FPL

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L-2002-214
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Addition of Cask Area Spent Fuel Storage Racks

In accordance with the provisions of 10 CFR 50.90, Florida Power and Light Company (FPL) requests that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4 be amended to incorporate the attached Technical Specification (TS) revisions. The proposed amendments would revise TS Section 5.6, Design Features – Fuel Storage to include the design of a new cask area spent fuel storage rack for each unit and increase each unit's spent fuel storage capacity by combining the cask area rack and existing SFP storage rack capacities.

Enclosure 1 is an evaluation of the proposed changes. Enclosure 2 is the Determination of No Significant Hazards Consideration. Enclosure 3 is a statement of Environmental Consideration. Enclosure 4 contains the affected Technical Specifications pages marked-up to show the proposed changes. Enclosure 5 contains a list of regulatory commitments resulting from the proposed change. Enclosure 6 contains the proposed Technical Specification clean pages.

The evaluation of proposed changes (Enclosure 1) contains proprietary and non-proprietary versions of an attached license amendment report for the new cask area racks prepared by the rack vendor. The affidavit required by 10 CFR 2.790 is also attached to the proprietary version of the report. FPL requests that the proprