necessary justification and limitations prior to review and approval.

5.4.2 If reasonable control and **traceability** can be maintained, a **conditional release** may be issued to permit limited use, installation, or testing of an item. The item shall be clearly tagged or otherwise traceable to show the status and the permitted actions.

5.5 MATERIAL STORAGE AND RELEASE

5.5.1 Items shall be stored in designated storage areas. Identification tags or marks and the inspection status shall be retained on items or on records which are traceable to the items. Release of accepted items shall be controlled to prevent damage, deterioration, or unauthorized storage and release.

5.5.2 Nonconforming items shall be segregated and stored in a designated storage area, when practical, to await disposition. When it is not practical to segregate nonconforming items, they shall remain tagged and held in storage areas until properly dispositioned.

5.5.3 Items shall be controlled to assure that they are properly dispositioned at the end of their specified shelf life or qualification period.

5.5.4 The appropriate handling equipment shall be provided and controlled to assure safe and adequate handling. Designated equipment shall be periodically inspected and tested to criteria established in procedures.

5.6 STORAGE INSPECTION PROGRAM

5.6.1 Inspection shall be maintained over items in storage areas. This program shall include:

5.6.1.1 Periodic inspections to assure that items are properly controlled, maintained, and protected. Inspections shall be documented.

5.6.1.2 The identification and control of nonconforming items until proper disposition is made.

6.0 PROCEDURES AND DRAWINGS

6.1 SCOPE

This section establishes requirements for preparation, review, approval, and control of procedures and drawings for **activities affecting quality.**

6.2 RESPONSIBILITY

Each organization performing **activities affecting quality** is responsible for ensuring this section is properly implemented in their area of responsibility.

6.3 REGULATORY COMMITMENTS

This section utilized in conjunction with Regulatory Guide 1.33 and American National Standards Institute N18.7 as committed in Sections 1.8 and 17.3 of the (U)FSAR, establishes the requirements essential to comply with the associated portions of 10CFR50 Appendix B.

6.4 PROCEDURES AND DRAWINGS

6.4.1 Appropriate procedures shall be developed for the preparation, review, approval, and issue of procedures and drawings.

6.4.2 The accomplishment of **activities affecting quality** shall be in accordance with approved procedures and/or drawings which are appropriate to the circumstances.

6.4.3 Procedures and drawings shall include the following elements in their content as applicable:

6.4.3.1 Prerequisites.

6.4.3.2 Precautions.

6.4.3.3 Qualitative/quantitative **acceptance criteria**.

6.4.3.4 Inspection points.

6.4.3.5 Checklists.

6.4.4 Measures shall be established to assure that procedures for **activities affecting quality** are reviewed prior to issue to ensure appropriate criteria have been specified. Appropriate criteria to be met include the Final Safety Analysis Report, Technical Specifications, operating license, commitments to regulatory agencies, regulations, and the Quality Assurance Program.

6.4.5 The approved, current revision, of procedures and drawings shall be strictly followed in accomplishment of work and shall be available at the work location where the activity will be performed (when applicable) prior to commencing work. Measures shall be established to assure continued use of approved, current revision documents.

6.4.6 Provisions shall be made for the review of procedures (Those procedures described in Reg. Guide 1.33) as required by plant commitment by an individual knowledgeable in the area affected to determine the need for changes.

7.0 INDOCTRINATION AND TRAINING

7.1 SCOPE

This section establishes the requirements for providing indoctrination and training for personnel performing **activities affecting quality**.

7.2 RESPONSIBILITY

Each department head responsible for **activities affecting quality** shall ensure the requirements of this section are implemented within his area of responsibility.

7.3 REGULATORY COMMITMENTS

This section utilized in conjunction with Regulatory Guides 1.8, 1.33 and 1.58, American National Standards Institute ANSI N3.1, ANSI N18.7 and ANSI N45.2.6 as committed in Sections 1.8 and 17.3 of the (U)FSAR, establishes the requirements essential to comply with the associated portions of 10CFR50 Appendix B.

7.4 GENERAL REQUIREMENTS

7.4.1 Training procedures shall be developed and implemented which encompass training, retraining, qualifications, and certifications of qualification, as required. Scope, method and objective of indoctrination and training shall be documented.

7.4.2 Personnel, both on-site and off-site, within the Carolina Power & Light (CP&L) organization performing **activities affecting quality** shall be indoctrinated and trained such that they are knowledgeable in the applicable quality-related procedures and requirements. Provisions to assure that these personnel remain proficient shall be made. The indoctrination and training program assures that:

7.4.2.1 Personnel responsible for performing **activities affecting quality** are instructed as to the purpose, scope, and implementation of the quality-related manuals and procedures.

7.4.2.2 Personnel performing/verifying **activities affecting quality** are trained and qualified in the principles and techniques of the activity being performed.

7.4.2.3 Proficiency and knowledge of personnel performing **activities affecting quality** is maintained by retraining, reexamining, and/or recertifying.

7.4.2.4 Formal training and qualification programs require documentation which includes objective, content of program, attendees, and dates of attendance.

7.4.3 Temporary personnel, both CP&L and/or contractors, are also trained in the categories in Section 7.4.2 to the extent necessary to assure safe execution of their duties.

7.4.4 Personnel within the operating organization performing duties of a licensed operator are indoctrinated, trained, and qualified as required by 10CFR55.

7.4.5 When specified in procedures, personnel performing welding, weld repair, brazing, heat treating, or other special processes shall be qualified and certified as set forth in the American Society of Mechanical Engineers Code, Section IX, and/or other applicable requirements.

7.5 TRAINING RECORDS

Records of personnel qualification and certification shall be maintained as required by procedures.

7.6 QUALIFICATION AND CERTIFICATION OF INSPECTION AND NONDESTRUCTIVE EXAMINATION (NDE) PERSONNEL

7.6.1 Personnel performing inspection, review, examination and testing, evaluations of inspection data, and reporting of inspection and test results will be qualified and certified, based on CP&L's commitment to Regulatory Guide 1.58.

7.6.2 Prior to certification, NDE personnel shall have satisfactorily passed an examination administered under the jurisdiction of a certified Level III in accordance with Recommended Practice SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing." Authority to certify CP&L Level III NDE personnel will be specified in CP&L's NDE procedures.

8.0 CALIBRATION CONTROL

8.1 SCOPE

This section sets forth the requirements to establish those measures which will assure that measuring and test equipment (M&TE) is properly controlled and calibrated.

8.2 RESPONSIBILITY

Each organization has the responsibility for the calibration of the items in the Calibration Program. As a minimum, the following types of items shall be included in the Calibration Program:

8.2.1 Instruments and control equipment required to be calibrated by the plant Technical Specifications

8.2.2 Instruments and equipment used to verify data points required by the plant Technical Specifications

8.2.3 Active safety-related instrumentation

8.2.4 Special tools: e.g., torque wrenches, micrometers, etc.

8.2.5 Portable measuring and test equipment (P-M&TE)

8.2.6 Calibration standards

8.2.7 Nondestructive examination (NDE) equipment utilized for NDE examinations/inspections. (NDE instruments used by NDE are not traceable to NIST or any other nationally recognized standard and do not fall under the chart depicted in Section 8.4.2).

NOTE: UT calibration blocks (used to set up ultrasonic NDE equipment); hardness test blocks (used to verify proper operation of the QC Receipt Inspection portable hardness testers); and Alloy Analyzer Test Specimens (used to verify proper operation of the QC Receipt Inspection Alloy Analyzers), are not considered as M&TE under CP&L's QAP, and as such, are not included in the calibration program. This does not preclude controls necessary to assure that each block meets applicable requirements prior to their initial release for use.

The responsibility for carrying out the requirements of this section shall be established by the appropriate section manager in procedures. The following sections define the structure within which **calibration** shall be controlled.

8.3 REGULATORY COMMITMENTS

This section utilized in conjunction with (U)FSAR Section 17.3 establishes the requirements essential to comply with the associated portions of 10CFR50 Appendix B.

8.4 GENERAL

8.4.1 **Calibration** frequency

For those instruments and devices required to be calibrated by the Technical Specifications, the frequency shall be at least as frequent as the Technical Specification frequency. Special tools shall be calibrated at specified frequencies or prior to use. Frequency of **calibration** for the other items in the **Calibration** Program shall be based upon one or more of the following:

8.4.1.1 Required accuracy

8.4.1.2 Purpose

8.4.1.3 Degree of usage

8.4.1.4 Stability characteristics

8.4.1.5 Other conditions affecting the measurement

8.4.1.6 Manufacturer's recommendation

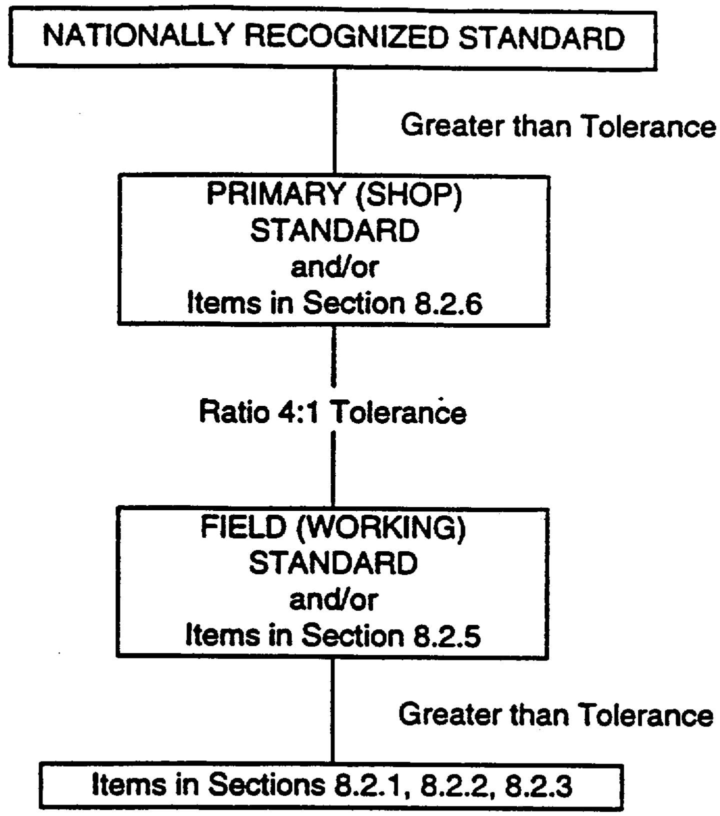
8.4.1.7 Governing Codes and Specifications

Special **calibrations** shall be performed when the accuracy of either installed or calibrating equipment is questionable

8.4.2 **Calibration** standards

Items in the **Calibration** Program shall have a known valid relationship to a nationally recognized standard; or where national standards do not exist, the basis for the **calibration** shall be documented.

The relationship between items in the Calibration Program and the devices that they are calibrated to shall be:



Equipment used both for **calibration** of installed instruments and to calibrate other standards shall be considered as **P-M&TE.**

Special tools (8.2.4) shall be calibrated to within the accuracy of the manufacturer's guarantee. Standards used in the **calibration** of special tools shall have a relationship to a calibrating standard which is equal to or greater than the accuracy of the special tool calibration standard.

In those cases where the given accuracy is not achievable or practicable, an evaluation shall be performed and documented to justify acceptability of the **calibration** accuracy in question.

8.5 CALIBRATION PROGRAM

A **Calibration** Program shall be developed and maintained up to date. The program will cover the type of equipment identified in Section 8.2 and, as a minimum, the program shall include:

8.5.1 Provisions for the review and approval of **calibration** procedures or instructions to include the review and approval of any vendor technical manual/document used in lieu of step-by-step directions for **calibration.**

NOTE: Procedures used by vendors on the **Approved Supplier's List** for **calibration services** have been reviewed as part of the original qualification. For specific applications, requesting organizations should ask for and review **calibration** procedures prior to use. This section allows a CP&L organization to incorporate vendor instructions (after suitable review) instead of writing their own should they desire to perform the **calibration**.

8.5.2 Provisions to properly control **calibrations** performed by vendors and contractors. The vendor/contractor shall certify the **traceability** in accordance with the procurement document.

8.5.3 Provisions for performing the required **calibration** at the prescribed frequencies.

8.5.4 Provisions for the establishment and maintenance of a master schedule reflecting the status of planned **calibrations**.

8.5.5 Unique identification of each item in the **Calibration** Program so that **traceability** to the **calibration** data is possible. This identification shall be clearly visible on or with the equipment; e.g., Test Gauge 8, 2-CAC-AQH-I264.

8.5.6 Status of **calibration** for **M&TE** is provided for through the use of tags, stickers, labels, routing cards, computer programs, or other suitable means. The status indicators indicate the date recalibration is due or the frequency of recalibration.

8.5.7 Provisions as to the action required if **P-M&TE** is out of **calibration**. Such action shall include a documented review to determine the validity of past **calibrations**, measurements, or monitored parameters.

8.5.8 Action to be taken in the case of repetitive out of **calibration** of any **M&TE** and the cause of the out-of-calibration conditions shall be determined. Corrective action such as **repair**, replacing the equipment, or increasing the frequency of its **calibration** shall be taken to prevent recurrence. Identification of the condition, the cause, and the corrective action taken shall be documented and reported to the appropriate foreman/supervisor.

8.5.9 Provisions to provide for the evaluation of the **calibration** data to ensure conformance to **acceptance criteria** by a responsible group or individual.

8.5.10 Provisions to require and document corrective actions required following **calibrations** which do not meet the **acceptance criteria**.

8.5.11 Provisions to document the permission by operating personnel to remove from service installed items to be calibrated.

8.5.12 Provisions for providing the environmental conditions required for the performance of the **calibration** (e.g., location, cleanliness requirements, temperature, etc.).

8.5.13 Provisions to ensure items in the **Calibration** Program that are calibrated by CP&L are calibrated in accordance with procedures. These procedures shall:

8.5.13.1 Identify the item(s) to which it applies.

8.5.13.2 Contain a description of objectives.

8.5.13.3 Contain the **acceptance criteria** that will be used to evaluate the results.

8.5.13.4 Contain prerequisites for performing the **calibration** including any special conditions to be used to simulate normal or abnormal operating conditions.

8.5.13.5 Contain limiting conditions.

8.5.13.6 Specify special equipment or **calibrations** required to conduct the **calibration.**

8.5.13.7 Prescribe the appropriate documentation requirements (e.g., data forms to be used, test data to be recorded, etc.).

8.5.13.8 Contain step-by-step instructions in the degree of detail necessary for performing the **calibration.**

8.5.13.9 Require the recording of:

8.5.13.9.1 **Calibration** date.

8.5.13.9.2 Identification of those performing **calibration.**

8.5.13.9.3 As-found condition.

8.5.13.9.4 As-left condition.

8.5.13.9.5 The standard or other item of **M&TE** used to perform the **calibration** in order to maintain **traceability**, including the **calibration** date and serial number or unique instrument identification number.

8.6 RECORDS

The following documents shall be filed as QA Records:

8.6.1 Out-of-**calibration** documentation.

8.6.2 **Calibration** certificates for reference standards.

8.6.3 Completed **calibration** document(s).

9.0 SURVEILLANCE

This section was deleted with the implementation of the assessment program.

10.0 PLANT OPERATIONS CONTROL

10.1 SCOPE

This section sets forth requirements for the control of plant operations. Plant operations control assures that the quality of installed plant items is not degraded and that the quality of operations is not compromised.

10.2 REGULATORY COMMITMENTS

This section utilized in conjunction with (U)FSAR Section 17.3 establishes the requirements essential to comply with the associated portions of 10CFR50 Appendix B.

10.3 OPERATIONAL CONTROL

10.3.1 Plant operations shall be controlled and conducted in accordance with procedures. These procedures shall be contained within the Plant Operating Manual and shall provide for normal and emergency plant operations including response to abnormal operating conditions and the conditions described in the emergency plan.

10.3.2 Procedures shall be developed and approved which prescribe those measures to be employed when the operating capability of plant items is restricted or limited. These conditions shall be positively identified by tagging or other controls as a means to prevent inadvertent operation or use.

10.4 OPERATING LOGS AND RECORDS

Applicable logs and records shall be maintained to support the reporting and record-keeping requirements of the plant Technical Specifications. Completed logs and records relating to plant operations shall be reviewed for accuracy and completeness and maintained in accordance with Section 14.0.

10.5 INSTALLED PLANT ITEMS

10.5.1 Installed items shall be tested in accordance with procedures.

10.5.2 When an installed item does not conform to test, design, installation specifications, or other requirements, action shall be initiated as set forth in Sections 3.0, 11.0, or 12.0 to correct or replace the item.

10.5.3 Whenever an installed item is inoperative, nonconforming, or malfunctioning, the system and/or item shall be tagged or otherwise identified in accordance with procedures to prevent erroneous operation and, if necessary, inadvertent use. Corrective action shall be documented in accordance with Sections 3.0, 11.0, or 12.0.

11.0 MAINTENANCE CONTROL

11.1 SCOPE

This section sets forth requirements for procedures to be applied at operating plants for maintenance. This section includes requirements for work planning and preparation to assure that maintenance procedures are adequate, prerequisites are met, maintenance is accomplished under suitably controlled conditions, and the functional capability and quality intended by the design is maintained.

11.2 REGULATORY COMMITMENTS

This section utilized in conjunction with (U)FSAR Section 17.3 establishes the requirements essential to control maintenance activities in accordance with the associated portion of 10CFR50 Appendix B.

11.3 MAINTENANCE PROCEDURES

11.3.1 Procedures shall be applied to control maintenance of safety-related items. Maintenance procedures will include the following information, as appropriate:

11.3.1.1 Requirements for indoctrination, training, and skills.

11.3.1.2 Prerequisites for special environments, equipment, tools, and material preparation.

11.3.1.3 Provisions for data collection and reporting.

11.3.1.4 Instructions for documentation of work performed.

11.3.1.5 Requirements for verification of functional capability and quality by inspection, witnessing, examination, testing including specified mandatory holdpoints, and special processes.

11.3.1.6 Quantitative and qualitative criteria for determining that important steps or functions have been satisfactorily accomplished.

11.3.2 Certain maintenance activities which involve skills normally possessed by qualified personnel may not require detailed step-by-step delineation in a procedure. The following types of activities are among those that may not require detailed step-by-step written procedures:

11.3.2.1 Gasket replacement.
11.3.2.2 Troubleshooting electrical circuits.

11.3.2.3 Changing chart or drive speed gears or slide wires on recorders.

11.3.2.4 Packing adjustment or replacement.

It is the responsibility of maintenance supervision to determine if the job is within the skill of the craftsman.

11.4 CORRECTIVE MAINTENANCE

11.4.1 Maintenance activities at the plant which affect the quality of items shall be prescribed in procedures and accomplished as prescribed therein. To meet this requirement, one or more procedures are necessary for:

11.4.1.1 Processes of rework or repair that establish the functional capability or quality of items which require step-by-step delineation.

11.4.1.2 Tests and examinations that determine or verify the functional capability or quality of items.

11.4.1.3 Material protection measures that prevent damage and deterioration of items during handling, storage, and other maintenance activities.

11.4.1.4 Processes, tests, and handling which, unless controlled, may degrade the functional capability or quality of an item.

11.4.2 Work planning.

11.4.2.1 Maintenance programs shall prescribe the preplanning and preparation necessary to ensure the required materials and equipment are available and that work procedures are adequate.

11.4.3 Work execution.

11.4.3.1 Maintenance of items shall be accomplished as prescribed in procedures. Work execution shall include, as a minimum:

11.4.3.1.1 Assurance that prerequisites have been satisfied prior to performance.

11.4.3.1.2 The establishment of prescribed environmental conditions for accomplishing the activity such as adequate cleanliness or an inert atmosphere.
11.4.3.1.3 The use of appropriate equipment.

11.4.3.1.4 Control and accomplishment of special processes by qualified personnel and procedures.

11.4.3.1.5 Provisions for assuring that proper item identification for **traceability** is maintained.

11.4.3.2 Tests shall be conducted when necessary to determine that a new, reworked, or **repaired** item will perform satisfactorily in service.

11.5 PREVENTIVE MAINTENANCE

A preventive maintenance program shall be developed and implemented in accordance with procedures. Preventive maintenance procedures shall consider manufacturer recommendations and plant operating and maintenance experience.

11.6 USE OF MATERIAL

Items used for maintenance shall be in accordance with requirements contained in controlled documents or as specified in a controlled **information management system**. If items cannot be determined to be correct for the intended application, the responsible personnel shall request engineering determination of the adequacy of the item for its intended use. This determination shall be documented and referenced on or attached to the work request, or shall be documented in accordance with approved procedures.

12.0 CONDITIONS ADVERSE TO QUALITY (CATQ) AND CORRECTIVE ACTION

12.1 SCOPE

This section sets forth requirements for reporting, controlling, and dispositioning CATQ.

12.2 RESPONSIBILITY

The responsibility for carrying out the requirements of this section shall be established by procedures.

12.3 REGULATORY COMMITMENTS

This section utilized in conjunction with (U)FSAR Section 17.3 establishes the requirements essential to comply with the associated portions of 10CFR50 Appendix B.

12.4 GENERAL

12.4.1 Personnel are responsible for reporting to their supervision **conditions adverse to quality (CATQ),** discovered as a result of inspections, observations, surveillance, assessments, monitoring, audits, tests, checks, and review of documents.

12.4.2 **CATQ** shall be documented, controlled, and dispositioned in accordance with this section. In-process control documents may be used provided:

12.4.2.1 The condition is corrected before final acceptance of the work.

12.4.2.2 Work does not go beyond a holdpoint to the point of prohibiting the required inspections.

12.4.2.3 The condition does not adversely affect work previously accepted.

12.5 IDENTIFICATION, CONTROL, AND DISPOSITION

12.5.1 Procedures to control **CATQ** shall provide for the following:

12.5.1.1 Identification of nonconforming items by tags, labels, or other appropriate status indicators. This status identification shall remain with the item or in records traceable to the item until the disposition is complete and accepted.

12.5.1.2 Segregation of uninstalled nonconforming items, if practical, to prevent inadvertent use pending proper disposition and/or reinspection.

12.5.1.3 Identification and prompt notification of individuals or organizations responsible for disposition of the condition.

12.5.1.4 Preparation of appropriate documents which identify and describe the condition; provide for proper evaluation; and provide for disposition including reinspection, testing, or other verification to determine the acceptability and proper implementation of the disposition.

12.5.1.5 Verification of the acceptability of **rework/repair** of items by reinspection or testing of the item as originally performed or by methods equivalent to the original inspection or testing methods.

12.5.1.6 Assurance that corrective action appropriate for the condition is determined and scheduled for timely implementation.

12.5.1.7 Initiation of stop-work action in the event an activity or condition presents a threat to personnel safety or plant equipment.

12.5.1.8 Escalation to appropriate levels of management to obtain resolution of disagreements between responsible organizations.

12.5.2 Documents identifying **CATQ** should be reviewed in a timely manner and, if a **CATQ** is confirmed, evaluated for significance and issued. Guidance for this evaluation is provided in Section 12.7. Action to determine appropriate disposition and corrective measures should be initiated.

If the condition is not confirmed, the initiating document shall be canceled, the basis for cancellation noted on the document, and the document shall be placed in a permanent file.

12.5.3 For **significant conditions adverse to quality (SCATQ)**, the root cause of the condition, corrective action, and action to preclude repetition shall be determined, documented, and reported to appropriate levels of management.

12.6 REPAIR OR USE-AS-IS DISPOSITIONS

12.6.1 When it is proposed to **repair** or to **"use-as-is"** a nonconforming item, an **engineering evaluation** shall be conducted before performing the **repair** or using the item.

12.6.2 **Engineering evaluations** to **repair** or **"use-as-is"** shall include documentation verifying the acceptability of the nonconforming item or condition being repaired or used as is.

12.6.3 **Engineering evaluations** shall be performed in accordance with procedures by personnel technically competent in the area of the nonconforming item or condition.

12.7 SIGNIFICANCE EVALUATION GUIDANCE

12.7.1 In determining if a condition is significant, the following criteria should be considered:

12.7.1.1 Adverse condition.

A deficiency, failure, malfunction, deviation, abnormal occurrence, defective material or equipment, or nonconformance in an item or activity which has affected or reasonably could affect:

12.7.1.1.1 Nuclear safety or quality.

12.7.1.1.2 Compliance with other regulations not included in nuclear safety or quality above.

12.7.1.1.3 Personnel safety.

12.7.1.1.4 Plant reliability.

12.7.1.1.5 Commercial concerns.

Adverse conditions may be performance-based, reliability-based, dimensional, material properties, testing, supporting documentation, etc.

12.7.1.2 **CATQ.**

An adverse condition associated with activities affecting the quality of structures, systems, components, programs, procedures, or documents that are subject to this QAP (i.e., Q-List, FP-Q, RW-Q, 10CFR71-Q, 10CFR72-Q, seismically qualified, equipment used to verify technical specification requirements, etc.)

12.7.1.3 **SCATQ.**

A **CATQ** which is important to the degree that action to preclude repetition is deemed appropriate by management. At a minimum, the following **CATQs** shall be considered **SCATQs**:

12.7.1.3.1 Severe or unusual plant transients.

12.7.1.3.2 A significant degradation in the ability of a safety system to perform its function.

12.7.1.3.3 Events involving nuclear safety that had a strong potential to be more severe if different conditions that could be reasonably expected had been present.

12.7.1.3.4 Discovery of a deficiency in an area such as design, analysis, operation, maintenance, testing, procedures, or training that is likely to cause a **SCATQ** in any of the items above.

13.0 ASSESSMENTS

13.1 SCOPE

The Assessment Program is designed to independently evaluate those company functions which have potential Nuclear Safety, Reliability or Quality Implications. The process is performance based using real time observation, interview and review techniques. Included in the program is the determination of each Nuclear Generation Group organization's ability to self evaluate its activities, identify needed improvements and deficiencies, and accomplish the appropriate corrective action.

13.2 QUALIFICATIONS

Personnel performing assessments shall have appropriate training and qualifications. They shall have no direct responsibilities in the areas they assess.

13.3 REGULATORY COMMITMENTS

This section used in conjunction with Regulatory Guides 1.144 and 1.146, American National Standards Institute N45.2.12, N45.2.23 and N18.7 as committed in Section 1.8 and Section 17.3.3.3, Independent Assessments, of the (U)FSAR, establishes the requirements essential for compliance with the associated portions of 10CFR50 Appendix B.

13.4 GENERAL

Assessments will be performed at nuclear plants and CP&L locations where functions affecting safety-related activities are performed. Assessments are regularly scheduled on the basis of the status and safety importance of the activity being performed. Assessments will verify compliance, determine the effectiveness, and evaluate the QAP against performance objectives and QAP requirements. Assessment frequencies are based on the Plant Technical Specifications, (U)FSAR commitments, and QAP Manual requirements and are maintained in commitment matrices by each assessment organization. Assessments shall be planned, conducted, and reported in accordance with procedures.

13.4.1 Assessments will focus on areas of potential improvement based on indicators such as previous assessment data, industry experience, regulatory sensitivity, and input from NGG Management.

13.4.2 The assessment process may include objective evaluation of line management's self assessment effectiveness, inspection of areas, observation of work activities and processes, interviews with personnel and review of documentation and procedures.

13.4.3 Assessment personnel are to maintain their independence of activities for which they are responsible for assessing. This independence should be sufficiently clear so as to avoid even the perception that they are in some way not independent. This, however, should in no way impede or dilute meaningful dialogue between assessors and assessed individuals and organizations.

13.5 ADVERSE CONDITIONS

Adverse conditions identified during the assessment process will be documented in accordance with Section 12.0.

13.6 REPORTS

Assessment results will be documented and distributed to appropriate levels of management.

Each assessment report will include documentation indicating the areas assessed, the appropriate QAP Manual requirement, and the commitment met by the assessment, as appropriate.

13.7 FOLLOW-UP

Follow-up is accomplished to assure that corrective action is taken as a result of the assessment and that deficient areas are reassessed, when necessary, to verify effectiveness of corrective actions.

14.0 QUALITY ASSURANCE (QA) RECORDS AND DOCUMENT CONTROL

14.1 SCOPE

This section establishes the requirements for accumulation, maintenance, and retention of QA Records associated with the nuclear plants and establishes requirements for control of documents relative to **activities affecting quality**. QA Records are those records which furnish documentary evidence of the quality of items and of **activities affecting quality**.

14.2 REGULATORY COMMITMENTS

This section used in conjunction with American National Standards Institute N45.2.9 and N18.7, and Sections 1.8 and 17.3 as committed in the (U)FSAR establishes the requirements for compliance with the associated portions of 10CFR50 Appendix B.

14.3 QA RECORDS

14.3.1 Requirements for implementation.

As required by procurement documents, vendors and contractors shall provide for accumulation and organization of those documents generated in their work that are required to be submitted for retention as QA Records. Upon completion of work by vendors and contractors, these records shall be transferred to Carolina Power & Light (CP&L) or its agent unless, by contractual agreement, the vendor or contractor will retain the records for CP&L for the required periods mutually agreed upon. Collection, storage, and maintenance of records shall be in accordance with commitments to Regulatory Guide 1.88 and/or ANSI N45.2.9 and the plant Technical Specifications.

14.3.2 QA Records accumulation, maintenance, and retention.

14.3.2.1 The responsible individual or organization shall provide for accumulation and organization of documents designated as QA Records in accordance with procedures. Documents shall be identifiable and retrievable.

14.3.2.2 Documents designated for retention shall be transferred in an organized manner for filing as QA Records. The documents shall be checked to verify that they are complete, properly identified, and that required documents are included.

14.3.2.3 QA Records shall be retained as part of the records system at the plant, unless by contractual agreement they are retained by an agent, vendor, or contractor for CP&L. QA Records may be maintained by other CP&L organizations as agents for a nuclear plant department as long as the agreement is covered by an approved document. The QA Records will be filed and maintained in facilities that prevent deterioration or damage to documents and shall be controlled to prevent loss. QA Records shall be organized and filed so that each document is identifiable, retrievable, and shall preclude deterioration of the records. QA Records shall be indexed.

14.4 DOCUMENT CONTROL

14.4.1 Appropriate document control procedures shall be established to identify those individuals or groups responsible for reviewing, approving, revising, and issuing documents.

Examples of documents which are to be controlled include:

14.4.1.1 **Design documents** (e.g., calculations, drawings, specifications, and analyses) including documents relating to computer codes.

14.4.1.2 Procurement documents

14.4.1.3 Quality Assurance Program Manual

14.4.1.4 Maintenance, modification, engineering, and operating procedures

14.4.1.5 Final Safety Analysis Report

14.4.1.6 **Conditions adverse to quality.**

14.4.1.7 Operating license/Technical Specification.

14.4.2 Procedures shall require that changes to documents be reviewed and approved prior to implementation by the same organization that performed the original review and approval or by other designated, qualified responsible organizations.

14.4.3 Procedures shall establish measures that assure current approved documents are used in accomplishment of work activities as well as in procedure and design document development or changes. Methods shall be implemented that preclude the use, or inadvertent use, of obsolete or superseded documents.

14.4.4 Controlling procedures for document control shall specify methods for identifying the current revision and status of plant procedures, **design documents**, modifications and change documents. These procedures shall also provide methods for identifying outstanding changes to procedures and **design documents.**

15.0 QUALITY ASSURANCE (QA) PROGRAM FOR FIRE PROTECTION SYSTEMS

15.1 SCOPE

This section sets forth the QAP requirements for permanent plant fire protection related systems, equipment, and administrative programs.

15.2 REGULATORY COMMITMENTS

The QAP delineated in this section incorporates the appropriate requirements of Branch Technical Position 9.5.-1, Appendix A; Appendix R to 10CFR50; commitments to the 1977 Nuclear Regulatory Commission "Administrative Controls for Fire Protection for Nuclear Power Plants" letter; and appropriate National Fire Protection Association codes and standards to the extent required by plant commitments. This section is to be used in conjunction with the (U)FSAR and Technical Specifications of each nuclear plant. Sections 1.0 through 14.0 of the QAP Manual apply only to the extent referenced in this section.

15.3 PROGRAM MANAGEMENT AND OBJECTIVES

15.3.1 The Plant General Managers at the three nuclear plants are responsible for the overall administration of the Fire Protection Program and provide the plant point of control and contact for contingencies. They may delegate their authority as appropriate to others; however, they shall not delegate their responsibility. The Plant General Managers/Director - Site Operations shall direct a documented program of QA for items designated as fire protection related. The program shall accomplish the following:

15.3.1.1 Provide controls for inspection, installation, corrective maintenance, modifications, and material acceptance activities for designated fire protection related items.

15.3.1.2 Verify compliance with governing procedures of the Fire Protection Program.

15.3.1.3 Provide adequate QA controls for designated fire protection related items to ensure the maintenance of an effective Fire Protection Program.

15.3.2 The Fire Protection Program shall include procedures and controls to accomplish the following:

15.3.2.1 Coordinate Fire Protection Program activities.

15.3.2.2 Prepare procedures and instructions which implement the Fire Protection Program.

15.3.2.3 Assure development and technical adequacy of training materials and sources related to fire protection related and assign qualified fire protection instructors.

15.3.2.4 Listing those fire protection items which are subject to the Fire Protection Program.

15.3.2.5 Periodic monitoring of fire protection related activities.

15.3.2.6 Assure that corrective maintenance and modifications of the fire protection related systems comply with Technical Specifications and as appropriate, applicable NFPA Codes and Standards; 10CFR50.48; and 10CFR50, Appendix R Sections III.G, III.J, and III.O.

15.3.2.7 Coordinate the arrangements for off-site fire company support and training.

15.3.2.8 Schedule and implement the Fire Drills Program.

15.3.2.9 Establish and maintain minimum equipment for the fire brigade teams.

15.3.2.10 Assign personnel to fire brigade teams.

15.4 DESIGN AND MODIFICATION CONTROL AND DOCUMENTATION

Design activities shall be accomplished in accordance with procedures that assure the applicable design requirements are included and that appropriate reviews are conducted. Design change of fire protection related items shall be prepared, approved, accomplished, and documented in accordance with Section 3.0.

15.5 PROCEDURES AND DRAWINGS

Activities such as design, installation, inspection, tests, maintenance, and modification of fire protection related systems shall be accomplished in accordance with procedures and drawings controlled in accordance with Sections 6.0 and 14.0.

15.6 CONTROL OF PURCHASED MATERIALS, EQUIPMENT, AND SERVICES

Control of plant purchased materials, equipment, and services with respect to fire protection related items shall be accomplished in accordance with the following for procurement, receiving, and storage:

15.6.1 Procurement.

Procurement documents for fire protection related items shall be completed in accordance with plant procedures. These procedures shall require:

15.6.1.1 Items to be either Underwriters Laboratories, Inc., listed and/or Factory Mutual approved or accepted, or accepted by American Nuclear Insurance, formerly NEL-PIA or Nuclear Mutual Limited.

<u>or</u>:

15.6.1.2 The item(s) technical and quality requirements are established during the review and approval process for the purchase requisition.

15.6.2 Receiving.

15.6.2.1 Fire protection related items shall be receipt inspected in accordance with procedures, noting in particular:

15.6.2.1.1 Any damage to the item.

15.6.2.1.2 Item identification and marking.

15.6.2.1.3 Any required vendor-supplied documentation.

15.6.2.1.4 Conformance with purchase requirements/ specifications.

15.6.2.2 A **receipt inspection** report shall be completed for received fire protection related items. Noted deficiencies shall be documented in accordance with Section 12.0.

15.6.2.3 Fire protection related items shall be tagged in accordance with procedures.

15.6.3 Storage.

Fire protection related items shall be stored in accordance with plant procedures.

15.7 FIRE PROTECTION SYSTEM TAGOUTS

Fire protection related system tagouts shall be accomplished in accordance with procedures.

15.8 CONDITIONS ADVERSE TO QUALITY (CATQ)

CATQ of fire protection related items shall be identified, reported, dispositioned, and corrected in accordance with Section 12.0.

15.9 QUALITY CONTROL INSPECTIONS

A documented program of quality control inspections is required when rework or design changes to those items can impair the ability of the system, equipment, component, or installation to accomplish its intended function.

15.10 FIRE PROTECTION INSPECTIONS

The Plant General Manager is responsible for implementing a documented program of periodic inspections which verifies compliance with governing procedures for the following fire protection related activities:

15.10.1 Housekeeping.

15.10.2 Surveillance tests of the fire protection related systems.

15.10.3 Control of ignition sources.

15.10.4 Use of fire watches.

15.10.5 Control of combustibles.

15.10.6 Fire protection related training documentation.

15.10.7 Preventive Maintenance Program.

15.11 PREVENTIVE MAINTENANCE

A Preventive Maintenance Program for designated fire protection related items shall be established and implemented in accordance with Section 11.0.

15.12 TESTING

Corrective maintenance which affects the function of designated fire protection related items requires post-maintenance testing except where such testing would be destructive. The specific test requirements shall be delineated in accordance with procedures and applicable NFPA codes and standards.

Design changes to fire protection related items require testing to demonstrate that design criteria and the function of the modification are met. The specific test requirements will be delineated in accordance with procedures.

15.13 ASSESSMENTS

Assessments shall be conducted in accordance with Section 13.0.

15.14 AUDITS

Fire protection related audits will be performed in accordance with plant Technical Specification.

15.15 RECORDS

Those records required to verify compliance with criteria of the Fire Protection Program shall be identifiable and retrievable and shall be assigned retention requirements.

15.16 MATERIAL UPGRADING

Items not originally procured for application in fire protection related systems shall be evaluated for the intended use prior to installation in accordance with procedures.

16.0 QUALITY ASSURANCE (QA) PROGRAM FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS (Harris Plant Only)

16.1 SCOPE

This section establishes the QAP requirements for radioactive waste management systems for use at the Harris Nuclear Plant only.

Maintenance and operation of radioactive waste management systems shall be in accordance with procedures to assure the original design requirements or evaluated alternatives are not compromised.

16.2 REGULATORY COMMITMENTS

This section establishes the QAP requirements of Regulatory Guide 1.143 for radioactive waste management systems. These requirements should be applied for design, installation, and initial testing of new radioactive waste management systems when specified by the **design organization.**

16.3 RESPONSIBILITIES

The responsibility for implementation of these requirements is assigned to the applicable department head unless specified in an interface agreement.

16.4 DESIGN AND PROCUREMENT ACTIVITIES

16.4.1 Design and procurement activities shall be accomplished in accordance with procedures. These procedures shall assure that the applicable design requirements are included in design and procurement documents and that appropriate reviews of these documents are conducted. Design changes shall be prepared, approved, accomplished, and documented in accordance with Section 3.0.

16.4.2 Procurement procedures shall include measures for evaluation of the supplier to assure an appropriate QA system is in place for the items or services to be provided. As an alternative to such a system for American Society of Mechanical Engineers (ASME), Section VIII, or American National Standards Institute (ANSI) B31.1 items, the supplier shall only be required to have a QA system which satisfies the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, or ANSI B31.1 for boiler external piping.

16.5 MATERIAL CONTROL

16.5.1 Material purchased for radioactive waste management systems shall be receipt inspected to assure conformance to technical and QA requirements of the procurement document.

16.5.2 Measures shall be established to control handling, storage, and preservation of material to prevent damage or deterioration.

16.5.3 Measures shall be established to provide for identification of material which has satisfactorily passed required inspections or tests. These measures may include tags, labels, stamps, computer programs, or other suitable means.

16.5.4 Material not originally purchased for radioactive waste management systems may be used providing appropriate evaluations and inspections are performed in accordance with procedures.

16.6 CONTROL OF MEASURING AND TEST EQUIPMENT (M&TE)

Control of M&TE used to support activities described in this section shall be in accordance with Section 8.0.

16.7 PROCEDURES AND DRAWINGS

Activities described in this section shall be accomplished in accordance with procedures and drawings. These procedures and drawings shall be controlled in accordance with Sections 6.0 and 14.0.

16.8 CORRECTIVE ACTION

Conditions adverse to quality shall be identified, reported, dispositioned, and corrected in accordance with Section 12.0.

16.9 RECORDS

Measures shall be established to assure sufficient records are maintained to furnish evidence that the activities described in this section are being implemented. These records shall be identifiable and retrievable.

17.0 IF-300, IRRADIATED FUEL SHIPPING CASK

17.1 SCOPE

This section specifies the quality assurance (QA) requirements for the IF-300 irradiated fuel shipping cask.

17.2 REGULATORY COMMITMENTS

This section provides for the implementation of the IF-300 Irradiated Fuel Shipping Cask QA program criteria required to comply with the "QA Program Approval for Radioactive Material Packages," Docket 71-0345 which complies with the QA Program requirements of 10CFR71 Subpart H.

17.3 GENERAL

17.3.1 Classification of cask components

- The following components and parts are 10CFR71 "Important to Safety" and are classified as safety-related. They are subject to QA program requirements contained in this section.

17.3.1.1 Containment

- Cavity End Plate
- Inner Shell
- Vent Pipe Assembly
- Locating Key
- Body Flange
- PWR Head Forging
- PWR Head Subassembly
- BWR Head Forging
- BWR Head Liner
- Trunnion Assembly
- Valve Boxes
- Rupture Disk Device
- BWR Head End Plate
- BWR Head Liner Ring
- BWR Sleeve Nuts
- PWR Sleeve Nuts
- Studs
- Cavity Globe Valves
- Valve Pipe Cap or Plug
- Valve Hardware
- Grayloc Seal Ring
- Fins
- Cavity Drain Line Assembly

17.3.1.2 Nuclear shielding:

Uranium shield (cask barrel, closure head, bottom, basket shield), neutron shield (corrugated barrel, valve boxes, expansion tank, piping, valves, blind flanges, liquid).

17.3.1.3 Criticality control:

- BWR basket

- PWR basket

17.4 ORGANIZATION

Organization and responsibilities are as described in Section 2.0.

17.5 DESIGN CONTROL

Design control shall be controlled as required by Section 3.0.

17.6 PROCUREMENT DOCUMENT CONTROL

The procurement of safety-related items and services shall be per the requirements of Section 4.0. To the extent necessary, suppliers are required to implement a QA program that meets Appendix B to 10CFR50 or Subpart H, 10CFR71. Safety-related parts shall be inspected, stored, and handled in accordance with plant procedures which meet Section 5.0.

17.7 PROCEDURES AND DRAWINGS

Cask loading, unloading, tests, and inspections are performed in accordance with procedures and drawings which are approved in accordance with Section 6.0. These procedures and drawings shall implement the requirements of the cask Certificate of Compliance.

17.8 DOCUMENT CONTROL

Documents relative to IF-300 activities shall be controlled as required by Section 14.0.

17.9 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

The identification and control of materials, parts, and components for the IF-300 cask shall be as described in Section 5.0.

17.10 CONTROL OF SPECIAL PROCESSES

Special processes shall be performed per the requirements of Section 11.0.

17.11 INSPECTIONS AND TEST CONTROL

Inspections and tests required by the IF-300 Certificate of Compliance shall be performed in accordance with the applicable portions of Section 11.0.

17.12 CONTROL OF MEASURING AND TEST EQUIPMENT (M&TE)

M&TE used for the IF-300 shall be calibrated and controlled as required by Section 8.0.

17.13 HANDLING, STORAGE, AND SHIPPING

Shipping of the IF-300 shall be in accordance with 49CFR and 10CFR71. Handling and storage shall be performed in accordance with procedures.

17.14 INSPECTION, TEST, AND OPERATING STATUS

Prior to placing the IF-300 cask in operation, certain preliminary tests shall be performed in accordance with procedures. These functional tests are designed to meet the requirements of the Certificate of Compliance for the IF-300 and Carolina Power & Light administrative controls.

Routine inspection of cask systems and components shall be accomplished in accordance with procedures. Where applicable, the manufacturer's recommended inspection intervals should be followed.

17.15 NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Conditions adverse to quality (CATQ) shall be controlled and dispositioned in accordance with Section 12.0.

17.16 CORRECTIVE ACTION

CATQ shall be controlled and dispositioned in accordance with Section 12.0.

17.17 QA RECORDS

Measures shall be established to assure sufficient records to furnish evidence that the activities described in this section are being implemented and records required by 10CFR71 are maintained in accordance with Section 14.0.

17.18 AUDITS/ASSESSMENTS

Audits/Assessments of the IF-300 cask activities shall be conducted in accordance with Sections 4.0 & 13.0.

18.0 RADIOACTIVE MATERIAL PACKAGES QUALITY ASSURANCE (QA) PROGRAM

18.1 SCOPE

This section establishes the QA requirements for activities associated with the procurement, testing, maintenance, repair, and use of "non LSA greater than Type A" packages as required by 10CFR71, Subpart H, other than the IF-300. The Quality Assurance Program (QAP) for the IF-300 is provided in Section 17.0 of the QAP Manual. This section also provides for management-controlled audits as required by 10CFR20.311.

18.2 REGULATORY COMMITMENTS

This section provides the implementation criteria required to comply with the "QA Program Approval for Radioactive Material Packages," Docket 71-0345 which complies with the QA Program requirements of 10CFR71 Subpart H.

18.3 ORGANIZATION

Organization and responsibilities are provided in Section 2.0.

18.4 DESIGN CONTROL

Design control shall be the responsibility of the package owner or manufacturer in the case of packages that are purchased by CP&L.

18.5 PROCUREMENT DOCUMENT CONTROL

Procurement documents:

Procurement documents shall:

- Require the package owner/manufacturer/ to have a Nuclear Regulatory Commission (NRC) approved QA program that meets the requirements of 10CFR71, Subpart H.
- Require the owner/manufacturer to submit current documentation attesting that the packaging was designed, procured, fabricated, assembled, tested, modified, repaired, and maintained in accordance with an NRC-approved quality assurance program.
- Designate other pertinent documentation to be furnished with the packaging (e.g., certificate of compliance, as-built drawings, photographs, sketches, use and maintenance manuals)

Approval of Vendors

Where procurement documents require the vendor to implement a quality assurance (QA) program that complies with 10CFR71, Subpart H, approval of this program by the NRC shall be confirmed prior to issuance of the purchase order or contract. Monitoring of supplier performance and continued qualification shall be documented in accordance with procedures. In the event replacement parts for the packagings are required, procurement of the parts shall be made by the packaging owner in accordance with the packaging owner's QA program.

18.6 PROCEDURES AND DRAWINGS

Package loading, unloading, filling, and inspections are performed in accordance with procedures and drawings which are approved in accordance with Section 6.0 and comply with the package Certificate of Compliance.

18.7 DOCUMENT CONTROL

Documents relative to activities performed by Carolina Power & Light (CP&L) shall be controlled as required by Section 14.0.

18.8 CONTROL OF PURCHASED MATERIALS, EQUIPMENT, AND SERVICES

Inspections shall be performed upon receipt of packaging to verify compliance with procurement documents. The criteria for acceptance of each of these inspections and the action to be taken if noncompliance is encountered is established in applicable plant procedures.

18.9 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Reference Section 18.5.

18.10 CONTROL OF SPECIAL PROCESSES

Special processes required shall be performed by the package owner.

18.11 INSPECTIONS

While packages are at CP&L facilities, inspections required by the Certificate of Compliance and by the package owner/manufacturer will be performed in accordance with Section 11.0. Packages owned by CP&L shall be inspected and maintained in accordance with procedures and drawings that are approved in accordance with Section 6.0.

18.12 TEST CONTROL

Tests required by the Certificate of Compliance shall be performed by the package owner/manufacturer in accordance with their NRC-approved QA program while the package is in their physical possession. Prior to a shipment, CP&L shall perform tests as required by the Certificate of Compliance and 10CFR71.

18.13 CONTROL OF MEASURING AND TEST EQUIPMENT (M&TE)

M&TE used by CP&L shall be calibrated and controlled as required by Section 8.0.

18.14 HANDLING, STORAGE, AND SHIPPING

Shipping of packages shall be in accordance with 10CFR71. Handling shall be performed in accordance with procedures. In the event CP&L stores a package, storage shall be in accordance with the package owner instructions.

18.15 INSPECTION, TEST, AND OPERATING STATUS

While a package is located on CP&L property, the status of the package shall be in accordance with the applicable sections of the QAP Manual.

18.16 CONDITIONS ADVERSE TO QUALITY (CATQ) AND CORRECTIVE ACTION

CATQ identified while on CP&L property shall be controlled and dispositioned in accordance with Section 12.0.

18.17 QA RECORDS

Those records required by 10CFR71 which are generated by CP&L shall be retained in accordance with Section 14.0.

18.18 AUDITS/ASSESSMENTS

Audits/Assessments of the Radioactive Material Packaging Program including the package contractor's activities shall be conducted in accordance with Sections 4.0 and 13.0.

19.0 SOFTWARE QUALITY ASSURANCE (QA) PROGRAM

19.1 SCOPE

19.1.1 This section provides detailed requirements which establish the Graded Approach to Quality for Software. The purpose of this section is to prescribe software configuration control requirements, define controls applicable to **quality software/computing systems**, and identify software life cycle requirements that establish the Nuclear Generation Group (NGG) Software Quality Assurance Program. Procedures that meet the requirements delineated in this section for **quality software/computing systems** shall be used to implement these controls. Controls are applied in a graded manner to **software/computing systems** which are outside safety related processes.

19.1.2 Items which have "self contained" digital processors or software (i.e. components that do not rely upon or interface with other software) that are periodically verified and validated are exempted from the scope of this program. This includes the following examples:

1. Digital instrumentation and control equipment (e.g. digital transmitters) subject to technical specification surveillance testing or periodic testing (e.g. periodic maintenance route).

2. Measuring and Testing Equipment (M&TE) which has in-process calibration to recognized standards or is in a site's calibration program.

3. Laboratory instruments utilizing software recognized as a national standard.

4. Changes to the simulator computer are exempted from this program because they are controlled by direct regulatory commitments.

19.1.3 Industry guidance documents on software quality assurance such as IEEE, ANSI, ANS, ASME, NUSMG, and NIRMA may be used for reference purposes.

19.2 REGULATORY COMMITMENTS

None.

19.3 RESPONSIBILITIES OF CAROLINA POWER & LIGHT (CP&L)

The responsibility for implementing this section is assigned to each nuclear organization that procures, develops, tests, verifies, uses, changes, maintains or retires **quality software/computing systems.** All **quality software/computing systems** shall be developed, tested, verified, used, and maintained under controlled conditions as appropriate based upon its importance to nuclear safety. A list of **quality software/computing systems** that support safety related work shall be maintained.

19.4 PROGRAM

19.4.1 Graded Approach to Software Quality Level Determination

The Software Quality Level is commensurate with the software's importance to nuclear safety. The most rigid controls are applied to **quality software/computing systems** based on the need for compliance with regulations, equipment reliability, or other factors. These controls are applied to the Software Life Cycle, which is the systematic approach to software development, maintenance, use, and retirement. The extent to which the individual requirements are applied depends upon the importance of software/computing systems as explained by Software Quality Assurance Program implementing procedures.

19.4.2 Software Life Cycle

1. **Procurement**

   Procurement of **quality software/computing systems** shall be in accordance with Section 4.0 of this manual as specified in the Software Quality Assurance Program implementing procedures.

2. **Development Phase**

   a. Baseline software life cycle documents are produced during the Development Phase in accordance with the Software Quality Assurance Program implementing procedures.

   b. Vendor supplied life cycle documents for **quality software/computing systems** shall meet the requirements of this section.

3. **Installation Phase**

Installation of **quality software/computing systems** shall be controlled by Software Quality Assurance program implementing procedures.

4. **Operation and Maintenance Phase**

   a. During the Operation and Maintenance Phase the software has been approved for operational use and the computing environment is established.

   b. Maintenance (changes to **quality software/computing systems**) to remove latent errors, to respond to new or revised requirements, or to adapt to changes in the operating environment shall be performed under the established change control process as defined by Software Quality Assurance program implementing procedures.

5. **Retirement Phase**

   a. During the Retirement Phase the support for a software product is terminated, and the routine use of the software is prevented.

   b. The software is de-installed from CPUs, Servers, etc. and returned to the software librarian who controls software. Diskettes, tapes and other media shall be labeled "RETIRED".

19.4.3 Configuration Management

1. Configuration management refers to the controls for hardware and software items that constitute a system. This includes the release and change of those items throughout the system life cycle including the documentation of modification activities.

2. The baseline version of **quality software/computing systems**, source code and life cycle documentation shall be stored per Records Management procedures and Software Quality Assurance Program implementing procedures.

3. Error notification, evaluation and resolution information shall be controlled for **quality software/computing systems** used in NGG.

4. Life Cycle Baseline documentation requirements shall be imposed upon modification of the existing product.

5. **Quality software/computing systems** that exist and are in production prior to the effective date of this program shall be included on a controlled list. Minimal information shall be software name, version, software quality level, software quality level justification and software owner name.

6. Each organization utilizing **quality software/computing systems** shall keep track of safety-related work resulting from the use of such software, including version numbers or issue date of the software. If significant errors are later determined, safety-related work can be reviewed to determine any impact of software errors.

19.4.4 Qualification Requirements

Qualification is the process of demonstrating that for a given input which may be defined in an Acceptance Test Plan, Benchmark Test Case or Calibration Test Procedure, the software produces the expected output

1. Acceptance Test Requirements

Plant computer systems shall undergo acceptance testing per applicable plant procedures to demonstrate required performance over the range of operation of the controlled function or process. The results of tests that are Quality Assurance or Vital records shall be designated and stored in Records Management in accordance with procedures.

2. Calibration and Control Test Requirements

Computer based calibration and control equipment shall be calibrated, adjusted, and maintained at prescribed intervals or prior to use per applicable plant procedures. The results of tests that are Quality Assurance or Vital records shall be designated and stored in Records Management as required by procedures.

3. Benchmark Test Requirements

The benchmark test process is typically used to demonstrate design analysis software products perform as expected. This includes development of appropriate test cases to access the software functionality and execution of these test cases. The benchmark test proves, for a given input, a known result is obtained.

19.4.5 Error Management

1. A method of describing user-identified errors or problems to the developer or the owner of the software shall be established. Errors for **quality software/computing systems** shall be identified and documented per Software Quality Assurance Program implementing procedures.

2. A list of errors, error reports to users, resulting evaluation and corrections, error impact statement, and error resolutions shall be maintained in a controlled manner.

3. Impact on NGG shall be determined and appropriate corrective action taken with any related errors introduced into NGG and with the software source code, if appropriate. In case of misuse, the causes of misuse shall be clarified and positive action taken to prevent future misuse.

4. Error reports shall document error impact and will be identified as a Quality Assurance or Vital records upon error resolution. These records shall be designated and stored in Records Management as required by procedures.

19.4.6 Self Assessment

Self assessments of the Software Quality Assurance Program shall be performed.

19.4.7 Records Management

Documentation resulting from the development, modification or use of **quality software/computing systems** shall be maintained as QA records as required in procedures.

20.0 NONSAFETY-RELATED COMPUTER SOFTWARE QUALITY ASSURANCE.

THE CONTENTS OF THIS SECTION HAVE BEEN INCLUDED IN SECTION 19.0.

21.0 QUALITY ASSURANCE (QA) PROGRAM REQUIREMENTS FOR QUALITY CLASS B ITEMS

21.1 SCOPE

This section sets forth the QA requirements for **quality Class B Items** and activities. Items subject to these requirements shall be identified in appropriate plant procedures.

Sections 1.0 through 14.0 of the QAP Manual apply only to the extent referenced in this section.

21.2 REGULATORY COMMITMENTS

This section is to be utilized in conjunction with Regulatory Guides 1.29 and 1.97 as committed in Section 1.8 of the (U)FSAR.

21.3 MANAGEMENT RESPONSIBILITIES

Each department head has responsibility for determining if this section applies to plant activities being performed, for implementation of these requirements, and for establishing the necessary interfaces with other organizations.

21.4 DESIGN ACTIVITIES

Design change activities shall be accomplished in accordance with Section 3.0. (Not applicable to BNP/RNP)

21.5 PROCUREMENT

Preparation, review, and approval of procurement documents shall be in accordance with Section 4.0 or acceptable alternatives delineated in procedures.

21.6 MATERIAL CONTROL

Receiving inspection, storage, and equipment control shall be in accordance with Section 5.0. (For BNP and RNP, these items are not required to be stored in specifically designated storage areas.)

21.7 CONDITIONS ADVERSE TO QUALITY (CATQ)

CATQ shall be identified, reported, dispositioned, and corrected in accordance with Section 12.0.

21.8 OPERATIONS CONTROL

Plant operations of these items shall be in accordance with Section 10.0.

21.9 CALIBRATION CONTROL

Calibration activities shall be in accordance with Section 8.0.

21.10 MAINTENANCE

Maintenance activities shall be in accordance with Section 11.0. (Not applicable to BNP or RNP.)

21.11 ASSESSMENTS/AUDITS

Assessments/Audits may be conducted in accordance with Sections 4.0 & 13.0.

21.12 QA RECORDS

Measures shall be established to assure sufficient records are maintained to furnish evidence that the activities described in this section are being implemented. These records shall be identifiable and retrievable. These records shall be maintained in accordance with Section 14.0.

22.0 QUALITY ASSURANCE PROGRAM FOR NONSAFETY RELATED SYSTEMS AND EQUIPMENT USED TO MEET THE STATION BLACKOUT RULE

22.1 SCOPE

This section sets forth the QAP requirements for nonsafety related systems and equipment used for meeting the Station Blackout (SBO) Rule (10CFR50.63) which are not otherwise covered by a quality assurance program.

22.2 REGULATORY COMMITMENTS

The QAP delineated in this section incorporates the appropriate requirements of 10CFR50.63 and Reg. Guide 1.155. Sections 1.0 through 14.0 of the QAP Manual apply only to the extent referenced in this section.

22.3 PROGRAM MANAGEMENT AND OBJECTIVES

The Plant General Manager is responsible for the overall administration of the Station Blackout Quality Assurance Program and provides the plant point of control and contact for contingencies. He may delegate his authority as appropriate to others; however, he shall not delegate his responsibility.

22.4 DESIGN CONTROL

Design activities shall be accomplished in accordance with procedures that assure the applicable design requirements are included and that appropriate reviews are conducted. Design changes of Station Blackout items shall be prepared, approved, accomplished, and documented in accordance with Section 3.

22.5 INSTRUCTIONS, PROCEDURES AND DRAWINGS

Activities such as design, installation, inspection, tests, maintenance, and modification of non-safety systems used to meet the Station Blackout Rule shall be accomplished in accordance with instructions, procedures and drawings in accordance with Section 6.

22.6 PROCUREMENT DOCUMENT CONTROL AND CONTROL OF PURCHASED MATERIAL,EQUIPMENT, AND SERVICES

Control of plant purchased materials, equipment, and services with respect to nonsafety related systems and equipment used to meet the SBO rule (SBO Items) shall be accomplished in accordance with the following for procurement, receiving, and storage:

22.6.1 Procurement

Procurement documents for SBO items shall be completed in accordance with plant procedures. These procedures shall require the item(s) technical and quality requirements be established during the review and approval process for the purchase requisition.

22.6.2 Receiving

22.6.2.1 Material, equipment and **services**, including spare and replacement parts for the nonsafety related systems and equipment used to meet the Station Blackout Rule shall be inspected, stored, issued, and controlled in accordance with procedures, noting in particular:

22.6.2.1.1. Any damage to the item.

22.6.2.1.2 Item identification and marking.

22.6.2.1.3 Any required vendor-supplier documentation.

22.6.2.1.4 Conformance with purchase requirement/specification.

22.6.2.2 A **receipt inspection** report shall be completed for received SBO Items. Noted deficiencies shall be documented in accordance with Section 12.

22.6.2.3 SBO Items shall be tagged in accordance with procedures.

22.6.3 Storage

SBO Items shall be stored in accordance with plant procedures.

22.7 NONCONFORMING ITEMS AND CORRECTIVE ACTIONS (CONDITIONS ADVERSE TO QUALITY)

Conditions Adverse To Quality (CATQ) of SBO items shall be identified, reported, dispositioned, and corrected in accordance with Section 12.

22.8 INSPECTIONS

Independent inspections of activities will be performed in accordance with procedures to verify compliance with documented installation drawings and test procedures for accomplishing activities related to the Station Blackout program.

22.9 TESTING AND TEST CONTROL

Testing will be performed and verified by inspection to demonstrate conformance with design and system readiness requirements. These tests will be performed; test results properly evaluated, and appropriate action taken in accordance with plant procedures.

22.10 ASSESSMENTS

Assessments shall be conducted in accordance with Section 13.0.

22.11 RECORDS

Those records required to verify compliance with criteria of the Station Blackout program shall be identifiable and retrievable and shall be assigned retention requirements.

22.12 MATERIAL UPGRADING

Items not originally procured for application in SBO systems shall be evaluated for the intended use prior to installation in accordance with procedures.

23.0 INTERPRETATIONS

23.1 SCOPE

This section sets forth requirements for issuing official CP&L Quality Assurance Program interpretations by the Manager - Performance Evaluation and Regulatory Affairs (PERAS). These interpretations are issued on an as-needed basis for the purposes of clarifying Company policy in areas pertaining to this program. It includes requirements for the issuance, control, and removal of interpretations and a listing of current interpretations.

23.2 REGULATORY COMMITMENTS

None

23.3 REQUIREMENTS

Interpretations shall be requested and issued in accordance with the following requirements:

23.31 Requests for interpretation shall require the signature of a section manager or above.

23.3.2 The Manager - PERAS is the sole authority for determining whether a response to a request for interpretation is to be included in this section.

23.3.3 A request for an interpretation will be responded to regardless of whether or not it will be included in this section.

23.3.4 Current and historical interpretations shall be listed in Section 23.5 by sequential number and by subject.

23.3.5 Interpretations, once issued, shall remain a part of this section until the manual is revised. After manual revision, interpretation will be withdrawn and a line added to the listing in Section 23.4 identifying that portion of the manual revised to incorporate that interpretation or noted for clarification only.

23.4 CONTENTS

The following is a listing of interpretations contained in Section 23.4. Both current and historical interpretations are included by number and subject line. Historical interpretations can be found in the historical files.

23.5 LISTING OF INTERPRETATIONS

Current and Historical

INTERPRETATION NUMBER	SUBJECT	DATE	STATUS
23.5.1	Interpretation of Inspector Qualification/Independence Requirements	7/19/89	Deleted by Revision 18
23.5.2	Interpretation of UT Block Calibration Requirements (QAP Manual Sections 4.5.6 & 8.0) for clarification only	1/15/90	Incorporated into Section 8.0 in Revision 18
23.5.3	Interpretations to Commitments to ANSI 18.7 via Final Safety Analysis Reports - (Requirements for Inspection By Other Than Quality Control Personnel) (QAP Manual Section 7.6) for clarification only	1/6/92	Deleted by Revision 18
23.5.4	Interpretation of the QAP Manual, Paragraph 6.4.7 of the Procedures and Drawings Section	1/17/92	Canceled Section 6.4.7 Deleted
23.5.5	Interpretation of Calibration accuracy requirements for M&TE (QAP Manual Section 8.4)	9/2/93	Incorporated in Rev. 17

(Note: This policy statement is a reprint of the CP&L Quality Assurance Program Policy statement on the Intranet. Any changes to this document requires the policy statement to be revised on the Intranet).

Document title

CP&L Quality Assurance Program Policy

Document number

REG-CPL-000

Keywords

Policy; CPL; Regulatory; Quality Assurance; Program

It is the policy of Carolina Power & Light Company to operate and maintain nuclear power plants to safeguard the health and safety of its employees and the public. The operation of nuclear power plants is in accordance with the facility operating license issued by the Nuclear Regulatory Commission (NRC). A Quality Assurance (QA) program is implemented and updated as necessary to assure that systems used for generating electricity using nuclear fuel are designed, constructed, and operated in a safe manner. Deviations from the requirements of this program are permitted only with written authority from the corporate management position which originally approved the program or implementing procedures.

CP&L's QA Program ensures compliance with NRC regulations specified in Title 10 of the US Code of Federal Regulations. The CP&L QA Program for 10CFR50 Appendix B requirements is established by and defined in Sections 1.0-14.0 of NGGM-PM-0007, *Quality Assurance Program Manual.* A description of this program is also contained in Section 17.3 of each nuclear plant's (U)FSAR. Implementing procedures are contained in the Plant Operating Manuals (POMs). CP&L QA programs for Fire Protection Systems, Radioactive Waste Management Systems, IF-300 Shipping Cask, Radioactive Material Packages, Computer Software, Quality Class B items, and Station Blackout are also contained in the NGGM-PM-0007.

The Senior Vice President - Nuclear Generation/Chief Nuclear Officer has the ultimate company responsibility for the safe operation of the nuclear power plants. Plant Vice Presidents and the Manager - Performance Evaluation and Regulatory Affairs (PERAS), have the responsibility and authority to identify and correct quality problems and are responsible for monitoring the effectiveness of quality assurance activities through a system of planned assessments and inspections. Plant Vice Presidents and the Manager - PERAS effect this responsibility by maintaining a strong self evaluation culture in the line organization supplemented with independent monitoring and systematic assessments performed by the Nuclear Assessment Sections and Performance Evaluation Support.

The Manager - PERAS is responsible for maintaining and monitoring the overall effectiveness of QA Program implementation and communicates directly with Senior Management up to and including the President/Chief Executive Officer, and if appropriate, with the Board of Directors, to resolve any quality concerns which cannot be resolved satisfactorily at a lower management level.

PES and NAS Managers review the effectiveness of the QA Program on a regular basis with the Senior Vice President - Nuclear Generation/Chief Nuclear Officer.

Although specific position and responsibilities are delineated in this policy statement the achievement of quality is the responsibility of each individual involved in nuclear generation.

W. S. Orser
Executive Vice President - Energy Supply

APPENDIX I

This Appendix provides a cross-reference between the QAP Sections I.0 through I4.0 and Title I0, Code of Federal Regulations, Part 50, Appendix B (I0CFR50, Appendix B), titled, "Quality Assurance Criteria for Nuclear Power Plants."

The references to the QAP contained in this appendix are limited to those that have a direct connection, or describe the immediate activity addressed by I0CFR50, Appendix B. The program references are identified by paragraph number, indicating a description of implementation somewhere within the text of the referenced paragraph. References to the 18 criteria of I0CFR50, Appendix B, are identified by roman numerals. Sentences within each Appendix B criterion are further subdivided by Arabic numbers and aspects of each sentence by lower case letters.

10CFR50 (APPENDIX B)			CP&L QA PROGRAM MANUAL
I.	**ORGANIZATION**		
1.	The applicant shall be responsible for:		
	a.	Establishment of the QA Program.	1.1
	b.	Execution of the QA Program.	1.1, 1.2, 2.1
2.	The applicant may delegate to others, such as contractors, agents, or consultants the work (or any part thereof) of:		
	a.	Establishing the QA Program.	2.2
	b.	Executing the QA Program.	1.1, 2.1, 2.2
	But the applicant shall retain responsibility therefore.		
3.	The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing.		2.2
4.	These activities include both the performing functions of attaining quality objectives and the quality assurance functions.		2.2, 2.3, 2.4
5.	The QA functions are those of:		
	a.	Assuring that an appropriate QA Program is established and effectively executed.	1.3, 2.2, 4.11, 13.1
	b.	Verify, such as by checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed.	2.2, 4.11, 13.1

6. The persons and organizations performing QA functions shall have sufficient authority and organizational freedom to:	
a. Identify quality problems.	2.2, 4.11, 13.1
b. To initiate, recommend, or provide solutions.	2.2, 4.11.6, 13.5
c. To verify implementation of solutions.	2.2, 4.11.8, 13.7
7. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations are provided.	2.2
8. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the QA Program may take various forms provided that the persons and organizations assigned the QA functions have this required authority and organizational freedom.	2.2
9. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the QA Program at any location where activities subject to this Appendix are being performed shall have direct access to such levels of management as may be necessary to perform this function.	2.2

II. QUALITY ASSURANCE PROGRAM

1. The applicant shall establish at the earliest practical time, consistent with the schedule for accomplishing the activities, a QA Program which complies with the requirements of	1.1, 1.2, 2.2

this Appendix.

2.	This Program shall be:	
	a. Documented by written policies, procedures or instructions.	2.2
	b. Carried out throughout plant life in accordance with those policies, procedures or instructions.	2.2, 6.0
3.	The applicant shall identify:	
	a. The structures, systems and components to be covered by the QA Program.	1.2
	b. The major organizations participating in the Program, together with the designated functions of these organizations.	2.2
4.	The QA Program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety.	1.2
5.	Activities affecting quality shall be accomplished under suitably controlled conditions.	5.3, 6.4, 8.5, 10.3, 11.1, 11.3
6.	Controlled conditions include:	
	a. The use of appropriate equipment.	5.2, 6.3, 8.4, 10.2, 11.1, 11.3
	b. The use of suitable environmental conditions for accomplishing the activity, such as adequate cleanliness.	5.3, 6.4, 8.5, 10.3 11.1, 11.3
	c. Assurance that all prerequisites for the given activity have been satisfied.	4.9, 4.11, 5.3, 6.4, 8.5, 13.1

7.	The Program shall take into account the need for:	
	a. Special controls.	5.3, 5.5, 10.3, 11.3
	b. Processes.	10.3, 10.5, 11.3, 11.4
	c. Test equipment.	8.2
	d. Tools.	5.4, 8.2
	e. Skills to attain the required quality.	7.0
	f. Verification of quality by inspection.	4.5, 4.6, 5.3, 5.5, 11.3, 11.4
	g. Verification of quality by test.	3.4, 3.5, 4.5, 11.3
8.	The Program shall provide for:	
	a. Indoctrination.	Section 7.0
	b. Training.	Section 7.0
	Of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained.	
9.	The applicant shall regularly review the status and adequacy of the QA Program.	1.3
10.	Management of other organizations participating in the QA Program shall regularly review the status and adequacy of that part of the QA Program which they are executing.	2.1, 2.2
III.	**DESIGN CONTROL**	
1.	Measures shall be established to assure that applicable regulatory requirements and the design bases, as defined in l0CFR50.2 and as specified in the license application for those structures, systems, and components to which this Appendix applies, are correctly translated into	3.3, 3.4

specifications, drawings, and instructions.

2.	These measures shall include provisions to assure that appropriate quality standards:	
	a. Are specified and included in design documents.	3.3, 3.4
	b. Deviations from such standards are controlled.	3.3, 3.4, 3.9
3.	Measures shall also be established for:	
	a. Selection.	3.4, 3.5
	b. Review.	3.4, 3.5
	For suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.	
4.	Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations.	3.10
5.	These measures shall include the establishment of procedures among participating design organizations for:	
	a. Review of documents involving design interfaces.	3.10
	b. Approval of documents involving design interfaces.	3.10
	c. Release of documents involving design interfaces.	3.10
	d. Distribution of documents involving design interfaces.	3.10
	e. Revision of documents involving design interfaces.	3.10
6.	The design control measures shall provide for verifying or checking the adequacy of design, such as by one or more of the following means:	

	a. Performance of design reviews.	3.4, 3.5, 3.6
	b. Use of alternate or simplified calculational methods.	3.4, 3.5, 3.6
	c. Performance of a suitable testing program.	3.4, 3.5, 3.6
7.	The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization.	3.4, 3.5
8.	Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking process, it shall include suitable qualification testing of a prototype unit under the most adverse design conditions.	3.3, 3.4, 3.5
9.	Design control measures shall be applied to items such as the following: reactor physics; stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for in-service inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.	3.3, 3.4, 3.5
10.	Design changes, including field changes, shall be:	
	a. Subject to design control measures commensurate with those applied to the original design.	3.5
	b. Approved by the organization that performed the original design unless the applicant designates another responsible organization.	3.5

IV. PROCUREMENT DOCUMENT CONTROL

1.	Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or references in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its Contractors or Subcontractors.	4.5
2.	To the extent necessary, procurement documents shall require Contractors or subcontractors to provide a QA Program consistent with the pertinent provisions of this Appendix.	4.4

V. INSTRUCTIONS, PROCEDURES AND DRAWINGS

1.	Activities affecting quality shall be:	
	a. Prescribed by documented instructions, procedures or drawings of a type appropriate to the circumstances.	6.0
	b. Accomplished in accordance with these instructions, procedures or drawings.	6.0
2.	Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.	6.0

VI. DOCUMENT CONTROL

1.	Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality.	6.1, 6.34, 14.4
2.	These measures shall assure that documents, including changes:	
	a. Are reviewed for adequacy.	6.4, 14.4

b. Approved for release by authorized personnel. 6.4, 14.4

c. Distributed to the location where the prescribed activity is performed. 6.4, 14.4

d. Used at the location where the prescribed activity is performed. 6.4, 14.4

3. Changes to documents shall be reviewed and approved by the same organization that performed the original review and approval unless the applicant designates another responsible organization. 6.4, 14.4

VII. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

1. Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through Contractors and Subcontractors, conform to the procurement documents. 4.8, 5.3

2. These measures shall include provisions, as appropriate, for:

a. Source evaluation and selection. 4.4, 4.8

b. Objective evidence of quality furnished by the Contractor or Subcontractor. 4.4, 4.8

c. Inspection at the Contractor or Subcontractor source. 4.4, 4.8

d. Examination of products upon delivery. 5.3

3. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear power plant site prior to installation or use of such material and equipment. 5.5

4. This documentary evidence shall be:	
a. Retained at the nuclear power plant site.	5.3
b. Sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment.	5.3
5. The effectiveness of the control of quality by Contractors and Subcontractors shall be assessed by the applicant or designee at intervals consistent with the importance, complexity, and quantity of the product or services.	2.2, 4.8, 4.11

VIII. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

1. Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies.	5.3, 5.4, 5.5
2. These measures shall assure that identification of the item is maintained by heat number, part number, serial number or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item.	3.5, 3.7, 5.3, 5.4 5.5, 10.5
3. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.	5.3, 5.4, 5.5, 10.3, 11.4, 11.6

IX. CONTROL OF SPECIAL PROCESSES

1. Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing are:	
a. Controlled.	7.4, 11.3

b. Accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.	7.4, 11.3

X. INSPECTION

1. A program of inspection of activities affecting quality shall be:	
a. Established.	2.2, 5.3, 5.6
b. Executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures and drawings for accomplishing the activity.	2.2, 5.3, 5.6
2. Such inspection shall be performed by individuals other than those who performed the activity being inspected.	2.2, 5.3
3. Examinations, measurements, or tests of material or products processed shall be performed for each work operation where necessary to assure quality.	2.2, 3.4, 4.5, 4.6, 4.8, 5.3, 5.5, 6.4
4. If inspection of processed material or products is impossible or disadvantageous, indirect control by monitoring processing methods, equipment, and personnel shall be provided.	2.2, 4.4, 4.8, 5.3, 5.6
5. Both inspection and process monitoring shall be provided when control is inadequate without both.	2.2, 4.4, 4.8, 5.3, 5.6, 6.4
6. If mandatory inspection holdpoints, which require witnessing or inspecting by the applicant's designated representative are required:	
a. Work shall not proceed beyond these holdpoints without the consent of the designated representative.	2.2, 3.4, 3.7

b.	Specific holdpoints shall be indicated in appropriate documents.	3.4, 4.5

XI. TEST CONTROL

1. A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is:

a.	Identified.	3.4, 3.7, 4.5
b.	Performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.	3.4, 3.7, 4.5

2. The test program for structures, systems and components shall include, as appropriate:

a.	Proof tests prior to installation.	3.4, 3.7, 4.5, 11.3
b.	Preoperational tests.	3.4, 3.7, 4.5, 11.3
c.	Operational tests during nuclear power plant operation.	3.4, 3.7, 4.5, 11.3

3. Test procedures shall include provisions for assuring that:

a.	All prerequisites for the given test have been met.	3.4, 3.7, 4.5, 11.3
b.	Adequate test instrumentation is available and used.	3.4, 3.7, 4.5, 11.3
c.	The test is performed under suitable environmental conditions.	3.4, 3.7, 4.5, 11.3

4. Test results shall be:

a.	Documented.	3.4, 3.7, 4.5, 11.3
b.	Evaluated to assure that test requirements have been satisfied.	3.4, 3.7, 4.5, 11.3

XII. CONTROL OF MEASURING AND TEST EQUIPMENT

1.	Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly:	
	a. Controlled.	8.4, 8.5
	b. Calibrated and adjusted at specified periods to maintain accuracy within necessary limits.	8.4, 8.5

XIII. HANDLING, STORAGE AND SHIPPING

1.	Measures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.	5.3, 5.5, 5.6
2.	When necessary for particular products, special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels, shall be specified and provided.	5.3, 5.5, 5.6

XIV. INSPECTION, TEST, AND OPERATING STATUS

1.	Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the nuclear power plant.	5.3, 5.4, 5.5, 5.6
2.	These measures shall provide for the identification of items which have satisfactorily passed required inspection and tests, where necessary to preclude inadvertent bypassing of such inspections and tests.	5.3, 5.4, 5.5, 5.6

3.	Measures shall also be established for indicating the operating status of structures, systems, and components of the nuclear power plant, such as by tagging valves and switches, to prevent inadvertent operation.	5.3, 5.4, 5.5, 5.6

XV. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

1.	Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation.		12.4, 12.5, 12.6, 12.7
2.	These measures shall include, as appropriate, procedures for:		
	a.	Identification.	12.4, 13.5
	b.	Documentation.	12.5
	c.	Segregation.	12.5
	d.	Disposition.	12.5
	e.	Notification to affected organizations.	12.5
3.	Nonconforming items shall be:		
	a.	Reviewed.	5.3.8, 12.4, 12.5, 12.7
	b.	Accepted or rejected.	5.3.8, 12.4, 12.5, 12.7
	c.	Repaired or reworked in accordance with documented procedures.	5.3.8, 12.4, 12.5, 12.7

XVI. CORRECTIVE ACTION

1.	Measures shall be established to assure that conditions adverse to quality such as failures, malfunctions, deviations and defective material and equipment are promptly identified and corrected.	12.4, 12.5
2.	In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken	12.4, 12.5

to preclude repetition.

3.	The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be:	
	a. Documented.	2.4, 12.5, 12.7
	b. Reported to appropriate levels of management.	12.4, 12.5, 12.7

XVII. QUALITY ASSURANCE RECORDS

1.	Sufficient records shall be maintained to furnish evidence of activities affecting quality.	14.3
2.	The records shall include at least the following:	
	a. Operating logs.	14.3, 14.4
	b. Results of reviews.	14.3, 14.4
	c. Inspections.	14.3, 14.4
	d. Tests.	14.3, 14.4
	e. Audits.	14.3, 14.4
	f. Monitoring of work performance.	14.3, 14.4
	g. Material analyses.	14.3, 14.4
3.	The records shall also include closely-related data such as qualifications of personnel, procedures, and equipment.	5.2, 7.4, 14.3, 14.4
4.	Inspection and test records shall, as a minimum, identify:	
	a. The inspector or data recorder.	5.2, 14.3, 14.4
	b. The type of observation.	5.2, 14.3, 14.4
	c. The results.	14.3
	d. The acceptability.	14.3

e.	The action taken in connection with any deficiencies noted.	14.3

5. Records shall be identifiable and retrievable. — 14.3

6. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as:

a.	Duration.	14.3, 14.4
b.	Location.	14.3, 14.4
c.	Assigned responsibility.	14.3, 14.4

XVIII. AUDITS

1. A comprehensive system of planned and periodic audits shall be carried out to:

a.	Verify compliance with all aspects of the QA Program.	4.11, 13.1, 13.4
b.	Determine the effectiveness of the Program.	4.11, 13.1, 13.4

2. The audits shall be performed:

a.	In accordance with written procedures or checklists.	4.11, 13.1, 13.4
b.	By appropriately trained personnel not having direct responsibilities in the areas being audited.	4.11, 13.2, 13.4

3. Audit results shall be:

a.	Documented.	4.11, 13.6
b.	Reviewed by management having responsibility in the area audited.	4.11.7, 13.6

4. Follow-up action, including reaudit of deficient areas, shall be taken where indicated. — 4.11, 13.7

APPENDIX II

This appendix provides a matrix of specific plant commitments to *QA Program related Regulatory Guides and referenced documents* and is included in this manual as a "quick reference". This appendix is not intended to be all inclusive of all commitments and has no specific relationship with the other sections of this manual.

It must be noted that the information depicted in the appendix reflects only those commitments found in Section 1.8 of each plant's (U)FSAR. It must also be noted that exceptions/clarifications taken by CP&L to these Regulatory Guides are not indicated in the reference and must be obtained from the applicable plant (U)FSARs.

Regulatory Guides are listed by number, title, and revision applicable to each plant. The document primarily endorsed by each Regulatory Guide are also listed with revision or date of issue, where applicable, for each plant. If a Regulatory Guide and/or an endorsed document is not applicable to a particular plant, it is so noted as "N/A" in the appropriate Commitment columns. In cases where Regulatory Guides do not endorse specific documents, "none" appears in the corresponding Primary Endorsed Documents column.

CP&L COMMITMENT MATRIX

APPENDIX II

Reg. Guide	Commitments			Regulatory Guide	Primary Endorsed Documents	Commitments		
	Reg. Guide Rev.					Document Rev		
	SHNPP	BSEP	HBR			SHNPP	BSEP	HBR
1.8	R2 2/79 Draft	3/71	9/75	Personnel Selection and Training	ANSI N18.1 ANSI 3.1	N/A 9/79	N/A N/A	N/A N/A
1.29	R3	R3	R3	Seismic Design Classification	None	-	-	-
1.30	R0	8/72	8/72	QA Requirements for the Installation and Testing of Instrumentation and Electrical Equipment	ANSI N45.2.4	3/72	3/72	3/72
1.33	R2	11/72	2/78	QA Program Requirements (Operation)	ANSI N18.7	2/76	2/76	2/76
1.37	R0	3/73	3/73	QA Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	ANSI N45.2.1	2/73	2/73	2/73
1.38	R2	3/73	3/73	QA Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants	ANSI N45.2.2	12/72	12/72	12/72
1.39	R2	3/73	3/73	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	ANSI N45.2.3	3/73	3/73	3/73
1.54	R0	6/73	-	Quality Assurance Requirements for Protective Coatings Applied to Water-cooled Nuclear Power Plants	ANSI N101.4	1972	-	-
1.58	R1	9/80	9/80	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel	ANSI N45.2.6	8/78	8/78	8/78
1.64	R2	10/73	10/73	QA Requirements for the Design of Nuclear Power Plants	ANSI N45.2.11	6/74	6/74	6/74
1.74	R0	2/74	2/74	QA Terms and Definitions	ANSI N45.2.10	5/73	5/73	5/73

CP&L COMMITMENT MATRIX

APPENDIX II

Reg. Guide	Commitments — Reg. Guide Rev.			Regulatory Guide	Primary Endorsed Documents	Commitments — Document Rev		
	SHNPP	BSEP	HBR			SHNPP	BSEP	HBR
1.88	R2	8/74	N/A	Collection, Storage, and Maintenance of Nuclear Power Plant QA Records	ANSI N45.2.9	6/74	6/74	5/79
1.94	R1	N/A	4/76	QA Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	ANSI N45.2.5	7/74	7/74 (Std. Only)	-
1.116	R0-R	N/A	N/A	QA Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	ANSI N45.2.8	5/75	5/75 (Std. Only)	5/75 (Std. Only)
1.120	N/A	N/A	N/A	Fire Protection Guidelines for Nuclear Power Plants	None	-	-	-
1.123	R1	N/A	N/A	QA Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	ANSI N45.2.13	2/76	Draft 2 Rev. 4 4/74	Draft 2 Rev. 4 4/74
1.143	R1	N/A	N/A	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water-Cooled Nuclear Power Plants	None	-	-	-
1.144	RO	1/79	1/79	Auditing of QA Programs for Nuclear Power Plants	ANSI N45.2.12	11/77	11/77	11/77
1.146	RO 8/80	RO 8/80	RO 8/80	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	ANSI N45.2.23	4/78	4/78	4/78

Supplemental Quality Assurance Requirements for the Design Change Packages Associated with Completion of the Units 2 & 3 Spent Fuel Pool Cooling System

SUPPLEMENTAL QA REQUIREMENTS

The following is a set of supplemental QA requirements developed for the implementation and turnover of Code items associated with the completion and activation of the Unit 2 & 3 Spent Fuel Pools at Harris Nuclear Plant. This document will be incorporated directly into the "Design Requirements" section of the design change packages for the pertinent modifications, and then by specific instructions in the appropriate sections (installation, testing, turnover, etc) as necessary to ensure that its requirements are implemented.

1.0 GENERAL

1.1 Scope

This document defines the set of QA requirements which will be used to govern the engineering, construction and startup of the Section III, Class 3 portions of the Spent Fuel Pool Facilities originally intended to service HNP Units 2 & 3. This portion of the plant was partially installed during original plant construction, but was suspended subsequent to cancellation of these units. The development of a supplement specific to this scope is necessitated by the following concerns:

- The original N-certificate associated with this program has long since been discontinued, and no partial turnover was conducted for the partially installed piping and equipment.
- The field construction documentation packages for partially installed piping have been discarded and are no longer available

As a result of the above, it is not possible to complete these systems in full compliance with Section III utilizing the previously installed piping and equipment. Since the N stamping process is the prescribed method for demonstrating quality assurance in construction activities, it is necessary to define a suitable alternate program which will ensure that the requisite level of quality exists upon completion and turnover. Generally, the corporate Nuclear Generation Group's Quality Assurance Manual is of suitable rigor to accomplish this. However, the program defined in the corporate QA manual was developed to comply with 10CFR50 Appendix B as it concerns operating plants, and was not intended to specifically conform to the requirements of Section III. For example, the corporate QA program outlines condition reporting requirements which govern field activities and meets the requirements of Appendix B in this regard. However, this program does not integrate involvement of the ANI in documenting adverse conditions, nor does it require the ANI to participate in the closeout of adverse condition reports. In addition, the current site procedures pertaining to field activities are generally oriented towards meeting the requirements of Section XI for inservice inspection, rather than Section III.

To address this issue, a set of QA requirements have been developed and are presented herein to supplement the corporate Appendix B QA Program. Generally, these requirements were the result of a review of the current corporate Appendix B Quality Assurance Program against the requirements of the approved ASME Section III QA

Manual utilized for construction of the Harris Nuclear Plant. These requirements are not intended to delete or revise any requirements in the corporate QA manual, but rather are to provide additional criteria in supplement of the existing program. These criteria will be implemented in one of the following manners:

Revision of site procedures: Since this supplement is not intended to contradict approved site procedures, this might be necessary to reconcile conflicts between the Supplemental QA Requirements and that of existing site procedures.

Incorporation through the work control process: When criteria are stipulated that are not already reflected in site procedures, it may be more suitable to add these through work planning and specific instructions in the work package. The requirements for additional involvement of the ANI would be an example of this.

Procedure revisions will be reflected by markups and inclusion on the Document Update Form (DUF), while work package implementation will be accomplished by specific instruction in the appropriate section of the modification package (implementation, testing, etc.).

1.2 Responsibilities

General - Programmatic responsibilities for implementation of the Corporate Appendix B program, including the site's Section XI Repair and Replacement Program, are as defined in the Corporate Quality Assurance Program Manual and supporting documents, including site procedures. The involvement of site organizations as pertains to the implementation of these supplemental requirements will be subject to their review and approval during the modification approval process.

AIA (ANI) - The Authorized Inspection Agency is responsible for providing the support necessary for implementation of the supplemental requirements described in this ESR. Acceptance of these requirements will be based upon NRC review and approval of the 10CFR50.55a Alternative Plan. Formal AIA endorsement of these supplemental requirements from a programmatic perspective will accomplished by review and approval of the modification packages which incorporate them.

Modification Engineer - The Modification Engineer for the affected ESR is responsible for implementing the requirements found herein in the most appropriate manner. This would include either revision of site procedures or through direct incorporation into the modification package, as described above.

Modification Responsible Engineer - This supplement pertains only to modification activities completing construction of the spent fuel cooling systems originally intended to service Units 2 & 3. As such, the ultimate responsibility for adherence for this rests with the RE for these modifications. Since this supplement will be incorporated into the modification packages, the RE is responsible for ensuring that the modification package contains sufficient instructions and guidance to implement it as written.

2.0 DESIGN AND DOCUMENT CONTROL

2.1 Design Control

Design Control over the modification design is directed and coordinated by CP&L in accordance with corporate and site procedures governing the modification process and design activities by outside organizations. This process results in rigorous design review process (including independent design verification) by the A/E and detailed owner's reviews by CP&L engineering personnel.

This supplement pertains only to modification activities completing construction of the spent fuel cooling systems originally intended to service Units 2 & 3. Generally, it is intended that completion of this portion of the plant will be governed by the same revisions of the Code that were utilized for original design and construction. To that end, the applicable version of the Code associated with a particular aspect of construction, and the boundaries of that applicability shall be clearly defined as design inputs in the modification packages. Later versions of the Code may be used only with reconciliation of any differences between it and the Code that was utilized for original design and construction.

2.2 Design Specifications

2.2.1 Design specifications will be prepared for all Code stamped items, in accordance with corporate and site procedures, and will be subject to the following requirements:

- The specification shall clearly delineate Code classification and boundaries and the pertinent code revision associated with the item.
- The specification shall address Code requirements for data reports, including any that may pertain to transmittal to enforcement authorities.
- The specification shall fully conform to Section III design requirements.
- The design specification shall be certified to be correct, complete, and in compliance with the code by one or more Registered Professional Engineers competent in the applicable field of design of components and related nuclear power plant requirements. It is noted that some of site's existing design specifications date back to the construction era, but may have been revised since the plant began operation. In these instances, it is acceptable to use previous certified revisions of design specifications, so long as a reconciliation of any subsequent revisions is performed to assess design impact and integration into the current the Appendix B Program.

2.3 Design Control

2.3.1 Design control shall be as directed in the corporate QA program as implemented by corporate and site procedures.

2.3.2 Design of Code stamped items shall conform to the version of the Code which would have been utilized during original plant construction. Later versions can be utilized only with documented reconciliation. Design criteria of Section III, Subsection ND shall apply to all Class III piping, equipment and components.

2.3.3 Subsequent revision to the affected modification packages shall also be subject to the supplemental requirements defined herein through completion of construction and the turnover process.

2.3.4 This supplement is "frozen" as it is incorporated into the 10CFR50.55a Alternative Plan and approved by the NRC. Design changes and modification revision packages shall not delete or revise the content or applicability of these supplemental requirements, in whole or part, without NRC approval.

2.4 Applicability of existing site procedures

2.4.1 It is appropriate to use the site Section XI Repair and Replacement as a guide for integration of site procedures with the construction of Code related items. Generally, existing site procedures shall apply as if the Code portions of construction were being performed as a Section XI Repair and Replacement activity. However, where this supplement contradicts existing procedure or program requirements, the requirements in this supplement shall take precedent and the affected procedure or program be revised as appropriate.

2.4.2 Welding, including weld procedures, welder qualification, weld material control, use and control of welder ID symbols and preparation of Weld Data Reports, will be done using the Corporate Welding Manual as invoked and implemented through site procedures.

2.4.3 The ANI shall have the opportunity to review procedures, including those for welding and QC, which will be utilized for Code related construction activities during the review of work packages prior to field issuance. Likewise, any revisions to these procedures which is intended to be utilized in the work package subsequent to the initial ANI review shall also be identified to the ANI for his review prior to its use.

2.5 Document Control

2.5.1 Document Control will be as currently defined in the corporate Appendix B QA program for quality related activities and implemented through site procedures.

2.6 Identification of ASME code Documents

2.6.1 Purchase requisitions, purchase orders, procedures and other documents generated and / or used at the site for fabrication and installation of Code items shall be identified as "ASME Section III".

3.0 PROCUREMENT

3.1 General

3.1.1 The A/E may provide input into the procurement process, however, all procurement will be performed by CP&L under its existing Appendix B Quality Assurance Program and implemented by corporate and site procedures.

3.1.2 Procurement of all code stamped items will be accomplished using approved design specifications certified by a Registered Professional Engineer competent in nuclear power plant design.

3.2 Service Contracts

3.2.1 Service Contracts intended to obtain services associated with the engineering or construction of piping and equipment affected by this supplement shall be subject to all the rules and requirements of this supplement.

3.3 Code Stamped Items

3.3.1 It is intended to complete construction to the version of the Code to which the system was originally designed and specified, which governed construction of the existing portion of piping and equipment installed during initial plant construction. The applicable version of the code associated with a particular aspect of procurement or construction and the boundaries of that applicability shall be clearly defined in the modification package. Code stamped items shall be clearly identified as such in the modification BOM or the Equipment Commissioning List. Code stamped items shall be specified and procured so as to fully comply with Code requirements, including the use of qualified suppliers with appropriate Code certification, and shall be stamped in accordance with code requirements.

The BOM or the Equipment Commissioning List shall, as a minimum, contain the following information regarding Code stamped items:

Commercial information which sets forth items, quantities, terms, conditions, etc. as appropriate, as well as the approved Design Specification(s) which defines the engineering and quality requirements.

3.3.2 Any exceptions to the Design Specifications taken by the supplier with regard to a Code stamped item shall be reconciled by revision to the affected Design

Specification prior to proceeding with procurement. Any such revision to the Design Specifications would be prepared, reviewed and approved as set forth for the original specification.

3.4 Qualification of Suppliers

3.4.1 Qualification of Suppliers of materials and services shall be accomplished in accordance with the existing CP&L Appendix B Program in accordance with approved plant procedures. All suppliers must be verified as being on the approved supplier's list for the scope of supply and holding active certification from the ASME for any Code items being procured.

4.0 RECEIVING INSPECTION

4.1 Code stamped items

Inspection, examination and acceptance of Code items shall be accomplished in accordance with corporate and site procedures. Receipt activities shall be documented in the form of a Receipt Inspection Report (RIR). Items accepted shall be appropriately tagged / labeled.

Nonconformances noted during receipt inspection shall be reported via Condition Report (nonconformance) initiation, and the affected items placed on hold or rejected. When the vendor's data package is missing or deficient, the item will be placed on hold pending the delivery of the missing information or resolution of the deficiency.

When conditions warrant, Conditional Release requests may be granted to permit progression of work involving a nonconforming item awaiting resolution. When this occurs, it will be processed and approved in accordance with existing site procedures. The ANI will be provided with the closure documentation for any conditional releases affecting Code stamped items or Code related construction.

5.0 STORAGE AND PROCESS CONTROL

5.1 Storage

Storage requirements for Code stamped items will be clearly identified in the Design Specification. Storage control through manufacture and shipment will be governed by the procurement process.

5.2 Equipment Commissioning Plan

5.2.1. General

This section prescribes the methodology which will be followed in commissioning previously installed equipment in support of completing and activating the C & D Spent Fuel Pools. The subject equipment was installed during the original site construction effort for Unit 2 & 3 fuel storage and handling activities, and was spared in place when these units were cancelled. This equipment was never incorporated into the operating unit nor has it been formally maintained under controlled storage conditions since that time. Note that the equipment in question (including Code related equipment) was procured to applicable design and quality assurance requirements, and this plan does not take exception to any of these requirements. Rather this plan prescribes a set of criteria which will ensure that the equipment in question will meet the applicable requirements of Appendix B and is capable of performing its intended function in the completed design.

5.2.2 Field Walkdown / Scope Development

Scope development is accomplished by performing a detailed field walkdown and comparing the modification design to the field condition. The entire list of previously installed equipment (both Code and non-Code related) which is anticipated to be used in the completed design will be compiled to comprise the scope of the Equipment Commissioning Plan. Note that this plan is not limited to mechanical equipment, and will include civil (pipe supports, penetrations), I&C (instrument racks, instrumentation, tubing) and electrical (cables, conduit, cable trays, equipment ground connections) as well. Each item in scope will be identified and individually dispositioned in the modification package.

5.2.3 Document Review / Retrieval

A document retrieval and review process will be included in the matrix of commissioning requirements to ensure that required quality assurance information is on hand. Generally, equipment commissioning matrix documentation requirements will be consistent with that of the original procurement effort. In particular, all Code documentation requirements (including Code data reports) must be satisfied for Code items. Records required for commissioning fall into one of two categories, which are discussed as follows:

(a) Procurement Documentation
This documentation pertains to the information which was originally used to procure the equipment in question and the vendor quality packages which were supplied with the item in response. These records are required to establish traceability and verify that required vendor quality assurance documentation and

quality releases are on file. Generally, this information is available in the Receipt Inspection Report (RIR) generated at the time the item was received. It is <u>not acceptable</u> to assume that the necessary information must have been received and is in order by virtue of its being installed in the field under control of the construction program, as it would have been possible to have issued the item to the field with a conditional release with outstanding quality related issues pending. All Code equipment must have traceability to the Code Data Report(s) for its construction.

(b) Field generated records
Construction records must be reviewed to ascertain to what extent the existing field condition was documented as being complete and satisfactory. Generally, this information exists in the equipment installation packages and has been maintained in document control for the major pieces of equipment in question. Once the equipment installation records have been retrieved, these must be compared against the field condition to verify that the installation as accepted has not been subsequently altered. Previous construction activities can be accepted for use in the modification implementation effort to the extent that required installation documentation exists and is verified to conform to the field condition.

In the event that records are found to be missing or deficient, an assessment is performed to determine what installation can be accepted by virtue of retest or re-inspection, or by use of alternate methods of verification. Alternately, the implications of the documentation deficiency can be evaluated to determine the potential impact to quality. Any such evaluation used to accept field conditions in the absence of required information must be formally documented and subject to design review as appropriate. Except as specifically provided in the 10CFR50.55a Alternative Plan for records of field installation of piping, this equipment commissioning plan is not intended to take exception to Code requirements pertaining to equipment installation or documentation requirements. Given this single exception, an evaluation of a deficiency is not allowed to stand in lieu of installation records which are deemed to be specifically required by Section III of the ASME B&PV Code.

5.2.4 Development of examinations, tests and acceptance criteria

The Equipment Commissioning Matrix shall specify any additional activities necessary to ensure the requisite level of quality assurance in light of the lack of formal controls on storage and handling since this equipment was initially installed. Development of these activities will include the following:

- Field verification of equipment identification against procurement documentation. In the case of Code related equipment, traceability will be established to the Code Data Report(s) and National Board Registration.
- Physical inspections, testing, etc., as required to verify that lack of controlled storage conditions and regular maintenance has not caused any condition affecting

quality. Commissioning criteria shall include consideration of corrosion, fouling, aging, radiation exposure, etc. For Code requirements, any degradation identified would be assessed in terms of Code requirements, with acceptability based on demonstrated compliance with those requirements.

- Physical inspections and considerations necessary to ensure that plant activities since construction have not resulted in any condition potentially adverse to quality (scavenging of parts, introduction of foreign material, damage from personnel and equipment traffic, etc). For Code equipment and piping, these criteria will specifically consider Code required attributes, with acceptability based on full Code compliance.

5.2.5 Repair of Deficiencies

Repair of any deficiencies shall be done in accordance with approved procedures. Since Code items in the scope of this equipment commissioning plan are supplied as completed Section III components from the vendor under that vendor's NPT Stamp Program, repairs to these items meet the definition of "Repairs" in ASME Section XI and shall be accomplished under the site's Section XI Repair and Replacement Program.

5.2.6 ANI Involvement

Code stamped equipment and related commissioning requirements will be specifically identified as such in the modification package in order to facilitate the system certification process. Provisions shall be made to ensure that any work packages generated to commission Code equipment are made available for ANI review subsequent to work completion.

5.2.7 Revising or Altering the Equipment Commissioning Plan

Generally, this equipment commissioning plan does not take exception to Code or quality requirements, but rather prescribes a dedication process which will ensure that all such requirements are met in light of the lack of storage control for the equipment it addresses. The sole exception is with regard to field installation records for Code related piping, which are no longer available and are the subject of a 10CFR50.55a Alternative Plan currently under review by the NRC. Acceptance of the field installation of this piping is contingent upon approval of this Alternative Plan by the NRC, and revising the Equipment Commissioning Plan with regard to piping acceptability may require prior notification of the NRC. Otherwise, this plan does not take exception from design or quality requirements (including ASME Code requirements), and authorization for its use and any revisions to it are provided under 10CFR50.59.

5.3 Process Control

Process control sheets are utilized to establish measures to ensure that processes, including welding and heat treating, are controlled in accordance with the Code and are accomplished by qualified personnel using qualified procedures. Generally, process control sheets for Code related construction activities will be as provided for under the site's procedures. Additional process control sheets are found in the Corporate Welding Manual and Corporate NDE Manual, as invoked and implemented by site and corporate procedures.

The ANI will review process control sheets for code related construction activities before they are issued to the field for construction. The ANI will have the opportunity to add any inspection hold points deemed necessary at this time. All process control sheets for Code related construction activities will be reviewed and accepted by the ANI subsequent to completion of field activities.

The hydrostatic test pressure used for pressure testing shall be required to meet Section III requirements, as opposed to those specified in Section XI. The process control sheets for hydrostatic testing shall reflect the more stringent test criteria.

Nonconforming field conditions will be controlled by site work process control and condition reporting procedures. The ANI will be notified of any condition reports initiated against code related construction activities, and will verify any such items are resolved prior to signing off the process control sheets for final acceptance.

Identification tags or markings shall be retained on each code item. When it is necessary to cut or transfer an item during code related construction, material identification shall be transferred to the affected piece prior to cutting. This activity shall be witnessed by QC and appropriately documented in the work package.

5.4 Modification Implementation Procedures

5.4.1 Modification procedures are being utilized for code construction (in the context of this ESR) will be those presently existing for use with the site's Section XI Repair and Replacement Program, subject to the supplemental requirements prescribed herein.

5.5 Start-up Procedures

5.5.1 Detailed start-up procedures will be developed and included in the affected modification package. Review of start-up procedures, including QC review, will be documented by review and signature approval as part of the modification approval process.

6.0 WELDING CONTROL

6.1 General

Welding activities associated with Code construction, including welding procedure qualification, weld materials procurement and control, welding equipment control, qualification of welders, weld process control and post weld heat treatment activities shall be controlled in accordance with the Corporate Welding Manual by the Plant Welding Engineer and the Plant Operating Manual. Welding may be performed by Contractors provided that the contractor is fully qualified to CP&L's welding program for the specific welding or welding related activity being performed.

Contractor's not qualified to and working under CP&L's Corporate Welding Program may only be used for Code welding activities for which they maintain their own program having the appropriate ASME certification. In this case, a service contract must be provided which authorizes the Contractor to invoke his program for the subject scope of work.

Work packages involving welding activities associated with Code construction will be reviewed by QC and the ANI prior to field issuance to ensure that appropriate hold points are included. Weld Data Reports shall be generated for any such welds per the Corporate Welding Program, and hold point inspections shall be accepted by QC and the ANI by signature and date on the WDR.

7.0 CONTROL OF EQUIPMENT, TOOLS, GAUGES AND INSTRUMENTS

7.1 General

Equipment, tools, gauges and instruments specified for calibration control shall be identified, stored, calibrated, and maintained in accordance with site procedures. Calibrations and adjustments shall be accomplished at prescribed intervals and against certified standards having known valid relationships to national standards. If no national standard exists, the equipment manufacturer's recommended standard shall be used. Recalibration shall be performed any time the accuracy of an instrument is suspect.

Traceability shall be maintained between the instrument and equipment or item being tested. The instrument identification number shall be recorded on the appropriate process control documentation. In the event an instrument is found to be out of calibration, a Condition Report must be initiated and an evaluation shall be performed to identify and disposition any suspect inspections, examinations, and test results.

8.0 INSPECTION, TESTS and NONDESTRUCTIVE EXAMINATION (NDE)

8.1 General

NDE activities associated with Code construction, including NDE procedures, qualification of personnel and control of inspection and test equipment shall be accomplished as provided in the Corporate NDE Manual. NDE procedures and acceptance criteria are provided in the Corporate NDE Manual for both original construction code and Section XI requirements. NDE shall be performed on all Code related construction activities in these modifications consistent with Section III requirements, and all such NDE shall utilize Section III acceptance criteria.

8.1.1 Process Control

Inspection, test and examination requirements shall be defined in the work packages and documented on appropriate process control sheets. These packages will be reviewed by the QC and ANI prior to field issuance. Work will not progress past established QC and ANI hold points until the hold point is accepted by signature and date by the QC inspector or ANI.

8.1.2 ANI Review and Approval of NDE Documentation

Records of inspections, tests and examinations containing QC and ANI hold points will not be considered completed until all such hold points are satisfied and the ANI has completed his inspection and signed and dated the process control sheets.

9.0 CODE DATA REPORT AND CERTIFICATION

9.1 General

The piping systems completed under these modifications will not be eligible for N stamping due to issues pertaining to the discontinuance of the original construction program and missing documentation. However, these systems will undergo a certification process similar to N stamping. Installation of Code piping, equipment and components will be documented on an ASME Section III data report "equivalent form". This form will be comparable to an NIS-2 form associated with Section XI repair / replacement activities, and PLP-605 can be used as a guideline for its completion. All work packages for installation of Code equipment shall be clearly identified as such, and provided to the ANI for review prior to field issuance and again upon completion of work activities. Completed and approved documentation pertaining to Code related construction, including field generated records and vendor data packages, shall be compiled in packages pending the review of the ANI for system turnover.

The ANI will review the documentation and certify completeness and conformance with the requirements of the corporate Appendix B Manual and these supplemental requirements prior to system turnover. Since these supplemental requirements will be implemented either by procedure revision or modification instruction, this certification will be accomplished by verifying that all Code related activities were conducted and documented in accordance with site procedures and the requirements of the modification package. The specific list of items reviewed to determine completeness and conformance will be provided as an attachment to this certification. Similar to the N-5, this listing will constitute the boundaries of the completed construction which would have normally been N-stamped.

The completed certification of the affected piping, equipment and components will be included in the modification documentation package as a permanent QA record.

10.0 NONCONFORMANCE AND CORRECTIVE ACTION

10.1 Nonconformance and corrective actions will be addressed within corporate and site procedures, including those associated with procurement, work control and condition reporting. Satisfactory resolution of any non-conformances or adverse conditions associated with code stamped items or code related construction activities will be verifiable by the ANI and all other responsible parties prior to turnover.

11.0 RECORDS CONTROL AND RETENTION

11.1 Records control and retention will be as directed by site work control and document control procedures, except as related to the ANI's role in certification as described herein.

12.0 AUTHORIZED NUCLEAR INSPECTOR

12.1 The services of an AIA shall be used as described herein. It is noted that a qualified ANI would be necessary for Section III construction activities, while an ANII is involved when performing repair and replacement activities under Section XI. Since elements of both are associated with this modification, dual qualification will be required for the AIA's site representative involved with this modification. Signoffs for this individual will reflect this dual qualification (ANI / ANII).

13.0 REVIEW, CONTROL AND REVISION OF SUPPLEMENTAL QA REQUIREMENTS

13.1 These supplemental requirements as incorporated into the modification design and approved therein will become part of a 10CFR50.55a Alternative Plan and therein subject to NRC review and acceptance. Since NRC acceptance for the alternative plan represents the authorization for these supplemental QA requirements, revision to these requirements can only be accomplished by submittal and review of the NRC as a revision to the Alternative Plan. Exceptions would be allowed only for revision to items which comply with all Code and Regulatory requirements and are provided for completeness and clarity (see Equipment Commissioning Plan), or administrative or clerical changes which do not affect technical requirements.

Comparison of CP&L ASME Section III QA Manual
vs.
Present QA Program Requirements.

Introduction

The basis for the overall quality assurance program used by Carolina Power & Light Company for the design and construction of the Shearon Harris Nuclear Power Plant is described in the PSAR. PSAR Section 1.8 states that "The Carolina Power & Light Company Quality Assurance Program for the engineering and construction of the Shearon Harris Nuclear Power Plant, which includes the quality assurance programs for both Ebasco and Westinghouse by reference, is structured with regard to safety-related equipment in accordance with the eighteen criteria of Appendix B to 10CFR50. In addition, the subject Program is structured in accordance with ANSI N45.2 and thereby Regulatory Guide 1.28 ...". The PSAR further states that the "Shearon Harris Nuclear Power Plant Quality Assurance Plan" was replaced by the "CP&L Corporate Quality Assurance Program" on Aprill, 1974, and provides a cross reference on how the subject plan met the criteria of 10 CFR50 Appendix B.

Certain aspects of Shearon Harris Nuclear Power Plant construction were subject to QA requirements beyond those outlined in the CP&L Corporate QA Manual. Since CP&L was not only the Owner, but also the constructor, installer and a fabricator of Code items for the Shearon Harris Nuclear Power Plant, an additional set of QA requirements were required to be developed, reviewed, approved and implemented specifically in order to obtain the required ASME Certificates of Authorization. ASME Code Section III, Subsection NA requires that an applicant for a Certificate of Authorization develop a QA program and implementing procedure specific to the proposed scope of work, and that the "the applicant shall request the Society to evaluate this procedure and Program prior to the issuance of a Certificate of Authorization." For construction of the Shearon Harris Nuclear Power Plant, CP&L met this requirement by the formalization of its "ASME Quality Assurance Manual". Section 1.1 of this manual (Scope) states that

> "This manual provides measures to assure compliance with the requirements and rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components. This Manual shall be applied to activities associated with plant items and services for which compliance with the rules of the ASME Code, Section III, is applicable".

It is important to note that, while the CP&L ASME Quality Assurance Manual may have shared certain common facilities, procedures, personnel, etc. with the overall site QA program, it did not rely on the larger program to demonstrate compliance with Code requirements. The CP&L ASME Quality Assurance Manual was specifically the QA Program reviewed and approved by the ASME for the purpose of granting N, NA and NPT Certificates of Authorization to CP&L for the Shearon Harris Nuclear Power Plant, and the program regularly subjected to ASME audit in order to maintain those authorizations. Therefore, in formalizing a QA Program for the completion of Construction of the Unit 2 Spent Fuel Pool Cooling Systems, it is appropriate that the requirements of this CP&L ASME QA Manual be compared against those of the current Corporate Appendix B QA Program. The results from this comparison would provide the basis for a set of "Supplemental QA Requirements", which would be used to facilitate completion of construction in accordance with Section III to the extent feasible, given the issues of missing documentation and no partial turnover for previously installed equipment.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
1.0 Scope			
1.1.1	The Construction QA manual was intended to provide measure to assure compliance with the requirements and rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, and was structured in accordance with the requirements of Section NA, Subsection NA-4000. This manual was applicable to activities associated with plant items and services for which compliance with Section III was mandatory.	The Corporate (Appendix B) QA Manual (QAM) establishes measures for assuring that organizations performing safety-related activities perform their responsibilities in a manner which results in safe nuclear power production. This manual also establishes QA programs for certain non-safety related areas of the plant, such as Rad-Q, FP-Q and Quality Class B. The Corporate QA Manual is not inclusive, but is intended to be used in conjunction with Section 1.8 and 17.3 of the FSAR to define the overall program and effect the development of procedures that implement that program.	The CP&L Corporate QA Program meets the eighteen QA criteria in Appendix B and is also the umbrella QA program for the site ASME Section XI Repair and Replacement Program. Much as would have been done at original construction, it is CP&L's intention to use the Corporate QA Program as the umbrella program to complete and activate the Unit 2 Spent Fuel Pools, augmenting this program with supplemental requirements extracted from the ASME QA Manual with the intent to achieve compliance with Code requirements to the extent feasible. NA-4133.2 defines the requirement for the AIA to review any significant changes to the ASME QA program. The design change package(s) for this activity will be subject to ANI review / approval. This will include review of the supplemental QA requirements and the turnover / certification process.
1.1.2	Identifies CP&L as the Owner, as well as the constructor, Installer and Fabricator	Written with CP&L as Owner / Operator (Ref. REG-CPL-000; CP&L Quality Assurance Program Policy)	No ongoing construction program in place. CP&L proposes to complete construction under Appendix B program much as would be done if repaired / replaced under Section XI, but using more stringent Section III criteria
1.1.3	Specifies that supporting companies shall operate in accordance with QA programs which are in compliance with this manual	Supporting company's activities will be directed either by contractural agreement or the supplier's QA program reviewed / approved by CP&L before issuance of PO or contract. (QAM 4.4).	QAM requires that an ASL be maintained, this is accomplished under MCP-NGGC-0406. MCP-NGGC-0406 also requires that contracts either utilize a CP&L approved program or that the CP&L program be invoked Existing program ensures that supporting companies operate either in accordance with the CP&L QA manual or their CP&L approved program. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
1.1.4	Specifies that the Constructor shall operate in accordance with this manual (no separate QA program)	See 1.1.3, above	All construction activities in the scope of the Alternative Plan shall be accomplished in accordance with the CP&L QA program or a CP&L approved QA program.
1.2 Responsibility for the QA Program	Responsibility for the Quality Assurance Program with Senior Vice President - Power Supply and Engineering & Construction.	QA Program approved by the Senior Vice President, Nuclear Generation Group (Ref. NGGM-PM-0007; QAM 2.2)	Comparable level of management responsibility. No supplement requirements needed.
1.3 Organization and Responsibilities	As shown in organization chart of the era. Predictably, this chart reflects the departments and personnel typical of a construction oriented organization, such as the Harris Plant Construction Section, and numerous management positions specific to the construction effort.	As described in FSAR 17.3, an organization fairly typical of operating plants. One noteworthy change from the construction organization is the transition to relying on the principle that the line organization has the primary responsibility for quality and safety. As such, the functions of the QA / QC Section which existed during the construction era are now largely satisfied by continual self-assessment, with evaluation / oversight of this program being provided by the Nuclear Assessment Section. (Ref. FSAR Section 17.3.1.1)	No changes to the site organization to be implemented as a result of this activity.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
1.4 Training and Qualification	Each Dept Manager for the Construction Site and General Office responsible for developing procedures for training and indoctrination. As a minimum, personnel will be trained in this QA Manual, supporting procedures and subsequent changes.	Each Dept head responsible - personnel performing activities affecting quality shall be indoctrinated and trained such that they are knowledgeable in the applicable quality related procedures and requirements. (QAM 7.4.2)	During the construction era, procedure TP-25, "Training of Supervisory and Technical Personnel in Implementation of ASME N Stamp Program" was developed to indoctrinate personnel in the CP&L N Certificate Program. The scope of the SFP Activation Project is not such that a large scale training effort is warranted, rather, training classes shall be held for supervisory and technical personnel which are directly responsible for the design, installation, startup and turnover of the Unit 2 Spent Fuel Pools. The purpose of this training will be to indoctrinate these key personnel on the Alternative Plan and its impact on the construction effort.
1.5 Delegation of Responsibility	Allows delegation of responsibility for any activity delineated in the manual	Requires that the authorities and duties of persons and organizations performing activities affecting quality be clearly established and delineated in writing. (QAM 2.4)	No specific delegation of responsibility proposed for this activity. Any special roles or duties of key personnel responsible for implementing the Alternative Plan shall be defined in the "Supplemental QA Requirements" and incorporated into the modification package for that activity.
2.0 Design and Document Control			
2.1 Design Control by Engineering Organization	Specifies CP&L participation in design, including maintaining control over engineering activities, reviewing, approving A/E and selected NSSS designs, directing document distribution, generating / updating design documents in accordance with authorizing procedures	Defines requirements for design control, including interface with design organizations (QAM 3.10)	CP&L responsible for design control for out-sourced design work, and performance of reviews as necessary to accept design products from outside organizations and assume responsibility for the design. (Ref. EGR-NGGC-0005) For this activity, design performed by A/E (Bechtel) through approved interface agreements. Implementation of the Alternative Plan integrated into the design change packages through Bechtel and subject to CP&L owner's review.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
2.2 Design Specifications, Calculations, Stress and / or Design Reports Section 2.2.1	Lists specific requirements for Design Specifications, calculations, design reports	Requires that measures be established to assure that applicable requirements are translated into design documents. (QAM 3.3, 3.4)	Overall design requirements for the design change package provided in EGR-NGGC-0005. Procedural requirements for content of design specifications found in ENP-013, and for calculations in ENP-011. ENP-013 requirements pertaining to content of specifications of similar rigor to that found in the ASME QA Manual. Also, note that per ENP-013, procurement specifications for Q-List equipment shall comply with the applicable sections of ANSI N45.2.13, Section 3.2.
2.2.2	Requires certification of the Design Specification by one or more Registered Professional Engineers competent in the applicable field of design of components and related nuclear power plant requirements.	Requires that measures be established to assure that applicable requirements are translated into design documents. (QAM 3.3, 3.4) Has no comparable requirement regarding certification of design specifications by Professional Engineers	Requires review/ approval of design specifications and design change packages; PE certification of specifications, calcs or design change packages not required under ENP-013, ENP-011 or EGR-NGGC-0005. Supplemental QA requirements for implementing Alternative Plan to require that design and procurement specifications associated with Code portions of design change be subject to PE certification. Note that this is generally not a significant issue, as most of these specifications have not been revised since the construction era.
2..2.3	CP&L has responsibility for assuring that copies of the certified Design Specifications are maintained and made available for the ANI and the NC enforcement authority having jurisdiction before Code items are placed into service	Requires that Design Specifications, as QA records , be maintained and retrievable in facilities that prevent deterioration, damage or loss. No ongoing requirements for reviews by the ANI or state enforcement authorities.(QAM 14.3)	Design specifications, design change packages, calculations, etc. are available in Document Control for review by ANI and other authorities and agencies. NC State Dept of Labor Boiler and Pressure Division has been briefed on the Alternative Plan and will conduct an independent review for the purpose of granting variance, relief from State requirements as deemed appropriate.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
2.2.4	The approved design organization responsible for Code items shall provide Design Specifications that are in accordance with Code.	Requires that measures be established to assure that applicable requirements are translated into design documents. (QAM 3.3, 3.4)	ENP-013 requires that codes and standards to be utilized in the design, fabrication, testing, delivery, and inspection of specified equipment, components and materials be appropriately identified, and that codes or standards and their effective dates are consistent with regulatory and plant modification requirements. Supplemental QA requirements developed for this project specifically require that the modification design fully conforms to Section III design requirements.
2.2.5	CP&L as the N Certificate Holder, is responsible for the design of piping systems, etc., and the adequacy and completeness of design documents. CP&L shall be responsible for assuring that Stress and or Design reports are prepared as set forth in the Code.	No N Certificate requirements associated with current program; however, CP&L as the licensee does maintain ultimate responsibility for configuration / design control issues.	Code portions of scope are Class 3. No formal stress reports required per ASME Section III requirements. Design inputs and parameters are delineated in the design change per EGR-NGGC-0005. Piping stress calculations are provided for the design as appropriate.
2.2.6	Requires review of the Certified Stress Report	N/A	No formal stress reports required. Design, including piping stress calculations, subject to plant review and approval per site procedures.
2.2.7	Lists requirements associated with modifications of any design document from the revision used in preparing a Stress Report	N/A	No stress report is required. Nonetheless, modification, review and approval of any design document at HNP is accomplished in accordance with applicable plant procedures which ensure the appropriate level of scrutiny is applied. Also, note that an electronic records management system (NRCS) serves to track document revisions and impacts to affected documents (Ref. NGGD-0300, PLP-202).

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
2.2.8	Addresses requirements for Code Class 1 and 2 steam and feedwater systems.	N/A	N/A, since scope is limited to Section III, Class 3 only.
2.3 Design Changes 2.3.1	Design changes shall be controlled in accordance with design control measures applied to the original design and require review / approval by the organization performing the original review. Design change approval required prior to final acceptance by QA/QC and the ANI. The design organizations and CP&L are responsible for design changes	Design changes controlled in accordance with design control measures which require consideration of design requirements. (QAM 3.3, 3.4)	Appropriate level of design change control exists in current program. Design change packages from outside suppliers subject to CP&L reviews. (Ref. EGR-NGGC-0003, 0005) Relative to the "Supplemental QA Requirements" associated with the Alternative Plan and subject to NRC approval, these will not be changed without notification / submittal to the NRC as appropriate. Also, note that the turnover process integrated into the modification package in the form of "Supplemental QA Requirements" requires that the ANI certify that all Code related activities were conducted and documented in accordance with applicable procedures and the modification package. These measures will ensure that design changes are fully approved prior to final turnover and declaration of operability.
2.3.2, 2.3.3, 2.3.4	Defines process for generating design changes (Field Change Request / Permanent Waiver)	Defines process for generating design changes (QAM 3.5)	Changes to design package controlled per EGR-NGGC-0005, will be processed as a revision and subject to appropriate level of reviews. Appropriate level of design change control exists. No supplement necessary.
2.4. Site Generated Specifications, Drawings and Procedures 2.4.1	This section deals with requirements for generation of documents associated with assembly, fabrication and installation of Code items at the construction site.	Generation and approval of site documents per applicable site procedures (QAM 6.0)	Generation of documents accomplished per modification requirements and plant procedures consistent with Appendix B requirements. No supplement necessary.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
2.4.1.1	Measures shall assure that documents and changes are reviewed for adequacy by authorized personnel and are available for use at the location where the activity is performed.	Requires that measures be established to assure that activities affecting quality are reviewed prior to issue and to control the issuance of documents. (QAM 6.1, 6.4.4, 14.4)	Generation, review, approval and retention of design documents controlled per corporate and site procedures. NRCS provides tracking for revision level and outstanding impacts. Appropriate level of design review and control of design documents exists. No supplement necessary.
2.4.1.2	Documents shall be reviewed by appropriate personnel.	Requires that measures be established to assure that designs and procedures are reviewed to ensure appropriate criteria and design inputs have been specified. (QAM 3.4.2, 6.4.4)	Review requirements for documents defined in site and corporate procedures. (Ref. EGR-NGGC-0003) Appropriate level of design review exists. No supplement necessary.
2.4.1.3	Copies of documents applicable to Code items shall be made available to the ANI and enforcement authority.	Requires that Design Specifications, as QA records , be maintained and retrievable in facilities that prevent deterioration, damage or loss. No ongoing requirements for reviews by the ANI or state enforcement authorities.(QAM 14.3). All documents associated with Code activities will be available in Document Control..	The design change packages, including revisions associated Code activities, will be subject to ANI review and approval per the requirements of EGR-NGGC-0005. Records will be provided to the NC DOL Boiler and Pressure Division as needed to support their review of this activity.
2.4.2	The Discipline Managers have overall responsibility for control and development of site-generated specifications used for field procurement or fabrication of Code activities.	Defines responsibilities and requirements for the development and control of design documents. (QAM 3.3, 3.4, 3.5)	Responsibility for specifications in accordance with ENP-013. Adequate level of responsibility exists. No supplement necessary.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
2.4.2.1, 2.4.2.2, 2.4.2.3	Delineates specific requirements associated with content, review and approval of site generated specifications. Requires that design specifications for Code items be certified by PE	Requires that measures be established for selection and review for suitability of application of materials, equipment and processes that are essential to safety related functions of structures, systems and components. (QAM 3.4, 3.5)	Site generated specifications per ENP-013; requires that codes and standards to be utilized in the design, fabrication, testing, delivery, and inspection of specified equipment, components and materials be appropriately identified, and that codes or standards and their effective dates are consistent with regulatory and plant modification requirements. ENP-013 requirements pertaining to content of specifications are of similar rigor to that found in the ASME QA Manual. Also, note that per ENP-013, procurement specifications for Q-List equipment shall comply with the applicable sections of ANSI N45.2.13, Section 3.2. Supplemental QA requirements states that all design specifications for Code items will be PE certified.
2.4.3	The Manager -QA Services has overall responsibility for development and control of Corporate Quality Assurance Dept. procedures.	Quality assurance integrated into line organization procedures and processes. NAS provides oversight; evaluates performance / effectiveness. (FSAR 17.3.1.1)	No supplement necessary.
2.4.4, 2.4.4.1, 2.4.5	Defines responsibility for development and approval of Construction administrative, technical, work and startup procedures related to Code items.	Requires procedure development and adherence for items affecting quality (QAM 6.4). Also, CP&L complies with Reg. Guide 1.33 as described in FSAR Section 1.8.	Specific administrative requirements are included in the "Supplemental QA Requirements" including the role of the ANI in Code related activities and defining the turnover process. Startup procedures shall be provided as appropriate in the design change packages for the work involved.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
2.4.6	Defines responsibilities and process for preparation, review, approval and revision of instrument isometric sketches. Defines process for red ink changes to these drawings, as well as criteria for rerouting without "red-lining".	Defines requirements for design control (QAM 3.3, 3.4)	Preparation and control of design drawings / sketches accomplished by ENP-012. Changes to approved design sketches accomplished through ESR revision process. Latitude for rerouting without drawing change limited to tolerances; defined in MMP-003. Appropriate level of design control exists. No supplement necessary
2.5 Site Document Control Section 2.5.1	Defines records management methods for distribution and control of specifications, drawings and work packages. Requires that documents issued "for info only" be appropriately stamped to preclude using for construction.	Requires that measures be established to control the issuance of documents which prescribe activities affecting quality (QAM 6.1, 6.34, 14.4)	Records management processes for distribution and control defined in RMP-002, 006. Verification of working document requirements provided in PLP-202. No supplement necessary
2.5.2, 2.5.3	Requires that document revisions be controlled in accordance with measures, including review and approval authorities, applied to the original document. Requires that provisions be made to assure that current revisions of documents are available for use. Defines distribution transmittal requirements.	Requires that measures be established to control the issuance of documents which prescribe activities affecting quality (QAM 6.1, 6.34, 14.4)	Procedural requirements for verification of working documents found in PLP-202, use of NRCS allows real time verification of revision level and affected documents. Control and distribution of documents / revisions accomplished through RMP-002 and related procedures. Current program provides adequate assurance that revisions are properly controlled, and that design and construction activities are accomplished using the latest effective document revision. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
Section 2.6 Identification of ASME Code Documents 2.6.1, 2.6.2	Requires that Purchase Requisitions, purchase orders and procedures used for fabrication and installation of Code items be identified as ASME Section III.	Requires that procurement control measures be established such that applicable regulatory requirements, design bases, etc are suitably included or referenced in the procurement of material, equipment and services (QAM 4.5).	Material to be procured specified in design change package through BOM per EGR-NGGC-0005. Materials acquisition controlled through MCP-NGGC-0002, 0401.Procedural controls are in place to ensure that appropriate considerations are made in materials procurements. However, to ensure that all Code material is procured as such, the "Supplemental QA Requirements" states that all Code items to be procured for this project shall be clearly denoted as such on the procurement documents and the design change package BOM.
3.0 Procurement			
3.1 Service Contracts	Defines requirements for services contracts, including those for engineering consultants & A/Es and Constructor and / or Construction Manager	Requires that procurement control measures be established such that applicable regulatory requirements, design bases, etc are suitably included or referenced in the procurement of material, equipment and services QAM 4.5).	Development of contract and contract administration governed by MCP-NGGC-0001; qualification of suppliers and audits accomplished per MCP-NGGC-0406 Suppliers of Code items must be appropriately qualified and on the ASL for Section III materials. No supplement required.
3.2 Procurement by the A/E 3.2.1 - 3.2.12	States that the A/E is responsible for procurement of Code stamped items on behalf of CP&L, outlines bid and evaluation process for suppliers.	Per approved interface agreements for this project, the A/E is responsible for providing complete specification to facilitate procurement, but procurement process will be accomplished by CP&L under their program and procedures.	Suppliers of Code items must be appropriately qualified and on the ASL for Section III materials and services. The bid and evaluation process outlined in the ASME QA Manual has been supplanted by the process for identifying, qualifying and auditing of suppliers per MCP-NGGC-0406. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
3.3 Site Procurement 3.3.1 - 3.3.14	Defines requirements / responsibilities for controlling field purchase requisitions for Code items and services	Procurement will be accomplished by CP&L under their program and procedures.	All Code items procured for this project will be specified as such in the design change package BOM and procured to applicable Code requirements from appropriately qualified vendors. Ref. MCP-NGGC-0001, 0002, 0401, 0402, 0406. Appendix B procurement is well defined and adequate. No supplement required.
3.4 Reclassified Material	Lists requirements for upgrading materials	N/A - no upgrade of Code items will be utilized in support of this activity.	No supplement required
4.0 Receiving Inspection			
4.1 - 4.13	Outlines requirements for receipt inspection of Code items. Lists responsibilities for QA/QC to accomplish receipt inspection in such a manner as to ensure that Code items are in compliance with requirements, and to prevent damage, deterioration or loss. Includes requirements for inspection and examination, identification and resolution of nonconformances, conditional release requests, and item acceptability.	The Corporate materials control program includes procedures for receipt inspection, storage, issuance and control of items	The Corporate materials control program meets the requirements of Appendix B and is deemed suitable for control of safety related items, including Code items. It is noted that this program is currently utilized for materials procured for use in the site Section XI Repair and Replacement Program.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
5.0 Storage and Process Control			
5.1 - 5.1.1 Storage	Outlines requirements and responsibilities for storage of Code items.	Requires that measure be established to control handling, storage, cleaning and preservation of material and equipment (QAM 5.3, 5.5, 5.6)	The Corporate materials control program meets the requirements of Appendix B and is deemed suitable for control of safety related items, including Code items. It is noted that this program is currently utilized for materials procured for use in the site Section XI Repair and Replacement Program. For those items which were installed during original construction and which will now be utilized in the modified design, the Supplemental QA Requirements defines an Equipment Commissioning Plan which outlines dedication requirements. Notably, this plan does not provide any exception to Code requirements except as pertains to documentation of field installation of piping (Addressed in the Alternative Plan).

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
5.2 Process Control 5.2.1, 5.2.2	Requires that process control sheets be utilized to establish measures assuring that processes, including welding and heat treating, are controlled in accordance with the Code and accomplished by qualified personnel. Process control sheets have appropriate spaces for QA/QC signature and for the ANI.	Requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements. (QAM 7.4, 11.3)	Generally, process control sheets in existing site procedures have appropriate verification of quality, including ANI review and involvement. It is noted that those procedures commonly associated with Section XI activities may specify ANII, vs. ANI. For the purposes of this project, the authorized inspector will be qualified as both. In addition, "Supplemental QA Requirements will include a requirement that the ANI review all process control sheets for Code related construction activities before they are issued to the field for construction, giving the ANI the opportunity to not only review the work planning, but also to specify any additional reviews / hold points as deemed necessary. Process control sheets documenting Code required attributes will be reviewed and accepted by the ANI prior to turnover.
5.2.3	For fabrication and installation of Code items by welding, the Weld Data Report, tank fabrication report and the safety-related instrumentation report are the process control sheets utilized. For pipe spool modifications, the Pipe Spool Fabrication / Modification Record is used to supplement the WDR as a process control sheet.	Requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements. (QAM 7.4, 11.3)	Site and corporate procedures associated with welding of Code items provide reference to the Corporate Welding Manual (NGGM-PM-0003) which utilize the Weld Data Report as a process control sheet. WDR in Corporate Welding Manual is consistent with Code requirements. No tank fabrication associated with this project.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
5.2.3.1- 5.2.3.3	Defines requirements / usage of the Pipe Spool Fabrication / Modification Sheet.	Requires that measures be established to assure that special processes are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements (11.3).	The "Pipe Spool Fabrication / Modification Sheet utilized during the construction era is not applicable to this project. Piping will be installed using modification WR/JOs for planning and implementation, and process control sheets from applicable plant procedures for material traceability, identification and qualification of personnel, quality verification, etc. Piping fabrication and installation process control sheets are provided in MMP-002, the Corporate Welding Manual and other applicable procedures.
5.2.4 - 5.2.11	Specifies process control sheets for repair or rework of non-welding activities, flanged or threaded connections, pressure tests, instrumentation, tube bending, etc.	Requires that necessary process control sheets for these activities are provided in corporate and site procedures. (QAM 7.4, 11.3)	Existing process control sheets are the same utilized for site Section XI Repair / Replacement Program. These sheets are are adequate to direct and document this work.
5.2.12	Requires QA / QC notification of the ANI when a mandatory hold point is reached, and that hold point inspections be accepted by signature (or initials) and date on the process control sheet prior to work proceeding past that point.	Requires that work not proceed beyond hold points without the consent of the designated representative. (QAM 2.2, 3.4, 3.7)	Work planning procedures and process control sheets incorporate hold points as appropriate for independent craft verification, QC, and for Code activities, for the ANI as well. Procedures require that hold points be established and utilized as appropriate. (Ref. WCM-002, MMM-001, ADM-NGGC-0104) Existing procedures are adequate with respect to incorporating independent verification and QC hold points into work planning and process control sheets. Additional controls regarding ANI involvement will be accomplished by requiring that all work packages associated with Code items be clearly identified as "ASME Section III", and be reviewed by the ANI prior to field issuance to allow hold points to be added if he desires.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
5.2.13	See Section 10.0 for discussion on nonconformances	-	-
5.2.14	Requires that identification tags or markings be retained, and transferred when necessary to cut an item. This transfer of identification was verified and documented by QA/QC.	Requires that measures be established to assure that identification of items are maintained by heat number, part number or other appropriate means, on the item or records traceable to the item throughout the fabrication, erection, installation and use of the item. (QAM 3.5, 3.7, 5.3, 5.4, 5.5, 10.5)	NGGC-MCP-0402 requires that traceability be accomplished at issuance either by markings or on issue documentation, as appropriate. For piping, verification of material identification is documented on process control sheets found in MMP-002. Additional controls regarding maintaining identification and traceability of materials is provided in ADM-NGGC-0104 and MMM-001. Corporate Welding Manual NW-04 also requires that material identification numbers be transferred when material is cut and that permanent markings be established in accordance with Code requirements. Adequate control of materials is provided with existing procedures and processes. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
5.2.15	Lists detailed requirements for labeling, identification requirements associated with maintaining traceability, particularly with regard to welding materials . Requires that hold or reject items be withheld from use, and that material protection procedures be implemented to prevent damage or deterioration.	Requires that measures be established to assure that identification of items are maintained by heat number, part number or other appropriate means, on the item or records traceable to the item throughout the fabrication, erection, installation and use of the item. (QAM 3.5, 3.7, 5.3, 5.4, 5.5, 10.5)	NGGC-MCP-0402 requires that traceability be accomplished at issuance either by markings or on issue documentation, as appropriate. The Corporate Welding Manual, NGGM-PM-0003, provides for issuance and control of welding materials in accordance with Code requirements. NGGC-MCP-0401 defines the receipt inspection / material disposition process, and precludes inadvertent issuance and installation of items not accepted except for conditional release, and in that case ensures that this material will be accepted prior to turnover. MMM-001 provides requirements regarding handling and storage of materials once they are issued for installation. Sufficient controls exist relative to materials identification and traceability, including welding materials. Items not accepted are precluded from issuance except for conditional release, and this process ensures that the material will be accepted prior to turnover. Work control procedures assure that Q material is properly handled, stored and segregated. No supplement required.
5.3 Construction Procedures Development	Defines responsibilities and requirements regarding development of construction procedures.	Defines requirements for preparation, review, approval and control of procedures (QAM 6.0)	Existing corporate and site procedures will be used to direct construction, with additional instructions / controls provided by the modification package as described herein.
5.4 Start-Up Procedures Development	Defines responsibilities and requirements regarding development of Start-Up procedures	Requires test control as required to demonstrate that structures, systems and components will perform satisfactorily in service. (QAM 3.4, 3.7, 4.5, 11.3)	As opposed to plant start-up, which covered a wide range of systems and equipment, start-up scope for this project is very limited, will be accomplished by including necessary instructions in the modification package for the affected scope of work. Start-Up procedures will be provided in the modification packages. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.0 Welding Control			
6.1 Procurement of Welding Material 6.1.1	Requires that welding material used in the construction Code items conform to Code requirements as detailed in the Site Specification SS-021, "Purchasing Welding Materials for Permanent Plant Construction" Provides requirements regarding material tests to be conducted by the manufacturer.	Requires that procurement control measures be established such that applicable regulatory requirements, design bases, etc are suitably included or referenced in the procurement of material, equipment and services (QAM 4.5).	Procurement and control of welding materials accomplished by Corporate Welding Manual, which invokes specification CPL-XXX-W-01, "Welding Filler Metals and Materials Procurement for Nuclear Power Plants, ASME Section III Applications". Program outlined in the Corporate Welding Manual provides a well-defined and specific process for specification and procurement of welding materials.
6.1.2	Requires that POs for weld materials include weld material classification and that testing and certification be performed to the requirements of ASME Code NB2400 for each heat and / or lot of material in accordance with the latest mandatory addenda of the ASME Code, Section II, Part C; and the 1974 Ed., 1976 Winter Addenda of Section III.	Requires that procurement control measures be established such that applicable regulatory requirements, design bases, etc are suitably included or referenced in the procurement of material, equipment and services (QAM 4.5).	Specification CPL-XXX-W-01 ensures that welding materials procured for Code applications conform to ASME Code Section II, Part C and Section III requirements; Use of this spec, invoked by the Corporate Welding Manual provides equivalent assurance of Code conformance. No supplement required

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.1.3	Specifically requires that welding materials received from a supplier without proper certification and complete documentation, as required by the Code, shall be tagged and placed on "Hold" status in a segregated area until the documentation has been received or corrected.	Procedures require that materials, parts, and components be identified and controlled to prevent the use of incorrect or defective items. Requires that items accepted or released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation of further work, and that items not meeting applicable requirements are identified and controlled until proper disposition is made. (FSAR 17.3.2.6)	Welding materials are received and inspected in accordance with NGGC-MCP-0401, which defines the receipt inspection / material disposition process, and precludes inadvertent issuance. Only when these materials are accepted, they are transferred to bin locations in the Weld Material Issue Station. Once there, the issuance and control of welding materials is strictly controlled by the Plant Welding Engineer in accordance with the Corporate Welding Manual. The Welding Material Control Procedure (NW-03) in the Corporate Welding Manual conforms to stringent requirements for welding in accordance with ASME Code requirements. No supplement is required.
6.2. Welding Procedure Qualification 6.2.1	Defines responsibilities for preparation, qualification and approval of CP&L welding procedures. Requires that welding procedures be qualified in accordance with ASME Code Section IX and meet the requirements of Section III. Requires that QA/QC be notified of procedure test schedules to allow QA/QC monitoring and documentation of the activity.	Requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements. (FSAR 17.3.2.11) Requires that if mandatory inspection hold points are required, work shall not proceed beyond these hold points without the consent of the designated representative. (QAM 3.4, 3.7)	Procedure NW-01 in the Corporate Welding Manual places responsibility for development, revision and qualification of welding procedures with the Welding Engineer. NW-01 also requires that WPS be in accordance with ASME Code Section IX and other referenced codes as applicable (includes Section III). Procedures in the Corporate Welding Manual provide compliance with requirements; no supplement required

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.2.2	Provides forms to record actual welding parameters, test results and Code required data, to be certified by the Welding Manager or his representative. Included QA/QC signature of these records after review against Code requirements and submittal to the ANI.	Requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements. (FSAR 17.3.2.11) Requires that if mandatory inspection hold points are required, work shall not proceed beyond these hold points without the consent of the designated representative. (QAM 3.4, 3.7)	QA/QC review is not required, but the monitoring and documentation of variables as the test proceeds is required. The test weld is subject to NDE and testing as required by codes and specifications, performed by certified personnel. The completed WPS is independently reviewed by a WE for approval. Current process does not require ANI involvement in the qualification process. Section III, Subsection NA-5252 requires that the Inspector assure himself that welding procedures have been qualified under the provisions of Section IX and Section III, and may request re-qualification as a requirement. The Code does not specifically require the ANI to review the WPS as it is developed.
6.3 Qualification of Welders and Welding Operators 6.3.1, 6.3.2	Defines responsibilities and requirements for testing, qualification and approval of welders and welding operators. Qualification is required in accordance with Section IX and the approved WPS. Tests will be performed in the weld shop under the Welding Manager. Welders are qualified on test coupons, with test results submitted to Document Control. Welder qualification status is maintained in a Welder Qualification Status List. Copies of the test records and Welder Qualification Status Report are made available to the ANI.	Not specific to qualification of welders and weld procedures, but requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements. (FSAR 17.3.2.11)	Qualification of welders and weld operators provided for in procedure NW-02 of the Corporate Welding Manual. Qualification specific to the requirements of the WPS; Welders qualifying to make Code weldments would be qualified per Code requirements. Testing is performed in a training / qualification area under the supervision of Weld Test Shop personnel. Test results are recorded on the Performance Qualification Test Record (PQTR) and reviewed and approved by the PWE. Testing is subject to monitoring by other organizations, including the ANI, "as applicable". The PWE maintains welder qualification records and the Welder Qualification Status Report. Qualification program meets Code requirements for applicable WPS, is the basis for the welding program used for the site Section XI Repair and Replacement Program.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.3.3	Welders shall be assigned a welder symbol, and a log shall be maintained for welding symbols. Upon termination or loss of the symbol stamp, that symbol stamp shall not be reassigned to another welder for a period of one year.	Not specific to qualification of welders, but requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel (FSAR 17.3.2.11)	Corporate Welding Manual procedures NW-02 and NW-10 provide requirements and instructions for the assignment of welder symbols. Existing program satisfactory; no supplement required
6.3.4, 6.3.5	Defines requirements for renewal and extension of welder qualification, requires re-qualification when the welder has not used the process for 3 months or more, except when the welder has been employed on some other welding process, the period may be extended up to 6 months by the Welding Manager. Re-qualification may also be required based on reason to question the ability of the welder.	Not specific to qualification of welders, but requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel (FSAR 17.3.2.11)	Corporate Welding Manual procedure NW-02 requires re-qualification when the process has not been used for 4 months, with no extension available based on use of other processes. Re-qualification may also be required based on reason to question the ability of the welder. Existing program satisfactory; no supplement required

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.4 Construction Welding 6.4.1	Defines responsibilities and requirements for preparation of Weld Data Report (WDR) based on design drawings, specifications and site procedures. WDRs are prepared by the weld manager, forwarded to QA/QC for review of essential requirements and hold points, then to the ANI who establishes his hold points and signs.	Requires that measures be established to assure that special processes, including welding, heat treating and NDE are appropriately controlled and are accomplished by qualified personnel using qualified procedures in accordance with applicable requirements. (FSAR 17.3.2.11)	Corporate Welding Manual Procedure NW-07 provides instruction on preparation of WDRs. WDRs are initiated by the PWE or his designees. WDRs are approved by the PWE, and Code WDRs forwarded to the ANII for review and to designate, at his option, additional hold points. QC hold points are designated by the preparer and the ANII. The existing program is satisfactory. However, since this work is not associated with Repair / Replacement activities, the Inspector must be qualified as ANI. Note that the Supplemental QA Requirements requires dual qualification (ANI, ANII) for this individual. The Supplemental QA Requirements also requires that <u>all</u> process control sheets associated with Code activities receives a review by the ANI prior to field issuance.
6.4.2	Requires all welding to be done using welders qualified by CP&L to CP&L WPS. All welding is to be accomplished using qualified procedures. Defines responsibilities for control of welding operation (including authority to assign or remove welders).	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11)	Corporate Welding Manual procedure NW-06 provides general welding instructions and technical requirements for carbon and low alloy steels, stainless steels and nonferrous welding at CP&L plants. This procedure requires that all such welding be performed by qualified welders (per NW-02) using qualified WPS (per NW-01) and with qualified materials (per NW-03). Under the Corporate Welding Manual, the PWE is responsible for welding program at site, including training, qualification and technical supervision of welders. Existing program satisfactory; no supplement required
6.4.3	Defines process for reviewing welder qualifications and assigning welders.	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11)	The PWE maintains and distributes the "Welder Qualification Status Report" (NW-02) The PWE is also responsible for reviewing and approving WDRs (NW-07) Existing program satisfactory; no supplement required

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.4.5, 6.4.6	Defines requires for control of welding materials, including returning unused welding materials and use / surveillance of heated ovens for coated electrodes.	Is not specific to weld materials, but requires that materials, parts and components be identified and controlled to prevent the use of incorrect or defective items (FSAR 17.3.2.6).	Corporate Welding Manual procedure NW-03 provides requirements for issuance and control of welding materials, usage of heated ovens and rod caddies for temperature controls of coated electrodes, returning unused and undamaged materials at the end of each shift, and for dispositioning / discarding welding materials. Existing program satisfactory; no supplement required
6.4.7	Defines responsibilities for notification of ANI; requires that hold point inspections be accepted by QA/QC and the ANI prior to any work proceeding past that point.	Requires that if mandatory inspection hold points are required, work shall not proceed beyond these hold points without the consent of the designated representative. (QAM 2.2, 3.4, 3.7)	Work planning procedures and process control sheets incorporate hold points as appropriate for independent craft verification, QC, and for Code activities, for the ANII as well. Procedures require that mandatory QC hold points be accepted prior to work proceeding. (Ref. WCM-102, MMM-001, ADM-NGGC-0104). Existing procedures are adequate with respect to incorporating independent verification and QC hold points into work planning and process control sheets. Additional controls regarding ANI involvement be accomplished by requiring that all work packages associated with installation of Code items be clearly identified as "ASME Section III", and be reviewed by the ANI prior to field issuance. Note that, to avoid procedural conflicts, the Supplemental QA Requirements requires dual qualification (ANI / ANII) for this individual. The Supplemental QA Requirements also requires that <u>all</u> process control sheets associated with Code activities receives a review by the ANI prior to field issuance.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.4.8	Provides QA/QC visual inspection requirements for weld preps, fit-up, tack welds, root pass, etc; notification requirements in the event that an unacceptable condition is observed.	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11). Requires that work shall not proceed past mandatory inspection hold points without the consent of the designated representative. (QAM 2.2, 3.4, 3.7)	Corporate Welding Procedure NW-07 provides instruction on preparation of WDRs. WDRs are prepared by the PWE or his designees based on WPS and weld joint requirements. WDRs are approved by the PWE, and when applicable, forwarded to the ANII to designate, at his option, additional hold points. QC hold points are designated by the preparer and the ANII. Existing program is satisfactory, no supplement required.
6.4.9	Requires the welder identification symbol be applied next to the weld	Requires that measures be established to assure that identification of items are maintained by heat number, part number or other appropriate means, on the item or records traceable to the item throughout the fabrication, erection, installation and use of the item. (QAM 3.5, 3.7, 5.3, 5.4, 5.5, 10.5)	NGGC-MCP-0402 requires that traceability be accomplished at issuance either by markings or on issue documentation, as appropriate. For piping, verification of material identification is documented on process control sheets found in MMP-002. Additional controls regarding maintaining identification and traceability of materials is provided in ADM-NGGC-0104 and MMM-001. Corporate Welding Manual NW-04 also requires that material identification numbers be transferred when material is cut and that permanent markings be established in accordance with Code requirements. Adequate control of materials is provided with existing procedures and processes. No supplement required.
6.5 Repairs to Welds and Base Material 6.5.1 - 6..5.4	Provides requirements for repairs to welds and base materials in the event that unacceptable defects are identified. Requires notification of the ANI and discipline Welding Engineer and preparation of a Repair Weld Data Report (RWDR), subject to essentially the same process as for the WDR for the original weldment.	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11).	Corporate Welding Manual procedure NW-09 directs activities associated with repairs to welds and base metals (including grinding and machining); incorporates the development of a RWDR in a process which parallels that associated with the WDR. The Corporate Welding Manual provides a comparable process for repairs to welds and base metals; no supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
6.6 Control of Welding Equipment 6.6.1, 6.6.2	Provides requirements for operational checks at least every 3 months, notification / disposition of machines out of tolerance (including investigation of the use of the machine since the last sat operational check), initiation of corrective measures.	Defines requirements for measuring and test equipment (M&TE) control program, including calibration to a standard, establishment of calibration frequency, , M&TE control, etc. (QAM 8.0)	Corporate Welding Manual procedure NW-14 directs control of welding equipment, requires operational checks and maintenance at least every 12 months. Welding equipment is not used as extensively as during construction, and is generally subject to better handling by a smaller group of permanent plant personnel (vs. a large contract construction force). Existing program is in accordance with Section IX. Therefore, current controls on inspections & operational checks of welding equipment is acceptable; no supplement required.
6.7 Additional Process Control Forms	Allows for continuation form for process control sheets as necessary	N/A	Similar continuation form exists in NW-07 for WDRs / RWDRs. Not a critical item, but no supplement required at any rate.
7.0 Heat Treating Heat 7.1 - 7.7	Provides requirements and responsibilities for performing heat treatment in accordance with the Code.	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11).	Comparable requirements for post-weld heat treatment are found in Corporate Welding Manual procedure NW-08. Although the existing program is sufficient, this item is not an issue for Code welding associated with this project, as the subject welds are exempt from mandatory PWHT.
7.8 Bending and forming 7.8.1	Prohibits bending and forming of Class 1 materials at the construction site	N/A	N/A - no Code Class 1 items associated with this project.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
7.8.2	Requires that bending of Code Class 2 & 3 instrument tubing be accomplished in accordance with the Code, refers to Sections 2.0 and 5.0 (in the ASME QA Manual) for procedure development / approval and process control requirements, including QC / ANI interface.	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures. (FSAR 17.3.2.11).	Process control sheets and bending requirements provided (either directly or by reference) in MMP-003. For Quality Class A material (such as Code related items), this includes independent craft and QC verification of critical attributes. No direct ANI involvement required per this procedure. Supplemental QA Requirements requires that Code related process control sheets are forwarded to the ANI / ANII prior to field issuance for his review and assignment of additional hold points.
8.0 Control of Equipment, Tools, Gauges and Instruments			
8.1 Calibration 8.1.1 - 8.2.6	Provides responsibilities and requirements for equipment, tools, gauges and instruments specified for calibration control, including calibration at prescribed intervals against certified standards. Requires traceability between the calibrated item and calibration equipment be recorded on process control sheets. Provides for issuance of "Out of Calibration Notification" forms to evaluate corrective action and review activities for which the tool was last used since a sat calibration. Requires shorter calibration intervals or replacement of instruments frequently found out of calibration, and that calibration status and calibration due date be shown on or with the instrument, except for pressure gauges which are calibrated before use and after being returned to the shop. Prescribes requirements for storage, maintenance and record keeping of calibrated equipment, includes use of a certification record form and calibration stickers. Requires that pressure gauges used for hydrostatic testing be calibrated before and after each test or series of tests.	Defines requirements for measuring and test equipment (M&TE) control program, including calibration to a standard, establishment of calibration frequency, M&TE control, etc. (QAM 8.0)	Existing program satisfies Appendix B requirements and is the basis for support of the operating unit; is judged to be acceptable for this activity. No supplement required The M&TE calibration and control program at Harris is prescribed in MMM-006. This program includes identification of M&TE equipment, specification of calibration interval, calibration against certified standards, use process control sheets, and traceability between calibration tool and calibrated instrument. This program also requires an evaluation of all equipment calibrated by an item found to be out of tolerance since its last satisfactory calibration. Relative to pressure gauges used for hydrostatic testing, MMP-012 requires that these gauges be calibrated before and after usage. Storage, maintenance and record keeping of M&TE equipment addressed in MMM-006, including the use and control of certification records and calibration stickers. Relative to pressure gauges used for hydrostatic testing, MMP-012 requires that these gauges be calibrated before and after usage.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
9.0 Inspection, tests and Nondestructive Examination			
9.1 Training, Qualification and Certification 9.1.1 - 9.1.3	Provides responsibilities and requirements for QA/QC personnel. QA/QC inspection personnel shall be trained and qualified in accordance with Section 1.0 (of the ASME QA Manual) and the relevant Corporate Quality Assurance procedure.	Requires that personnel performing inspection review, examination and testing, evaluations of testing data and reporting of inspection and test results be qualified and certified based on CP&L commitment to Reg. Guide 1.58 (QAM 7.6)	Training and qualification is prescribed in the Corporate Quality Assurance Manual and accomplished as directed in Nuclear NDE Manual procedure NDEP-A. Program meets the requirements of Section XI, is acceptable for this scope given that all piping is Class 3, and that NDE consists of surface exams only.
9.2 Inspections and Tests 9.2.1 - 9.2.7	Provides requirements and general requirements for performance of inspections and tests. Requires that personnel be appropriately trained in preparation and control of inspection and test records, that inspections and tests are performed in accordance with approved procedures, that process control sheets be utilized, that the status of the inspected item be identifiable and traceable, that the ANI be notified when ANI hold points are reached, that work will not proceed past hold points until accepted, and that nonconforming work shall be stopped and corrective action initiated.	Requires that personnel performing inspection review, examination and testing, evaluations of testing data and reporting of inspection and test results be qualified and certified based on CP&L commitment to Reg. Guide 1.58 (QAM 7.6). Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11).Requires that work shall not proceed past mandatory inspection hold points without the consent of the designated representative. (QAM 2.2, 3.4, 3.7)	Training and qualification addressed in NDEP-A (see item 9.1). This procedure also requires that for NDE procedures be based on ASME Section III and V as applicable, and requires that the ANI / ANII review and concur with any NDE procedures used for acceptance of Code work. Process control sheets for NDE activities are provided in the Nuclear NDE Manual procedure for that specific test or inspection, including hold points as appropriate. Corporate and site procedures ensure that hold points are accepted prior to work proceeding. NDEP procedures are provided to conform with ASME Code requirements as applicable. NDE procedures for LP and MT examinations are provided with acceptance criteria to ASME Section III requirements. No supplement required.
9.3 Nondestructive Examination 9.3.1	Requires personnel performing NDE to be trained, qualified and certified in accordance with SNT-TC-1A (1975), the Code and QA/QC procedures. Requires that only qualified personnel are assigned to perform NDE, and that procedures for NDE training, qualification and certification be prepared by a Level III. This section also provides an outline of the inspection procedure.	Requires that personnel performing inspection review, examination and testing, evaluations of testing data and reporting of inspection and test results be qualified and certified based on CP&L commitment to Reg. Guide 1.58 (QAM 7.6)	Nuclear NDE Manual procedure NDEP-A states that this NDE manual meets the requirements of SNT-TC-1A (1980, 1984 Ed) and Section XI. NDE Procedures require that personnel be appropriately trained. NDEP-A also includes a listing of minimum content for NDE procedures based on the type of activity being performed. Existing program is of comparable rigor. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
9.3.2, 9.3.3	Requires Level I and II personnel to be qualified and certified by an examination administered by a Level III; Level III to be qualified by an exam administered by a Level III and certified by the QA/QC manager. Allows the services of an outside agency to be used in the event that no Level III personnel exist within the organization.	States that prior to certification, NDE personnel shall have satisfactorily passed an examination administered under the jurisdiction of a certified Level III, and that CP&L Level III NDE personnel will be specified in CP&L's NDE Procedures (QAM 7.6).	NDEP-A requires certification of Level I and II to be performed by Level III, Level III certification to be performed by the Chief Mechanical / Materials Engineer. Use of an outside organization is not prohibited. Existing program is of comparable rigor. No supplement required
9.3.4 - 9.3.9	Requires training, qualification and certification of Levels I, II & III personnel to be in accordance with the applicable NDEPs and documented on appropriate certification forms. Requires records be maintained, that NDE personnel be re-certified at least once every 3 years, and that interpretation of results is accomplished by a Level II or III. Provides guidance for the preparation of NDE requests and reports.	States that prior to certification, NDE personnel shall have satisfactorily passed an examination administered under the jurisdiction of a certified Level III, and that CP&L Level III NDE personnel will be specified in CP&L's NDE Procedures. (QAM 7.6)	NDEP-A provides comparable requirements relative to training, qualification re-qualification and certification of personnel. Requires that records be maintained, and lists performance review requirements for maintenance of certification. Existing program is of comparable rigor. No supplement required
9.4 Inspection and Test Equipment	Requires that QA/QC inspection personnel be responsible for ensuring that inspection and test equipment is calibrated and has current calibration stickers.	Defines requirements for measuring and test equipment (M&TE) control program, including calibration to a standard, establishment of calibration frequency, M&TE control, etc. (QAM 8.0)	Inspection and test equipment is subject to the site M&TE control procedure, MMM-006. Existing program is sufficient regarding control of calibrated equipment.
9.5 Inspection and Test Records 9.5.1, 9.5.2	Requires that inspection and test records are prepared and maintained.	Defines requirements for maintain records of activities affecting quality, including inspection, test, audit and qualification records (QAM 14.3, 14.4)	NDEP-A requires that NDE records associated with Code activities be considered permanent QA records and be processed accordingly. Process control sheets and other required records are provided in the Nuclear NDE Manual as applicable. Existing program is of comparable rigor. No supplement required

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
9.5.3 - 9.5.7	Provides requirements for involvement of ANI, states that records are not considered complete until signed and dated by the ANI on the process control sheets, that the NDE Level III shall assure that NDE capability is proven by demonstration to the satisfaction of the ANI prior to the use of the procedure, and that the ANI may require re-qualification of NDE procedures or personnel as he deems necessary.	Requires that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures (FSAR 17.3.2.11 Requires that work shall not proceed past mandatory inspection hold points without the consent of the designated representative. (QAM 2.2, 3.4, 3.7)	NDEP-A requires that ANI review / concurrence be obtained for NDE procedures used for Code work. Requires that the Level III provides procedure qualification demonstration to the ANI when necessary. Also, provides that work may continue prior to ANI review of procedures, but that any such work would be at risk to ANI review. Existing program is of comparable rigor. No supplement required
9.5.8, 9.5.9	Provides responsibilities and requirements for pressure testing, including QA/QC and ANI involvement, establishment of hold points, and review / approval of the process control sheets.	Requires that a test program be established to assure that structures, systems and components perform satisfactorily in service, and that this program include pre-operational tests and proof test prior to installation. (QAM 3.4, 3.7, 11.3)	Pressure testing requirements provided in MMP-012. Generally, these pressure test procedures are intended to meet Section XI pressure test requirements. Existing pressure test procedures are adequate, except that the test pressure specified for Section XI may be less conservative. Therefore, the Supplemental QA Requirements specify that more stringent Section III criteria be employed for pressure testing.
9.6 Code Data Report and Nameplate Stamping 9.6.1 - 9.6.4	Provides the process and requirements for the development and review of Code Data Reports and N Stamping.	No partial N stamping of existing equipment and the original N certificate program has been discontinued, so that originally installed equipment cannot be subject to the stamping process. No provision for N stamping is provided.	Supplemental QA Requirements defines a certification process wherein data reports are used to document field activities towards an overall system turnover. Whenever possible (i.e., for completed Code items supplied by an NPT supplier), these data reports will be the actual Code Data Reports for the items in question. For new construction and documentation of installation of preexisting piping for which records are no longer available, a form comparable to an NIS-2 will be employed. The ANI will ensure that the required data reports are completed and certified.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
10.0 Nonconformance and Corrective Action			
10.1 Scope 10.2 Reporting Non-conformances	Provides responsibilities and requirements for identification, reporting, segregation, investigation and resolution of non-conformances relating to Code conditions. Requires deficiencies in documentation and construction control, including Start-Up procedures, be reported as non-conformances. Utilized hold tags and labeling as required to indicate limits of hold. Defined review requirements for NCRs	Requires that measures be established to assure identification and control of incorrect or defective material, parts and components (QAM 5.3, 5.4, 5.5, 10.3, 11.4, 11.6). Requires that measures be established to conditions adverse to quality are promptly identified and corrected. (QAM 12.4, 12.5)	Procedures and processes provide measures (i.e., process control sheets, independent verification, STAR) to ensure that construction deficiencies are precluded from occurring. For conditions that are identified, CAP-NGGC-0001 provides direction on the initiation and processing of condition reports, such as would be generated in the event of non-conformances. Relative to receipt and control of materials, MCP-NGGC-0401 & 0402 ensure that defective items are not accepted and issued. These condition reporting and materials control processes provide effective programmatic means to ensure that discrepancies and non-conformances are captured and resolved.
10.3 Corrective Action 10.3.1 - 10.3.7	Provides instructions and guidance relative to the process for dispositioning NCRs. Requires verification and disposition of corrective action be performed by QA/QC prior to signing and closing the NCR.	Requires that measures be established to conditions adverse to quality are promptly identified and corrected. (QAM 12.4, 12.5)	CAP-NGGC-0001 provides instructions and requirements for dispositioning CRs. Incorporates requirements for event categorization, causal evaluation, disposition and corrective action Existing program provides an effective means to capture and resolve non-conformances.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
10.4 Review of Nonconformance Report 10.4.1 - 10.4.4	Requires corrected items or documents to be re-inspected by a QA/QC inspector, and acceptance documented. Requires that closed out NCRs become QA records and transferred to the QA records vault. Requires that the ANI be apprised of any NCRs pertaining to the Code, and requires ANI signature prior to closing any such NCRs	Requires that measures be established to conditions adverse to quality are promptly identified and corrected. (QAM 12.4, 12.5). Does not require ANI involvement in review of construction related conditions adverse to quality.	The stated purpose of CAP-NGGC-0001 is to implement the NGG Corrective Action Management Policy and the requirements of 10 CFR 50, Appendix B, Criterion XVI. Disposition of any CRs related to construction requires that the item be corrected or formally evaluated as being acceptable. Review and approval of CRs goes up to and includes PNSC review, as appropriate. The CR process does not specifically require notification of the ANI for Code related items, although the formal evaluation process would tend to ensure his cognizance of any such issue. To further ensure the ANI's involvement on CRs related to Code items, the Supplemental QA Requirements requires that any such items be available to the ANI for verification of satisfactory resolution prior to turnover.
10.5 Receiving Inspection Software Deficiencies	Provides requirements for identification by QA/QC at receipt of documentation deficiencies, requires that an NCR be initiated for any such discrepancies that cannot be resolved by routine measures.	Requires that measures be established to assure identification and control of incorrect or defective material, parts and components (QAM 5.3, 5.4, 5.5, 10.3, 11.4, 11.6).	A similar receipt inspection process, including requirements for documentation review, are provided in MCP-NGGC-0401. Existing process provides sufficient assurance regarding resolution of documentation discrepancies. No supplement required.
11.0 Record Retention			
11.1	Defines responsibilities and requirements for records retention, requires that records generated by suppliers and contractors be transferred to CP&L for retention. Requires restriction of access to records storage areas and the use of a records sign-out log.	Defines requirements for maintaining records of activities affecting quality, including inspection, test, audit and qualification records (QAM 14.3, 14.4)	RMP-006 provides requirements for classification of QA records. Design change package, work records and other quality related documentation generated as a result of this project would be classified therein as a QA record and subject to permanent retention. Existing process is equivalent to the construction program. No supplement required.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
11.2 Records Index	Requires preparation of a record index to facilitate timely retrieval of records	Defines requirements for maintaining records of activities affecting quality, including inspection, test, audit and qualification records (QAM 14.3, 14.4)	CP&L maintains a computer-based index of records (NRCS) for indexing and retrieval of records Existing process is equivalent of the construction program. No supplement required.
11.3 Accumulation and Maintenance of Records	Provides requirements and responsibilities for accumulation and maintenance of records, including identification of retention period, prevention of loss, damage, etc. Requires access to records by the ANI	Defines requirements for maintain records of activities affecting quality, retention period and prevention of loss, damage, etc. (QAM 14.3, 14.4)	RMP-006 provides requirements for classification, submittal, control and maintenance of records. No supplement required.
12.0 ANI			
12.1.1 - 12.1.7	Summarizes the interface and requirements associated with the ANI for compliance with the Code. Requires that the ANI be given free access to all work locations under his jurisdiction, that he be provided adequate facilities and assistance, that he witness or otherwise verify required examinations and inspections, and that inspection services be subject agreement between CP&L and the AIA as required.	Requires that a program of inspection of activities be established. Does not address the ANI role in construction process (QAM 2.2, 5.3, 5.6).	Individual procedures address the role of the ANI in work activities and reviewing / approving process control sheets. Generally, these procedures are associated with Section XI activities and requirements for the ANII. Contractural agreement for the Inspector's services is provided as required for Section XI. Supplemental QA Requirements require that ANI be provided process control sheets for Code activities and items associated with this project prior to field issuance of the associated work package. Supplemental QA Requirements requires that the Inspector for the SFP project be dual qualified, as ANI / ANII.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
12.2 Document Accessibility 12.2.1 - 12.4.3	Requires that the ANI be provided free access to all information and records related to Code items, that the ANI review procedures utilized to implement Code requirements, that the ANI monitor the QA Program. Requires that the ANI be provided an opportunity to select holdpoints, and that he is provided sufficient notification of Code related work and testing. Requires that the ANI has authority to require re-qualification of procedures and personnel, that the ANI may witness or verify records of NDE, and that the ANI shall witness final hydrostatic testing required by the Code.	Requires that a program of inspection of activities be established. Does not address the ANI role in construction process.(QAM 2.2, 5.3, 5.6).	Role of ANI is provided in procedures and contractural agreements in accordance with Section XI requirements. Supplemental QA Requirements require that ANI be provided process control sheets for installation of Code items prior to field issuance of the associated work package.
13.0 Audits			
13.1 - 13.3	Provides responsibilities and requirements for Corporate QA audit activities. Defines the approach for auditing of the engineering, construction and start-up as being a comprehensive system of planned audits. Requires regularly scheduled audits on the basis of status and importance to ensure Code compliance. Requires written audit reports, that corrective action be taken as appropriate and verified as complete, and that follow-up audits and monitoring be conducted as necessary	Requires that a comprehensive system of audits be carried out. (QAM 4.11, 13.1, 13.4).	For internal assessment, the Corporate approach towards auditing and quality assurance is founded on the principle that the line organization has the primary responsibility for quality and safety. Nuclear Assessment Section evaluates the performance and effectiveness of this process through independent assessment, and the Performance Evaluation Support Unit (PES) provides oversight of each plant's NAS by reviewing NAS assessment reports and perform a NAS effectiveness assessment at least once every 24 months. External audits of suppliers are performed in accordance with MCP-NGGC-0406. Existing program meets Appendix B requirements and is sufficient rigor for completion of construction.

ASME QA Manual Section No.	ASME QA Manual	Corporate Appendix B QA Program	Reconciliation
13.1.4	Requires that audit reports be maintained, and be made available to the ANI at his request.	Requires that audit results be documented and reviewed by management (QAM 4.11, 13.6)	Supplemental QA requirements require that all CRs associated with Code activities within this project be available to the ANI for verification of satisfactory resolution prior to turnover
13.2 Supplier Audits 13.2.1 - 13.3	Provides responsibilities and requirements for the auditing of activities by suppliers. For Code items, requires audits at least every 3 years. Requires audit results be made available to the ANI upon his request.	Requires that a comprehensive system of audits be carried out, and that audit results be documented and reviewed by management (QAM 4.11, 13.1, 13.4, 13.6)	MCP-NGGC-0406 provides requirements for audits of outside suppliers, includes specific requirements for suppliers of Code items, and requires auditing of suppliers at least every 3 years. Existing program provides equivalent assurance and rigor, no supplement required.
14.0 Review and Control of Manual			
14.1 - 14.6	Provides responsibilities and requirements for issuance, review and control of the ASME QA Manual. Requires that controlled copies be kept and maintained, and that revisions be reviewed and approved by the ANI.	Requires that activities affecting quality be prescribed by documented instructions, procedures etc. (QAM 6.0) and that measure be established to control the issuance of those documents (QAM 6.1, 6.34, 14.4)	Control, distribution and accountability of QA documents accomplished in accordance with RMP-002. Relative to the Alternative Plan, the AIA has formally reviewed and endorsed this plan as submitted to the NRC. The implementation of the Alternative Plan will be subject to ANI review as part of the modification review / approval process, including the Supplemental QA Requirements and the turnover / certification process it defines. However, since the authorization for the Alternative Plan comes from NRC approval, any revisions outside of typographical or minor administrative changes will require the review and approval of the NRC.

CP&L

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

SERIAL: HNP-99-094

JUN 1 4 1999

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

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE
HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Dear Sir or Madam:

By letter dated April 29, 1999, the NRC issued a request for additional information (RAI) regarding the Harris Nuclear Plant (HNP) license amendment request, submitted by CP&L letter Serial: HNP-98-188, dated December 23, 1998, to place spent fuel pools C and D in service. The HNP response to the NRC RAI is enclosed. The enclosed information is provided as a supplement to our December 23, 1998 license amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

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

Sincerely,

Donna B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosure

A/H

bc:

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

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE
HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Requested Item 1

Although the burnup criteria for storage in Pools C or D will be implemented by administrative procedures to ensure verified burnup prior to fuel transfer into these pools, an administrative failure should be assumed and evaluation of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per Technical Specifications (TS) Figure 5.6.1) should be analyzed.

Response to Requested Item 1

The presence of soluble boron in the spent fuel pool water will assure that the reactivity is maintained substantially less than the design limitation in the event of a misloading event as described above. The Double Contingency Principle provides that neither the utility nor the staff is required to assume two unlikely, independent, concurrent events. Therefore, a failure of the administrative controls related to fuel assembly placement and the inadvertent dilution of the spent fuel pool water need not be considered to occur simultaneously. As a result, credit for the presence of soluble boron in the spent fuel pool water may be taken for an assembly misloading event as described. A minimum spent fuel pool boron concentration of 2000 ppm is maintained in accordance with HNP chemistry procedure CRC-001. This minimum boron concentration is more than adequate to offset the reactivity addition from a postulated fuel assembly misloading event. Based on analysis performed by Holtec International, it has been determined that a soluble boron concentration of 400 ppm would be sufficient to maintain k_{eff} less than 0.95 in the event of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1).

Requested Item 2

How will the burnup requirements needed to meet TS Figure 5.6.1 be ascertained for fuel assemblies shipped from other PWR plants (Robinson)?

Response to Requested Item 2

The burnup curve (proposed TS Figure 5.6.1) applies to the Robinson 15 x 15 fuel assembly types identified in Table 4.3.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98.

The selection of spent fuel for shipment to Harris is made in accordance with procedure NFP-NGGC-0003, entitled "Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask." The purpose of this procedure is to assure that the requirements of the IF-300

Cask Certificate of Compliance No. 9001 are met with regard to the selection of irradiated fuel to be shipped and that the fuel selected for shipment is acceptable for storage at CP&L's Harris plant. This procedure has been in use since 1990 for Robinson spent fuel shipments.

A computer program, which has also been in use since 1990 for Robinson spent fuel shipments, is used in conjunction with the above-referenced fuel selection procedure. For candidate assemblies to be shipped, the program retrieves the fuel type, enrichment, burnup, and decay heat from the special nuclear materials database. The initial enrichment data for each fuel assembly is contained in this database along with the other fuel data, and this data is based on manufacturing records. The burnup data for each fuel assembly is also included in the database along with the other isotopic inventories, and this data is obtained from the core monitoring software used for the Robinson plant. The special nuclear material database and core monitoring software have also been in use since 1990 for Robinson shipments.

The burnup curve proposed as TS Fig. 5.6.1 for pools C and D has already been programmed into the software for use in conjunction with fuel selection procedure NFP-NGGC-0003; however, this version is not yet in production as testing and documentation per CP&L's computer code quality assurance requirements are in progress. This new version will screen candidate PWR (Robinson) fuel against the burnup curve.

Revision to fuel selection procedure NFP-NGGC-0003 to reflect criticality screening requirements for fuel to be stored in Harris pools C or D has begun, but will not be completed until after: (1) the software changes identified above have been tested and the revised software placed in production status, and (2) the NRC has approved CP&L's license amendment application to place spent fuel pools C and D in service.

Requested Item 3

The fuel enrichment tolerance is specified in Section 4.5.2.5 as +0.0/-0.05. Why isn't a positive tolerance of +0.05 assumed (i.e., 5.0+0.05 weight percent U-235)?

Response to Requested Item 3

A maximum U-235 enrichment of 5.0 weight percent was specified, because it is the maximum enrichment allowed by both the Robinson and Harris Technical Specifications. Robinson TS 4.3.1.1.a states that the spent fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Robinson TS 4.3.1.2.a states that the new fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Harris TS 5.3.1 states that the initial core loading shall have a maximum enrichment of 3.5 weight percent U-235 and that reload fuel shall have a maximum enrichment of 5.0 weight percent U-235.

Also, the manufacturing facility of Siemens Power Corporation (SPC), the current fuel supplier for both the Robinson and Harris plants, is limited by license to a maximum U-235 enrichment of 5.0 weight percent. The SPC manufacturing tolerance is 0.05 weight percent U-235. Therefore, for enrichments with a tolerance of +/- 0.05%, the nominal design enrichment may not exceed

4.95 weight percent U-235 to ensure that the nominal plus the tolerance does not exceed 5.0 weight percent. The fuel enrichment and density tolerances specified in Section 4.5.2.5 appropriately supports a maximum allowable enrichment of 5.0 weight percent U-235.

Requested Item 4

Justify that the allowance that was assumed for possible differences between the fuel vendor and the Holtec calculations is sufficient to also encompass burnup calculational uncertainties.

Response to Requested Item 4

The Criticality Safety Calculations for the BWR Fuel Racks are summarized in Table 4.2.2 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98. An uncertainty on depletion was not explicitly included in the uncertainties summarized in Table 4.2.2. Instead, the 0.01 additive allowance for comparisons to vendor calculations discussed in Section 4.4.2.2 also accounts for burnup uncertainty. This practice is acceptable for the following two reasons:

First, the BWR calculations consider the peak reactivity during burnup. The k_{inf} in the rack corresponding to a peak k_{inf} in the Standard Cold Core Geometry (SCCG) of 1.32 was calculated in the analysis. The burnup corresponding to this peak reactivity value is simply a by-product of this calculation and, in contrast to PWR analysis, burnup is not used as a criteria for establishing acceptability for fuel storage. Any uncertainty in the burnup calculation would simply decrease or increase, with burnup, the location of the peak reactivity. However, the k_{inf} in the SCCG and the k_{inf} in the rack would remain the same at the peak in reactivity. As a result, an additional uncertainty on depletion is not necessary.

Second, the fuel vendor performs similar depletion calculations to those discussed in Section 4. Therefore any uncertainty in depletion is an inherent part of the comparison between those calculations in Section 4 and those performed by the vendor to determine the peak k_{inf} in SCCG as a function of burnup. Again, it is noted that the actual burnup at which the peak occurs is not used in the BWR acceptable fuel storage criteria.

Requested Item 5

The summary of criticality safety calculations shown in Tables 4.2.1 and 4.2.2 indicates that the total uncertainty is a statistical combination of the manufacturing tolerances but do not indicate methodology biases and uncertainties. Were these included?

Response to Requested Item 5

Section 4.4.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98, discusses the fact that CASMO-3, because it is a two-dimensional code, can not be directly compared to critical experiments and as a result a calculational/methodology bias is not available for CASMO-3. This section also discusses MCNP, which is a full three-dimensional Monte Carlo code, which has been benchmarked against critical experiments. CASMO-3 was used as the primary method of calculation and the results from CASMO-3 were compared to the regulatory limit of $k_{eff} \leq 0.95$ in Tables 4.2.1 and 4.2.2. As noted, the methodology bias and uncertainty were not included in these tables. However, these factors were implicitly included in a code-to-code comparison between CASMO-3 and MCNP shown in Table 4.5.1.

As discussed above, a methodology bias can not be developed for CASMO-3. Therefore, CASMO-3 results were compared to MCNP results to either verify that it produces conservative results relative to the benchmarked MCNP, or to determine a code-to-code bias. This comparison is discussed in Sections 4.5.1 and 4.6.1 with the results presented in Table 4.5.1. In the comparison between MCNP and CASMO-3, the methodology bias, uncertainty on the bias, calculational statistics, and a correction from 20°C to 4°C were added to the MCNP results. These results indicate that CASMO-3 is conservative relative to the benchmarked code MCNP and therefore the code-to-code bias was 0.0 for CASMO-3. Since the code-to-code bias was 0.0, it was not included in Tables 4.2.1 and 4.2.2. In conclusion, it can be stated that even though a methodology bias and uncertainty were not directly included in the final results shown in Tables 4.2.1 and 4.2.2, they were implicitly included through comparison of CASMO-3 and the benchmarked MCNP, provided in Table 4.5.1.

CP&L

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

SEP 3 1999

SERIAL: HNP-99-129

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

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

Dear Sir or Madam:

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

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

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

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

Sincerely,

for D. B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosures:

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

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

bc: (all w/ Enclosure 1)

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

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

Requested Information Item 1:

In September 1983, the staff issued NUREG-1038, "Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," which included a review of the spent fuel storage facility. The review of the spent fuel storage facility, including the two spent fuel pool cooling systems (SFPCSs) and the four fuel storage pools, was performed in accordance with the applicable sections of NUREG-0800, "Standard Review Plan." The U. S. Nuclear Regulatory Commission (NRC) staff's review found the design of the Unit 1 and Unit 2 fuel storage facilities acceptable. At the time NUREG-1038 was issued, construction of the Unit 2 SFPCS was still ongoing and expected to be completed. In November 1983, plans to complete Unit 2 were canceled and construction of the partially completed Unit 2 SFPCS was placed on hold.

On December 23, 1998, Carolina Power & Light Company (CP&L) requested a license amendment to activate spent fuel pools (SFPs) C and D. The submittal provided information to the staff regarding the activation of the pools; however, no information was provided on the design of the SFPCS supporting SFPs C and D. Given that the SFP C and D SFPCS was never placed in operation, and that significant changes to the design were proposed in the December 23, 1998 submittal, please provide information to show how the portions of the Unit 2 spent fuel storage facility (e.g., SFPs C and D, and the fuel pool cooling and cleanup system - as built), that are not already addressed in the December 23, 1998 submittal, meet the guidance in Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," and NUREG-0800. You may reference the NRC's acceptance of those portions of the fuel storage facility that have not changed from the design the staff previously accepted.

Response 1:

This requested information item is addressed below by a matrix that shows how the portions of the spent fuel storage facility originally intended to support Unit 2 (i.e., SFPs C and D, and the Fuel Pool Cooling and Cleanup System - as built) meet the guidance in Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," and NUREG-0800, Standard Review Plan. The matrix provides a cross-reference listing of the relevant NUREG-0800/NUREG-1038 sections associated with spent fuel storage, fuel pool cooling, and fuel pool area ventilation; identifies the proposed changes to the portions of SFPs C and D and associated Fuel Pool Cooling and Cleanup System previously accepted by the NRC staff (Reference: NUREG-1038, "Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2, dated November 1983); and provides the reason / basis for the proposed changes.

Reconciliation of SFP Activation Project with SER (NUREG-1038)

SRP/SER Section	Section paragraph number	Change from Design As Documented in SER	Reason / Basis for Change
9.1.2	1	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	2	Change with regard to the following: Infers that two units will be completed Discussion of storage capacity has been revised per the license amendment request Discussion of rack arrangements has been revised Note that discussion pertaining to maintaining K_{eff} at or below 0.95 has <u>not</u> been affected.	Unit 2 was not completed. A single new fuel pool is provided for Unit 1, with the remaining 3 pools representing existing or proposed spent fuel storage capacity. A current description of the completed facility at this point in time is provided in the FSAR, Section 9.1.2.2. Information relative to the proposed number and type of storage locations associated with the proposed configuration is provided in Enclosure 1 of the license amendment request (HNP-98-188, dated 12/23/98).
9.1.2	3	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	4	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	5	Assumes that two units are completed and addresses shared portions of facilities, stating that a loss of offsite power will not impair the ability to safety store spent fuel.	Since Unit 2 was not completed, there are no shared facilities between units. Relative to the proposed change, redundancy is provided so that an accident or loss of power in the operating unit will not impair the ability to safely store spent fuel in any of the fuel storage pools.

9.1.2	6	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	7	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	8	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	9	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	10	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	11	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.2	12	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	1	No change from design previously accepted by NRC Staff as documented in the SER.	N/A

9.1.3	2	Assumes that two units are completed. Refers to one fuel pool cooling system being provided "for each unit". States that each fuel pool cooling pump is capable of being loaded to onto a separate emergency power supply in case of loss of offsite power, and that each cooling train is a 100% subsystem servicing the new and spent fuel storage pool in that unit.	The proposed configuration completes the FPCCS as described in the SER for the two unit site. Two separate fuel pool cooling systems are provided, one for the two pools currently in service, and one for the two additional pools which were originally intended for Unit 2 (i.e., the C and D pools). Consistent with the description in the SER, each FPCCS will contain two cooling trains: each train including a heat exchanger, strainer, and fuel pool cooling pump, with each pump capable of being manually loaded onto a separate emergency power supply in the event of loss of offsite power. Each cooling train is a 100% subsystem, servicing both pools in that system.
9.1.3	3	No change from design previously accepted by NRC Staff as documented in the SER. The equipment in this discussion is the sum of that which would be provided for the entire facility, not just one unit.	N/A
9.1.3	4	Assumes two units were completed. Describes the facility as having new storage pools at either end, and the spent fuel pools being connected to the fuel transfer canal "in its unit." Also states "Makeup to the pools may be provided from a seismic Category 1 source (the refueling water storage tank) by means of the fuel pool cooling pumps."	See FSAR Section 9.1.2.2 for a description of the facility. The RWST for Unit 1 is available as a source of makeup water, will be connected to the fuel pool cooling pumps for both FPCCS, and has been evaluated and found sufficient to perform this function for all four pools in the proposed configuration.

9.1.3	5	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	6	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	7	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	8	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	9	Identifies the commitment by CP&L to provide two cooling pumps and two heat exchangers for Unit 1 and to provide a similar arrangement for Unit 2.	The proposed configuration is consistent with the commitment made for the Unit 2 design. The new FPCCS will be completed to essentially the same design as originally proposed to service Unit 2, including two fuel pool cooling pumps and two heat exchangers. The detailed description of the Unit 2 FPCCS is essentially the same as that provided in FSAR Section 9.1.3.

9.1.3	10	Discusses transfer of fuel between units with regard to meeting requirements of GDC 5.	Since Unit 2 was not completed, there can be no transfer of fuel between units. Relative to fuel pool cooling capacity, the license amendment request proposes a new Technical Specification to limit fuel pool (C and D) heat loads to no more than 1.0 MBtu/hr. This relatively low heat load limit is determined sufficient to support spent fuel storage needs until analyses associated with Steam Generator Replacement and Power Uprate progress to the point at which the integrated effect on CCW can be quantified. Once this integrated assessment is made, a subsequent license amendment request will be required to increase the heat load limit to reflect full spent fuel storage capacity in pools C and D.
9.1.3	11	Assesses temperature of Unit 2 spent fuel pool with the assumption that this unit was completed and that this pool has the greatest heat load. This assessment is not valid, because Unit 2 was not completed and pools C and D will only be used for "colder" spent fuel meeting specific burnup limitations.	Under the license amendment request to place pools C and D in operation, spent fuel storage in the Unit 2 spent fuel pools (i.e., pools C and D) will be limited to 1.0 MBtu/hr. As a result, the heat load in these pools is bounded by that which might exist in the Unit 1 spent fuel pools. The current temperature limit associated with the operating Unit 1 (i.e., A and B) pools is 137 °F. That limit is not affected by this license amendment request. Relative to the Unit 2 (C and D) pools, peak temperatures are anticipated to be well below this value at the maximum allowable pool heat load of 1.0 MBtu/hr.
9.1.3	12	No change from design previously accepted by NRC Staff as documented in the SER.	N/A

9.1.3	13	Describes makeup being provided from two Refueling Water Storage Tanks, one from each unit. Staff acknowledges that only one RWST will be available while Unit 2 is being built, and SER states that the single Refueling Water Storage Tank (RWST) is sufficient. SER states that ESW is available through valved and flanged emergency connections as a backup seismic Category 1 water source.	Since Unit 2 will not be completed, no separate RWST exists with regard to the Unit 2 FPCCS. The Unit 1 RWST will be connected to both FPCCS. This single RWST has been evaluated and determined adequate for providing a seismic Category 1 makeup source for all four pools. Other sources of makeup water are also available, including the seismic Category 1 ESW System, the Demineralized Water System, and the Reactor Makeup Water Storage Tank. The emergency connections described in the SER will not be provided on the Unit 2 FPCCS, since there is no ESW supply in the proximity to which they can be connected. Rather, ESW is available at a location in the Unit 1 RAB, where a cross-tie to both FPCCS can readily be made.
9.1.3	14	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	15	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.1.3	16	States that spent fuel pool water will be sampled weekly. Chemical impurity limits are to be maintained in accordance with Westinghouse WCAP-7452, Revision 2, 1977.	Fuel pool water chemistry limits are consistent with guidelines and specifications established by the NSSS vendor, fuel manufacturer, and EPRI standards. Fuel pool water is monitored routinely by chemical and radiochemical analysis of grab samples.
9.1.3	17	No change from design previously accepted by NRC Staff as documented in the SER.	N/A

9.1.3	18	Reiterates the commitment to provides two fuel pool cooling pumps and heat exchangers for the Unit 2 FPCCS.	The proposed configuration of the Unit 2 FPCCS is essentially the same as that for Unit 1, and includes two separate, 100% subsystems. Each train includes a fuel pool cooling pump, strainer and heat exchanger. The fuel pool cooling pump for each train is powered by a separate emergency power supply to provide spent fuel pool cooling capability even in the event of a loss of offsite power. The completed design is essentially the same as that shown in FSAR Figures 9.1.3.1, 9.1.3.2, 9.1.3.3 & 9.1.3.4 as being on "Construction Hold."
9.4.2	1	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	2	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	3	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	4	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	5	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	6	No change from design previously accepted by NRC Staff as documented in the SER.	N/A

9.4.2	7	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	8	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	9	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	10	No change from design previously accepted by NRC Staff as documented in the SER, except to note that this section contains a statement that precautions will be taken during construction of Unit 2 to protect operating features of the spent fuel pool area ventilation system.	Control of all site work activities (including those associated with completion of the Unit 2 spent fuel storage facilities) is controlled under site procedures for work control and screened for potential impact on the operating and licensed portions of the plant.
9.4.2	11	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	12	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	13	No change from design previously accepted by NRC Staff as documented in the SER.	N/A

9.4.2	14	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	15	No change from design previously accepted by NRC Staff as documented in the SER, except to note that a typo exists. Should reference Position C.4 of RG 1.13, not RG 1.14.	N/A
9.4.2	16	No change from design previously accepted by NRC Staff as documented in the SER.	N/A
9.4.2	17	No change from design previously accepted by NRC Staff as documented in the SER.	N/A

Requested Information Item 2:

As shown on Table 9.2.2-1 of the SHNPP Final Safety Analysis report (FSAR), each of the two Component Cooling Water (CCW) heat exchangers (HXs) has a design heat transfer rate of 50 MBtu/hr. Table 9.2.1-3 of the SHNPP FSAR shows the maximum service water heat load from the CCW HX following a loss-of-cooling accident (LOCA) to be 273 MBtu/hr. The unreviewed safety question (USQ) analysis in Enclosure 9 of the license amendment appears to indicate that one RHR HX (a single failure assumption) and one CCW train could remove 111.1 MBtu/hr. Discuss the differences among FSAR Tables 9.2.1-3, 9.2.2-1, and the USQ analysis.

Response 2:

HNP FSAR Section 9.2.1 addresses Emergency Service Water (ESW) capabilities. HNP FSAR Table 9.2.1-3 shows the maximum ESW system heat loads estimated to exist during post-LOCA conditions. In order to postulate the maximum potential heat load on the ESW system, both RHR loops are considered to be in service under worst case conditions. Since a single train of RHR is analyzed to remove up to 111.1 MBtu/hr in the post-LOCA scenario, the RHR contribution to the CCW portion of the total ESW heat load is 222.2 MBtu/hr. An additional 50.4 MBtu/hr of station auxiliary loads cooled by CCW is added to this value, resulting in the 272.6 MBtu/hr value shown in FSAR Table 9.2.1-3 for the total CCW contribution to ESW heat loads. These cumulative heat loads represent the maximum estimated ESW heat loads existing under post-LOCA conditions, whereas the 111.1 MBtu/hr is the heat removal requirement for a single train of RHR based on containment analyses.

HNP FSAR Section 9.2.2 describes the design and operation of the CCW system. The 50.5 MBtu/hr value shown in FSAR Table 9.2.2-1 is the "design heat transfer rate" for CCW. This value is consistent with design requirements for normal plant operation, and is based on a CCW outlet temperature of 105° F (vs. 120° F for post-LOCA conditions). Since heat transfer varies with flow rates and inlet conditions, the heat exchanger is capable of a wide range of heat removal rates. Analyses show that both 50.5 MBtu/hr (normal operation) and 111.1 MBtu/hr (post-LOCA) are within the capability of the CCW heat exchanger under the conditions associated with the scenario being evaluated.

Relative to the USQ analyses, the thermal-hydraulic calculations which support using the Unit 1 CCW system to provide cooling to the C and D spent fuel pools did not change any assumptions regarding maximum sump temperatures or RHR heat removal requirements under post-LOCA containment conditions. The analyses did, however, identify that fluid properties at the higher RHR temperatures associated with the post-LOCA scenario would result in an increase in heat exchanger heat transfer coefficient (HTC) values over the fixed value currently assumed. For the purpose of this analysis, the RHR HTC value was therefore allowed to vary as a function of fluid properties in order to ensure that the adequacy of downstream heat sinks was demonstrated under most limiting conditions. The CCW flow rate which corresponds to the requisite 111.1 MBtu/ hr heat removal rate was calculated in this manner and used to prescribe the CCW system flow rebalance associated with the additional spent fuel pool heat load.

Requested Information Item 3:

In Enclosure 6, Section 5.0, "Thermal-Hydraulic Considerations," of the license amendment request, Holtec provides a summary of the methods, models, analyses, and numerical results to demonstrate the compliance of the SHNPP SFPs C and D with the provisions of Section III of the NRC OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications (April 1978). Section 5.3 discusses the bulk pool temperature analysis. Holtec's conclusion is that the cooling water system must meet the design flow versus inlet water temperature specifications shown on Figure 5.3.1 of LAR Enclosure 6. Given that the SFPCS is already designed and constructed, and that the licensee has proposed to limit the heat load in SFPs C and D to 1.0 MBtu/hr (proposed Technical Specification (TS) 5.6.3.d), provide a thermal-hydraulic analysis using the system parameters for the SFPCS that support SFPs C and D that show the maximum bulk pool temperature for SFPs C and D will not exceed 137 °F assuming a single active failure.

Response 3:

The Holtec scope of supply included a single analysis to support operation with up to full pool conditions in both the C and D spent fuel pools. The Holtec analysis considers 15.63 MBtu/hr and establishes the system conditions (spent fuel pool cooling flow and temperature) required to maintain the current spent fuel pool limit at that heat load. Section 5.0 of the Holtec report is pertinent to the consideration of forced cooling requirements at 1.0 MBtu/hr in that it includes a review of pool design and layout, and shows that short-circuiting of cooling flow will not occur. Actual requirements for the 1.0 MBtu/hr heat load are established in HNP calculation SF-0040, which considers not only spent fuel pool cooling requirements, but also the cooling requirements of downstream heat sinks (i.e., CCWS, ESWS, and the ultimate heat sink). The body of calculation SF-0040 is provided in Enclosure 2. Note that the same spent fuel pool cooling pumps and piping will be used for the initial phase at 1.0 MBtu/hr as for the full pools, such that the same spent fuel pool cooling flow rates will exist for both scenarios. Calculation SF-0040 demonstrates the adequacy of plant heat sinks to maintain SFPs C and D at or below 137 °F given a single active failure for all plant conditions which require that assumption.

Requested Information Item 4:

Table 5.2.3 of Enclosure 6 states that bounding decay heat input from stored fuel in spent fuel pools C and D assumed in the thermal-hydraulic analysis totals 15.63 MBtu/hr. The proposed TS 5.6.3.d limits the heat load in spent fuel pools C and D to 1.0 MBtu/hr. Explain the difference between the maximum heat load requested in the license amendment and the heat load calculated and used in the Enclosure 6 decay heat analysis.

Response 4:

CP&L is proceeding with a phased approach to licensing HNP spent fuel pools C and D. The first phase will complete the spent fuel pool cooling systems and other supporting systems and provide for a maximum heat load of 1.0 MBtu/hr. This phase is now being reviewed by the NRC. The second phase will assess conditions necessary to utilize C and D pools at full capacity and consider the impacts of power uprate and steam generator replacement projects (scheduled for implementation in the fall of year 2001) on plant heat sinks. This phased approach is necessary, because spent fuel generation and storage requirements dictate that construction begin to complete spent fuel storage facilities expeditiously; however, the analyses supporting the power uprate and steam generator replacement projects have not progressed to the point that a comprehensive evaluation can be conducted. Therefore, based on a review of spent fuel generation and storage plans, an interim heat load of 1.0 MBtu/hr was chosen as a limit which supports fuel handling operations at HNP through the year 2001.

Requested Information Item 5:

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

Response 5:

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

Requested Information Item 6:

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

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

Response 6:

The RHR heat exchangers provide long-term cooling during the containment sump recirculation phase of a LOCA. This function is accomplished by aligning the RHR system to take reactor coolant from the containment sump, circulating the reactor coolant through the RHR heat exchangers, and then returning the reactor coolant back to the RCS cold legs. Thermal performance of the RHR heat exchangers at HNP has been analyzed using the dynamic RHR heat exchanger performance model and shown to remain comparable to that calculated by the current containment pressure/temperature analyses. The dynamic RHR heat exchanger performance model, however, yields a slight reduction in heat transfer relative to long-term post LOCA environmental conditions.

An assessment of the dynamic RHR heat exchanger performance model has been completed for long-term containment heat removal and equipment qualification. From this assessment, it is noted that the heat removal rate calculated by using the dynamic RHR heat exchanger performance model is 111.9 MBtu/hr at peak containment sump temperature (occurring at t = 3600 seconds into the event), which is marginally higher than the 111.1 MBtu/hr value obtained using the fixed HTC model. At 10^4 seconds into the event, the calculated heat removal rate using the dynamic RHR performance model is 92.2 MBtu/hr, still slightly higher than the 92.1 MBtu/hr associated with the fixed HTC model used in the current containment analysis. By 10^5 seconds into the event, the calculated containment sump temperature has decreased from 244.1 °F to 167.8 °F, and the heat removal rate calculated by using the fixed HTC model has now become slightly higher than that calculated by the dynamic HTC model (42.8 MBtu/hr compared to 41.9 MBtu / hr, respectively). At 10^6 seconds into the event (approximately 12 days), calculated containment sump temperature has decreased to 142.7 °F, and the heat removal rate using the fixed HTC model is 20.3 MBtu/hr, compared to 19.7 MBtu/hr calculated by the dynamic HTC model.

The table shown on the following page provides the requested heat exchanger performance parameters.

Comparison of RHR Heat Exchanger Performance for Long-Term Post LOCA Environmental Qualification

	@ t = 3600 secs.		@ t = 10^4 secs.		@ t = 10^5 secs.		@ t = 10^6 secs.	
	Fixed HTC	Variable HTC	Fixed HTC	Variable HTC	Fixed HTC	Variable HTC	Fixed HTC	Variable HTC
No. U-Tubes	592	592	592	592	592	592	592	592
Surface Area (ft^2)	4280	4280	4280	4280	4280	4280	4280	4280
UA (BTU/hr-°F)	1.635E6	1.758E6	1.635E6	1.734E6	1.635E6	1.665E6	1.635E6	1.627E6
Q (MBTU/hr)	111.1	111.9	92.1	92.2	42.8	41.9	20.3	19.7
RHR Flow (10^6 lbm/hr)	1.846	1.846	1.846	1.846	1.846	1.846	1.846	1.846
RHR Inlet Temp (°F)	244.1	244.1	222.9	222.9	167.8	167.8	142.7	142.7
CCW Flow (gpm)	5600	4874	5600	4874	5600	4874	5600	4874
CCW Inlet Temp (°F)	120	120	120	120	120	120	120	120

Because the relationship between heat exchanger flow rates and heat transfer is not linear, the analysis summarized above shows that the reduction in CCW flow from 5600 gpm to 4874 gpm yields no reduction in heat transfer at the earlier stages of the event associated with the highest postulated containment sump temperatures. Comparably, only a minimal reduction in heat transfer (about 3%) occurs much later into the event when containment sump temperatures have decreased significantly. Both the fixed and dynamic HTC models consider design fouling conditions and 0% tube plugging.

Requested Information Item 7:

The USQ analysis also indicates the need to increase the current minimum required emergency service water (ESW) flow to the CCW HX from 8250 gpm to 8500 gpm, which is said to have been verified to be within the capacity of the current system. Will FSAR Table 9.2.1-1 be revised to require the ESW flow through the CCW HX to be 8500 gpm?

Response 7:

Yes, FSAR Table 9.2.1-1 will be revised accordingly to reflect the revised system flow requirements.

Requested Information Item 8:

HNP FSAR Sections 9.2.2.2 and 9.2.2.2.2 provide CCW system and component descriptions, including information about the CCWS surge tank. This surge tank accommodates changes in the fluid volume of the CCWS from thermal expansion and contraction and accommodates water which may leak into the system from cooled components. The surge tank also provides the CCWS with a limited water supply until a leaking cooling line can be isolated. Discuss the effects of the additional system volume and heat load (from the piping and components added to support SFPs C and D activation) on the capability of the CCWS surge tank to perform its design function.

Response 8:

Section 9.2.2.2.2 of the HNP FSAR provides a discussion about four specific functions of the Component Cooling Water (CCW) surge tank. The impact on each of these functions by placing spent fuel pools C and D in service is discussed below:

a) Accommodates changes in CCWS water volume due to changes in operating temperature.

To place spent fuel pools C and D in service, the required modifications to the CCW system will add approximately 2000 gallons of water volume to the CCW system, including the shell side volumes of both spent fuel pool heat exchangers. Assuming this volume of water undergoes a temperature increase from 60 °F to 105 °F, the incremental volume increase in CCW inventory would be about 15 gallons. Since normal CCW surge tank level is at approximately 1000 gallons and the tank has a 2000 gallon capacity, this represents only about 1.5% of the tank's available surge volume. Even then, the tank is fitted with overflow and overpressure protection, sized to provide adequate relief from comparatively high volume makeup sources. Based on these considerations, an increase in CCW volume brought about by an abrupt rise in temperatures would be of no consequence to either plant operation or nuclear safety. The same can be said of the impact of abrupt temperature decreases, where the volume of water maintained in the tank and the makeup capability to the system are adequate to compensate for shrinkage. Finally, the CCW surge tank is equipped with high and low level instrumentation, which alerts the operator in the control room to significant changes in level so that any needed corrections can be readily made.

b) Accommodates, for 20 minutes, the maximum flow from either makeup water supply.

The activation of spent fuel pools C and D will not change makeup water supply capabilities, normal water level, or the capacity of the CCW surge tank. There is no impact with regard to the capability to accommodate makeup flow.

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

The design and construction of the piping and equipment being added to support activation of spent fuel pools C and D is similar to that already installed, such that the size and nature of leaks which might occur and the isolation capability of the equipment is consistent with that which already exists. Given these considerations, the adequacy of the reservoir to provide time to locate and terminate a leak is not adversely impacted.

d) Accommodates, for about 2 hours, the Technical Specification maximum identified reactor coolant leakage of 10 gpm.

As with item b) above, there is no impact regarding the capability to accommodate leakage from the RCS, because the activation of spent fuel pools C and D will not affect the capacity of the tank or the normal water level.

Requested Information Item 9:

Will the SFPCCS and makeup system(s) for SFPs C and D be included in the inservice inspection program or an inspection program similar to those used with systems that support SFPs A and B?

Response 9:

Portions of the SFPCS will be included in the site ISI / IST program, consistent with the treatment of equipment and support systems associated with spent fuel pools A and B. Specifically, the piping within Code boundaries will be included in the ISI program, and subject to regular inspections per the requirements of that program. In addition, the following spent fuel pool cooling system components will be added to the site IST program:

- spent fuel pool cooling pumps
- spent fuel pool cooling system relief valves
- spent fuel pool cooling pump discharge check valves

Requested Information Item 10:

Enclosure 2, Part 1 of the significant hazards consideration determination discusses the probability or consequences of an accident previously evaluated. In the fourth paragraph, you allude to the movement of fuel assemblies "... required to be performed to support this activity (e.g., installation of racks) ..." Since SFPs C and D are currently empty, and no reracking of SFPs A and B are included in this licensing amendment, what fuel movements do you anticipate will be required during the course of the modifications authorized by this license amendment?

Response 10:

There is currently no fuel in the C (or D) spent fuel pool, nor will any be installed until such time as the initial installation of racks are completed and approval from the NRC to place the pool in service is obtained. No fuel movement will be involved with the installation of racks performed in support of this license amendment request.

Requested Information Item 11:

In Section 4.6, "CCWS Hydraulic Margins," the first paragraph refers to a modification to the CCWS piping to SFP HXs C and D to be able to throttle flow to 2.03% for the Hot Shutdown (350°F) alignment. CP&L staff note that the CCWS valves to these HXs must be heavily throttled and will require a suitable sized bypass line with a smaller throttle valve in order to achieve acceptable throttling characteristics. Will these modifications be performed as part of the system activation? If not, how will operators throttle flow to the SFP HXs to meet the design conditions specified in SF-0040 Table 6?

Response 11:

A 6" bypass line and a 6" throttle valve will be installed as part of the modifications performed as part of system activation. This arrangement has been sized to provide acceptable throttling characteristics at the relatively low flow rates required to accommodate a 1.0 MBtu/hr heat load. The requisite throttle position will be set in an initial flow balance. Thereafter, system alignment will consist of opening and closing isolation valves. It is not anticipated that subsequent adjustments to throttle valve position will be necessary.

Requested Information Item 12:

Tables 7a through 7j present the results of a CCWS flow analysis to determine the hydraulic margins for various CCWS lineups. In its summary in Section 4.6, CP&L states that the evaluation of the system thermal analysis results during the "LOCA: Recirculation (RHR and SFP) alignment" (Table 7i) shows that the steady state equilibrium temperature of the fuel pool A/B does not exceed 136 °F even with degraded CCWS pump flow and design fouling of all. HXs. Please provide a copy of Attachment (Z), "Containment Sump Recirculation (RHR and SFP) Alignment Thermal Performance," or provide the basis, assumptions, and results for this calculation, including the assumed decay heat load, the duration of time the CCWS system is providing insufficient flow to SFP HX A, and the maximum SFP bulk temperature for all SFPs.

Response 12:

As requested, a copy of Attachment (Z) to calculation SF-0040 is provided herein as Enclosure 4.

Requested Information Item 13:

CP&L's description of the refuel-normal and abnormal full core offload analysis results (Tables 7e and 7f) indicate that the SFP HX A (or B) can just accommodate an assumed full core offload heat load of 31.7 MBtu/hr at design SFPCS thermal conditions; therefore, the negative CCW flow margin is acceptable under these extreme thermal-hydraulic conditions; however, no basis is provided for this conclusion. Please provide your justification for concluding that operating the SFPCS with CCWS flow 7% less than the minimum flow stated in Tables 7e and 7f assures the design limits for the SFPs are not exceeded.

In addition, in table 7f, the minimum CCWS flow to RHR pump A with a 6% uncertainty is calculated to be 8% less than the minimum required flow rate, yet no justification is given why this deficient condition is acceptable. Please provide the basis why this condition is acceptable.

Response 13:

Calculation SF-0040 Attachments (M) and (N) document the SFP Hx A (or B) thermal-hydraulic analysis at the design SFPCS flow of 3750 gpm at 137 °F, a design CCWS supply temperature of 105 °F, minimum ESWS flow and maximum ESWS temperature, a design fouling of 0.0005 hr-ft^2-F/BTU on both the inside and outside tube surfaces and assuming no tubes plugged. This analysis shows that the SFP heat exchanger(s) can accommodate a heat duty of 31.69 MBtu/hr. The estimated CCWS supply temperature for this system alignment is 104.7 °F with the CCW heat exchanger operating design fouling factors and ESW flow of 8500 gpm at 95 °F.

The assumption of no tubes plugged in the SFP heat exchangers is valid since these heat exchangers are being placed into service for the first time, and this analysis will be utilized for only a single operating cycle, after which system thermal/hydraulic performance will be re-evaluated in support of power uprate and steam generator replacement projects.

In addition, current operating practice at HNP is to evaluate spent fuel pool heat loads each cycle, and specify a minimum time prior to offloading fuel to ensure adequacy of the CCWS. In this instance, core offload would not be performed if SFP heat load exceeded 31.69 Btu/hr.

Relative to the adequacy of CCWS flow to the RHR pumps, Table 7f reflects that the minimum CCWS flow to RHR pump A with a 6% uncertainty is calculated to be 8% less than the minimum design flow rate. This table is applicable to Mode 6, wherein the RHR pumps are used intermittently for volume control rather than heat removal. In this scenario, the spent fuel pool cooling system rejects the heat associated with the offloaded core. For the purposes of SF-0040, it was assumed that both trains of RHR were in operation, even though no design requirement exists for doing so and as it would not be likely as a matter of practicality. This is a conservative approach in that it ensures that flow which might be diverted through the RHR seal coolers is not considered available to other heat loads.

During Mode 6, the CCWS trains are required to be separated to prevent CCW pump run out. In this case, the "B" CCW train, which supplies only the safety related header, has significant flow margin. The "A" CCW train supplies the other safety related header along with the nonessential header, and has slightly less than the 5 gpm design flow to the RHR pump seal cooler (-8%) under these conditions. Assuming the design RHR pump seal cooler heat load (0.07 MBtu/hr), this deficiency in flow would be expected to result in a slight temperature increase in the seal water returning from the cooler. However, this maximum heat load is associated with maximum RHR operating temperature of 350 °F, considerably higher than the RCS temperatures that would exist during defueled conditions (well below 200 °F). As RHR temperatures decline, the heat removal requirements on the seal cooler would also diminish. In fact, HNP Operating Procedure OP-111 does not include a requirement for any seal cooling at RHR temperatures below 225 °F. It is concluded that the 8% flow deficit listed for the RHR pump seal cooler in Table 7f is of no consequence to the performance or reliability of the RHR pumps.

Requested Information Item 14:

On page 28 of SF-0040, you state that the SFPs are conservatively assumed to be at the maximum temperature limit of 105 °F prior to the beginning of the transient. What administrative controls are used at SHNPP to assure that the SFP bulk coolant temperature will not exceed 105 °F during normal operation?

Response 14:

HNP Operating Procedure OP-116, Fuel Pool Cooling and Cleanup, includes a fuel pool heat exchanger outlet temperature limitation of 105 °F under normal operating conditions. A control room alarm is provided to alert the operator if this value is exceeded, and spent fuel pool temperatures are recorded every 4 hours in the reactor operator log books. Operating experience has found that the system is capable of maintaining temperatures well below this value, even under full core offload conditions.

CP&L

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

OCT 29 1999

SERIAL: HNP-99-172

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

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOLS C & D COOLING AND CLEANUP SYSTEM PIPING

Dear Sir or Madam:

By letter HNP-98-188, dated December 23, 1998, Carolina Power & Light Company (CP&L) submitted a license amendment request to increase fuel storage capacity at the Harris Nuclear Plant (HNP) by placing spent fuel pools C & D in service. The U. S. Nuclear Regulatory Commission (NRC) issued letters dated March 24, 1999, April 29, 1999, June 16, 1999, and August 5, 1999 requesting additional information regarding our license amendment application. HNP letters HNP-99-069, dated April 30, 1999, HNP-99-094, dated June 14, 1999, HNP-99-112, dated July 23, 1999, and HNP-99-129, dated September 3, 1999 provided our respective responses.

By letter dated September 20, 1999, the NRC issued a fifth request for additional information (RAI) regarding our license amendment application to place spent fuel pools C & D in service. The September 20, 1999 NRC RAI specifically requests additional information on the proposed alternative plan 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 Enclosures to this letter provide the HNP response to the NRC staff's September 20, 1999 RAI.

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

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

Sincerely,

Donna B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosures:

1. HNP Responses to NRC Request For Additional Information (RAI)
2. Technical Report: HNP - Material Identification of Chips from Carbon Steel Welds Associated with the Spent Fuel Pool Activation Project (1 page total)
3. Chemistry Sample Data Sheets (2 sheets total)
4. QCI-19.1, Revision 1, entitled "Preparation & Submittal of Weld Data Report, Repair Weld Data Report, Tank Fabrication Weld Record & Seismic I Weld Data Report" (25 pages total)

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

bc: (all w/ Enclosure 1)

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

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOL COOLING AND CLEANUP SYSTEM PIPING

Requested Information Item 1:

Explain how the Metorex X-Met 880 Alloy Analyzer discriminates between the different standards that you used in your analysis described in Enclosure 3, "Metallurgy Unit Report for Spent Fuel Pool Weld Metal Composition analysis," of your April 30, 1999, RAI response. What are the chemical element ranges associated with the different standards that you used? What determines a match on a particular standard? What chemical elements are not included in the "Match" determination and how are these elements reconciled?

Response 1:

Background:

The primary objective of the field alloy analysis was to confirm with reasonable assurance that the as-deposited weld material for the spent fuel pool piping field welds is an austenitic stainless steel material compatible with Type 304 stainless steel piping material. The chemical composition of the stainless steel filler materials are specified in ASME Section II, Part C, SFA-5.4 / 5.9. The elements controlled under this specification for stainless steel filler materials are: carbon, chromium, nickel, molybdenum, columbium plus tantalum, manganese, silicon, phosphorus, sulfur, nitrogen, and copper.

The Alloy Analyzer was used in a comparison / identification mode. In the comparison / identification mode, the unknown is compared to reference materials which are input by a specific measurement technique and stored in a memory location of the instrument. This method of analysis was selected to provide reasonable assurance that the chemical compositions of analyzed field welds are consistent with an austenitic stainless steel having a chromium content in the range of 18 to 24 weight percent and a nickel content in the range of 8 to 14 weight percent.

Explain how the Metorex X-Met 880 Alloy Analyzer discriminates between the different standards that you used in your analysis described in Enclosure 4, "Metallurgy Unit Report for Spent Fuel Pool Weld Metal Composition Analysis," of your April 30, 1999, RAI response.

The Metorex X-Met 880 Alloy Analyzer utilizes a Cadmium-109 isotopic source to excite the analyzed material and measure the secondary radiation produced by the source excitation. This instrument can detect elements that range between and include chromium and molybdenum on the periodic chart of the elements. (The elements between and including terbium and uranium are also detected by this instrument with a cadmium source.)

The instrument was configured to detect six specific elements using the following pure element standards: (1) chromium, (2) manganese, (3) iron, (4) nickel, (5) copper, and (6) molybdenum. Iron was selected because austenitic stainless steels are considered to be iron-based alloys; chromium, nickel, and molybdenum were selected because they are primary alloying elements; manganese was selected because it is a secondary alloying element; and copper was selected because it is a potential "tramp" (i.e., unwanted) element in this material that is detectable by this instrument. A backscatter standard was used to determine the background spectrum. The pure element standards and the backscatter standard were supplied with the instrument by the manufacturer. A series of comparison standards were loaded into the instrument for this analysis. These standards included: (1) Type 304 stainless steel, (2) Type 309 stainless steel, (3) Type 310 stainless steel, (4) Type 316 stainless steel, and (5) NIST SRM 1154a. These four secondary standards and one National Institute of Standards and Technology (NIST) Standard Reference Material (SRM) were used because: (1) the instrument was used in a comparison mode, and (2) none of the SRMs available from NIST have compositions consistent with either Type 304, Type 308, or Type 309 stainless steels. NIST SRM 1155 (Type 316 stainless steel) and NIST SRM C1287 (Type 310 stainless steel - modified) were used also, as independent reference checks of the instrument during the field analysis.

In the comparison / identification mode, the unknown is compared to reference materials which are input by a specific measurement technique and stored in a memory location of the instrument. The alloy analyzer has a multi-channel analyzer (MCA) having 256 micro channels. These micro channels represent a specific X-ray energy range (e.g., Channel 1 - 1 to 2 eV, Channel 2 - 2 to 3 eV, etc.). Each element has an average value for its excitation X-ray energy and, in practice, the actual response has a Gaussian distribution. Each pure element has a range, or window, consisting of several micro channels based on the full width at half maximum value of the Gaussian distribution. Therefore, counts detected in an element window are due to a detectable and measurable concentration of this element. The pure element standards and the austenitic stainless steel standards have different compositions. The response of the instrument varies with the concentration of a given element in a standard. The counts obtained for a standard by this instrument are proportional to the elemental concentration(s). Each standard will have a unique pattern (or "fingerprint") of counts in the selected element windows based on its chemical composition. The instrument discriminates between standards and unknowns based on the similarity of the instrument response (or counts detected) to the element windows for the stored standards.

What are the chemical element ranges associated with the different standards that you used?

The chemical element ranges for the standards used are shown below in Table 1. The NIST SRM (1154a) that was used to set-up the Alloy Analyzer has a chemical composition that is not within the chemical composition range for any standard UNS stainless steel alloy. However, the nickel and chromium contents of the NIST 1154a standard are similar to the nickel content of the Type 309 comparison standard and the chromium content of the Type 304 comparison standard, respectively. The remaining detectable elements in these three comparison standards are comparable and cannot be used to accurately differentiate between the various unknowns.

TABLE 1

Chemical Element Ranges for Standards Used to Set-up the Metorex Alloy Analyzer

Standard	Composition, Weight Percent					
	Chromium	Manganese	Iron	Nickel	Copper	Molybdenum
Type 304	**18.28**	1.48	bal.	8.13	0.19	0.17
Type 309	22.60	1.63	bal.	**13.81**	--	--
Type 310	24.87	1.94	bal.	19.72	0.11	0.16
Type 316	16.74	1.44	bal.	10.07	0.11	2.06
NIST 1154a	**19.31**	1.44	bal.	**13.08**	0.44	0.068
Chemical Element Ranges for Standards Used to Check the Alloy Analyzer						
NIST C1287	23.98	1.66	bal.	21.16	0.58	0.46
NIST 1155	18.45	1.63	bal.	12.18	0.169	2.38

The tolerances for the chemical element ranges for the secondary standards (nominal Type 304, Type 309, Type 310, and Type 316 stainless steels) are not known. These secondary standards were provided with mill test reports for their chemical compositions, but the precise accuracy of these standards is not known because they are not certified as traceable to primary reference standards. However, the applicable ASTM standards for these alloys permit a major alloying element range of between 1 and 2.5 weight percent (e.g., carbon content - 0.08 weight percent maximum; silicon content - 1.00 weight percent maximum; nickel content - 8.00 to 10.50 weight percent maximum; etc.) without the applicable product analysis tolerances that depend upon the specific element and its relative concentration.

What determines a match on a particular standard?

During a test, the Alloy Analyzer detects, measures, and compares the counts obtained for the specified elements in the unknown to those for the standards that have been loaded into the instrument (the specified elements are those that were loaded as pure element standards during the instrument set-up). The X-ray energy detection range for each of the specified elements is pre-set in the instrument and is based on physical constants related to the energy difference between electron shells in atomic structures. The number of counts in each pure element range is measured and compared to the counts for these elements in the known comparison standards. The difference in counts between the unknown and the comparison standards is measured. The instrument is configured with three thresholds (or limits) for the difference in counts between the closest standard and the unknown. The least amount of difference between a comparison standard and the unknown is indicated by "GOOD MATCH." If there are differences between the unknown and standard that do not meet the "GOOD MATCH" criteria, but the unknown is similar to one or more standards, the alloy analyzer will indicate "POSSIBLE MATCH." If the difference in counts is too large, the instrument will indicate "NO GOOD MATCH."

What chemical elements are not included in the "Match" determination and how are these elements reconciled?

The primary objective of the field alloy analysis was to confirm with reasonable assurance that the as-deposited weld material was an austenitic stainless steel material compatible with the Type 304 stainless steel piping material. The chemical compositions of stainless steel filler materials are specified in ASME Section II, Part C, SFA-5.4 / 5.9. The elements controlled under this specification for stainless steel filler materials are: carbon, chromium, nickel, molybdenum, columbium plus tantalum, manganese, silicon, phosphorous, sulfur, nitrogen, and copper.

The alloy analyzer was set up to detect the primary alloying elements: chromium, nickel, and molybdenum. In addition, the alloy analyzer was also set up to detect the secondary alloying element manganese, the tramp element copper, and the alloy base iron. The remaining elements addressed in the specification, but not detected by the alloy analyzer, are: carbon, columbium plus tantalum, silicon, phosphorous, sulfur, and nitrogen. None of these elements are capable of being detected with the Metorex Alloy Analyzer using a Cadmium-109 source either due to their relative concentration or their X-ray excitation energy. These secondary alloying elements, while important to the weldability characteristics of the filler material, are not as important to the performance of the weld in service with regard to strength and corrosion resistance.

Samples of three spent fuel pool cooling piping field welds were obtained by plant personnel and submitted to an external commercial laboratory for chemical analysis. The elements that were not determined by field analysis and those that were used in the identification mode of the field welds were measured by this laboratory and are shown in Table 2. Laboratory analysis of this representative sample substantiates the results of the field analysis and provides additional assurance that the chemical compositions of spent fuel pool field welds are satisfactory.

TABLE 2

NSL Chemical Analysis Results			
Identification	**2-SF-36-FW-450**	**2-SF-38-FW-451**	**2-SF-71-FW-329**
Alloy Analyzer Results	304 SS Possible	NIST 1154a Possible	NIST 1154a Possible
NSL Chemical Analysis Results			
Carbon	0.13	0.10	0.064
Niobium	< 0.05	< 0.05	< 0.05
Chromium	20.08	20.11	19.06
Copper	0.054	0.10	0.093
Manganese	1.46	1.39	0.79
Molybdenum	0.12	0.10	0.085
Nickel	9.30	9.24	9.63
Phosphorus	0.021	0.021	0.026
Sulfur	0.007	0.005	0.013
Silicon	0.37	0.39	0.25
Titanium	< 0.01	0.011	< 0.01

In summary, the alloy analyzer was set up to confirm with reasonable assurance that the as-deposited weld material for the spent fuel pool piping field welds is an austenitic stainless steel material compatible with the reported Type 304 stainless steel piping material and the chemical composition requirements specified in ASME Section II, Part C, SFA-5.4 / 5.9. The programmatic and procedural controls which existed at the time of construction, augmented by the testing and analysis effort described above, provide reasonable assurance that the weld material for the spent fuel pool piping field welds is the proper weld material and will perform satisfactorily in service.

Requested Information Item 2:

Provide assurance that the ferrite numbers are acceptable for A-No. 8 weld wire (ND-2433) used in welds with missing weld wire documentation.

Response 2:

Ferrite numbers have been measured for 18 of the 19 accessible field welds remaining in the scope of the Alternative Plan (one field weld is located underneath a grating which could not be removed at the time the measurements were taken). The results of this work show mean ferrite numbers ranging from approximately 4 to 9 FN. SFA 5.9, Section A4.12 states that the ferrite potential for 308, 308L, and 347 is approximately 10 FN, but notes that the ferrite content may vary by +/- 7 FN or more around these midpoints and still be within the limits of the chemical specification. Furthermore, Section A4.13 also states that the ferrite potential of a filler metal is usually modified downward in the deposit due to changes in the chemical composition caused by the welding process and technique used.

Ferrite is know to be beneficial in reducing the tendency for cracking or fissuring in weld metals; however, it is not critical, particularly under the mild service conditions associated with the spent fuel pool cooling system. Assurance that the ferrite numbers are acceptable is demonstrated by the following: (1) the measured ferrite numbers are reasonably consistent with those expected for the type of filler material used, (2) all of the exposed field welds in the scope of the Alternative Plan have successfully completed a liquid penetrant examination which noted no evidence of cracks or fissures, (3) a strict materials control program governed issuance and control of weld materials, and (4) there is no evidence that incorrect or uncontrolled filler material might have been used.

Requested Information Item 3:

Explain the chemical analysis in the Table associated with PQR 6(c), dated 11/15/84, page 2 of 2, laboratory test No. 9-2-149 described in Enclosure 6, "Lab Test Reports," of your April 30, 1999, RAI response. What row(s) are associated with the base material, weld, and standard(s)? What criteria was used to determine acceptability?

Response 3:

Welding Procedure Specification (WPS) 8B2, Revision 16 is supported by four Procedure Qualification Records (PQRs). The original procedure qualification test, as documented on PQR 6, was performed in 1976. The procedure qualification test coupon for this test was prepared from 10 inch schedule 40 pipe, which has a wall thickness of 0.365 inches. This test coupon thickness supports a qualified base metal thickness range of 3/16 (0.1875) inches to 0.730 inches. In 1981, an additional procedure qualification test, as documented in PQR 6(A), was performed to support the extended thickness range of 3/16 inches to 8 inches. This new qualified range was achieved by welding a 1.5 inch thick weld test coupon. In 1982, another procedure qualification test was performed, as documented in PQR 6(B), to expand the thickness range qualified to include a base material thickness as thin as 0.049 inches. This extended range was achieved by welding a 0.049 inch wall thickness test coupon. In 1984, the final procedure qualification test, as documented in PQR 6(C), was performed to extend the qualified thickness range to include materials as thin as 0.031 inches. This new thickness range was achieved by welding a weld test coupon with a thickness of 0.031 inches.

The portion of WPS 8B2, Revision 16 that was used to fabricate the fuel pool piping, based on base metal thickness range, is supported by PQR 6 and PQR 6(A). The fuel pool piping has a nominal wall thickness of 3/8 (0.375) inches, which is within the qualified base metal thickness range of 3/16 (0.1875) inches to 0.730 inches for PQR 6 and 3/16 (0.1875) inches to 8 inches for PQR 6(A).

Relative to the chemical analysis in the Table associated with PQR 6(c), dated 11/15/84, page 2 of 2, laboratory test No. 9-2-149, referenced WPS 8B2 addresses welding of a SA240 TP 304 test coupon with a thickness of 0.031 inch. The documented mechanical test results reference two test specimens having a thickness of 0.031 inch (E&E Laboratory Test Number 9-2-149, specimen numbers 699 and 700). PQR 6(c) references an Arcos welding filler material, which according to the Certified Material Test Reports (CMTRs) attached to PQR 6(c) is Type 316 stainless steel filler material.

A definitive explanation for all of the entries on the data sheet in question, page 2 of 2 of the chemical analysis results, can not be provided due to insufficient documentation. However, based on the documentation supporting the procedure qualification test for PQR 6 (C), Metallurgy Unit test records and anecdotal information, it appears that Harris Welding Engineering personnel requested the E&E Laboratories to perform mechanical testing and chemical analyses for a completed welding procedure qualification coupon performed using 0.031 inch thick Type 316 stainless steel base material. It is believed that the chemical analysis requested was to be performed on a sample of the material taken from the item that was to be welded in production and which provided the impetus to perform the additional weld procedure qualification. This is supported by the fact that chips of the supplied material were provided to the Analytical Chemistry Laboratory on November 12, 1984 (sampled on November 9, 1984) while the PQR is dated November 15, 1984. This indicates that the chemical analysis was performed prior to the welding of the procedure qualification test coupon and should not be considered a part of the procedure qualification test.

Requested Information Item 4:

For the piping and welds examined internally, provide a discussion of the examination results. What inspection criteria is used for evaluating the piping and welds for corrosion and fouling? Describe the corrosion and fouling inspection procedure and inspection personnel qualification process. For the embedded welds not examined internally, describe what is preventing their examination. Discuss why the decision not to inspect all of the embedded welds will result in an acceptable level of quality and safety.

Response 4:

An initial visual inspection of the embedded piping and welds was completed using a pneumatically-powered crawler carrying a high resolution camera. This crawler employed two sets of pneumatic cylinders which expanded and contracted in coordination with a single cylinder between them to produce an "inch worm" effect. Inspections of four of the eight embedded spent fuel pool cooling lines were performed using this crawler, including six embedded field welds. Camera resolution was excellent and the visual inspection of the lines was thorough. This arrangement proved unsuitable, however, for longer lines having multiple elbows, and a decision was made to investigate other possible methods of inspecting the balance of embedded piping. An arrangement was eventually selected which used flexible fiberglass rods to manually drive a camera on rollers through the pipe. A second inspection effort, only recently completed, used this crawler to successfully inspect all 9 of the remaining embedded field welds and associated piping.

The remainder of this response will focus on the initial inspection of four SFP cooling lines and six embedded welds. The results of the inspection of the remaining lines and nine embedded welds is still in the review process. Our preliminary evaluation is that the results of the second visual inspection are consistent with those of the first inspection and demonstrate that the piping and welds have not measurably degraded and are acceptable for their intended purpose.

The pneumatically-powered crawler provided a stable base from which to successfully complete a visual examination of the piping and welds which could be reached using this equipment. Each inspection was preceded by a resolution check wherein the camera was required to discern a 1.0 mil wire at the appropriate focal length, and the level of detail provided of the internal pipe surfaces was excellent. These inspections were conducted in accordance with Special Plant Procedure SPP-0312T, which provided specific acceptance criteria, as well as qualification requirements for the equipment and inspectors. The inspection included welds on four of the eight embedded cooling lines connected to Spent Fuel Pools C & D. All of the lines inspected were 12 inch, schedule 40 stainless steel (304) piping.

The initial inspection included the following field welds:

Field Weld Number	Piping Function
2-SF-8-FW-65	C SFP Cooling Supply
2-SF-8-FW-66	C SFP Cooling Supply
2-SF-143-FW-512	D SFP Cooling Supply
2-SF-144-FW-515	D SFP Cooling Supply
2-SF-144-FW-516	D SFP Cooling Supply
2-SF-159-FW-408	D SFP Cooling Supply

In accordance with the acceptance criteria in Special Plant Procedure SPP-0312T, welds which can be accepted without further evaluation must be completely free of the following defects:

- no Cracks
- no Lack of Fusion
- no Lack of Penetration
- no Oxidation
- no Undercut greater than 1/32"
- no Reinforcement ("Push Through") greater than 1/16"
- no Concavity (Suck Back") greater than 1/32"
- no Porosity greater than 1/16"
- no Inclusions

In addition, any indications not included in the above list of weld attributes but potentially pertinent to the condition of the piping and welds were required by the inspection procedure to be reviewed and formally evaluated by Harris Nuclear Plant Engineering staff. Such indications would include arc strikes, foreign material, evidence of mishandling, pipe mismatch, pitting, and evidence of corrosion.

The inspection procedure requires that personnel performing visual examinations be CP&L Visual Weld Examiners, certified in accordance with the Corporate NDE Manual. In addition, they are required to have successfully completed the CP&L training course on remote camera equipment and/or have demonstrated their capability to utilize the equipment to the satisfaction of the NDE VT Level III. Vendor personnel operating the closed circuit television system were not required to be certified visual weld examiners, but were required to be familiar with their equipment and proficient in its use.

Generally, the inspection results were good. It is noted that the welds in question were not subject to volumetric examination, and were sufficiently far from the open end of the pipe at the time of welding that an internal visual examination would not have been performed at the time of welding. Relative to the inspection criteria pertaining to weld attributes provided above, five of the six field welds were accepted based on the qualified examiner's review of the camera inspection video. A single weld, 2-SF-144-FW-516, was identified as having areas where portions of a consumable insert could be discerned. This weld, which exists in the horizontal piping on the supply line to SFP D, had several locations where a consumable insert had been utilized but was not fully consumed. Generally, these locations were limited to several very small areas where a small portion of the insert could be discerned, but included one area about 1.5 inches long where a continuous portion of the insert could be seen.

The presence of a small amount of unconsumed insert is not considered to be an indication of an unqualified welder, inadequate procedures, or inappropriate materials. The small amount of unconsumed insert is a relatively insignificant imperfection which is not unusual on field welds such as 2-SF-144-FW-516, which was only subject to surface examination and does not lend itself to internal visual examination. ASME Section III, Subsection ND design rules recognize the potential for imperfections of this nature in welds not subject to volumetric examination, and require that a reduction in joint efficiency be assumed for butt welds which are subject to surface examination only (ref. ND-3552.2).

The root pass associated with the indication of unconsumed insert is backed up by multiple weld passes, any one of which would be adequate to establish a leak tight pressure boundary under these conditions. Hydrostatic test records show that field weld 2-SF-FW-144-516 successfully completed hydrostatic testing at 32 psi during construction prior to the line being embedded, and that this test was witnessed by both QC and the ANI. Procedures and processes at the time required that both these field welds were subject to multiple inspections and documentation reviews during construction. Given this, and considering that this weld was subject to multiple inspections at the time of construction, it is highly unlikely that the indications noted on field weld 2-SF-144-FW-516 extend into the root pass, let alone the multiple passes that followed it.

Since field weld 2-SF-144-FW-516 is on a line which connects directly to atmospheric spent fuel pools, hydraulic pressure at the welds is limited to static head and a small amount of friction losses. (The effect of velocity head would be sufficiently small as to be negligible, but would actually tend to reduce the effective pressure.) At the location of field weld 2-SF-144-FW-516, static head due to the elevation difference is approximately 286 - 277.5 = 8.5 feet. Piping friction losses per 100 ft for 12 inch steel piping is only about 3 feet at 4000 gpm, so even considering the effect of elbows in the line, the 55 foot length of piping between this field weld and SFP C would only contribute another few feet for a total head of about 10 feet (i.e., less than 5 psi).

Operation of the SFP cooling and cleanup system for the C & D pools will be at a relatively low temperature and very low pressure. Accordingly, the minimum wall thickness needed to retain this pressure over a localized area of reduced wall is only a very small percentage of the 0.375 inch wall thickness in this piping. The piping in the vicinity of field weld 2-SF-FW-516 is completely embedded in concrete, located approximately at the center of a six foot thick, seismically-designed wall. As such, this piping is not subject to externally induced movement or stresses. Since the SFP cooling and cleanup system operates at a relatively low temperature with little variation, thermally induced stresses and thermal cycling are not of appreciable concern. Given the lack of externally induced stresses or thermal cycling, the small pieces of unconsumed insert will not initiate a crack or otherwise propagate a piping failure.

Based on all of the above considerations, the indications of an unconsumed insert identified on field weld 2-SF-144-FW-516 are acceptable, and no rework or repair to the weld is required.

Videotapes of the first six embedded field welds and associated piping to be visually inspected have been reviewed by CP&L engineering and metallurgical personnel. Aside from localized occurrences of loosely adhering surface film (principally boron deposits from boric acid added to the water), the videotape provides clear evidence that the piping was free from fouling or foreign materials. Where necessary, deposits were removed with pressurized water before the visual inspection. It is the consensus of the reviewers that the condition of the piping and welds is very good. Several inconsequential stains and small pits were noted, indicating that a small amount of minor corrosion may have occurred at some time in the past. Videotapes of all 15 embedded field welds and associated piping have been forwarded to corrosion experts both within CP&L and in the industry.

Requested Information Item 5:

What are the chemical analyses for steel welds 2-CC-3-FW-207, 2-CC-3-FW-208, and 2-CC-3-FW-209?

Response 5:

Chemical analyses for the carbon steel chips have been completed and are provided as Enclosure 2 to this RAI response. The results of these analyses substantiate that the filler material used for these welds is generally consistent with chemical composition requirements found in SFA 5.1 for ER70S-6 and SFA 5.18 for E7018.

Requested Information Item 6:

Describe the paper trail that identifies a specific weld material to a specific weld on the isometric drawings, i.e., show that the weld material being verified with the Metorex X-Met 880 was specified for that location. Identify missing documentation that breaks the paper trial, if any.

Response 6:

The weld metal to be used on a given weld was prescribed by the Weld Procedure Specification. The Weld Data Report (WDR) documented the Weld Procedure Specification to be used, as well as the AWS Classification of filler material. For the field welds for which WDRs are no longer available, it is not possible to directly document the Weld Procedure Specification and filler metal that was used. However, since the vendor data sheets are available on the pipe spools, a review has been done of the Weld Procedure Specifications available at that time and which would have been applicable for this type piping, material, and end prep. These Weld Procedure Specifications were provided to the NRC as Enclosure 6 to HNP-99-069, dated April 30, 1999, the HNP response to the March 24, 1999 NRC RAI on the Alternative Plan.

The pipe spools utilized in the HNP spent fuel pool cooling system are Type 304 stainless steel, a P-8 material. The Weld Procedure Specifications for P-8 to P-8 piping welds such as these in the spent fuel pool cooling system would have used filler metals conforming to SFA No. 5.4 / 5.9, including ER308, ER308L, ER316, ER316L and ER347. For Type 304 to Type 304 piping, ER308 would have typically been specified on the WDR. Given that some chemical changes in composition will be caused by the welding process and that blending of the base metal and filler metal would occur, the Metorex X-Met 880 testing is not intended to confirm the that chemical composition conforms to chemical composition requirements for each element, but rather to assure that weldments are sound by substantiating that the filler metal used was compatible with the piping material and generally consistent with composition requirements of the Weld Procedure Specification. Additional details on the use of the Alloy Analyzer to evaluate filler metal is provided in the HNP response to Requested Information Item 1 above.

Requested Information Item 7:

Discuss the chemical analysis and any other analysis performed on the water in the fuel pool cooling and cleanup system (FPCCS) and component cooling water system (CCWS) for spent fuel pools (SFPs) C and D. Where did the water come from? Discuss any differences between the chemical analysis and the original water source. Provide the staff with a representative analysis of the water.

Response 7:

A review of plant documentation substantiates that the embedded lines connected to SFPs C & D had water in them on two separate occasions during the construction process. Water samples were collected from seven of the eight lines associated with the embedded piping. * Analysis results of those water samples substantiate that the water in these lines originated from the spent fuel pools. Specifically, chloride and fluoride concentrations were very low, and generally consistent with specifications for spent fuel pool chemistry. Sulfate levels and conductivity, while not typically analyzed for spent fuel pool chemistry, were also very low and consistent with high purity water. The water samples also showed low levels of tritium, at a concentration similar to that of the spent fuel pools. Enclosure 3 to this RAI response provides a representative analysis of water samples taken from both the C and D SFP piping.

Initially, these lines were filled with water for hydrostatic testing prior to pouring concrete. Potential sources of hydrotest water included potable water and lake water, although procedures did require that the piping be drained and vented subsequent to test completion. Since these lines could not be isolated from their respective fuel pool liners, they would have been filled again in support of pool liner leak testing. The procedure for liner leak testing required test water to have a chloride content of no more than 100 ppm, which effectively precluded the use of either potable water or lake water for this evolution. Furthermore, procedures required the pools to be drained after testing, then rinsed with distilled or demineralized water. Subsequent to liner leak testing, there was no reason to introduce water into the pools again until they were filled and put into service (1989 - 1990 time frame). Several of these lines were drained one additional time in 1995 - 1996, when drain valves were added to the exposed portions of several of the embedded lines. Since that time, these lines refilled with water from the spent fuel pools. The water samples that were collected and analyzed, as discussed above, were samples of water that leaked past "plumbers plugs" in the pool nozzles since this last evolution.

* One of the eight lines has no drain line with an isolation valve for taking water samples, and was not represented in the initial set of water samples.

Requested Information Item 8:

In Enclosure 8, "Hydrotest Records for Embedded Spent Fuel Pool Cooling Piping and Field Welds," of your April 30, 1999, RAI response, you provided signed hydrostatic test reports for 13 embedded welds. Starting with the signed hydrostatic test report, back track through procedures and program requirements to the point where the missing document(s) were verified as being complete. In other words, identify the specific procedural and program controls requiring verification of completion of the missing documentation (manufacturing/fabrication records, weld data records, updated isometric drawings, and inspections) starting backward from the hydrostatic test report.

Response 8:

Construction procedure WP-115, "Pressure Testing of Pressure Piping (Nuclear Safety Related)," governed the hydrostatic testing of the embedded lines connected to HNP SFPs C and D. This procedure specifically required, prior to hydrotesting, the Mechanical QA Specialist verify that:

1) all required piping documentation is complete, and
2) all required weld documentation is complete.

Reference to piping and weld documentation is found in WP-102, "Installation of Piping." Specific requirements found in this document include:

1) that each weld joint for Code piping receive a WDR, and that these WDRs receive a QA and ANI inspection.
2) that weld procedures utilized be qualified in accordance with MP-01, "Qualification of Weld Procedures."
3) that welders and welding operators be qualified in accordance with MP-02, "Procedure for Qualifying Welders and Weld Operators."
4) that welds be stamped in accordance with MP-05, "Stamping of Weldments."
5) that weld material be controlled in accordance with MP-03, "Welding Material Control."

Generally, items 2 - 5 above ensure that Code welds were performed to appropriate procedures in the plant's Section IX weld program. Relative to item 1, WP-102 provided reference to CQC-19, "Weld Control" which again required that all Code welds received a WDR, and referenced procedure CQI-19.1, "Preparation & Submittal of Weld Data Report & Repair Weld Data Report," for detailed instructions on the use of WDRs. As prescribed by this procedure, the WDR included essentially all of the required attributes and documentation for welds within Code boundaries. Enclosure 4 provides a copy of CQI 19.1 at a revision level existing at or about the time most of the welds in question were made. Similarly, WP-102 contained requirements for layout and dimensional tolerances, as well as references to appropriate procedures for other piping installation processes, such as performance of cold pulls and torqueing of flanged connections. Therefore, in order to satisfy the prerequisites of procedure WP-115, the Mechanical QA Specialist would be required to verify that all the WDRs and RWDRs were complete and approved, dimensional and tolerance inspections had been completed, and all other piping installation processes had been completed and appropriately documented.

Requested Information Item 9:

Identify the concrete pouring procedure that requires checking for the welder symbol and a successful hydrostatic test before pouring.

Response 9:

Since embedding a line in concrete represented a point at which piping was no longer accessible for inspections, rework, etc., 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. Procedure WP-05, "Concrete Placement", included a pre-placement requirement for a craft superintendent sign-off on the concrete placement report to signify completion of the craft's installation and superintendent inspection thereof. This procedure required that this sign-off be made by all craft superintendents, as a safeguard against omissions, whether or not they had material in a particular placement. Subsequently, procedure WP-05 required that the Construction Inspection Unit (QC) be notified when the installation was complete and ready for pre-placement inspection.

Procedure TP-24, "Mechanical Pipe Installation Inspection" provided requirements for the Construction Inspection Unit relative to inspection of piping, and included separate sections on embedded piping inspection. This procedure specifically required the CI inspector to inspect the installation and documentation prior to concrete placement. The CI inspector was required to verify the specific installation attributes:

1) that piping installation was performed in accordance with design drawings and documents, notably including verification of pipe spool identification
2) that piping was free from physical damage, and had no missing parts, and
3) that all piping leak tests were complete and documented.

It can be seen that procedures associated with concrete placement did provide assurance that piping embedded in concrete was the correct piping and was correctly installed. Furthermore, since the hydro-test was generally the final milestone for completion of a pipe segment, verification that all piping leak tests were complete and documented provided assurance that all test and inspection requirements were met. Procedures WP-05 and TP-24 do not specifically require a verification of the welder symbol. Rather, this assurance is provided by the review of weld documentation prior to hydro-testing, as well as the programmatic controls in CQC-19 and related procedures discussed above.

Requested Information Item 10:

Describe how the liner leak tests support weld integrity for welds 2-SF-8-FW-65 and 2-SF-8-FW-66 (Enclosure 3 of your response to NRCs RAI). For these two welds, back track through procedures and program requirements to the point where the missing documents were verified as being completed.

Response 10:

Leak testing of the liner was accomplished under procedure TP-057, "Hydrostatic Testing of Fuel Pool Liners." This procedure provided specific steps to be completed prior to performance of the liner leak test. The procedure required that Engineering prepare the test package, including identification of all boundaries and all isolation points to be utilized. For the north spent fuel pool liner hydrostatic test, the documented test boundaries included the piping runs containing 2-SF-8-FW-65 and 2-SF-8-FW-66.

Subsequent to preparation of the test package, QC was required to complete the "Prerequisites" section of the test form. Similar to the discussion of piping hydro-test procedures provided in the response to Requested Information Item 8 above, these prerequisites included a line item for the QC Inspector to verify "all weld documentation complete." Although the test procedure was specifically concerned with inspection of the liners, this verification would have necessarily extended to the entire pressurized boundary to ensure that no external leakage occurred, that partially completed welds were not overstressed, etc.

Although hydrostatic test packages have not been located at this time for welds 2-SF-8-FW-65 and 2-SF-8-FW-66, plant documentation does support that this hydrostatic test was done. For example, QA Deficiency and Disposition Report (DDR) 794 was initiated to assess hydrostatic test requirements for the plate rings reinforcing the piping to pool nozzle connections. The resolution to this DDR acknowledged that the pipe spools adjacent to these welds had been subject to hydrostatic testing, even going so far as to include the dates of test performance. Four of the ten spools listed are included in the scope of the SFP C and D embedded piping, and two of these spools are in the line in which welds 2-SF-8-FW-65 and 2-SF-8-FW-66 are located. The other two spools referenced are on isometric drawing 2-SF-159, and are specifically included in a hydrostatic test package for which records have been located (provided previously to the NRC as Enclosure 7 to HNP-99-069, dated April 30, 1999). Comparison of the dates listed on DDR 794 against those associated with piping on isometric drawing 2-SF-159 verify that the test dates on these documents are in agreement.

Therefore, even though hydrostatic test records specifically listing welds 2-SF-8-FW-65 and 2-SF-8-FW-66 as inspection items have not been located, it can be established with a high level of confidence that these welds were hydro-statically tested, and that documentation associated with these welds was reviewed and verified as being complete.

Requested Information Item 11:

Describe precautions that were taken to protect system components (e.g., pumps, valves, heat exchangers, piping) from deleterious environmental effects during layup. Describe the layed up condition of the partially completed piping system and how this was determined. How would these layup conditions be different if it was known that SFPs C and D would be put in service later?

Response 11:

The location of system components (e.g., pumps, valves, heat exchangers, piping), the 236' elevation area of the Fuel Handling Building, is fully enclosed and serviced by a safety related HVAC system. This area is also the location of the operating Unit 1 spent fuel pool cooling pumps and heat exchangers, and is completely suitable for the long term storage of piping and equipment. It was anticipated that at some time it would be necessary to place C and D pools into service, and consideration was given to specific requirements for equipment protection. The spent fuel pool cooling pump motors were removed and placed in controlled storage conditions with heaters energized and shafts periodically rotated. The spent fuel pool heat exchangers were capped to preclude introduction of foreign material, and provided with a nitrogen blanket on the shell (CCW) side to prevent moisture and other contaminants from inducing corrosion. Spent Fuel Pool Cooling piping not connected to the spent fuel pools, which had never been wetted and was not connected to any active water systems, also received Foreign Material Exclusion (FME) type covers. Notably, the spent fuel pool cooling pumps and strainers were protected by FME covers on adjacent piping.

Through conversations with cognizant personnel, it is known that when it became necessary to fill the C and D spent fuel pools, the exposed ends of the connected spent fuel pool piping were fitted with leak tight covers and flooded as well. At some point, "plumber's plugs" were fitted in the C and D spent fuel pool cooling nozzles, although it is not clear whether these plugs were installed before or after the lines were flooded by the spent fuel pools The primary purpose of these plugs was not for equipment protection but instead for ALARA considerations, i.e., to preclude collection of radioactive material in the piping.

Requested Information Item 12:

Why was visual inspection rather than ultrasonic inspection chosen to examine the integrity of the embedded welds?

Response 12:

Examination requirements for the embedded spent fuel pool cooling piping at the time of construction consisted of a surface visual and liquid penetrant examination of the piping OD, consistent with design rules and NDE requirements in ASME Section III, Subsection ND. Numerous programmatic and documentation assurances exist to confirm that these required inspections were indeed completed. In reviewing options for inspection of embedded piping and associated welds under the Alternative Plan, the objective was to implement an inspection program which: (1) provided yet another measure of assurance of construction quality, (2) provided a means to inspect as much of the overall scope as possible, (3) allowed for inspection of not only discrete areas of interest (ie., field welds), but also for qualitative assessment of overall piping condition, including corrosion and fouling, and (4) had a high level of probability to produce meaningful results with existing, proven technology. These criteria are individually discussed as follows:

1) Provides additional measure of assurance of construction quality

A detailed inspection of the interior of the piping with a high resolution camera provides a means to discern and assess numerous attributes pertaining to construction quality, including fit-up and alignment, adequacy of purge, and fusion of the root pass. These attributes, while readily examined with the use of a remote camera, do not lend themselves to detection and evaluation through ultrasonic examination.

2) Provides a means to inspect as much of the overall scope as possible

Camera inspection provides a means to see as much of the overall inspection scope (piping interior surfaces) as possible, as well as focus on specific areas of interest. A number of vendors offer inspection services of piping using remote cameras and a variety of propulsion methods, providing the best probability of inspecting as much of the piping as possible. Using real time feedback, direct camera operators can move relatively quickly over long runs of piping which can be readily observed as clean and in good condition; however, considerable time is spent in adjusting focus, lighting and other parameters to provide a detailed examination of specific areas of interest. Although ultrasonic techniques are commonly used to detect wall thinning in steam piping, this process requires that the entire surface to be examined be mapped, with each grid location receiving an ultrasonic examination. Clearly, the lack of access in the embedded piping precludes the use of a similar technique to assess the overall condition of the embedded piping.

3) Allows for inspection of overall piping condition, but also macroscopic examination for fouling, corrosion, etc.

Camera inspection is the only viable means to identify and assess numerous attributes which pertain to the suitability of piping for service, including surface corrosion, fouling, foreign objects in the line, etc. Visual inspection with a high resolution camera can also detect visual evidence of corrosion (stains, discoloration) even when no loss of material or other degradation is obvious.

(4) Provide a high level of probability of producing meaningful results with existing, proven technology

While not deemed appropriate to evaluate macroscopic examination of piping quality for the reasons discussed above, CP&L has investigated the feasibility of using ultrasonic examination to disposition discrete, localized indications. The obstacles associated with remotely performing ultrasonic examinations of these 12 inch embedded lines are considerable, and include:

- Piping runs approaching 100 feet long
- Piping runs including 4 or more elbows
- Both horizontal and vertical runs
- Since pools are full, inspections must be done from the exposed piping end, meaning that all vertical runs are upward
- The weld joints themselves are irregular to the extent a direct beam method could not be used. In addition, these butt welds utilized consumable inserts with an end prep having a counterbore approximately ¾ inch from the weld joint. This configuration complicates the use of angle beam ultrasonic methods
- The piping surface must be clean and smooth, such that boron crystals or any other film or material which are in the area to be inspected must be removed.
- A means must be devised to inject couplant in the area to be inspected
- The technique must provide a means to precisely locate and control the detector transducers, which would invariably require the use of a remote camera

The device would need to be capable of propelling a camera, UT transducers, and all attendant cabling through long pipe sections with numerous elbows and risers to the location of interest, identify and focus on the indication to be examined, clean it as necessary, inject couplant on the area where the transducer will be placed, then precisely locate the transducer at that point, adjusting it as necessary to provide a good signal. Even then, since the back (outside) surface of the weld joints is irregular, it is not certain that the results will allow an accurate interpretation of the condition of the piping. In summary, while several vendors have expressed an interest in working on a cost and materials basis to provide the propulsion, robotics, and equipment necessary to perform ultrasonic examination of the embedded piping, none have been identified with the proven experience necessary to provide repeatable, reliable results under similar conditions.

Requested Information Item 13:

Describe the post modification testing to be performed to ensure that the system(s) will satisfy all design requirements. Include description of hydro-tests to verify the integrity of the system pressure boundaries, flushing to ensure unobstructed flow through the system components, and pre-operational functional testing under design flow/heat loads.

Response 13:

Post modification testing will include the following:

1) System Hydrostatic testing conforming to Section III requirements will be performed on the completed system. With the exception of embedded piping, components inside Code boundaries will be included in this test effort, including pumps, heat exchangers and strainers. In a previous HNP response to the NRC RAI on the Alternative Plan (ref. HNP-99-069, dated April 30, 1999), CP&L stated that Code Case N-240 would be used to exempt formal requirements for hydro-testing of the embedded piping connected to the atmospheric spent fuel pools. CP&L is continuing to investigate methods to provide additional assurance of the quality of embedded piping and field welds, including consideration of pressure testing. The final disposition of hydrostatic testing of embedded spent fuel pool piping will be provided to the NRC as part of the follow-up report on embedded piping and welds as discussed in the response to Requested Information Item 4 above.

2) A flush procedure will be developed which ensures that piping and components inside Code boundaries are free from fouling and debris which might affect system performance, reliability or spent fuel integrity.

3) Pre-operational testing will include a flow balance and verification which ensures that design flow requirements are met for the Spent Fuel Pool Cooling and Component Cooling Water systems, as well as those heat loads which rely on CCW (such as RHR) and heat sinks downstream of CCW (ESW, UHS). Given the lack of a heat load which would facilitate the performance of a meaningful heat duty test of the Spent Fuel Pool Cooling System, no such test will be performed. Moreover, at the 1.0 Mbtu / hr maximum heat load associated with this license amendment request, performance of such a test would not be viable even at the proposed licensed limit. Although the C and D spent fuel pool cooling heat exchangers were installed in the Fuel Handling Building nearly 20 years ago, they have never been placed into service and, from a design perspective, are still new. Moreover, these heat exchangers were layed up with a nitrogen blanket on the shell side, protecting it from moisture and corrosion. A pre-service inspection of the tubesheets and tubes has been performed on these heat exchangers to ensure that no foreign material or corrosion exists which might obstruct flow or otherwise reduce performance.

ENCLOSURE 2 to SERIAL: HNP-99-172

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOLS C & D COOLING AND CLEANUP SYSTEM PIPING

Carolina Power & Light Company
Material Services Section
Metallurgy Services
Technical Report

Subject: HNP - Material Identification of Chips from Carbon Steel Welds Associated with the Spent Fuel Pool Activation Project

(1 page total)

CAROLINA POWER & LIGHT COMPANY
MATERIAL SERVICES SECTION
METALLURGY SERVICES

TECHNICAL REPORT

To: Mr. Jeff Lane

Project Number: 99-134
Date: August 25, 1999

Investigators:
Robert Jordan
Danny Brinkley

Reviewed by: J. W. Wood

Distribution:
File/Metallurgy Services

Approved by: J. J. Block
Supervisor, Metallurgy Services

SUBJECT: HNP- Material Identification of Chips from Carbon Steel Welds Associated with the Spent Fuel Pool Activation Project.

On July 8, 1999 three samples of chips were received from HNP personnel for chemical analysis. The chips were removed from Welds 2CC-FW-207, 208, and 209 on ASME Section III, Class 3 Piping used on the Component Cooling Water (CCW) System. Metallurgy Services personnel were asked to perform chemical analysis on the three samples.

On July 27, 1999 the three samples of chips were sent to NSL Analytical Services, Inc., in Cleveland, Ohio for analysis. A report of the analyses was received from NSL on August 16, 1999. The results of the analysis for each sample are listed in the table below and a copy of the results from NSL is attached.

ELEMENT	SAMPLE 2CC-FW-207 (WEIGHT PERCENT)	SAMPLE 2CC-FW-208 (WEIGHT PERCENT)	SAMPLE 2CC-FW-209 (WEIGHT PERCENT)
Carbon	0.13	0.11	0.11
Chromium	0.028	0.031	0.027
Copper	0.035	0.018	0.018
Manganese	1.29	1.20	1.15
Molybdenum	0.014	0.004	0.003
Nickel	0.028	0.016	0.014
Phosphorus	0.021	0.014	0.013
Sulfur	0.011	0.012	0.013
Silicon	0.29	0.29	0.41
Vanadium	0.018	0.026	0.026

ENCLOSURE 3 to SERIAL: HNP-99-172

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOLS C & D COOLING AND CLEANUP SYSTEM PIPING

Chemistry Sample Data Sheets from HNP Procedure CRC-001

(2 sheets total)

CHEMISTRY SAMPLE DATA SHEET

SAMPLE PT SPENT FUEL POOL 'C' - 2SF-213		
SAMPLE COLLECTOR JERRY BOLEY	OP MODE I	DATE 4-27-99

SAMPLE TIME	PARAMETER	LIMITS	RESULTS	ANALYST
0930	F		13.1 ppb	LLB/MSJ
↓	Cl		28.2 ppb	↓
↓	SO_4		424 ppb	↓
↓	pH		5.33	↓
↓	KTOT		25.4	↓

Comments

Reviewed By: Terry Thompson Date 4-29-99

CHEMISTRY SAMPLE DATA SHEET

SAMPLE PT SPENT FUEL POOL D 2SF-212

SAMPLE COLLECTOR JERRY BOLEY | OP MODE I | DATE 4-27-99

SAMPLE TIME	PARAMETER	LIMITS	RESULTS	ANALYST
0900	F		120 ppb	MSJ/LCB
↓	Cl		70.5 ppb	↓
	SO4		676 ppb	
	pH		6.74	
	KTCT		93.3	

Comments

Reviewed By: Terry Thompson Date 4-29-99

CP&L

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

OCT 15 1999

SERIAL: HNP-99-156

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

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
SUPPLEMENTAL INFORMATION REGARDING THE
LICENSE AMENDMENT REQUEST TO PLACE HNP
SPENT FUEL POOLS 'C' AND 'D' IN SERVICE

Dear Sir or Madam:

Enclosure 8 of the HNP license amendment request (ref. SERIAL: HNP-98-188, dated December 23, 1998) provided a detailed Alternative Plan for demonstrating compliance with ASME Boiler & Pressure Vessel Code requirements for spent fuel pool cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i). By letter dated March 24, 1999, the NRC issued a request for additional information (RAI) related to the Harris Nuclear Plant (HNP) license amendment request to place spent fuel pools C and D in service. The March 24, 1999 RAI included a request to identify each of the embedded field welds within the scope of the Alternative Plan. The HNP response (ref. SERIAL: HNP-99-069, dated April 30, 1999) provided a field weld matrix which identified the field welds to be inspected by using a high resolution remote video camera. The sample size was selected based on a feasibility walkdown with the camera vendor. CP&L has continued, however, to investigate alternative inspection methods with other vendors. Through these efforts with another vendor, CP&L has successfully performed a remote camera inspection of all 15 embedded field welds included within the scope of the Alternative Plan. In the course of the inspection, two field welds (2-SF-1-FW-3 and 2-SF-1-FW-6) which were not embedded in concrete, but within the scope of the Alternative Plan, were cut out to facilitate removal of piping to provide access for the camera inspections. An updated field weld matrix will be provided to reflect the removal of these two welds and the inspection of all 15 embedded field welds.

In addition, by letter dated April 29, 1999, the NRC issued an RAI related to the criticality control provisions in the HNP license amendment request. Item 1 of this RAI requested information regarding a postulated fresh fuel assembly misloading event. As a supplement to our June 14, 1999 response (ref. SERIAL: HNP-99-094) to requested item 1 of the RAI, we had our vendor, Holtec International, perform additional fuel assembly misloading analyses. The results of these analyses are included as an Enclosure to this letter. These analyses demonstrate that criticality will not occur as a result of the postulated misloading of a fresh fuel assembly in the spent fuel storage racks for HNP pools C and D.

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

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

Sincerely,

JH. Eads for DBA

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosure:

c: (all w/ Enclosure)

Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator - Region II

bc: (all w/ Enclosure)

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

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (609) 797-0900
Fax (609) 797-0909

October 11, 1999

Mr. Steven Edwards
Manager of Projects
Carolina Power & Light Company
Harris Nuclear Plant
P.O. Box 165
New Hill, NC 27562

References: Holtec Project 70324
CP&L Contract XTA7000024

Subject: Additional Criticality Analysis Results

Dear Mr. Edwards,

Per your request, and in support of the recent NRC RAIs pertaining to the criticality evaluations performed for fuel storage in pools C and D, we have performed additional analyses.

RAI #1 from the NRC stated that an evaluation of a fuel assembly misloading event should be analyzed. Holtec's previous response drew upon earlier spent fuel rack evaluations and stated that the k_{eff} would remain below 0.95 with a minimum of 400 ppm soluble boron in the pool.

As a supplement to this response, Holtec International has performed additional analyses for the Harris Spent Fuel Pools C and D to determine the amount of soluble boron required to maintain k_{inf} below 0.95 with a misloaded fresh PWR fuel assembly. The results of this analysis are summarized here.

The inadvertent misloading of a fresh PWR fuel assembly into Harris Pools C and D was analyzed using MCNP-4A and CASMO-3. A delta-k_{inf} for the misloading event was calculated using MCNP and this delta-k_{inf} was applied to the maximum k_{inf} in the licensing amendment report (LAR) to determine the maximum k_{inf} under the misloading scenario. This accident scenario consisted of a single 5 wt.% ^{235}U PWR fresh fuel assembly misloaded into the PWR racks surrounded by fuel of maximum reactivity as determined by the burnup and enrichment curve in the LAR. The k_{inf} for the PWR racks with the misloaded fresh assembly, without taking credit for soluble boron, was determined to be 0.9916 with a 95%/95% confidence level.

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (609) 797-0900
Fax (609) 797-0909

A second scenario was also analyzed in which the fresh 5 wt.% ^{235}U PWR fuel assembly was placed in a PWR storage cell adjacent to the BWR storage racks. The PWR and BWR racks were filled with fuel of maximum permissible reactivity. The k_{inf} for this scenario with the misloaded fresh 5 wt.% ^{235}U PWR fuel assembly, without taking credit for soluble boron, was 0.9932 with a 95%/95% confidence level.

These results clearly demonstrate that the spent fuel pool will remain subcritical even with a fresh 5 wt.% ^{235}U PWR fuel assembly misloaded in the PWR racks.

The April 1978 NRC letter to All Power Reactor Licensees states that "The double contingency principle of ANSI N-16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident." Consistent with this approach, credit for soluble boron, which is normally in the spent fuel pool, was taken when the misloaded fresh 5 wt.% ^{235}U PWR fuel was analyzed. It was determined that the maximum k_{inf} for the misloading accident is 0.9352 with 400 ppm soluble boron in the spent fuel pool water. Therefore, the minimum amount of soluble boron required to maintain k_{inf} less than the regulatory limit of 0.95 under all postulated abnormal and accident conditions is 400 ppm.

Additional calculations were also performed to determine the k_{inf} for the misloading accident with 1000 and 2000 ppm soluble boron in the spent fuel pool water. The maximum k_{inf} was calculated to be 0.8671 and 0.7783 for the 1000 and 2000 ppm respectively. These results demonstrate that there is considerable un-credited margin in the criticality analysis of Harris Spent Fuel Pools C and D.

If you have any questions please feel free to contact me.

Sincerely,

Scott H. Pellet

Scott H. Pellet
Project Manager

cc: Holtec Engineering File 80964
Holtec Contracts file

Document ID: 80964SP1

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

PLANT OPERATING MANUAL

VOLUME 6

PART 2

PROCEDURE TYPE: System Description (SD)

NUMBER: SD-116

TITLE: Fuel Pool Cooling and Clean-Up System

Table of Contents

1.0 SYSTEM PURPOSE

The Fuel Pool Cooling and Clean-up System (FPCCS) is designed for the following purposes:

1. To maintain water quality by removing the particulate and dissolved fission and corrosion products resulting from spent fuel in the fuel storage pools and reactor cavity, and;

2. To remove residual heat loads generated by spent fuel stored in the fuel storage pools.

2.0 SYSTEM FUNCTION

2.1 General

The FPCCS maintains water quality in the fuel pools, transfer canals, cask loading pool and the reactor cavity and removes residual heat generated in the stored spent fuel. The FPCCS consists of three subsystems to perform these functions - a cooling system, a cleanup system, and a skimmer system. These systems will be discussed in later sections.

The Fuel Handling Building consists of five main pools. The south end of the FHB consists of New Fuel Pool "A" and Spent Fuel Pool "B". The north end of the FHB consists of Spent Fuel Pool "C", Spent Fuel Pool "D", and the Cask Loading Pool. The five pools are tied to each other by the main transfer canal and the South and North Transfer Canals. The spent fuel is placed in either the "A" or "B" fuel pool during refueling and stored until further disposition. Cooling of spent fuel can be accomplished in either "A" or "B" fuel pool since they are serviced by the fuel pool cooling system. Gates are provided to isolate the five pools and the transfer canals.

Spent Fuel Pools "C" and "D" will not be completed for several years. Consequently, much information is not available for them. The flow paths and mechanical components are discussed but specific electrical and instrumentation information is not.

The fuel pools, the cask loading pool, and transfer canals are furnished with stainless steel liners. The fuel pool liners are constructed to the applicable portions of Section III of the ASME Code. The Fuel Handling Building is designed to Seismic Category I and is tornado missile resistant. Piping in the FPCCS is welded except where flanged connections are used at the pumps, heat exchangers, blind flanges, and control valves to facilitate maintenance. Draining or siphoning of the spent fuel pools via piping or hose connections to these pools and transfer canals is precluded by the location of the penetrations, limitations on hose length and termination of piping penetrations flush with the liner. Control Room and local alarms are provided to alert the operator of abnormal pool level and high temperature. Local water level indication is provided at Spent Fuel Pools "A" and "B".

2.2 Cooling System

(Reference Figures 7.1 through 7.4 for flow diagrams)

The fuel pool cooling system is comprised of two separate systems. Each of these cooling systems consists of fuel pools "A" and "B", south transfer canal, and a cooling loop each with a fuel pool cooling pump, a fuel pool heat exchanger, and a fuel pool strainer. Each of these cooling loops is 100% capacity and independent. The fuel pool cooling pumps are powered from train separated power sources with the capability of being connected to the emergency diesel generator should a loss of offsite power occur.

2.2 Cooling System (continued)

The cooling loops are protected from externally generated missiles and the effects of high and moderate energy fluid piping ruptures.

The fuel pool cooling water return piping terminates at elevation 279'6". The spent fuel pool suction piping exits at 278'6" and the new fuel pool suction piping exits at 277'6". Normal water level in the pool is 284'6" with the top of the spent fuel racks at approximately 260.08'. This design thus precludes uncovering the fuel as a result of a suction line rupture since approximately 18' of water is over the fuel at all times. The location of the cooling inlet and outlet connections to the fuel pools preclude the possibility of coolant flow short circuiting the pool. If, due to a gross valve misalignment, one pool was aligned to the suction of both fuel pools cooling water pumps with no makeup the pumps would loose suction in approximately 7.5 minutes for the spent fuel pools "B" and "C", 2.7 minutes for new fuel pool "A", and 3.5 minutes for spent fuel pool "D".

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

In the event of a single failure in one spent fuel pool cooling loop, the other loop will provide adequate cooling. The concrete forming the pools is designed for 150 degrees F, however, HVAC considerations make 137 degrees F the upper allowable pool temperature. A low flow alarm is provided to warn of interruption of cooling.

2.3 Cleanup System

(Reference Figures 7.5 through 7.7 for flow diagram)

The fuel pool cleanup system is comprised of two separate systems. Each of the cleanup systems consists of a fuel pool demineralizer, a fuel pool demineralizer filter, a fuel pool and refueling water purification filter, and two fuel pool purification pumps. The cleanup system is not safety related nor is it designed to seismic Category I requirements. Valving is provided between the cooling system and cleanup system to permit isolation of this non safety related system.

The fuel pool cleanup system can be used to maintain the purity and clarity of fuel pool water by diverting approximately 6% of the cooling system flow through the cleanup system. The clean-up loop can also take a suction from the refueling cavity at elevation 246.00 ft. and clean the refueling water through the demineralizer and discharge back to the refueling cavity at elevation 285.00 ft. This is done, independently of the cooling loop. The cleanup system is also used to purify the reactor coolant drain tank heat exchanger effluent prior to discharging into the recycle holdup tank, to purify the contents of the RWST, and to drain and purify the reactor cavity.

2.4 Skimmer System

(Reference Figures 7.8 and 7.9 for flow diagram)

The fuel pool skimmer system consists of a skimmer pump, a fuel pool skimmer strainer, and a fuel pool skimmer filter. The skimmer system removes any floating debris from the surface of the various pools. Skimmers may be dispersed as follows:

Pool A:	3
Pool B:	5
South Transfer Canal:	2
Main Transfer Canal:	1
North Transfer Canal:	2
Cask Loading Pool:	1

3.0 COMPONENTS

3.1 Major System Components

(Reference Table 6.2 for Component Design Parameters and Table 6.3 for Electrical Power Supplies of applicable components)

The Fuel Pool Cooling and Clean-up System is comprised of the following components:

1. Fuel Pool Heat Exchanger

   Two fuel pool heat exchangers are provided (1&4A-SA and 1&4B-SB). The fuel pool heat exchangers are of the shell and straight tube type. Component cooling water supplied from the RAB Component Cooling Water System circulates through the shell, while fuel pool water circulates through the tubes. The installation of two heat exchangers assures that the heat removal capacity of the cooling system is only partially lost if one heat exchanger fails or becomes inoperative. The exchangers are located on elevation 236 of the FHB.

2. Fuel Pool Demineralizer

   The demineralizer (1&4X-NNS) is sized to pass approximately six percent of the cooling loop circulation flow to provide adequate purification of the fuel pool water and to maintain optical clarity in the pool. The demineralizer also cleans refueling cavity water by passing a maximum of 260 gpm through the demineralizer. The demineralizer is located on elevation 261 of the FHB, south end.

3. Fuel Pool Cooling Pump

   Two 4560 gpm horizontal centrifugal pumps are installed (1&4A-SA and 1&4B-SB). The use of two pumps installed in separate lines assures that pumping capacity is only partially lost should one pump become inoperative. This also allows maintenance on one pump while the other is in operation. The pumps are located on elevation 236 of the FHB, west of the heat exchangers.

3.1 Major System Components (continued)

4. Filters (Fuel Pool Demineralizer Filter, Fuel Pool and Refueling Water Purification Filter, and Skimmer Filter)

   Three filters are installed; Fuel Pool Demineralizer Filter - 1&4X, Fuel Pool and Refueling Water Purification Filter 1&4X, and Skimmer Filter 1&4X. These filters remove particulate matter from the fuel pool water and are cleaned by the backwash system.

5. Fuel Pool Cooling and Cleanup System Skimmers

   Fourteen skimmers are available for use; five for spent fuel pool "B", one for the main fuel transfer canal, one for the cask loading pool, three for new fuel pool "A", and two each for the south and north transfer canals.

6. Fuel Pool Skimmer Pump

   The 385 gpm fuel pool skimmer pump 1&4X-NNS takes suction from the selected surface floating skimmers and discharges through a filter to the selected pools/canals.

7. Fuel Pool and Refueling Water Purification Pump

   Two 325 gpm fuel pool and refueling water purification pumps are provided (1&4A-NNS and 1&4B-NNS). Each pump can take suction from and return fluid to the refueling water storage tank via Containment Spray System lines, the transfer canals, the new and spent fuel pools, the refueling cavity, or the cask loading pool. Fluids from these systems are purified by the fuel pool demineralizer and filter. Each pump can also take suction from the demineralized water system for line flushing. The pumps are located on elevation 216 of the FHB, south end.

8. Fuel Pool Cooling and Cleanup System Valves

   Manual butterfly valves are used to isolate equipment and lines and throttle valves provide flow control. Valves in contact with fuel pool water are of austenitic stainless steel or of equivalent corrosion resistant material.

9. Fuel Pool Cooling and Cleanup System Piping

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

10. Fuel Pool Gates

   The vertical steel gates with inflatable rubber seals, on the new fuel pools, spent fuel pools, fuel transfer canals, main transfer canal, and cask loading pool allow the spent fuel to be immersed at all times while being transferred without the necessity of filling all pools and canals. They also allow each area to be isolated for draining, if necessary. The pool gates are moved about by the 10-ton auxiliary crane and stored in a storage area in the main transfer canal.

3.1 Major System Components (continued)

11. Fuel Pool Skimmer Pump Suction Strainer

One duplex basket strainer is installed in the suction piping to the Fuel Pool Skimmer Pump. Large particles and debris are collected on the 100 mesh strainer baskets. One side of the strainer may be in operation while the other side is idle or under maintenance.

12. Fuel Pool Strainer

One simplex basket strainer is installed in the suction piping to each Spent Fuel Pool Cooling Pump. The stainer basket is a 40 mesh basket.

3.2 Instrumentation and Control

Each FPCCS has physically and electrically separate and independent instrumentation and controls.

CAUTION

Setpoints are for information only. The official set point document is Ebasco Drawing CAR 2166-B-508.

---

1. Instrumentation

a. FT-5100 (A and B)

These flow transmitters measure flow rate into New Fuel Pool "A" from the fuel pool cooling pumps with a signal to annunciators for low flow on panel F-P9 in the Fuel Handling Building (FHB) and ALB-23, 5-19, in the Main Control Room.

b. TE-5100 (A and B)

These temperature elements measure temperature of New Fuel Pool "A" water with a signal to annunciator panel ALB-23, 5-16 for high new fuel pool temperature. Monitor light boxes are provided to indicate operational status.

c. LS-5100 (A and B)

These level switches measure the level of New Fuel Pool "A" and compare the measured level to the normal water level of 284.50 ft. These annunciators are mounted on ALB-23, 5-18 and Panel F-P9 in the FHB. Monitor Light Boxes are provided for indication of operational status. Annunciators for levels are as follows: high level - 284.75 ft., lo level - 284.00 ft., lo-lo level - 282.00 ft.

d. FT-5110 (A and B)

These flow transmitters measure flowrate into Spent Fuel Pool "B" from the fuel pool cooling pumps with a signal to local indicators and annunciators for low flow on panel F-P9 in the FHB and ALB-23, 4-19.

3.2 Instrumentation and Control (continued)

e. TE-5110 (A and B)

These temperature elements measure temperature of the Spent Fuel Pool "B" water with a signal to an annunciator ALB-23, 4-16 for high pool temperature. Monitor Light Boxes are provided to indicate operational status.

f. LS-5110 (A and B)

These level switches measure the level of Spent Fuel Pool "B" and compare the measured level to the normal water level of 284.50 ft. These annunciators are mounted on ALB-23, 4-18 and panel F-P9 and in the FHB. Monitor Light Boxes are provided to indicate operational status. Level annunciators are as follows: Hi level - 284.75 ft., lo level - 284.00 ft., lo-lo level 282.00 ft.

g. PDT-5112

This pressure transmitter measures the differential pressure across the fuel pool skimmer filter with a signal sent to the Filter Backwash System for indication of need for filter cleaning.

h. PI-5111

This pressure indicator measures the discharge pressure from the fuel pool skimmer pump with local indication.

i. PDS-5109

This differential pressure switch measures the differential pressure across the fuel pool skimmer pump suction strainer and sends a signal to the Filter Backwash System for indication of need for strainer cleaning.

j. PDS-5130 (A and B)

These differential pressure switches measure the difference in pressure across the fuel pool cooling pump strainers, indicate differential pressure at local instrument rack F-R7, and alarms an annunciator mounted on panel F-P7 at high differential pressure.

k. PI-5130 (A and B)

These pressure indicators are locally mounted on the cooling pump discharge headers.

l. PT-5140 (A and B)

These pressure transmitters supply cooling pump discharge header pressure signals to the PIC Cabinets where a pressure switch alarms annunciators mounted on panel F-P7 for header low pressure.

3.2 Instrumentation and Control (continued)

m. PDS-5150

This differential pressure switch measures the differential pressure across the fuel pool demineralizers with indication locally mounted and an annunciator on panel F-P7.

n. PDT-5152 (A and B)

These differential pressure transmitters measure the differential pressure across the fuel pool and refueling water purification filter and the fuel pool demineralizer filter. The signal is fed to the Filter Backwash System for indication of need for filter cleaning.

o. FT-5154 (A and B)

These flow transmitter measure the flow rate of fuel pool filters discharge flow with a signal to an annunciator and indication on panel F-P7. An interlock allows purification pump startup and shutdown without annunciation.

p. TE-5160 (A and B)

These temperature elements measure inlet temperature of fuel pool water to the cooling loop heat exchangers. A local indication is given and annunciators alarm on panel F-P7 for high temperature.

q. TE-5170 (A and B)

These temperature elements measure outlet temperature of the fuel pool water from the cooling loop heat exchangers. On high temperature, an annunciator alarms on panels F-P9 and F-P7 in the FHB. Local indication is given on panel F-P7.

r. PI-5190 (A and B)

These pressure indicators measure the fuel pool and refueling water purification pumps discharge pressure and gives a local indication.

s. PS-5190 (A and B)

These pressure switches measure the fuel pool and refueling water purification pump discharge pressure and on low discharge pressure alarms an annunciator on panel F-P7 in the FHB. This switch is interlocked with the purification pump to allow for pump startup and shutdown without annunciation.

2. System Controls

a. Fuel Pool Cooling Pumps

Start-Stop controls and indication lights are provided for the fuel pool cooling pumps on the Auxiliary Equipment Panel (AEP #1) in the Main Control Room. Indication lights are also provided on panels F-P7 and F-P9. The pumps are sequenced off during a loss of offsite power event, but may be manually restarted.

3.2 Instrumentation and Control (continued)

b. Fuel Pool and Refueling Water Purification Pumps

Start-Stop controls and indication lights for the fuel pool and refueling water purification pumps are provided on panels F-P7, and F-P9. All panels are located in the FHB. Annunciators for pump overload are provided on panel F-P9.

c. Fuel Pool and Refueling Water Purification Filters and Fuel Pool Demineralizer Filters

Filter Backwash System Isolation Valves 1SF-134, 1SF-135, 1SF-124, and 1SF-125

Open-shut controls are provided to isolate the filters from FPCCS piping for cleaning through the filter backwash system. These controls are located in the Waste Processing Control Room on the BOP B/F Auxiliary Control Panel.

d. Fuel Pool Skimmer Filters

Isolation Valves (1SF-86, 1SF-85)

Open-shut controls are provided to isolate the filters from FPCCS piping for cleaning through the filter backwash system. These controls are located in the Waste Processing Control Room on the BOP B/F Auxiliary Control Panel.

e. Fuel Pool Skimmer Pumps

Start-stop controls and indication lights are provided on panel F-P7 and F-P9. All panels are located in the FHB. Annunciators for pump overload are provided on panel F-P9.

4.0 OPERATIONS

The FPCCS is manually controlled and may be shutdown safely for reasonable lengths of time for maintenance or replacement of malfunctioning components. The clean-up loop is normally run on an intermittent basis as required by chemistry analysis of fuel pool water conditions. It is possible to operate the clean-up loop with the demineralizer bypassed. Local samples are taken to permit analysis of demineralizer or filter efficiencies.

The operator may control the fuel pool cooling pumps from main control board only, with indication lights on panels F-P7 and F-P9. FPCCS valves must be manually aligned to take suction from the following locations for various modes of FPCCS operation:

1. Refueling Cavity - The clean-up loop is available for refueling water clean-up and draining through the filter and demineralizer during refueling operations.

2. RWST - The clean-up loop and cooling loop is available to take suction from the RWST and fill transfer canals and fuel pools.

3. North and South Transfer Canals - The clean-up loop is available to drain and clean water in the north and south transfer canals.

4.0 OPERATIONS (continued)

4. Reactor Coolant Drain Tank - The clean-up loop is available to filter and purify water through the demineralizer and filters.

5. Fuel Pools - The cooling loop is available to cool the fuel pool water and send water to the clean-up loop for purification.

6. Cask Loading Pool - The clean-up loop is used to fill and purify the cask loading pool.

7. Main Transfer Canal - The clean-up loop is available to drain and clean water on the main transfer canal.

8. Reactor Make-up Water Storage Tank - The cleanup loop is available to take suction from the RMWST and fill the transfer canals and fuel pools.

FPCCS valves may also be aligned to discharge to the following locations:

1. Refueling Cavity - The clean-up loop discharges to the refueling cavity during refueling operations.

2. RWST - Water can be pumped back to the RWST through the clean-up loop.

3. Boron Recycle Holdup Tank - The clean-up loop provides a path to the Boron Recycle System Holdup Tanks to store water from the transfer canal during fuel transfer system maintenance or recycle refueling water.

4. Fuel Pools - Cooled and cleaned fuel pool water is discharged to the fuel pools through cooling piping.

5. North and South Transfer Canals - The clean-up loop and cooling loop provide a path to fill the north and south transfer canals.

6. Cask Loading Pool - The clean-up loop provides a path to fill and purify the cask loading pool.

5.0 INTERFACE SYSTEMS

5.1 Systems Required for Support

1. Waste Processing System

The Filter Backwash System (FBWS) allows cleaning of the following filters:

a. Fuel Pool and Refueling Water Purification Filters

b. Fuel Pool Demineralizer Filters

c. Fuel Pool Skimmer Filters

In association with the FBWS, the Demineralized Water System, and the Nitrogen System are required to clean the above filters. The spent resin in the fuel pool demineralizer is pumped to the Waste Processing System spent resin header. Controls for isolation valves on the filters and demineralizer are on the Waste Processing System Control Board.

5.1 Systems Required for Support (continued)

2. Component Cooling Water

The Component Cooling Water System provides the heat sink for the fuel pool heat exchangers.

3. Demineralized Water

The Demineralized Water System provides make-up water for the fuel pools and canals by connection to the suction header of the Fuel Pool Purification Pumps. The Demineralized Water System provides priming water for the pool skimmers by connection to the suction piping of the Fuel Pool Skimmer Pump.

5.2 System to System Cross-ties

1. Waste Processing System

The reactor coolant drain tank pumps can transfer reactor coolant from the reactor coolant drain tank to the fuel pool and refueling water purification pumps through the fuel pool demineralizer filter and the demineralizer to the RWST or Boron Recycle holdup tank.

2. Boron Recycle System

The FPCCS purification loop has the capability of transferring borated water from the fuel pools, cask loading pool, transfer canals, refueling cavity, or reactor coolant drain tank to the Boron Recycle System recycle holdup tank.

3. Chemical and Volume Control System

For emergency purposes, CVCS supplies borated water to FPCCS through emergency connections.

4. Containment Spray System

The FPCCS can either take suction from or discharge to the RWST for filling or draining transfer canals and filling pools.

5. Primary Make-up System

The FPCCS can take suction from the RMWST for filling transfer canals and fuel pools.

6.0 TABLES

Table 6.1 - Fuel Pool Heat Load, Equilibrium Temperature, and Heat Inertia

Table 6.2 - Component Design Parameters

Table 6.3 - Fuel Pool Cooling and Clean-Up System Electrical Power Supply

Table 6.1
Fuel Pool Heat Load, Equilibrium Temperature, and Heat Inertia[1]

Fuel Pool Heat Load	
Incore Shuffle	16.84×10^6 Btu/hr
Full Core Offload Shuffle	35.06×10^6 Btu/hr
Post Outage Full Core Offload	35.87×10^6 Btu/hr
Fuel Pool Equilibrium Temperature[2]	
Incore Shuffle	≤137 °F
Full Core Offload Shuffle	≤137 °F
Post Outage Full Core Offload	≤137 °F
Combined Spent and New Fuel Pool Heat Inertia	
Incore Shuffle	4.37 °F/hr
Full Core Offload Shuffle	9.09 °F/hr
Post Outage Full Core Offload	9.30 °F/hr

Notes

1. Based on operation through the end of cycle 9 with the bounding heat load from post RFO-8 plus additional spent fuel shipments.

2. Administrative controls placed on the minimum cooling time prior to the transfer of irradiated fuel from the core to the storage facility to maintain the pools at less than or equal to 137 °F.

Table 6.2
Component Design Parameters

Fuel Pool Heat Exchanger	
Quantity	2
Type	Shell & tube, two pass
UA (per Heat Exchanger) Btu/hr-°F	21.1×10^5
Manufacturer	Yuba Heat Transfer Corp.
Heat Transfer Rate (Btu/hr)	1.506×10^6
Shell Side (Component Cooling Water)	
Inlet temperature, °F	105
Outlet temperature, °F	110.62
Design flowrate, lbm/hr	2.68×10^6
Design pressure, psig	150
Design temperature, °F	200
Material	Carbon Steel
Tube Side (Fuel Pool Water)	
Inlet temperature, °F	120
Outlet temperature, °F	112
Design flowrate, lbm/hr	1.88×10^6
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless Steel
Fuel Pool Cooling Pump	
Quantity	2
Type	Horizontal, centrifugal
Design flowrate, gpm	4560
TDH, ft. water	98.2
Motor horsepower	150
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless Steel
New Fuel Pool (Pool A)	
Volume, gallons at El. 284.5'	142,272
Boron concentration, minimum, ppm	2,000
Liner material	Stainless Steel
Spent Fuel Pool (Pool B)	
Volume, gallons at El. 284.5'	388,800
Boron concentration, minimum, ppm	2,000
Liner material	Stainless Steel
Fuel Pool Demineralizer Filter	
Quantity	1
Type	Back Flushable
Design pressure, psig	400
Design temperature, °F	200
Design flowrate, gpm	325
Maximum dp clean, psi	5
Maximum dp dirty, psi	60
Filter rating, microns	5
Filter type	Stacked, etched disks

Table 6.2

Fuel Pool Demineralizer	
Quantity	1
Resin type	Mixed bed (1:1, cation: anion)
Resin volume, cubic feet	85
Design pressure, psig	400
Design temperature, °F	200
Design flowrate, gpm	325
Maximum dp clean, psi	10
Maximum dp dirty, psi	25
Fuel Pool and Refueling Water Purification Filter	
Quantity	1
Type	Back Flushable
Design pressure, psig	400
Design temperature, °F	200
Design flowrate, gpm	325
Maximum dp clean, psi	5
Maximum dp dirty, psi	60
Filter rating, microns	5
Filter type	Stacked, etched disks
Fuel Pool Strainer	
Quantity	2
Type	Single Basket
Design pressure, psig	150
Design temperature, °F	200
Design flowrate, gpm	4560
Maximum dp at design flowrate, psi	1.4
Mesh size	40
Fuel Pool Skimmer Pump Suction Strainer	
Quantity	1
Type	Duplex Basket
Design pressure, psig	150
Design temperature, °F	200
Design flowrate, gpm	385
Maximum dp at design flowrate, clean, psi	5
Mesh size	100
Fuel Pool Skimmer Filter	
Quantity	1
Type	Back Flushable
Design pressure, psig	400
Design temperature, °F	200
Design flowrate, gpm	400
Maximum dp clean, psi	5
Maximum dp dirty, psi	60
Filter rating, microns	5
Filter type	Stacked, etched disks

Table 6.2

Fuel Pool Skimmer Pump	
Quantity	1
Type	Horizontal, centrifugal
Design flowrate, gpm	385
TDH, ft. water	210
Motor horsepower	40
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless Steel
Fuel Pool and Refueling Water Purification Pump	
Quantity	2
Type	Vertical In line centrifugal
Design flowrate, gpm	325
TDH, ft. water	320
Motor horsepower	60
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless Steel

Fuel Pool Skimmer Connections	Quantity	GPM Rating Each
Spent Fuel Pool (Pool B)	5	35
New Fuel Pool (Pool A)	3	30
North Transfer Canal	2	25
South Transfer Canal	2	25
Main Transfer Canal	1	20
Cask Loading Pool	1	50

Table 6.3
Fuel Pool Cooling and Clean-Up System
Electrical Power Supply

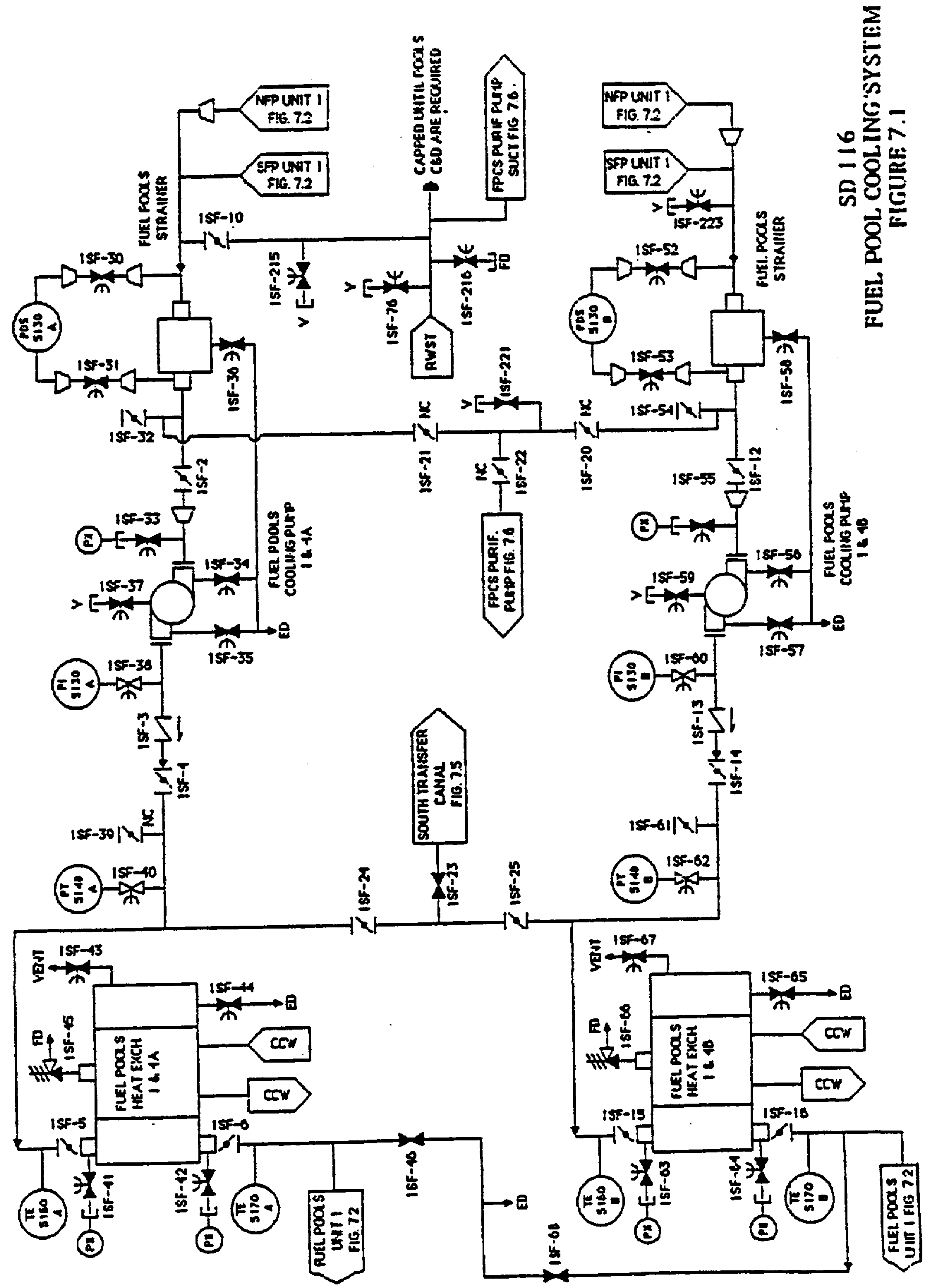
1. 480 VAC Motor Control Centers

LOAD	MCC#
Fuel Pool Cooling Pump 1A-SA	1A33-SA COMPT 4D
Fuel Pool Cooling Pump 1B-SB	1B33-SB COMPT 2D
Fuel Pool & Refueling Water Purification Pump 1A-NNS	1-4A1021 COMPT 1D
Fuel Pool Skimmer Pump 1X-NNS	1-4A1021 COMPT 5E
Fuel Pool & Refueling Water Purification Pump 1B-NNS	1-4B1021 COMPT 5E

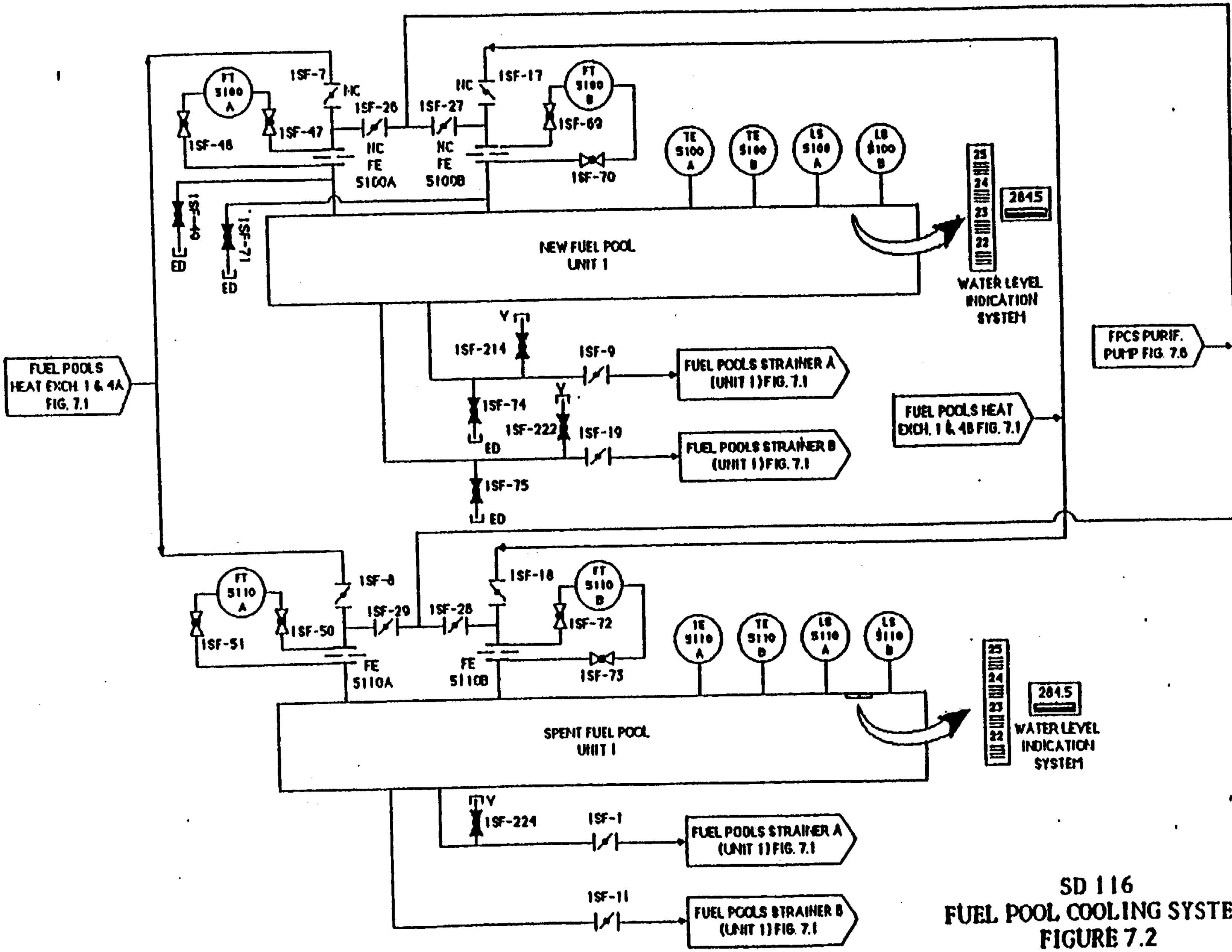
2. 125 VDC Power Panels

LOAD	DISTRIBUTION PANEL #
New Fuel Pool Level Annunciation Relays for LS-5100A(SA)	DP-1ASA CKT#15
New Fuel Pool Level Annunciation Relays for LS-5100B(SB)	DP-1BSB CKT#22
Cooling Pump 1A-SA Annunciation Relay for Low Disch Pressure, PS-5140A	DP-1A-2 CKT#30
Cooling Pump 1B-SB Annunciation Relay for Low Disch Pressure, PS-5140B	DP-1A-2 CKT#30
Spent Fuel Pool Temp Annunciation Relay for TS-5110A(SA)	DP-1ASA CKT#15
Spent Fuel Pool Temp Annunciation Relay for TS-5110B(SB)	DP-1BSB CKT#22
Spent Fuel Pool Level Annunciation Relay for LS-5110A(SA)	DP-1ASA CKT#15
Spent Fuel Pool Level Annunciation Relay for LS-5110B(SB)	DP-1BSB CKT#22
New Fuel Pool Temp Annunciation Relay for TS-5100A(SA)	DP-1ASA CKT#15
New Fuel Pool Temp Annunciation Relay for TS-5100B(SB)	DP-1BSB CKT#22
Fuel Pool Purification Pump 1A-NNS Space Heater	PP1-4A 10221 CKT #8
Fuel Pool Purification Pump 1B-NNS Space Heater	PP1-4B 10212 CKT #2
Fuel Pool Cooling Pump 1A-SA Space Heater	PP 1A 33-SA CKT #1
Fuel Pool Cooling Pump 1B-SB Space Heater	PP 1B 33-SB CKT #1
Local Control Panel F-P9	PP 1-4A10221 CKT #9
Local Control Panel F-P7	PP 1-4A111 CKT #27

7.0 FIGURES



SD 116
FUEL POOL COOLING SYSTEM
FIGURE 7.1



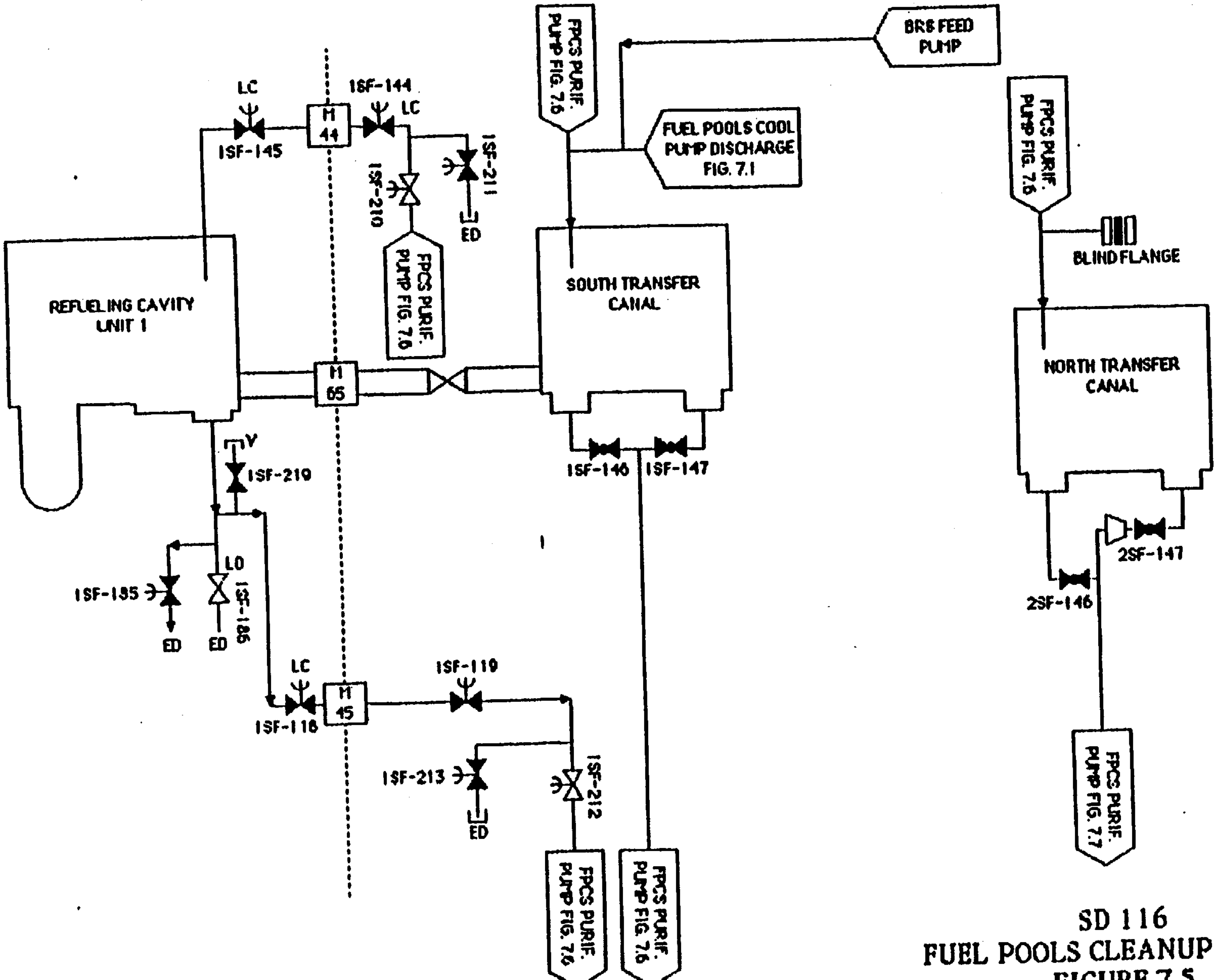
SD 116
FUEL POOL COOLING SYSTEM
FIGURE 7.2

UNIT 2 AND 3 FUEL POOL COOLING
SYSTEM CURRENTLY NOT INSTALLED

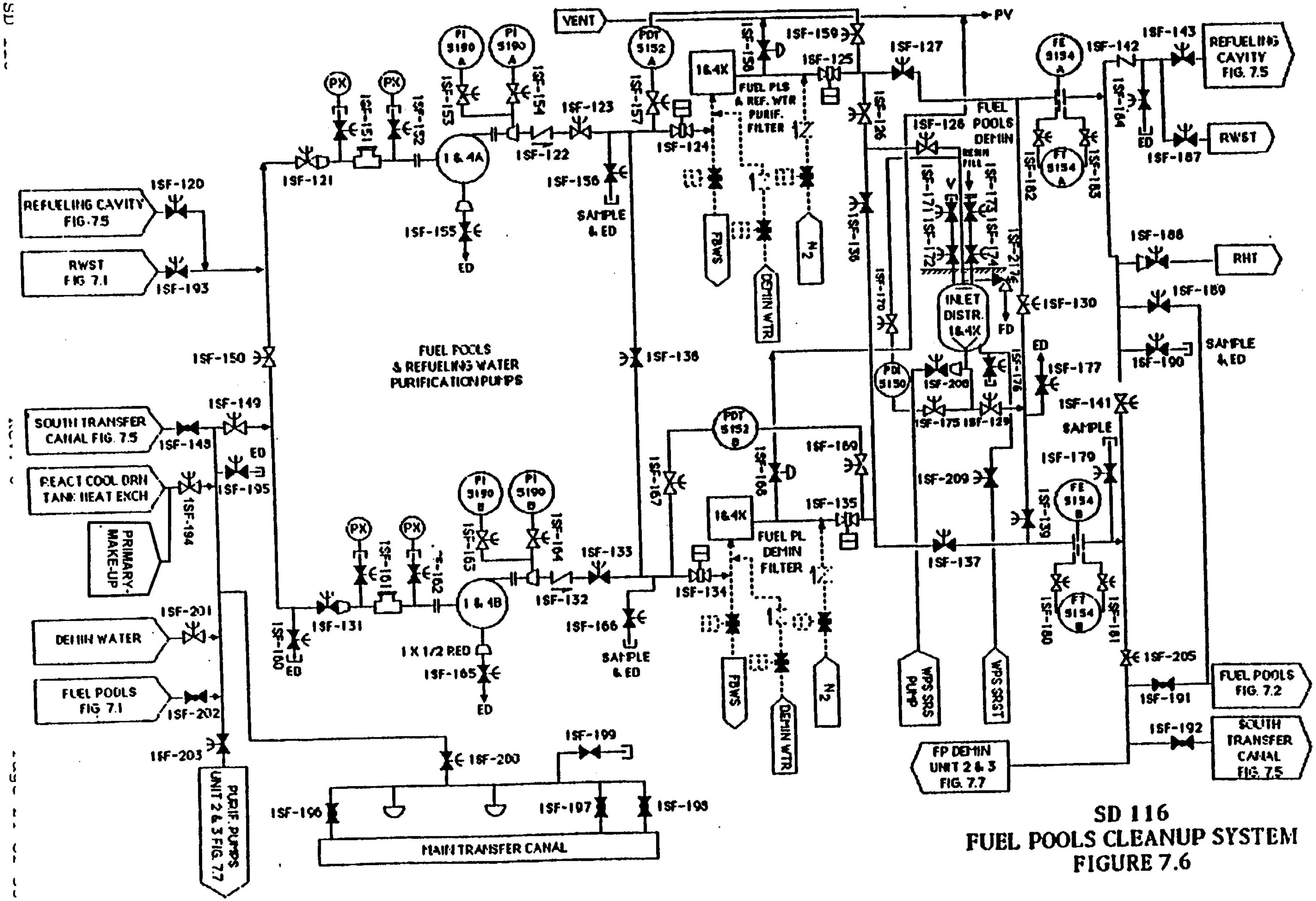
SD 116
FUEL POOL COOLING SYSTEM
FIGURE 7.3

UNIT 2 AND 3 FUEL POOL COOLING
SYSTEM CURRENTLY NOT INSTALLED

SD 116
FUEL POOL COOLING SYSTEM
FIGURE 7.4

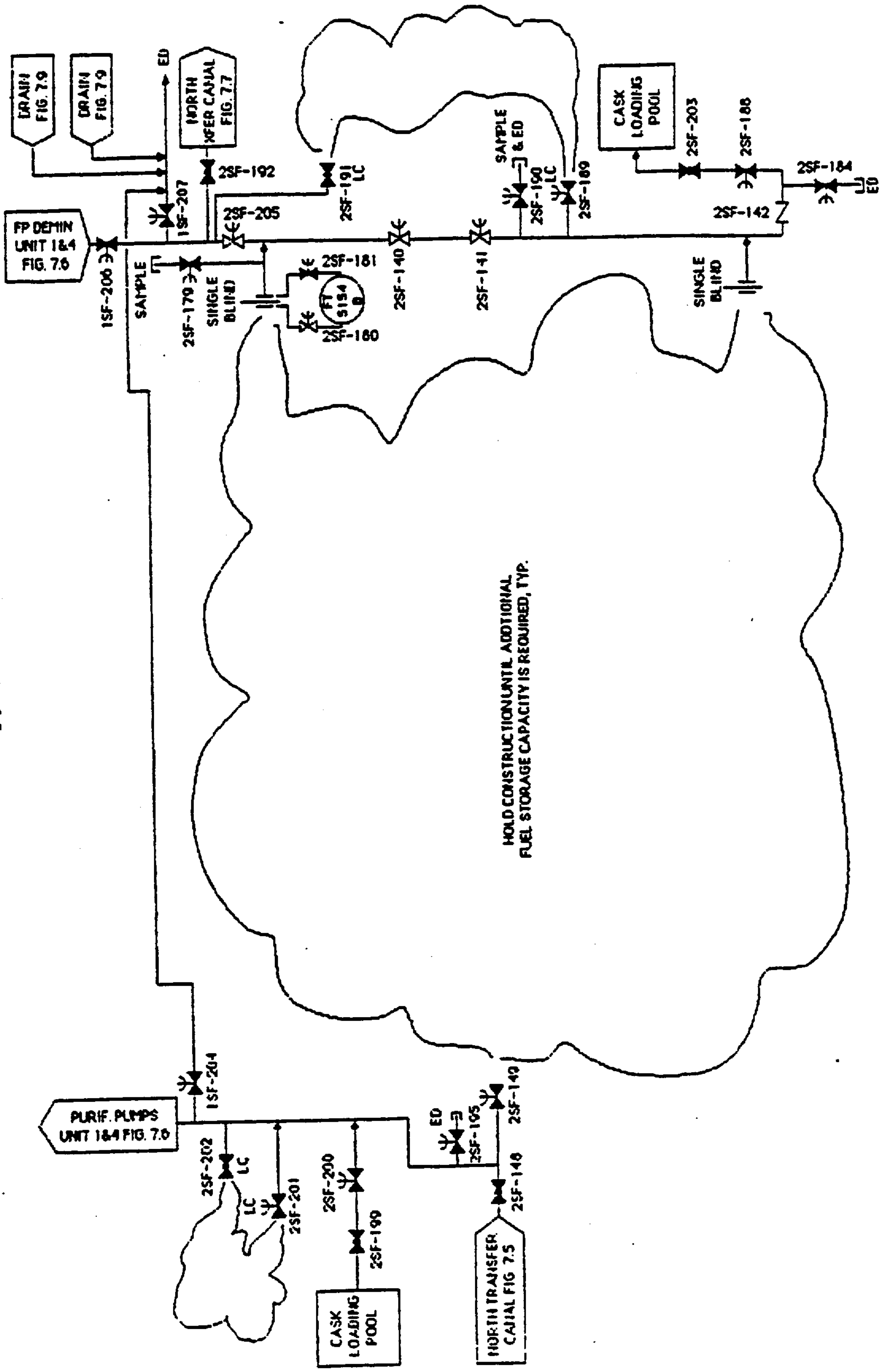


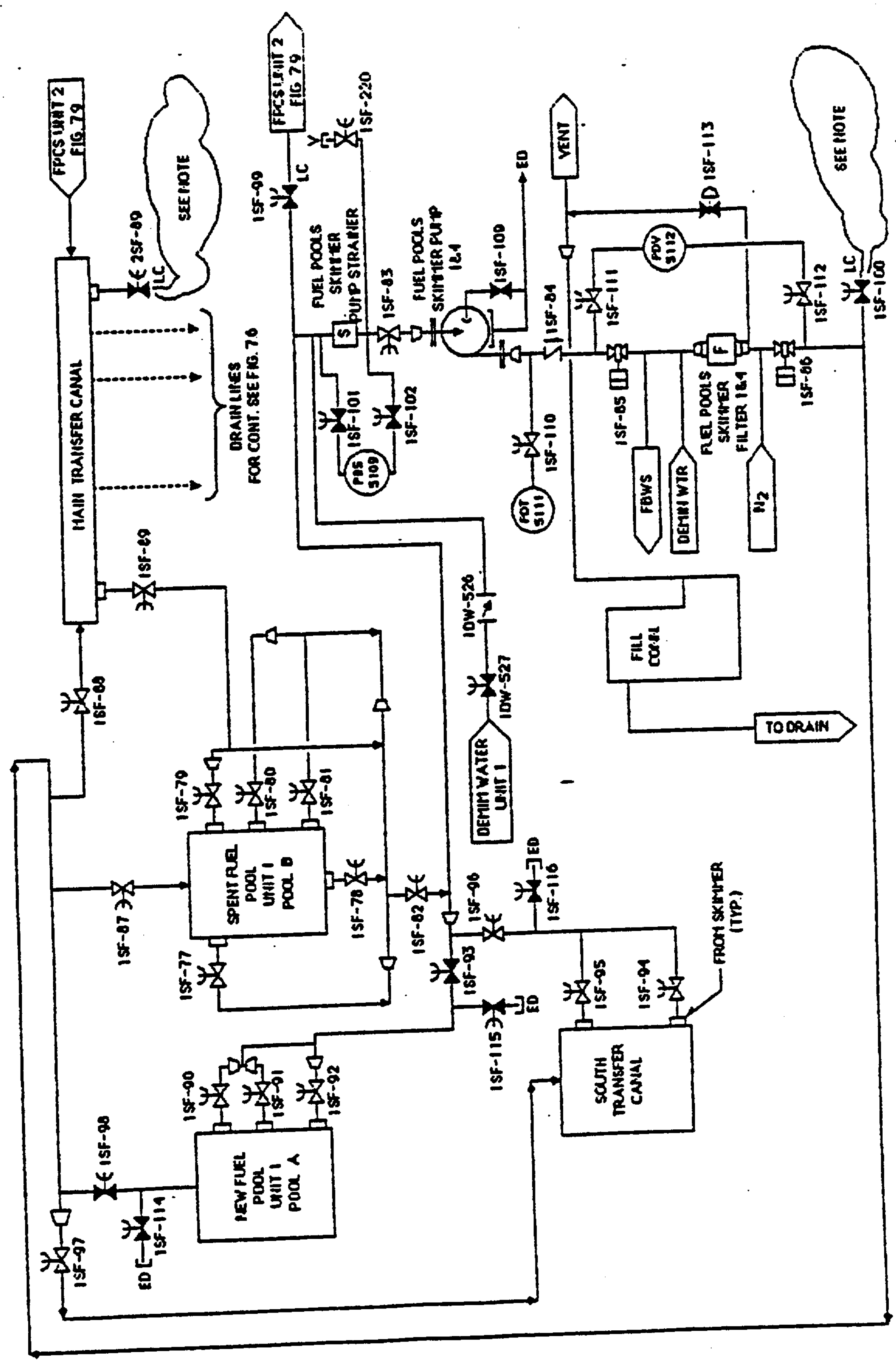
SD 116
FUEL POOLS CLEANUP SYSTEM
FIGURE 7.5



SD 116
FUEL POOLS CLEANUP SYSTEM
FIGURE 7.6

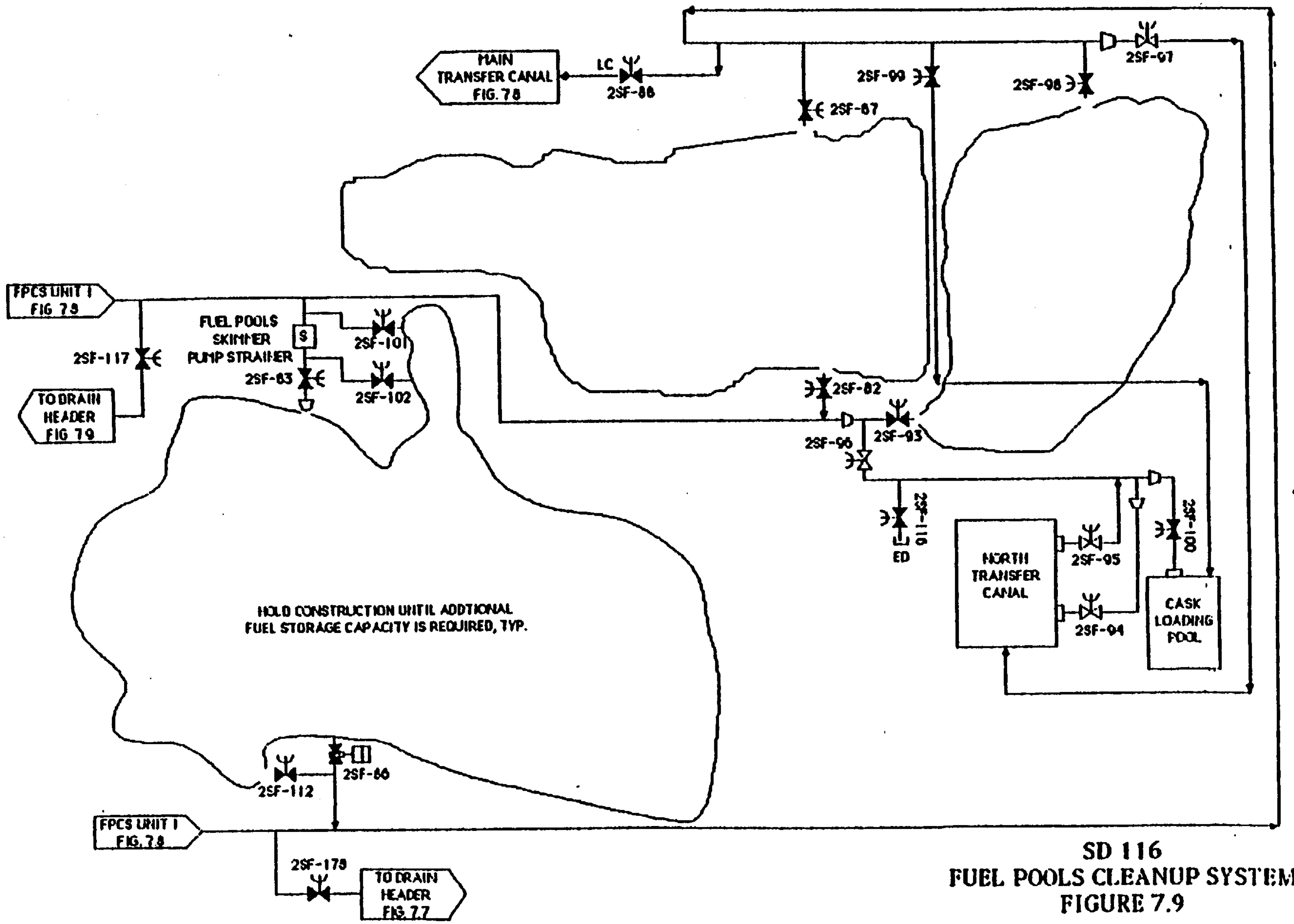
SD 116
FUEL POOLS CLEANUP SYSTEM
FIGURE 7.7



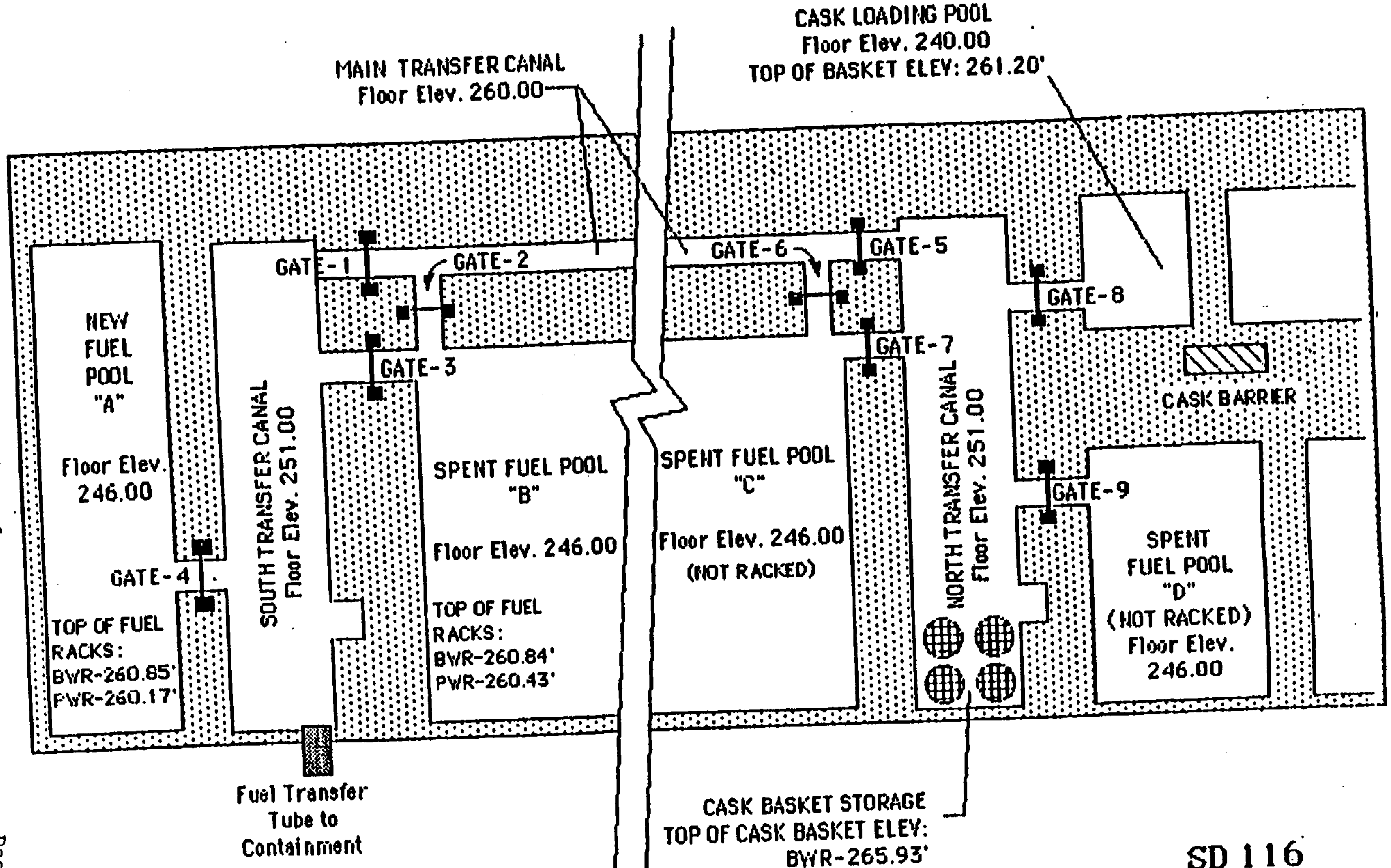


SD 116
FUEL POOLS CLEANUP SYSTEM
FIGURE 7.8

NOTE: HOLD CONSTRUCTION UNTIL ADDITIONAL FUEL STORAGE FUEL STORAGE CAPACITY IS REQUIRED.



SD 116
FUEL POOLS CLEANUP SYSTEM
FIGURE 7.9



SD 116
FUEL POOLS LAYOUT PLAN
Figure 7.10

8.0 REFERENCES

8.1 Drawings

1. Flow Diagrams

   CAR-2165-G-061, Flow Diagram Fuel Pools Cleanup Systems, Sheet 1, Units 1, 2, 3, and 4

   CAR-2165-G-062, Flow Diagram Fuel Pools Cleanup Systems, Sheet 2, Units 1, 2, 3, and 4

   CAR-2165-G-305, Flow Diagram Fuel Pools Cooling System, Units 1 and 4

2. Control Wiring Diagrams

   CAR-2166-B-401, Sheet 881, Fuel Pool Cooling Pump Heat Exchanger 1 & 4A-SA Instrumentation

   CAR-2166-B-401, Sheet 882, Fuel Pool Cooling Pump Heat Exchanger 1 & 4B-SB Instrumentation

   CAR-2166-B-401, Sheet 883, Spent Fuel Pool Instrumentation

   CAR-2166-B-401, Sheet 884, Spent Fuel Pool Water Level Alarm and Indication, Sheet 1

   CAR-2166-B-401, Sheet 885, Spent Fuel Pool Water Level Alarm and Indication, Sheet 2

   CAR-2166-B-401, Sheet 886, Spent Fuel Pool Water Level Alarm and Indication, Sheet 3

   CAR-2166-B-401, Sheet 887, New Fuel Pool Instrumentation

   CAR-2166-B-401, Sheet 888, New Fuel Pool Water Level Alarm and Indication, Sheet 1

   CAR-2166-B-401, Sheet 889, New Fuel Pool Water Level Alarm and Indication, Sheet 2

   CAR-2166-B-401, Sheet 890, New Fuel Pool Water Level Alarm and Indication, Sheet 3

   CAR-2166-B-401, Sheet 891, Fuel Pools and Refueling Water Purification Pump 1 & 4A-NNS

   CAR-2166-B-401, Sheet 892, Fuel Pools and Refueling Water Purification Pump 1 & 4B-NNS

   CAR-2166-B-401, Sheet 895, Fuel Pools Skimmer Pump 1 & 4X-NNS

   CAR-2166-B-401, Sheet 897, Fuel Pools Cleanup System Annunciation

   CAR-2166-B-401, Sheet 904, Fuel Pool Cooling Pump 1 & 4A-SA

   CAR-2166-B-401, Sheet 905, Fuel Pool Cooling Pump 1 & 4B-SB

8.1 Drawings (continued)

3. Foreign Prints

VENDOR: Ingersoll-Rand Company EQUIPMENT: FPRW Purification Pumps

P.O. No: NY435009

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
2005	3040PG5/SFPRWPP	FUEL POOLS 7 REFUELING WTR PURIF PUMP MOTOR OUTL
20545	SPAD-15B/AMEND5	FUEL POOL & REFUELING WTR PURIF PUMP O/L
22733	3318B34	SPENT FUEL POOL WTR PURIF PMP MOTOR O/L 2 SH
35461	214-70	MOTOR DATA FOR SPENT FUEL POOL PURIF PMP MTR
48683	10VOC-B	SPENT FUEL POOL PURIF PUMP PERFORMANCE CURVE

VENDOR: Hungerford & Terry, Inc. EQUIPMENT: Spent Fuel Pool Demineralizer

P.O. NO.: NY435028

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
1917	SK27609-C2	SPENT FUEL POOL DEMINERALIZER NAMEPLATE DETAILS
1918	27609-B3	SPENT FUEL POOL DEMINERALIZER UNDERDRAIN HUB DETA
1919	27609-B4	SPENT FUEL POOL DEMINERALIZER INTERIOR REFERENCE
1920	27609-B2	SPENT FUEL POOL DEMINERALIZER UNDERDRAIN LATERALS
1921	27609-B1	SPENT FUEL POOL DEMINERALIZER INLET DISTRIBUTOR D
1922	27609-A1	SPENT FUEL POOL DEMINERALIZER TANK DETAILS
2153	SHSS-E-32	SPENT FUEL POOL DEMINERALIZER STRAINER ASSEMBLY

VENDOR: Yuba Heat Transfer Company EQUIPMENT: Fuel Pools Heat Exchanger

P.O. NO.: NY435029

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
1907	73-N-003-1-1	FUEL POOLS HEAT EXCHANGER OUTL
2751	73-N-003-1	FUEL POOLS HT EXCH BILL OF MATERIAL SH 1 TO 6
2752	73-N-003-1-5	FUEL POOLS HT EXCH DIST. BELT DETAILS
2753	73-N-003-1-2	FUEL POOLS HT EXCH SHELL 7 CHANNEL DETAILS

8.1 Drawings (continued)

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
3020	A2587	3/4 X 1 IN 150LB REL VA FOR SPENT FUEL POOL HEAT EXCH

VENDOR: Vacco Industries

P.O. NO.: NY435106

EQUIPMENT: Flushable Filters, Fuel Pool Skimmer Filters, Fuel Pool and Refueling Water Purification Filter, Fuel Pool Demineralizer Filter

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
7769	76283/SH1	FILTER HOUSE ASSY FUEL POOL SKIMMER FHB-236
7670	76283/SH2	FILTER HOUSE ASSY FP SKIMMER FILTER FHB-236
7671	76282/SH1	FILTER HOUSE ASSY FP 7 REFUEL PURIF FILTER FHB-236
7672	76282/SH2	FILTER HOUSE ASSY FP & REFUEL PURIF FILTER FHB-236
7673	76281/SH1	FILTER HOUSE ASSY FUEL POOL DEMINERALIZER FHB-236
7674	76281/SH2	FILTER HOUSE ASSY FP DEMIN FILTER FHB-236
3790	N1E10109/SH1	FUEL POOL SKIMMER FILTER FHB-236
3791	N1E10108/SH1	FUEL POOL REFUEL WATER PURIF FILTER FHB-EL 236
3792	N1E10112/SH1	SPENT RESIN SLUICE FILTER WPB-236
3793	N1E10107/SH1	FUEL POOL DEMINERALIZER FILTER FHB-236

VENDOR: Zurn Industries

EQUIPMENT: Strainers

P.O. NO.: NY435163

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
5435	I-771102-B	FUEL POOLS STRAINER FHB EL 236
5980	I-780619-A	FUEL POOLS SKIMMER PUMP SUCT STRAINER FHB EL 236

8.1 Drawings (continued)

VENDOR: Gould Pumps, Inc. EQUIPMENT: Fuel Pool Skimmer Pumps

P.O. NO: NY435181

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
8276	C784889N01	FUEL POOL SKIMMER PUMP O/L FHB EL 236
7321	1592/C784888912	FUEL POOL SKIMMER PUMP PERF CURVE
8273	C784889N02	FUEL POOL SKIM PUMP CROSS SECT & BILL OF MATL
23810	A-26152/7806JH	FUEL POOL SKIMMER PMP 1&4X-NNS PC&TL - FINAL
23811	A-26153/7805JH	FUEL POOL SKIMMER PMP 2&3X-NNS PC&TL - FINAL
15516	IDENT-L0301	FUEL POOL SKIM PUMP SPD TO PRTOUT
14406	5631D92	FUEL POOL SKIMMER PUMP MOTOR O/L 4SHS

VENDOR: Goulds Pumps Inc. EQUIPMENT: Spent Fuel Pool Cooling Pumps

P.O. NO.: NY435042

EBASCO DWG. NO.	DRAWING NO.	FUNCTIONAL TITLE
2296	N232723#1	FUEL POOLS COOLING PUMP O/L UNITS 2 & 3 FHB EL 236
2679	N23723#1A	FUEL POOLS COOLING PUMP O/L UNITS 1 & 4 FHB EL 236
2295	N232723#2	FUEL POOLS COOLING PUMP - CROSS SECTION
16182	C-25551	SPENT FUEL POOL COOLING PUMP PERF CURVE
5238	A-11596	SFP COOLING PUMP MOTOR SPEED TORQUE

4. General Arrangements

   CAR-2165-G-022 through 026, General Arrangements Fuel Handling Building - Plans and Sections

5. Piping Plans

   CAR-2165-G-438S01, Miscellaneous Piping Containment Building, Unit 1

   CAR-2165-G-252, FHB Piping Plan - EL 216, Units 1 and 4

   CAR-2165-G-253, FHB Piping Plan - EL 216 and 261, Units 1 and 4

8.1 Drawings (continued)

CAR-2165-G-254, FHB Piping Plan - EL 236, Sheet 1, Units 1 and 4

CAR-2165-G-255, FHB Piping Plan - EL 236, Sheet 2, Units 1 and 4

CAR-2165-G-256, FHB Piping Plan - EL 286, Sheet 1, Units 1 and 4

CAR-2165-G-257, FHB Piping Section, Sheet 1, Units 1 and 4

CAR-2165-G-258, FHB Piping Sections, Sheet 2, Units 1 and 4

CAR-2165-G-259, FHB Pipings Sections, Sheet 3, Units 1 and 4

CAR-2165-G-260, FHB Pipings Sections, Sheet 4, Units 1 and 4

CAR-2165-G-261, FHB Pipings Sections, Sheet 5, Units 1 and 4

CAR-2165-G-262, FHB Pipings Sections, Sheet 6, Units 1 and 4

CAR-2165-G-263, FHB Pipings Sections, Sheet 7, Units 1 and 4

CAR-2165-G-266, FHB Piping Plan - EL 286, Sheet 2, Units 1 and 4

6. Instrument Schematics and Logic Diagrams

   CAR-2166-B-430, Fuel Handling (Fuel Pools) Sheet Nos. 4.1-4.8, 4.8A, and 4.9

7. Power Distribution and Motor Data Sheets CAR 2166-S-041, Sheets 177S01, 183S01, 227S01, 227S02, 254S01, 613, and 633

8.2 Specifications

Specification No. M-13, Spent Fuel Pool Cooling Pumps, Gould Pumps, Inc.

Specification No. N-15, Spent Fuel Pool Demineralizer, Hungerford & Terry, Inc.

Specification No. M-49Z, Series 514S Inlet Strainers, Zurn Industries

Specification No. M-24, Spent Fuel Pool Heat Exchangers, Yuba Heat Transfer Corporation

Specification No. M-10, Spent Fuel Pool Refueling Water Purification Pumps, Ingersoll Rand Company

Specification No. N-36, Flushable Filters, Vacco Industries

8.3 Technical Manuals

Spent Fuel Pool Cooling Pumps, Manual No. BHQ, Goulds Pumps, Inc.

Spent Fuel Pool Heat Exchangers, Manual No. MXJ, Yuba Heat Transfer Corporation

Series 514 Sinlex Strainer, Manual No. BRP, Zurn Industries

Spent Fuel Pool Demineralizer, Manual No. MXK, Vol. 2, Hungerford & Terry, Inc.

8.3 Technical Manuals (continued)

Spent Fuel Pool and Refueling Water Purification Pumps, BJH, Ingersoll Rand Company

Flushable Filters, Manual No. AYF, Vacco Industries

Fuel Pool Skimmer Pump, Manual No. IQY, Goulds Pumps, Inc.

8.4 Other References

Shearon Harris Nuclear Power Plant FSAR, Volume 16, Section 9.1.3, Fuel Pool Cooling and Clean-up System

Technical Specifications for Shearon Harris Nuclear Power Plant, Unit 1, Section 5.6, Paragraph 2

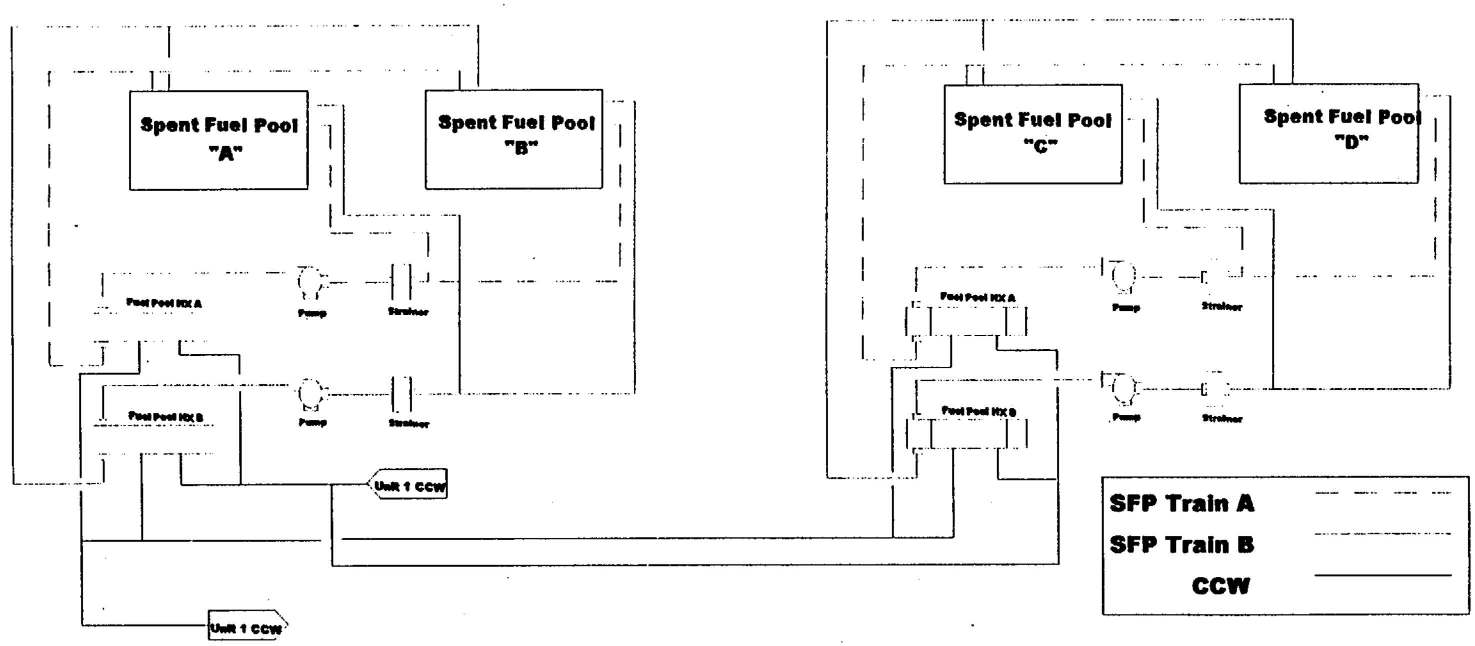
Shearon Harris Nuclear Power Plant Instrument List CAR-2166-B-432

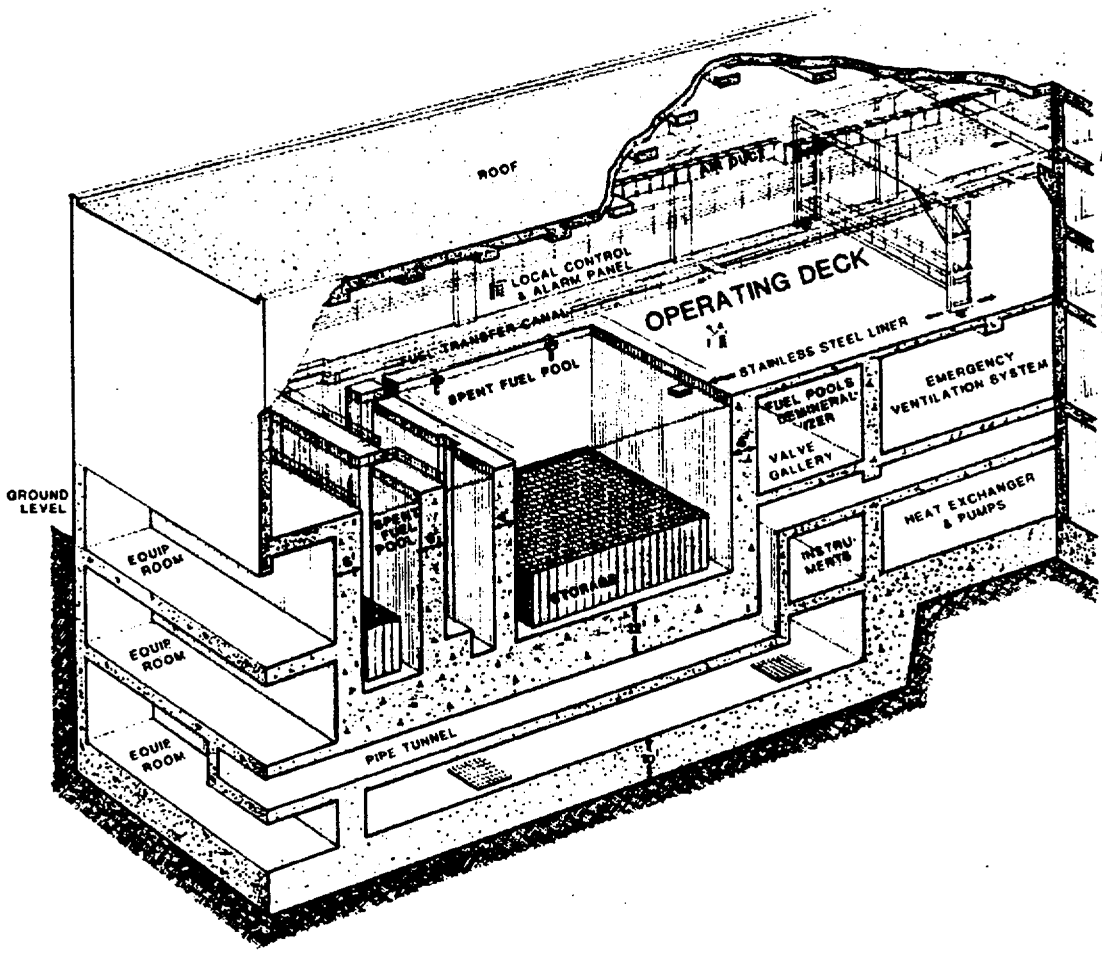
SUMMARY OF CHANGES TO SD-116, REV. 6

Page	Description
All	Changed revision number to 6.
4	Section 2.2, revised paragraph on makeup to fuel pools per current plant practices

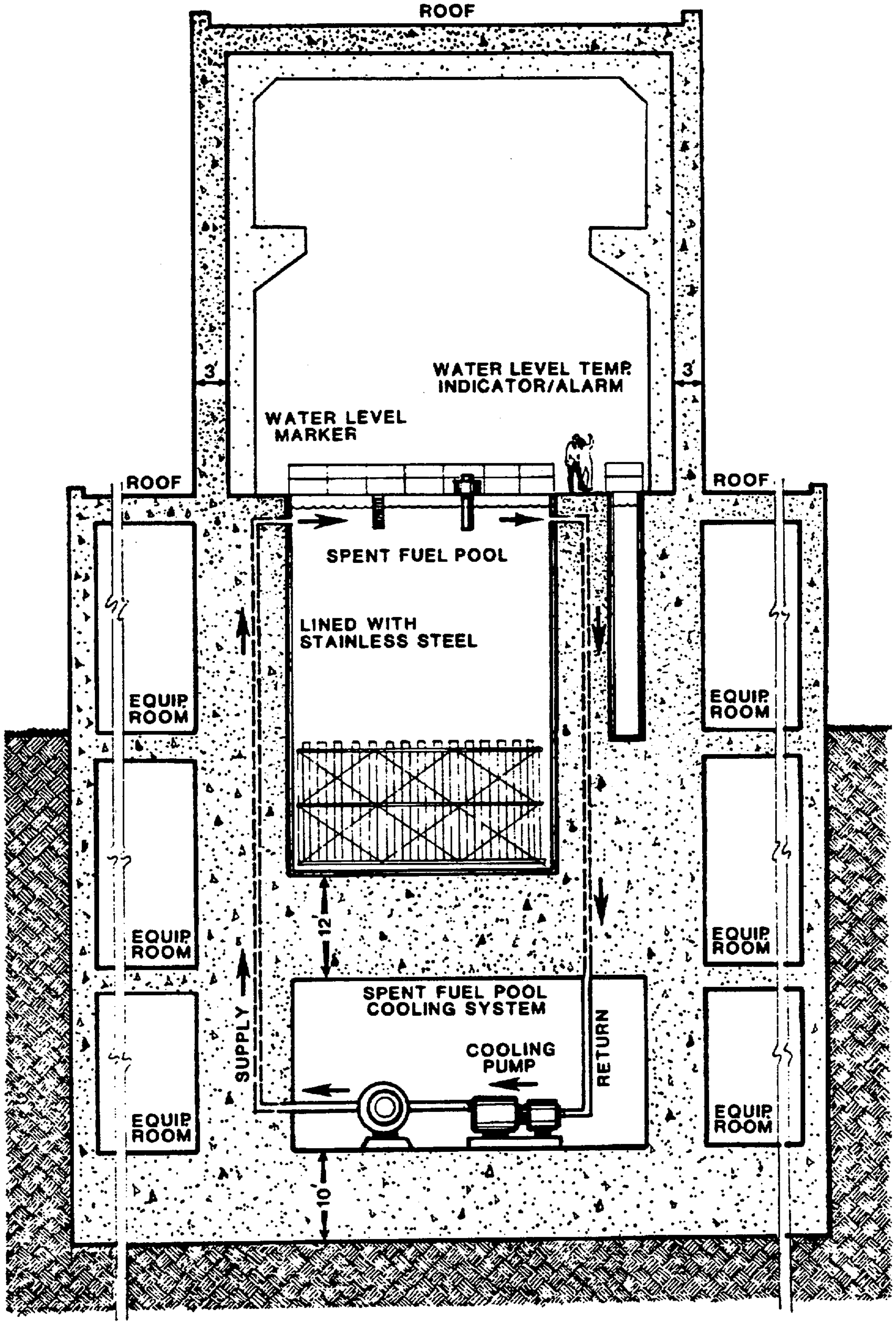
HNP Spent Fuel Pool Cooling System

Simplified Flow Diagram

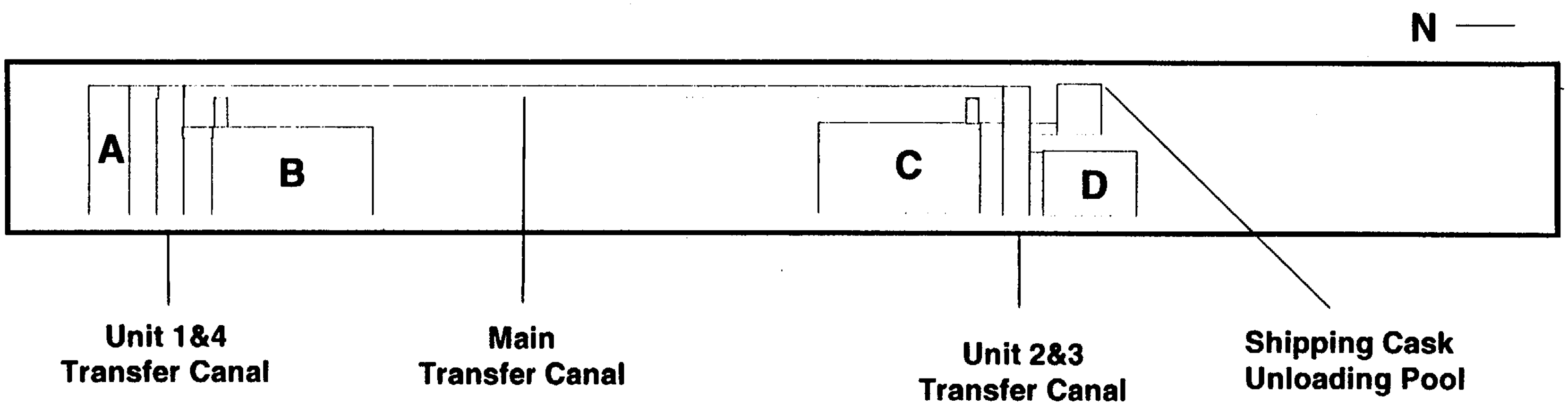




ROOF
LOCAL CONTROL & ALARM PANEL
OPERATING DECK
STAINLESS STEEL LINER
SPENT FUEL POOL
EMERGENCY VENTILATION SYSTEM
VALVE GALLERY
HEAT EXCHANGER & PUMPS
INSTRU-MENTS
EQUIPMENT ROOMS
GROUND LEVEL
EQUIP ROOM
EQUIP ROOM
EQUIP ROOM
PIPE TUNNEL


ROOF
3'
3'
WATER LEVEL TEMP.
INDICATOR/ALARM
WATER LEVEL
MARKER
ROOF
ROOF
SPENT FUEL POOL
LINED WITH
STAINLESS STEEL
EQUIP.
ROOM
EQUIP.
ROOM
EQUIP.
ROOM
EQUIP.
ROOM
EQUIP.
ROOM
EQUIP.
ROOM
12'
SPENT FUEL POOL
COOLING SYSTEM
SUPPLY
RETURN
COOLING
PUMP
10'

HNP Fuel Handling Building
Operating Floor (286' elevation)



CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

PLANT OPERATING MANUAL

VOLUME 1

PART 2

PROCEDURE TYPE: Plant Program (PLP)

NUMBER: PLP-616

TITLE: Fuel Handling Operations

REVISION 10

Table of Contents

1.0 PURPOSE

1. To identify the functions associated with the refueling of the reactor and other fuel handling activities.
2. To establish the responsibilities of organizations involved in fuel handling activities.

2.0 REFERENCES

2.1 Plant Operating Manual Procedures

1. AOP-013, Fuel Handling Accident
2. AP-110, Pre-Job Briefings
3. CM-M0072, IRVH Seismic Tie Rod Installation/Removal
4. CM-M0074, RV Cavity Seal Procedure
5. CM-M0093, RV Lower Internals Package Removal and Installation
6. CM-M0094, IRVH and Upper Internals Removal
7. CM-M0165, RVH and Upper Internals Installation
8. FHP-006, RCCA Change Fixture Operation
9. FHP-009, Control Rod Drive Shaft Unlatching/Relatching Tool Operation
10. FHP-010, Core Mapping Following Fuel Loading
11. FHP-014, Fuel and Insert Shuffle Sequence
12. FHP-015, New RCCA Handling Tool Operation
13. FHP-020, Refueling Operations
14. FHP-024, HNP Spent Fuel Handling Operations
15. FHP-025, HNP Insert Handling Operations
16. FMP-106, New Fuel Receipt Inspection
17. GP-008, Draining the Reactor Coolant System (Mode 5)
18. GP-009, Refueling Cavity Fill, Refueling, and Draindown of the Refueling Cavity (Modes 5-6-5)
19. MMM-011, Cleanliness, Housekeeping, Foreign Material Exclusion (FME) Classification and Work Practices
20. MMM-020, Operation, Testing, Maintenance and Inspection of Cranes and Special Lifting Equipment

2.1 Plant Operating Manual Procedures (continued)

21. PLP-100, Conduct of Infrequently Performed Tests or Evolutions

22. PLP-612, Special Nuclear Material Accountability Plan

23. PLP-625, Harris Nuclear Plant Spent Fuel Management Program

24. PM-I0009, Incore Instrumentation Thimble Insertion, Retraction, Removal and Replacement

25. SPP-0015, Unpacking and Handling of New Fuel Assemblies and New Fuel Shipping Containers

2.2 Final Safety Analysis Report

1. SHNPP FSAR Section 1.8

2. SHNPP FSAR Section 9.1

2.3 Technical Specifications

1. Section 3/4.9

2. Section 6.2.2

2.4 Other

1. Regulatory Guide 1.33

3.0 DEFINITIONS/ABBREVIATIONS

3.1 Refueling Coordinator

A person designated by the Manager - Site Engineering, with concurrence of the General Manager - Harris Plant, who is assigned the overall responsibility for coordination of refueling activities associated with a refueling outage. The refueling coordinator will be responsible to the General Manager - Harris Plant through normal lines of supervision and management. The refueling coordinator will typically be appointed from Engineering. When fuel is being moved in the FHB only, this position is not required but may be filled at the discretion of the General Manager - Harris Plant.

3.2 SRO-Fuel Handling

A person holding a senior reactor operator license or a senior reactor operator-limited to fuel handling license for SHNPP.

3.3 FHB Operator

A person holding a license for SHNPP assigned to supervise fuel handling activities in the Fuel Handling Building.

3.4 Outage Manager

A person assigned responsibility for overall coordination of outage activities.

4.0 GENERAL

Fuel handling at SHNPP requires the coordinated efforts of several plant groups. If fuel is to be handled safely and efficiently, each group must recognize and carry out its assigned responsibilities. Section 5.0 contains a detailed description of group and individual responsibilities. It is imperative that a "team" attitude be maintained at all times and that each individual contribute toward achieving the common goal.

5.0 IMPLEMENTATION

5.1 Program Requirements

R 1. All activities associated with the handling of new or spent fuel shall be performed with strict adherence to the limitations and precautions of established plant procedures (Reference 2.2.1 and 2.4.1).

2. Pre-job briefs shall be conducted prior to all fuel handling and fuel inspection activities. These briefs should be attended by all CP&L and contract personnel involved in the activity. The pre-job briefs should include, as a minimum, discussions on the following:

   ≡ An overview of the activity scope,
   ≡ Group responsibilities and expectations,
   ≡ Applicable regulatory requirements,
   ≡ Timely notice of adverse conditions, and
   ≡ Supervision of contract personnel.

R 3. During refueling, all activities involving the movement of new or spent fuel in Containment shall be under the direct supervision of the SRO-Fuel Handling. The SRO-Fuel Handling shall have no other duties while performing this function (Reference 2.3.2).

4. During refueling, all activities involving the movement of new or spent fuel in the Fuel Handling Building shall be under the direct supervision of the FHB Operator. The FHB Operator reports to the SRO-Fuel Handling while performing fuel handling functions. If fuel is being handled in the FHB only, these activities shall be directly supervised by either the SRO-Fuel Handling or the FHB Operator.

5. The SRO-Fuel Handling and FHB Operator shall have the authority to stop any action deemed unsafe or detrimental to plant equipment or fuel.

6. Bypassing of fuel handling equipment interlocks which is not specified in approved procedures shall require permission of the SRO-Fuel Handling and concurrence of the Superintendent - Shift Operations.

7. Handling of fuel assemblies and inserts within the spent fuel pools at times other than refueling shall be performed or directly supervised by licensed operators for SHNPP.

5.1 Program Requirements (continued)

8. New fuel assemblies may be handled by qualified personnel other than licensed operators until such time that the fuel is placed into the fuel pools or the new fuel elevator.

9. Material control shall be established in accordance with MMM-011 while the reactor vessel head is removed to prevent unauthorized material from entering the vessel. Each individual is responsible for maintaining control of his loose articles in designated areas as stated in MMM-011.

10. The number of personnel entering Containment during refueling shall be limited to facilitate evacuation if conditions require it.

11. Planning and scheduling of refueling activities, in conjunction with the Outage Management group, shall be assigned to a Refueling Coordinator.

NOTE: The status board can be the magnetic/hooked traditional metal tag board with tags, enlarged drawings or computer based video displays.

12. The official fuel assembly status board shall be located in the Control Room. The Control Room status board shall be updated as fuel is moved to always reflect the current location of each fuel assembly.

13. An activity specific Radiation Work Permit (RWP) is required for fuel movement.

14. All units are responsible for maintaining their radiation exposure as low as reasonably achievable (ALARA).

15. Personnel, such as the Fuel Handling Operator, operating the cranes and tools may be contract (non-CP&L) personnel. Operations personnel responsible for directly supervising fuel handling activities shall be licensed operators for SHNPP with the exception of new fuel receipt.

16. Contract personnel performing fuel handling and fuel inspection activities shall be provided continuous utility supervision.

5.2 Scope

The following activities are controlled by this procedure. A list of implementing procedures is contained in Section 5.4.

The scope and responsibilities of the Spent Fuel Management Program are outlined in PLP-625.

5.2.1 New Fuel Receipt

1. Arrival and unloading

2. New fuel handling tool operation and checkout

3. Receipt inspection

4. Storage of acceptable assemblies

5.2.1 New Fuel Receipt (continued)

5. Disposition of unacceptable assemblies
6. Empty shipping container removal
7. New fuel elevator operation and checkout
8. Special Nuclear Material accountability
9. New RCCA handling tool operation
10. Foreign material exclusion

5.2.2 Refueling

1. Reactor coolant system preparation for head removal
2. Spent fuel handling tool operation and checkout
3. PCSR hatch cover closure and testing
4. Reactor vessel head and upper internals removal
5. Refueling cavity flooding and drain down
6. Incore instrumentation thimble retraction, insertion, removal, and replacement
7. Reactor vessel head and upper internals installation
8. Lower internals removal and installation
9. IRVH seismic tie rod removal/installation
10. Control rod drive shaft unlatching tool operation and checkout
11. Fuel shuffle sequence
12. Fuel inspection activities
13. Manipulator crane, FHB bridge crane, and fuel transfer equipment operation and checkout
14. RCCA change fixture operation and checkout
15. Thimble plug handling tool operation and checkout
16. BPRA handling tool operation and checkout
17. Post-load core verification
18. Irradiated RCCA handling tool operation and checkout
19. Foreign material exclusion during refueling
20. Special Nuclear Material accountability

5.2.3 Normal Operation

1. Fuel shuffles in Fuel Handling Building
2. Fuel inspection activities
3. Special Nuclear Material accountability
4. Spent fuel handling tool operation and checkout
5. FHB bridge crane operation and checkout
6. New fuel elevator operation and checkout
7. Thimble plug handling tool operation and checkout
8. BPRA handling tool operation and checkout
9. Irradiated RCCA handling tool operation and checkout
10. Foreign material exclusion

5.3 Responsibilities

5.3.1 Unit Responsibilities

1. Operations

    a. Provide personnel to function as SRO-Fuel Handling.

    b. Provide personnel to function as FHB Operator.

    c. Provide personnel for operation of the manipulator crane, FHB bridge crane, FHB auxiliary crane, fuel transfer system, and fuel handling tools.

    d. Coordinate and conduct pre-job briefs for fuel handling activities.

    e. Initiate AOP-013 in the event of a fuel handling accident.

2. Maintenance

    a. Provide personnel for operation of the polar crane and FHB auxiliary crane.

    b. Provide personnel for movement and opening of new fuel shipping containers.

    c. Provide personnel for movement of new fuel assemblies during new fuel receipt/inspection.

    d. Provide general maintenance support as necessary during fuel handling activities.

5.3.1 Unit Responsibilities (continued)

3. Engineering

   a. Provide personnel with expertise in the areas of reactor engineering and fuel handling system engineering during refueling and fuel inspection activities.

   b. Coordinate and conduct pre-job briefs for fuel inspection activities.

   c. Coordinate and perform receipt inspection of new fuel.

   d. Ensure post-loading core verification is completed.

   e. Perform activities as required by MMM-011.

   f. Maintain accountability and records for new and spent fuel.

4. Environmental and Radiation Control

   a. Provide Health Physics support during fuel handling to ensure that good radiation control practices are followed and assist in maintaining exposure ALARA.

   b. Provide Chemistry support during fuel handling for sampling and analysis as required.

   c. Prepare and maintain procedures for shipment and receipt of radioactive material to ensure compliance with federal regulations.

   d. Provide support as stated in MMM-011.

   e. Coordinate and conduct radiation work permit (RWP) briefs.

5. Security

   a. Maintain access control to Containment, the Fuel Handling Building, and the Protected Area.

5.3.2 Individual Responsibilities

1. Refueling Coordinator

   a. Implement the overall outage plan concerning scheduling of activities directly involved with refueling.

   b. Interface with the Outage Manager and Superintendent - Shift Operations to coordinate the integration of fuel handling activities with the overall outage plan.

   c. Be cognizant of ongoing outage activities affecting the refueling operation.

   d. Maintain cognizance of Technical Specification requirements for fuel movement and assist the Superintendent - Shift Operations with verification of compliance.

5.3.2 Indiviual Responsibilities (continued)

2. Superintendent - Shift Operations

   a. Hold overall responsibility for all fuel handling activities.

   b. Provide input to the Refueling Coordinator concerning fuel handling activities.

   c. Ensure the Control Room fuel assembly status board is maintained and updated.

   d. Maintain cognizance of fuel handling requirements and limitations.

   e. Verify that the required Technical Specifications are satisfied prior to fuel movement and daily during fuel movement.

3. SRO-Fuel Handling

   a. Hold primary responsibility for the safe movement of fuel and core components inside Containment and the Fuel Handling Building.

   b. Supervise or oversees fuel handling activities in Containment.

   c. Supervise the FHB Operator.

   d. Has the authority to stop any action he deems potentially unsafe or detrimental to plant equipment or fuel.

   e. Maintain cognizance of fuel handling requirements and limitations.

4. FHB Operator

   a. Direct fuel handling activities in the Fuel Handling Building.

   b. Report to the SRO-Fuel Handling.

   c. Move, or directly supervise movement of, fuel and core components in the Fuel Handling Building.

   d. Maintain cognizance of fuel handling requirements and limitations.

   e. Has the authority to stop any action he deems potentially unsafe or detrimental to plant equipment or fuel.

5. Fuel Handling Operator

   a. Assist the SRO-Fuel Handling or the FHB Operator as necessary.

5.3.2 Indiviual Responsibilities (continued)

6. Reactor Engineer

   a. Provide Reactor Engineering support as necessary.

   b. Coordinate new fuel receipt inspection.

   c. Maintain cognizance of fuel handling requirements and limitations.

   d. Ensure fuel handling activities are conducted in a manner that prevents unplanned criticality.

   e. Maintain accountability and records for new and spent fuel.

   f. Coordinate fuel inspection activities

7. Fuel Handling Equipment System Engineer

   a. Provide system engineering support as necessary.

   b. Maintain cognizance of fuel handling requirements and limitations.

8. Maintenance

   a. Maintain current qualifications for operation of the polar crane and FHB auxiliary crane.

   b. Maintain cognizance of safe crane operating practices.

   c. Maintain cognizance of safe new fuel shipping container handling practices.

9. Chemistry

   a. Maintain knowledge of sampling methods and analysis techniques.

   b. Perform sampling and analysis as required by procedure or when requested by the SRO-Fuel Handling.

10. Health Physics

   a. Maintain knowledge of radiation control practices and procedures.

   b. Assist in conducting fuel handling activities in a manner that maintains radiation exposure ALARA.

   c. Has the authority to stop any work that violates or could violate Health Physics procedures or that could result in excessive radiation exposure.

11. Containment Coordinator

   a. Coordinate work activities in Containment to minimize interferences and maximize efficiency.

5.4 Procedures

The following plant procedures implement the indicated sections of this PLP.

Procedure		Sections Implemented
MMM-011	Cleaniless, Housekeeping, Foreign Material Exclusion (FME) Classification and Work Practices	4.3, 5.3
CM-M0072	IRVH Seismic Tie Rod Installation/Removal	5.2.2.9
CM-M0074	RV Cavity Seal Procedure	5.2.2.3
CM-M0093	RV Lower Internals Package Removal and Installation	5.2.2.8
CM-M0094	IRVH and Upper Internals Removal	5.2.2.4
CM-M0165	IRVH and Upper Internals Installation	5.2.2.7
FHP-006	RCCA Change Fixture Operation	5.2.2.14
FHP-009	Control Rod Drive Shaft Unlatching/Relatching Tool Operation	5.2.2.10
FHP-010	Core Mapping Following Fuel Loading	5.2.2.17
FHP-014	Fuel and Insert Shuffle Sequence	5.2.2.11, 5.2.3.1
FHP-015	New RCCA Handling Tool Operation	5.2.1.9
FHP-020	Refueling Operations	5.2.1.7, 5.2.2.2, 5.2.2.13, 5.2.3.5, 5.2.3.6
FHP-024	HNP Spent Fuel Handling Operations	5.2.2.2, 5.2.2.12, 5.2.3.4, 5.2.3.5
FHP-025	HNP Insert Handling Operations	5.2.2.15, 5.2.2.16, 5.2.2.18, 5.2.3.7, 5.2.3.8, 5.2.3.9
FMP-106	New Fuel Receipt Inspection	5.2.1.1, 5.2.1.3, 5.2.1.4, 5.2.1.5

5.4 Procedures (continued)

Procedure		Sections Implemented
GP-008	Draining the Reactor Coolant System (Mode 5)	5.2.2.1
GP-009	Refueling Cavity Fill, Refueling, and Draindown of the Refueling Cavity (Modes 5-6-5)	5.2.2.5 and coordination of 5.2.2
MMM-020	Operation, Testing, Maintenance and Inspection of Cranes and Special Lifting Equipment	All lifting by the polar and FHB auxiliary cranes
PLP-612	Special Nuclear Material Accountability Plan	5.2.1.8, 5.2.2.20, 5.2.3.3
PM-I0009	Incore Instrumentation Thimble Insertion, Retraction, Removal and Replacement	5.2.2.6
SPP-0015	Unpacking and Handling of New Fuel Assemblies and New Fuel Shipping Containers	5.2.1.1, 5.2.1.2, 5.2.1.6, 5.2.1.7, 5.2.1.8

6.0 DIAGRAMS/ATTACHMENTS

None Applicable

Revision Summary

Title page and footer Revision 10

5.1.15 removed "or performing"

added Revision Summary

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

PLANT OPERATING MANUAL

VOLUME 3

PART 7

PROCEDURE TYPE: Fuel Handling Procedure

NUMBER: FHP-014

TITLE: Fuel and Insert Shuffle Sequence

MULTIPLE USE
Reference Use Procedure Requirements are Utilized Until the Level of Use Classification is Changed.

NOTE: This procedure has been screened per PLP-100 criteria and determined to be a CASE III procedure. No additional management involvement is required.

Table of Contents

1.0 PURPOSE

1. This procedure is to be used along with the appropriate fuel movement procedure and provides instruction for the preparation and use of the data sheets that document the sequence and tracking of the following:

   - Fuel Assembly and Fuel Insert shuffle in the Storage Pools
   - Fuel movement between the Fuel Handling Building and Containment
   - Fuel Cask unloading
   - Placement of New Fuel into Spent Fuel Pool Storage Racks

2. This procedure also provides guidance and data sheets for Step deviations, temporary fuel storage, and the monitoring of source range counts during core reload.

2.0 REFERENCES

2.1 Plant Operating Manual Procedures

1. PLP-616
2. OP-105
3. FHP-020
4. FHP-024
5. FHP-025
6. FHP-030
7. FHP-040
8. PLP-114

2.2 Technical Specifications

1. Section 3/4.9

2.3 Final Safety Analysis Report

1. 14.2.10

2.4 ANSI 18.7

1. 5.3.4.5

2.5 Other

1. Westinghouse letter dated January 29, 1990 (90CP*-G-0013)
2. ESR 95-00278, Additional BWR Racks for Pool B
3. ESR 95-00633, Monitoring Source Range Counts During Reload

3.0 RESPONSIBILITIES

1. Operations - Overall control and responsibility during all fuel movement

2. Superintendent - Shift Operations or Designee - Approve fuel and insert shuffle data sheets

3. Crane Operators - Verify all fuel assemblies and fuel inserts that are moved, are removed from and placed in the correct assembly locations per the applicable shuffle data sheets

4. Responsible Engineering Supervisor - Provide verified fuel and insert shuffle data (Attachment 1 or 2) and insure the accuracy of all attachments to this procedure

5. Spent Fuel Shipment Director - Provide verified Cask to Storage Fuel Handling Data Sheets (Attachment 3) and ensure the accuracy of all attachments to this procedure

6. Environmental and Radiation Control - Provide health physics coverage, as required, during all fuel or core component movement

4.0 PREREQUISITES

None

5.0 PRECAUTIONS AND LIMITATIONS

1. The index marks for the spent fuel pools and bridge crane are for coarse indexing only. Correct assembly placement should be by visual verification.

2. Due to obstructions encountered during installation, Rack B-B7 is slightly misaligned with adjacent racks. The index marks will be more inaccurate for this Rack.

3. Fuel movement personnel must use the utmost care in ensuring that the fuel assemblies and inserts are properly moved per the applicable data sheets. This is to ensure that the correct assemblies/inserts are moved and that they are placed in their correct storage or core locations.

4. Concurrent verification of the location is required prior to latching a tool to an assembly or insert and prior to lowering an assembly or insert into a storage location.

5. If damage to a spent fuel assembly or the storage rack is observed, stop fuel movement and notify the Superintendent - Shift Operations and Spent Nuclear Fuel Shipment Director/ Reactor Engineer, as appropriate.

6. Spent Fuel Pool Locations B-A1A4 and B-A1K3 are damaged. B-A1A4 Boraflex lining has been damaged. The BNP dummy is stored in this location. Fuel storage is not permitted in either of these locations.

5.0 PRECAUTIONS AND LIMITATIONS (continued)

7. All fuel shuffles involving incore fuel will clear core locations RO8, RO9, PO8, and PO9. These locations will be emptied by the first Steps of a shuffle and filled by the last Steps. This is done so that in the loss of a Permanent Cavity Seal Ring or Nozzle Dam integrity, the operator has a place to lower any fuel latched to the manipulator crane.

8. All fuel movement shall be tracked on the tag boards located in the main control room while maintaining communication with the FHB.

9. When spent fuel is loaded in the four BWR racks installed in locations A-6, B-6, A-7, and D-8, the first rack to be loaded shall be Rack D-8. Start loading spent fuel in Rack D-8 by first loading the six cells directly surrounding cell K-10 which are not blocked by the boral coupon tree. After spent fuel has been loaded in these six cells, rotate the boral coupon tree 90 degrees within cell K-10. Continue loading the remaining two cells surrounding cell K-10. After this has been done, continue loading spent fuel per the spent fuel loading plan.

10. When spent fuel is loaded in the three BWR racks installed in locations B-7, A-8, and B-8, the first rack to be loaded shall be Rack A-8. Start loading spent fuel in Rack A-8 by first loading the six cells directly surrounding cell B-10 which are not blocked by the boral coupon tree. After spent fuel has been loaded in these six cells, rotate the boral coupon tree 90 degrees within cell B-10. Continue loading the remaining two cells surrounding cell B-10. After this has been done, continue loading spent fuel per the spent fuel loading plan.

6.0 TOOLS AND EQUIPMENT

Not Applicable

7.0 PROCEDURE

7.1 Fuel Assembly and Insert Shuffle Data Sheet

NOTE:
- Attachment 1 provides the shuffle sequence and sign-offs for the movement of fuel assemblies or fuel inserts in the FHB. All data is pre-approved before fuel or insert movement begins except for the time and date blocks, the initial blocks, and the comments blocks. A computer generated shuffle sequence is acceptable as long as it is similar and contains all the information in Attachment 1.
- The Steps listed in the left hand column of Attachment 1 should be performed in order. Evolutions such as fuel sipping can be performed out of order, if fuel sipping equipment requires such actions. Each Step is accomplished by performing the fuel or insert shuffle described by the data blocks to the right of each Step number.
- Steps that are the responsibility of the person preparing the shuffle sheet are denoted with an asterisk (*).

* 1. Prior to preparing the shuffle sheet, review the Precautions and Limitations for special considerations of spent fuel locations.

2. The Attachment 1 blocks (from left to right) contain the following information:

* a. STEP # - Shuffle Step number, normally numerical

b. START TIME AND DATE - Time and date Step was started

c. SFP BRIDGE CRANE:

* (1) FUEL SWITCH / OVERLOAD SELECTOR SWITCH - Required positions for the SFP Bridge Crane Fuel Selector and Overload Selector switch for the corresponding fuel assembly, per Attachment 9.

d. FROM:

* (1) ASSEM SER # - Fuel assembly serial number for the assembly to be moved or for the assembly containing the insert to be moved

* (2) LOCATION - Initial storage position that the fuel assembly or insert is being moved from

(3) POSIT - Initial space provided for positioner of crane prior to latching

(4) VERIFY - Initial space provided for person verifying crane position prior to latching

* (5) RCCA - Control Rod serial number if fuel assembly being moved has RCCA insert or if RCCA insert is being moved, otherwise space is blank

* (6) TP - Thimble Plug serial number if fuel assembly being moved has a thimble plug or if a thimble plug is being moved, otherwise space is blank

7.1 Fuel Assembly and Insert Shuffle Data Sheet (continued)

* (7) BPRA - BPRA serial number if fuel assembly being moved has a BPRA insert or if a BPRA is being moved, otherwise space is blank

* (8) OTHER - Serial number of another type of insert (such as a source assembly) that is being moved or is in a fuel assembly that is being moved, otherwise space is blank

e. TO:

* (1) ASSEM SER # - Fuel assembly serial number that an insert is being moved to, space is blank if moving fuel assembly to new location

* (2) LOCAT - Final storage location that the fuel assembly or insert is being moved to

(3) POSIT - Initial space provided for positioner of crane prior to lowering

(4) VERIFY - Initial space provided for person verifying crane position prior to lowering

f. TIME & DATE COMPLETE - Time and date Step was completed

g. INIT - Sign-off with initials for completion of Step

h. COMMENTS Y OR N - COMMENTS Yes or No, listed on Sheet 1 of 2

7.2 Core Offload/Reload Fuel Transfer Data Sheet

NOTE: • Attachment 2 provides the shuffle sequence and sign-offs for the movement of fuel assemblies from the Core to the FHB and then back. All data is pre-approved before fuel movement begins except for the time and date blocks, and the initial blocks. A computer generated shuffle sequence is acceptable as long as it is similar and contains all the information in Attachment 2.

- The Steps listed in the left hand column of Attachment 2 should be performed in order. Evolutions such as fuel sipping can be performed out of order, if fuel sipping equipment requires such actions. Each Step is accomplished by performing the fuel or insert shuffle described by the data blocks to the right of each Step number.

- Steps that are the responsibility of the person preparing the shuffle sheet are denoted with an asterisk (*).

* 1. Prior to preparing the shuffle sheet, review the Precautions and Limitations for special considerations of spent fuel locations.

2. The Attachment 2 blocks (from left to right) contain the following information.

* a. STEP # - Shuffle Step number, normally numerical

b. START TIME & DATE - Time and date Step was started

* c. ASSEMBLY SERIAL NUMBER - Fuel assembly serial number for the fuel assembly to be moved

* d. INSERT - Fuel insert serial number for the insert contained in the fuel assembly to be moved, otherwise space is blank

NOTE: SFP Bridge Crane Fuel switch and the Overload Selector positions are contained in Attachment 9.

* e. LOAD SWITCH POSITION - Required position of the Manipulator Crane Load Selector switch OR the positions for the Fuel switch and the Overload Selector switch on the SFP Bridge Crane for the corresponding fuel assembly

f. FROM:

NOTE: CORE LOCAT can also be the RCCA Change Fixture. RCCA1 is the cell closest to the FHB. RCCA2 is the cell furthest from the FHB.

* (1) CORE LOCAT - Initial position in the core or RCCA Change Fixture that the fuel assembly is being moved from, otherwise space is blank

* (2) SFP LOCAT - Initial Spent Fuel Pool position that the fuel assembly is being moved from, otherwise space is blank

* (3) UPEND - Contains a mark if the fuel assembly being moved is in the upender (Containment or FHB) at the beginning of the Step, otherwise space is blank.

7.2 Core Offload/Reload Fuel Transfer Data Sheet (continued)

(4) POSIT - Initial space provided for positioner of crane prior to latching

(5) VERIFY - Initial space provided for person verifying crane position prior to latching

g. TO:

NOTE: CORE LOCAT can also be the RCCA Change Fixture. RCCA1 is the cell closest to the FHB. RCCA2 is the cell furthest from the FHB.

* (1) CORE LOCAT - Final Core or RCCA Change Fixture position the fuel assembly is being moved to, otherwise space is blank

* (2) SFP LOCAT - Final FHB pool position the fuel assembly is being moved to, otherwise space is blank

* (3) UPEND - Contains a mark if the fuel assembly is being moved to the upender, otherwise space is blank

(4) POSIT - Initial space provided for positioner of crane prior to lowering

(5) VERIFY - Initial space provided for person verifying crane position prior to lowering

h. TIME & DATE COMPLETE - Time and date the Step was completed

i. INIT - Sign-off with initials for completion of Step

j. Attachment 2 Comments section - May contain comments or note indicating visual verification of assembly serial number

7.3 <u>Cask to Storage Fuel Handling Data Sheet</u>

<u>NOTE</u>:
- Attachment 3 provides the shuffle sequence and sign-offs for the movement of spent fuel assemblies from a shipping cask to a storage location in a Spent Fuel Pool. All data is pre-approved before fuel movement begins except for the time and date blocks, the initial blocks, and the comments blocks. A computer generated shuffle sequence is acceptable as long as it is similar and contains all the information in Attachment 3.
- A separate Attachment 3 package is needed for each spent fuel cask.
- The Steps listed in the left hand column of Attachment 3 shall be done in order. Each Step is accomplished by performing the fuel shuffle described by the data blocks to the right of each Step number. An exception to this is for damaged channel fasteners. FHP-030 contains guidance to place the assembly in a designated cell and document in the comments section the final location and date and time completed. This is acceptable for that evolution without an official change to Attachment 3.
- Steps that are the responsibility of the person preparing the shuffle sheet are denoted with an asterisk (*).

* 1. Prior to preparing the shuffle sheet, review the Precautions and Limitations for special considerations of spent fuel locations.

2. The Attachment 3 blocks (from left to right) contain the following information.

* a. STEP # - Shuffle Step number, normally numerical

b. START TIME & DATE - Time and date Step was started

c. SFP BRIDGE CRANE:

* (1) FUEL SWITCH / OVERLOAD SELECTOR SWITCH - Required positions for the SFP Bridge Crane Fuel switch and Overload Selector switch for the corresponding fuel assembly, per Attachment 9.

d. ASSEMBLY:

* (1) SERIAL # - Fuel assembly serial number for the fuel assembly being moved

* (2) INSERT - Fuel insert serial number for the insert contained in the fuel assembly being moved, otherwise space is blank

e. FROM:

* (1) SPENT FUEL CASK LOCATION - Cask position the fuel assembly is being moved from (See Attachment 3 for spent fuel cask loading diagrams), space is blank if moving new fuel

(2) POSIT - Initial space provided for positioner of crane prior to latching

(3) VERIFY - Initial space provided for person verifying crane position prior to latching

7.3 Cask to Storage Fuel Handling Data Sheet (continued)

f. TO:

* (1) SFP LOCATION - Final Spent Fuel Pool position the fuel assembly is being moved to

(2) POSIT - Initial space provided for positioner of crane prior to lowering

(3) VERIFY - Initial space provided for person verifying crane position prior to lowering

g. TIME AND DATE COMPLETE - Time and date the Step was completed

h. INIT - Sign-off with initials for completion of Step

i. COMMENTS Y OR N - COMMENTS Yes or No, listed on Sheet 1 of 2

7.4 Step Deviations and Temporary Storage Locations

7.4.1 General

1. When moving fuel or inserts per Attachments 1, 2, or 3, the shuffle Steps are to be performed in order and all Steps must be completed. A Step Deviation may be used to allow the performance of other Steps and temporary storage of a fuel assembly or insert if the following problems are encountered:

   a. Difficulty placing a fuel assembly into a specific location

   b. Difficulty removing a fuel assembly from a specific location

   c. Difficulty placing a fuel insert into a specific location

   d. Difficulty removing a fuel insert from a specific location

2. Attachment 7 will be used to document Step Deviations and temporary storage locations. Attachment 7 shall not be used for the following:

   a. To change the permanent Core loading arrangement for an approved Attachment 2

   b. To change permanent storage locations for approved Attachments 1, 2, or 3

3. If any changes need to be done to an approved data sheet that are not covered by this section, then these changes must be considered whole page replacements and the following must be done:

   a. Responsible Engineering Supervisor/Designee or Spent Fuel Shipment Director/Designee completes the revised data sheet page.

   b. The revised data sheet page shall be verified by a person designated by the Superintendent - Mechanical Systems.

   c. Superintendent - Shift Operations shall review and approve the revised data sheet page.

   d. The old data sheet page with the error shall be kept, with uncompleted Steps marked N/A, and the revised data sheet page shall be placed in the sequence with previously completed Steps marked N/A.

4. Step Deviation and temporary storage approval shall be obtained from the SRO - Fuel Handling, with concurrence from the Superintendent - Shift Operations and the Reactor Engineer/Shipment Director. This approval and concurrence shall be denoted by the SRO - Fuel Handling initials in the Approval of Temporary Storage block on Attachment 7.

5. To maintain a location to place fuel in the event of loss of refueling cavity integrity, locations R08, R09, P08, and P09 shall not be used as temporary storage locations.

7.4.2 Step Deviation Requiring a Temporary Storage Location

1. SRO - Fuel Handling selects a temporary storage location, per the requirements of Attachment 8, and documents the location in the TEMP STORAGE LOCATION block of Attachment 7.

2. SRO - Fuel Handling obtains Superintendent - Shift Operations and Reactor Engineer/Shipment Director concurrence, for approval of the Step Deviation/temporary storage location.

3. To document that approval has been obtained, the SRO - Fuel Handling fills in APPROVAL OF TEMP STORAGE (Initials) (Time/date) block on Attachment 7.

4. Assign a SHUFFLE DEVIATION NUMBER on Attachment 7.

5. Assign an ASSEMBLY FINAL PLACEMENT SEQUENCE on Attachment 7.

6. On the original Fuel Handling Data Sheet (Attachment 1, 2, or 3) enter the following:

   a. In the affected Step, mark the TIME AND DATE COMPLETE and INIT blocks N/A.

   b. In the affected Step place a Note in the Comments Section stating the SHUFFLE DEVIATION NUMBER from Attachment 7 and the reason for deviation.

   c. In the Comments section of the Step designated by the ASSEMBLY FINAL PLACEMENT SEQUENCE block of Attachment 7, place a note to complete the Step Deviation after this Step is completed.

7. Fill in the following blocks on Attachment 7 with the required data:

   a. SHUFFLE STEP NUMBER

   b. ASSEMBLY SERIAL NUMBER

NOTE: Crane switch positions are contained in Attachment 9.

   c. LOAD SELECTOR SWITCH (if needed)

   d. FINAL FUEL ASSEMBLY LOCATION

8. Align the assembly/insert with the temporary storage location and initial for TEMP STORAGE, POSIT and VERIFY on Attachment 7.

9. Place the assembly/insert in the temporary storage location and complete the TEMP STORAGE, COMPLETE (initials)(time/date) block on Attachment 7.

10. Continue with the unaffected Steps in the original Fuel Handling Data Sheet (Attachment 1, 2, or 3) until the Step that is designated by the ASSEMBLY FINAL PLACEMENT SEQUENCE block of Attachment 7 is complete.

7.4.2 <u>Step Deviation Requiring a Temporary Storage Location</u> (continued)

11. After the Step in the original Fuel Handling Data Sheet (Attachment 1, 2, or 3) that is designated by the ASSEMBLY FINAL PLACEMENT SEQUENCE block of Attachment 7 is complete, perform the following:

    a. Align the crane with the temporary storage location and initial for TO FINAL FUEL STORAGE, RETRIEVAL, POSIT and VERIFY on Attachment 7.

    b. Retrieve the temporarily stored assembly/insert and move to the FINAL FUEL ASSEMBLY LOCATION and initial for TO FINAL FUEL STORAGE, STORAGE, POSIT AND VERIFY.

    c. Place the temporarily stored assembly/insert in the FINAL FUEL ASSEMBLY LOCATION listed on Attachment 7.

    d. Complete the TO FINAL FUEL STORAGE, STORAGE, ASSEMBLY LOCATION (time/date) block on Attachment 7.

    e. Initial for the Step Deviation complete in the INITIALS block of Attachment 7.

    f. Send a copy of Attachment 7 to Responsible Engineer - Reactor Engineering.

7.4.3 <u>Step Deviation With No Temporary Storage Required</u>

1. SRO - Fuel Handling obtains Superintendent - Shift Operations and Reactor Engineer/Shipment Duty Engineer concurrence, for approval of the Step Deviation.

2. To document that approval has been obtained, the SRO - Fuel Handling fills in APPROVAL OF TEMP STORAGE (Initials) (Time/date) block on Attachment 7.

3. Assign a SHUFFLE DEVIATION NUMBER on Attachment 7.

4. Assign an ASSEMBLY FINAL PLACEMENT SEQUENCE on Attachment 7.

5. On the original Fuel Handling Data Sheet (Attachment 1, 2, or 3)

    a. In the affected Step, mark the TIME AND DATE COMPLETE and INIT blocks N/A.

    b. In the affected Step place a Note in the Comments Section stating the Shuffle Deviation Number from Attachment 7 and the reason for deviation.

    c. In the Comments section of the Step designated by the Assembly Final Placement Sequence block of Attachment 7, place a Note to complete the Step Deviation after this Step is completed.

7.4.3 <u>Step Deviation With No Temporary Storage Required</u> (continued)

6. N/A the following blocks on Attachment 7:

   a. TEMP STORAGE LOCATION

   b. TEMP STORAGE POSIT

   c. TEMP STORAGE VERIFY

   d. TEMP STORAGE COMPLETE (initials)(time/date)

7. Fill in the following blocks on Attachment 7 with the required data:

   a. SHUFFLE STEP NUMBER

   b. ASSEMBLY SERIAL NUMBER

<u>NOTE</u>: Crane switch positions are contained in Attachment 9.

   c. LOAD SELECTOR SWITCH (if needed)

   d. FINAL ASSEMBLY LOCATION

8. Continue with the unaffected Steps in the original Fuel Handling Data Sheet (Attachment 1, 2, or 3) until the Step that is designated by the ASSEMBLY FINAL PLACEMENT SEQUENCE block of Attachment 7 is complete.

9. After the Step in the original Fuel Handling Data Sheet (Attachment 1, 2, or 3) that is designated by the Assembly Final Placement Sequence block of Attachment 7 is complete, perform the following:

   a. Align the crane with the affected assembly/insert and initial TO FINAL FUEL STORAGE, RETRIEVAL, POSIT and VERIFY.

   b. Move the affected assembly/insert in the FINAL FUEL ASSEMBLY LOCATION listed on Attachment 7 and initial TO FINAL FUEL, STORAGE, POSIT and VERIFY.

   c. Place the affected assembly/insert in the FINAL FUEL ASSEMBLY LOCATION listed on Attachment 7.

   d. Complete the TO FINAL FUEL STORAGE, STORAGE, ASSEMBLY LOCATION (time/date) block on Attachment 7.

   e. Initial for the Step Deviation complete in the INIT (initials) block of Attachment 7.

   f. Send a copy of Attachment 7 to Responsible Engineer - Reactor Engineering.

7.4.4 <u>Temporary Fuel Storage Data Sheet</u>

1. The Attachment 7 blocks (from left to right) should be filled in with the following information (if applicable):

   a. CORE CYCLE - Core Cycle number - Filled in if Step Deviation is from Attachment 1 or 2.

   b. SHIPMENT NUMBER - Spent fuel or new fuel shipment number - Filled in if Step Deviation is from Attachment 3.

   c. SHUFFLE DEVIATION NUMBER - Assigned deviation number, to be used in the Comments section of the applicable Fuel Handling Data Sheet.

   d. SHUFFLE STEP NUMBER - Step number from the applicable Fuel Handling Data Sheet, which is unable to be completed at this time.

   e. ASSEMBLY SERIAL NUMBER - Serial number of the fuel assembly being moved (If an insert is to be moved, note this in the Comments Section of Attachment 7).

<u>NOTE</u>: Crane switch positions are contained in Attachment 9.

   f. LOAD SELECTOR SWITCH - Position of the Manipulator Crane Load Selector switch OR the positions for the Fuel switch and the Overload Selector switch for the SFP Bridge Crane for the corresponding fuel assembly.

<u>NOTE</u>: Temporary storage location selected must comply with temporary storage criteria as stated in Reference 2.1.0.03 and 2.5.0.01, if applicable.

   g. TEMP STORAGE LOCATION - Location assembly/insert is to be temporarily stored in.

   h. FINAL FUEL ASSEMBLY LOCATION - Final location assembly/insert will be placed in per the applicable Fuel Handling Data Sheet.

   i. ASSEMBLY FINAL PLACEMENT SEQUENCE - Step number, from the applicable Fuel Handling Data Sheet, that must be completed before the temporarily stored assembly/insert can be placed in its Final Fuel Assembly Location.

   j. APPROVAL OF TEMP STORAGE (initials)(time/date) - Initials of SRO - Fuel Handling and time/date indicating approval of temporary storage.

   k. TEMP STORAGE:

      (1) POSIT - Initial space provided for positioner of crane prior to lowering.

      (2) VERIFY - Initial space provided for person verifying crane position prior to lowering.

      (3) COMPLETE (initials time/date) - Initials, time, and date by SRO - Fuel Handling indicating the assembly/insert is placed in the Temporary Storage Location.

7.4.4 Temporary Fuel Storage Data Sheet (continued)

l. TO FINAL FUEL STORAGE:

(1) RETRIEVAL:

(a) POSIT - Initial space provided for positioner of crane position prior to retrieving from the TEMP STORAGE LOCATION.

(b) VERIFY - Initial space provided for verifier of crane position prior to retrieving from the TEMP STORAGE LOCATION.

(2) STORAGE:

(a) POSIT - Initial space provided for positioner of crane prior to lowering into FINAL FUEL ASSEMBLY LOCATION. If assembly/insert is to be moved to another temporary storage location, mark this block N/A.

(b) VERIFY - Initial space provided for person verifying crane position prior to lowering into FINAL FUEL ASSEMBLY LOCATION. If assembly/insert is to be moved to another temporary storage location, mark this block N/A.

(c) ASSEMBLY LOCATION (time/date) - Time and date assembly/insert placed in Final Fuel Assembly Location. If assembly/insert is to be moved to another temporary storage location, mark this block N/A and note a new shuffle deviation number in the Comment section.

m. INIT - SRO-Fuel Handling initials block, indicating assembly/insert is seated in FINAL FUEL ASSEMBLY LOCATION. If assembly/insert is to be moved to another temporary storage location, mark this block N/A.

n. COMMENT Y OR N - COMMENT Yes or No, listed on Sheet 1 of 2, Reason for the Step Deviation/temporary storage and comments.

7.5 Monitoring Source Range Data During Core Reload
(Reference ESR 95-00633)

CAUTION

When changing the Audio Count Rate Scaler Timer Multiplier Switch position or the Audio Count Rate Source Range Channel Selector, ensure personnel in containment are notified that a change in count level may be detected.

---

1. If ERFIS is not available, log NI-31 and NI-32 to compare source range response after each fuel assembly is loaded into the core and skip to the NOTE prior to Step 7.5.0.04. Send the log to Reactor Engineering at the completion of reload.

2. Using ERFIS computer points ANM0106, ANM0107 and ANM9107, make a plot with a 10 second update rate to monitor source range response.

3. Using ERFIS computer points ANM0106, ANM0107 and ANM9107, make a GTLOG with a 15 minute update rate to provide Reactor Engineering count rate data.

NOTE: During refueling the source range counts are excepted to increase and reach a new plateau between fuel assembly insertions. Counts should reach a stable level and each successive reading should be within 250 counts of the previous reading and not continuously increasing.

4. Monitor source range counts for stabilization after each fuel assembly is placed in the core for the first eighteen fuel assemblies.

NOTE: A doubling in source range counts should not occur from the nineteenth fuel assembly to completion of the reload.

5. Monitor the source range counts to verify the count increase is gradual and less than the previous fuel assembly.

6. If an unexpected increase is noticed in the source range counts, contact the Refueling Coordinator AND Reactor Engineering.

7. If the unexpected increase in counts cannot be resolved, perform the following:

   a. Request an additional boron sample.

   b. Verify the required refueling boron concentration is met.

   c. Verify the source range detectors and associated instrumentation are responding as expected.

   d. If the unexpected increase in counts continues, and the source range instrumentation and the boron concentration are satisfactory, it may be acceptable to continue to reload the core. Reactor Engineering and Operations should monitor the source range response during each fuel assembly insertion until the anomaly is resolved.

8.0 DIAGRAMS/ATTACHMENTS

CAUTION

Attachments 1, 2, and 3 are CONTINUOUS USE once they have been prepared and approved.

Attachment 1 - Fuel Assembly and Insert Shuffle Data Sheet

Attachment 2 - Core Offload/Reload Fuel Transfer Data Sheet

Attachment 3 - Cask to Storage Fuel Handling Data Sheet

CAUTION

The remaining Attachments are REFERENCE USE.

Attachment 4 - Spent Fuel Pool A Map

Attachment 5 - Spent Fuel Pool B Map

Attachment 6 - Inspection Pit New Fuel Dry Storage Map

Attachment 7 - Temporary Fuel Storage Data Sheet

Attachment 8 - Guidelines for Temporary Storage Locations

Attachment 9 - Crane Switch Positions

Fuel Assembly and Insert Shuffle Data Sheet

Data Sheet prepared by: ______________________________
Superintendent - Mechanical Systems or Designee/Date

Verified correct by: ______________________________
Designated by Superintendent - Mechanical Systems/Date

Data Sheet approved by: ______________________________
Superintendent - Shift Operations/Date

Fuel Movement Performed by:

Initials	Name (Print)	Initials	Name (Print)

COMMENTS (List Step numbers where applicable)

Send a copy of this completed attachment to Responsible Engineer - Reactor Engineering

Attachment Satisfactory Completed:

______________________________ Unit SCO

______________ Date

After receiving the final review signature, this FHP Attachment becomes a QA RECORD and should be submitted to Document Services.

Fuel Assembly And Insert Shuffle Data Sheet

CORE CYCLE:________ Page ________ of ________

STEP #	START TIME AND DATE	SFP BRIDGE CRANE		FROM						TO		TIME & DATE COMPLETE	INIT	COMMENT Y OR N
				ASSEM SER #	POSIT	INSERT				ASSEM SER #	POSIT			
		FUEL SWITCH	OVERLOAD SELECTOR SWITCH	LOCATION	VERIFY	RCCA	TP	BPRA	OTHER	LOCATION	VERIFY			

Prepared By: ________________ Verified By: ________________ S-SO Approval: ________________

Core Offload/Reload Fuel Transfer Data Sheet

Data Sheet prepared by: ______________________________
Superintendent - Mechanical Systems or Designee/Date

Verified correct by: ______________________________
Designated by Superintendent - Mechanical Systems/Date

Data Sheet approved by: ______________________________
Superintendent - Shift Operations/Date

Fuel Movement Performed by:

Initials	Name (Print)	Initials	Name (Print)

Core Offload/Reload Fuel Transfer Data Sheet

COMMENTS (List Step numbers where applicable)

Send a copy of this completed attachment to Responsible Engineer - Reactor Engineering

Attachment Satisfactory Completed:

Unit SCO

Date

After receiving the final review signature, this FHP Attachment becomes a QA RECORD and should be submitted to Document Services.

Core Offload/Reload Fuel Transfer Data Sheet

CORE CYCLE:________ Page________ of ________

STEP #	START TIME & DATE	ASSEMBLY SERIAL NUMBER	INSERT	LOAD SWITCH POSITION (NOTE 1)	FROM				TO				TIME & DATE COMPLETE	INIT
					CORE LOCAT	SFP LOCAT	UPEND	POSIT VERIFY	CORE LOCAT	SFP LOCAT	UPEND	POSIT VERIFY		

NOTE 1: Verify the Load Switches are placed on the corresponding number for the respective crane.

COMMENTS:____

____

____

Prepared By: ________________ Verified By: ________________ S-SO Approval: ________________

Cask to Storage Fuel Handling Data Sheet

Shipment No. ________ Cask No. ________

Data Sheet prepared by: ______________________________
Superintendent - Radiation Protection or Designee /Date

Verified correct by: ______________________________
Designated by Superintendent - Radiation Protection /Date

Data Sheet approved by: ______________________________
Superintendent - Shift Operations /Date

Fuel Movement Performed by:

Initials	Name (Print)	Initials	Name (Print)

COMMENTS (List Step numbers where applicable)

____
____
____
____
____
____
____
____
____
____

Send a copy of this completed attachment to Responsible Engineer -Reactor Engineering

Attachment Satisfactory Completed:

______________________________ ________
Unit SCO Date

After receiving the final review signature, this FHP Attachment becomes a QA RECORD and should be submitted to Document Services.

Cask to Storage Fuel Handling Data Sheet

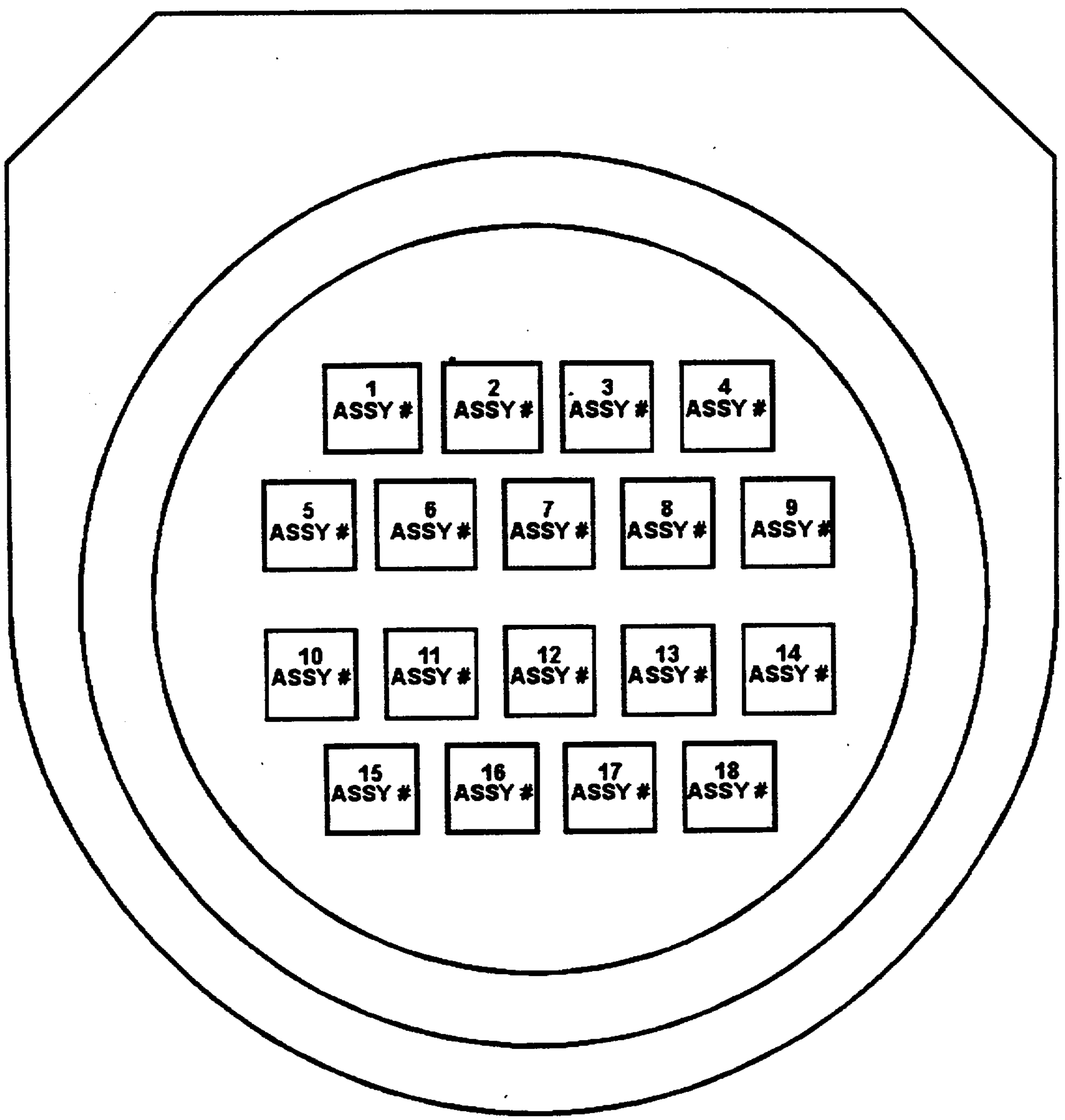
Shipment No. ________ Cask No. ________ Page ________ of ________

STEP #	START TIME AND DATE	SFP BRIDGE CRANE		ASSEMBLY	FROM		TO		TIME AND DATE COMPLETE	INIT	COMMENTS Y OR N
		FUEL SWITCH	OVERLOAD SELECTOR SWITCH	SERIAL # / INSERT #	SPENT FUEL CASK LOCATION	POSIT / VERIFY	SFP LOCATION	POSIT / VERIFY			

Prepared by: ____________________ Verified by: ____________________ S-SO Approval: ____________________

Cask to Storage Fuel Handling Data Sheet

BWR Configuration   Shipment No: __________   Cask No: ________

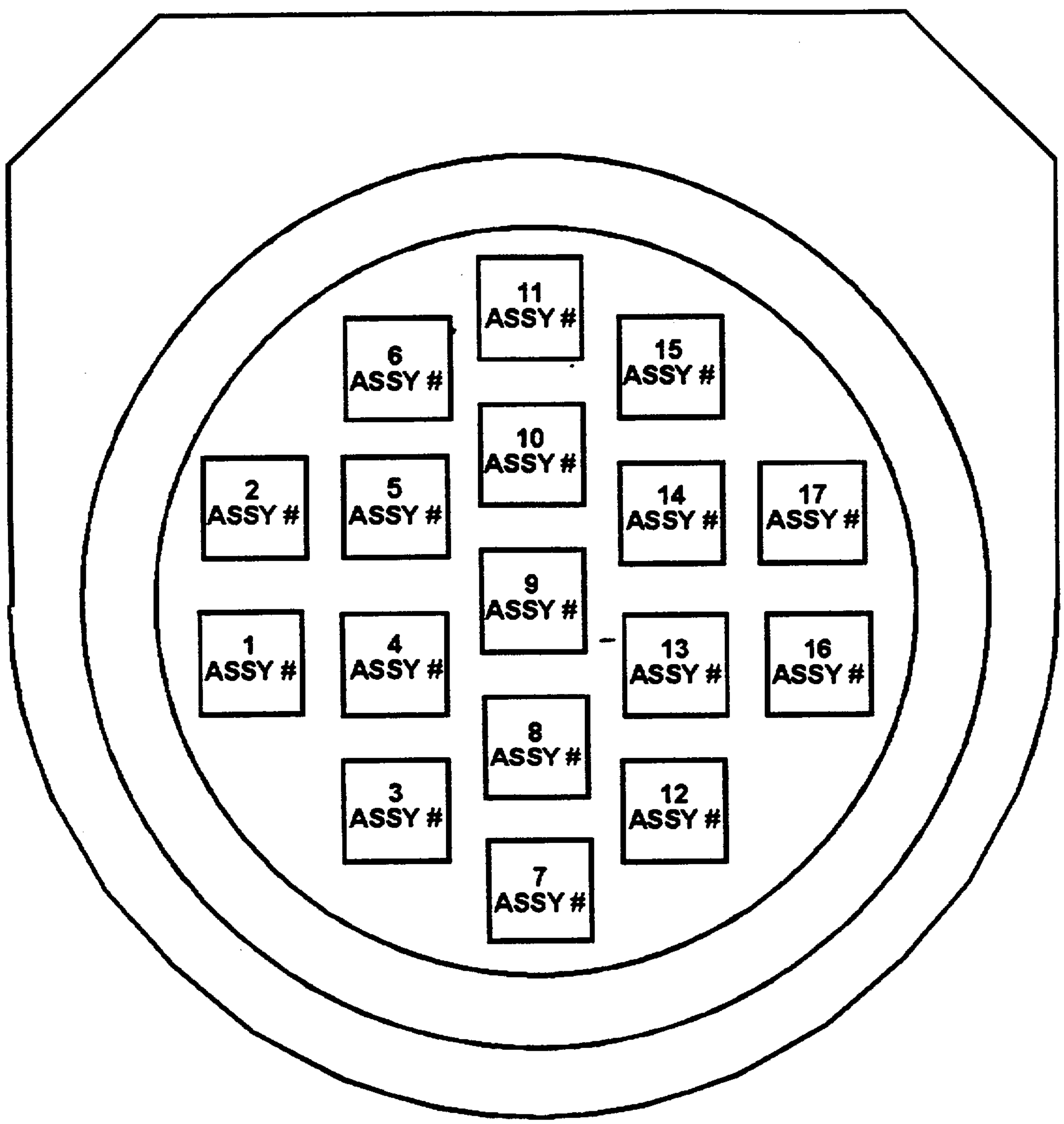


Cask to Storage Fuel Handling Data Sheet

BWR Configuration
Channeled Fuel

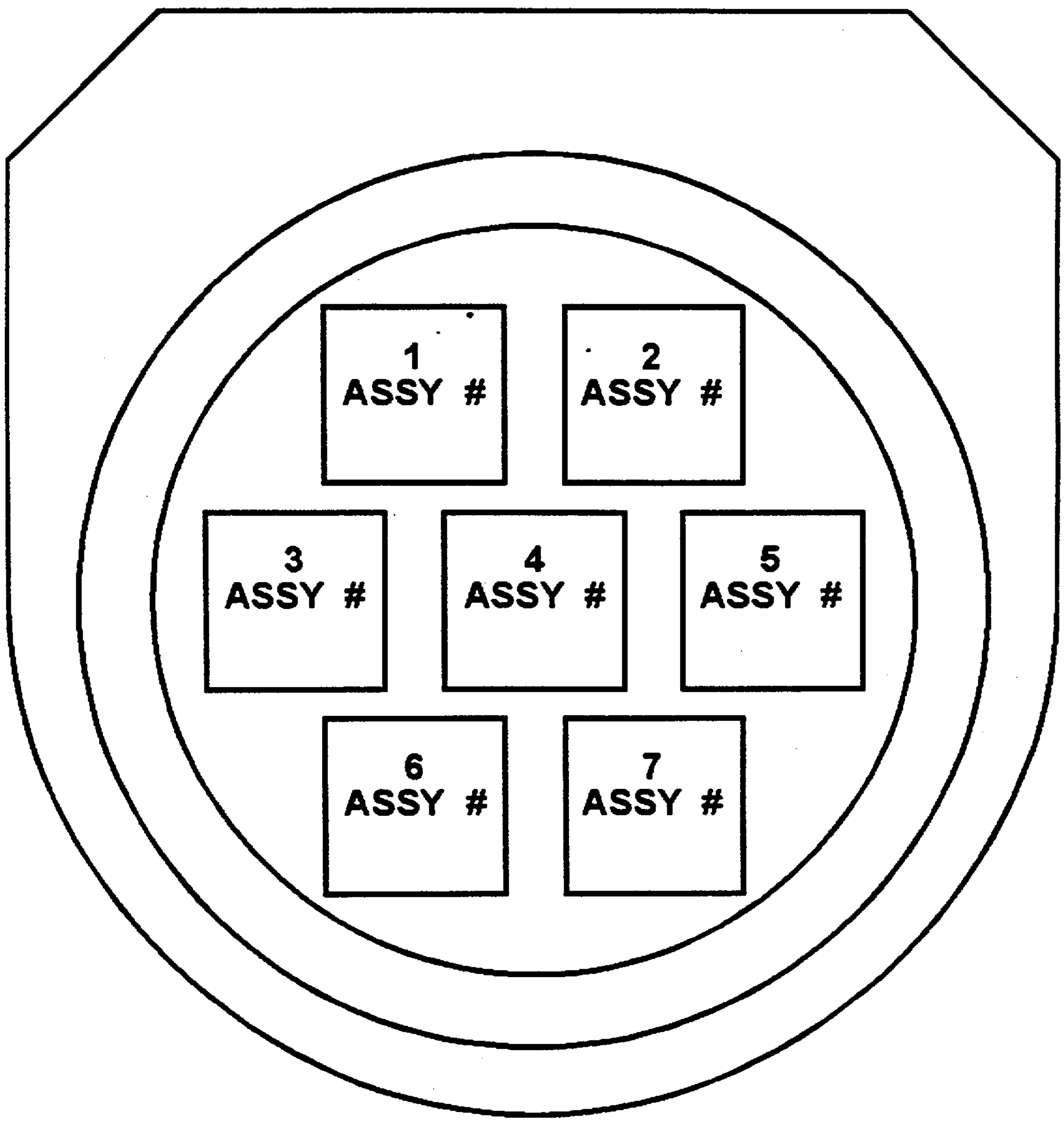
Shipment No: ________

Cask No: ________



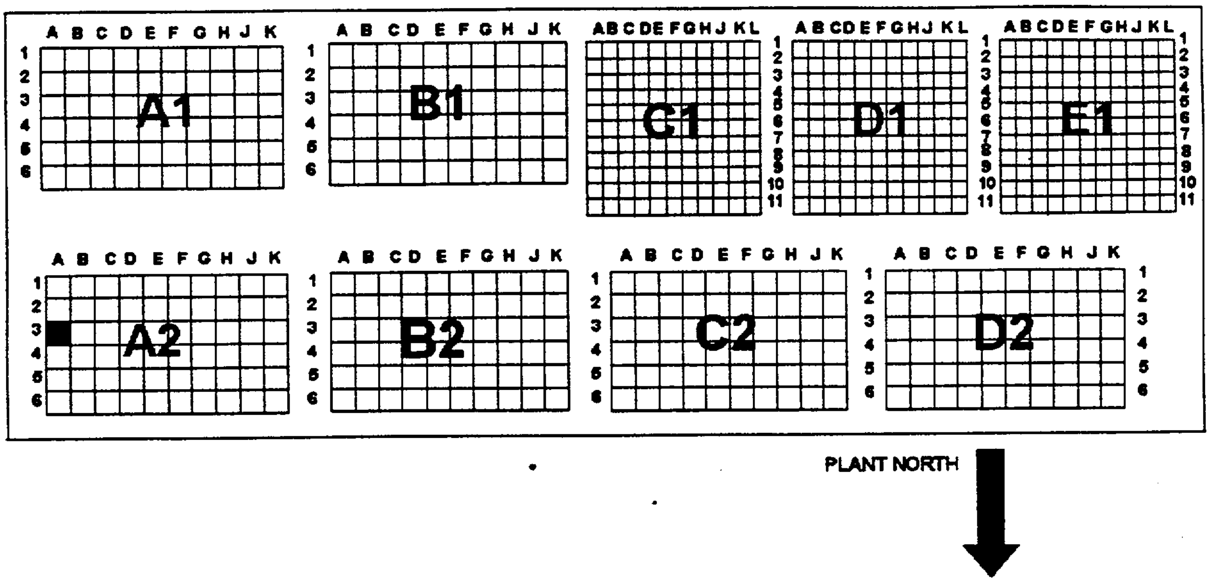
Cask to Storage Fuel Handling Data Sheet

PWR Configuration Shipment No: __________ Cask No: ________



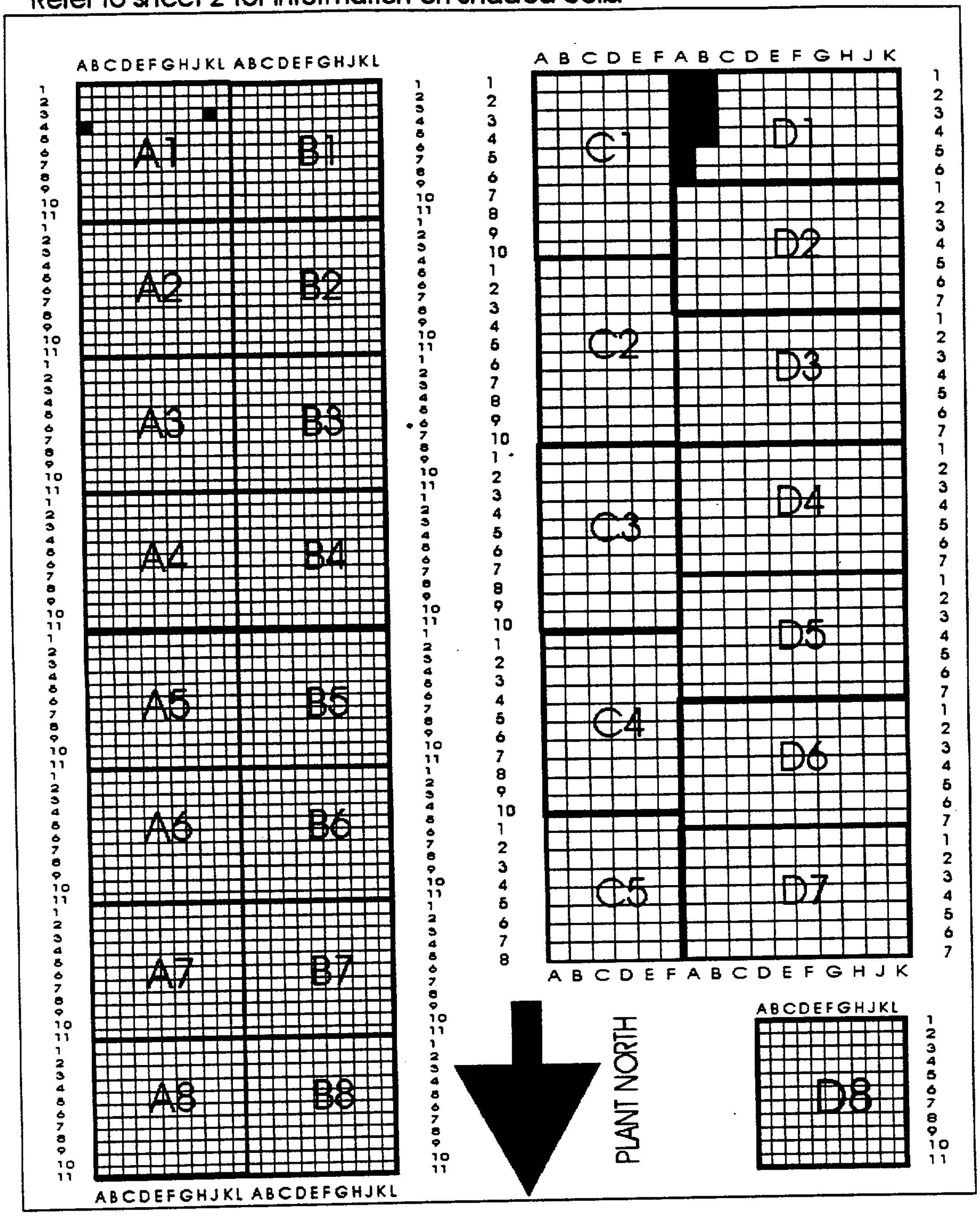
<u>Spent Fuel Pool A Map</u>
(New Fuel Pool Unit 1)

NOTE: TRASH BASKET A2A3



Spent Fuel Pool B Map
(Spent Fuel Pool Unit 1)

Refer to sheet 2 for information on shaded cells.

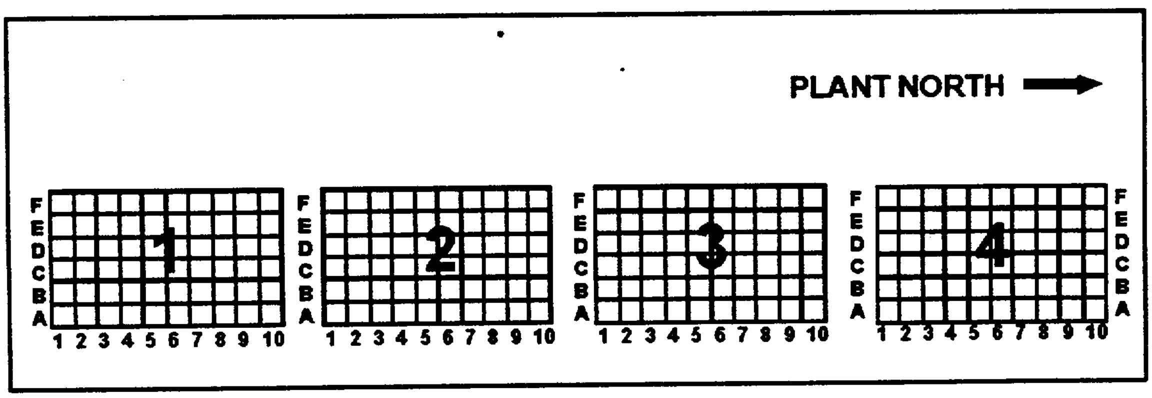


Spent Fuel Pool B Map
(Spent Fuel Pool Unit 1)

Explanation for Shaded Cells

1. DAMAGED CELLS: **A1K3**, **A1A4**
2. HNP MOCK: **D1A1**
3. HBR DUMMY: **D1A2**
4. TRASH BASKETS: **D1A3**, **D1A4**, **D1B3**, and **D1B4**
5. VENDOR SUPPLIED TRASH BASKET: **D1B2**
6. SPECIMEN BASKET: **D1A5**
7. HNP DUMMY: **D1A6**
8. BNP DUMMY: **A1A4**
9. FAILED FUEL ROD STORAGE BASKET: **D1B1**

Inspection Pit New Fuel Dry Storage Map



Temporary Fuel Storage Data Sheet

Core Cycle ________ Shipment No. ________

Fuel Movement Performed by:

Initials	Name (Print)	Initials	Name (Print)

COMMENTS:

Send a copy of this completed attachment to Responsible Engineer - Reactor Engineering

REVIEWED BY: ______________________________ ______________
Unit SCO Date

REVIEWED BY: ______________________________ ______________
Responsible Engineer - Reactor Engineering Date

After receiving the final review signature, this FHP Attachment becomes a QA RECORD and should be submitted to Document Services.

Temporary Fuel Storage Data Sheet

Core Cycle ________ Shipment Number ________ Page ________ of ________

								TEMP STORAGE		TO FINAL FUEL STORAGE				
										RETRIEVAL	STORAGE			
SHUFFLE DEVIATION NUMBER	SHUFFLE STEP NUMBER	ASSEMBLY SERIAL NUMBER	LOAD SELECTOR SWITCH	TEMP STORAGE LOCATION	FINAL FUEL ASSEMBLY LOCATION	ASSEMBLY FINAL PLACEMENT SEQUENCE	APPROVAL OF TEMP STORAGE (initials time/date)	POSIT VERIFY	COMPLETE (initials) (time/date)	POSIT VERIFY	ASSEMBLY LOCATION (time/date)	POSIT VERIFY	INIT	COMMENT Y OR N

NOTE: No off-site spent fuel shall be stored in the New Fuel Pool as a result of a sequence change or new temporary storage location. This will minimize crud accumulation in the New Fuel Pool.

Send copy of this attachment to Responsible Engineer - Reactor Engineering

Guidelines for Temporary Storage Locations

NOTE: Source assemblies refers to twice-burned fuel assemblies or fuel assemblies containing source inserts used in baffle locations adjacent to the source range detectors to provide baseline counts during core reload.

1. Source assemblies should be loaded in the baffle locations closest to the source range detector before loading any other assemblies. Nuclear coupling must be maintained between the source assembly and the detector.

2. Nuclear coupling must be maintained between the source assemblies and any assembly in the core which is face-adjacent to any other assembly. Temporary storage of an assembly or assemblies in baffle locations is permissible if no stored assembly is face-adjacent to any other stored assembly and there is at least one open location between the core assemblies and all inward faces and corners of the stored assembly.

3. After establishment of the baseline count rate from both source range detectors, assemblies that are added should bridge the core. Assemblies that are added after the bridge is formed should be added such that assemblies are coupled to the bridge until the final configuration is reached.

4. A fuel assembly shall be preferentially placed in its location in the final fuel loading configuration except when temporarily stored along the baffle or when used to construct temporary "boxes" which may be required to load a difficult assembly. The forming of temporary boxes has the potential for fueling configurations which may be more reactive than the final analyzed core configuration. To preclude these configurations, the following criteria are to be followed:

   a. The two methods below are preferable to other alternatives listed in these guidelines for forming boxes to load difficult assemblies:

      (1) Use of the fuel assembly loading guide.

      (2) Use of boxing configurations in which all the assemblies within the box are in their final core locations.

   b. If the methods in the Step 4a above are not practical, dummy fuel assemblies can be used as temporary assemblies to form boxes. There are no restrictions on the number or location of the dummy fuel assemblies within the box. Other fuel assemblies making up the box must either be assemblies in their final locations or assemblies meeting Step 4c below.

   c. If Step 4a and Step 4b above are not practical, twice-burned fuel assemblies containing control rods can be used as temporary assemblies to form boxes. There are no restrictions on the number or location of these twice-burned assemblies within the box. Other fuel assemblies making up the box must either be assemblies in their final locations or dummy fuel assemblies.

5. At all times the source range counts must be monitored for any "unexpected" increase (decrease) to preclude an inadvertent criticality. Furthermore, boron concentration analysis of refueling water shall be performed per Technical Specification requirements to assure no inadvertent dilution effects.

Crane Switch Positions

Type of Fuel	SFP Bridge Crane		Manipulator Crane
	FUEL SWITCH	OVERLOAD SELECTOR SWITCH	LOAD SELECTOR SWITCH
HNP LOPAR With RCCA	1	NORM	4
HNP LOPAR Without RCCA	2	NORM	2
HNP SIEMENS With RCCA	1	NORM	4
HNP SIEMENS Without RCCA	2	NORM	2
HNP VANTAGE 5 With RCCA	2	NORM	3
HNP VANTAGE 5 Without RCCA	3	NORM	1
RNP With RCCA	2	MAX	
RNP Without RCCA	3	MAX	
BSEP	4	NORM	

Revision Summary

General

This change is the result of CR 9900190, which requested that the steps for preparing the shuffle sheets be spelled out. It also asked that repetitive Precautions be deleted from this procedure. Load cell settings for the manipulator crane were added to Attachment 9 per request and input from David Baksa. This revision deletes reference to FSAR and ANSI commitments, and just lists the FSAR/ANSI Section originally referenced by the commitment. Finally, the procedure was reclassified as a multiple use procedure, with the majority of the procedure being reference use. Only the approved shuffle sheets are now continuous use.

Description of Changes

Page	Section	Change Description
All		Updated revision level.
1	Cover	Changed use classification to Multiple use, with the procedure starting as a reference use procedure.
3	1.0	Added the words "preparation and" dealing with the shuffle sheets, to more correctly state the purpose of the procedure.
	2.3, 2.4	Removed the word commitment. Deleted the piece of the FSAR and ANSI Section that denoted the commitment number.
4	3.0	Separated original Step 1 into 3 individual Steps. The crane operator may not be from Operations. Added new Step 5 for the Spent Fuel Shipment Director for spent fuel casks shuffle sheets (Attachment 3). Clarified that the Engineering Supervisor was responsible only for Attachment 1 and 2.
	4.0	Deleted prerequisites. They were of no extra value, since this procedure is used both in the preparation and implementation of the shuffle sheets.
5	5.0	Deleted Precautions that were fully covered in GP-009, FHP-020, or specific fuel handling procedures.
6 - 11	7.1, 7.2, 7.3	Added new note that steps that were the responsibility of the preparer were denoted by an asterisk. Added new step 1 to review the precautions for special considerations. Added asterisks to those steps that were done by the attachment preparer.
16	7.4.4	Changed Note before Step 1.f to reflect the fact that the manipulator switch positions are also on Attachment 9.
19	8.0	Added Caution that approved Attachment 1, 2, or 3 were classified as continuous use. Renamed Attachment 9 to reflect the manipulator load switches were now included.
37	Attachment 9	Added Manipulator load switch settings.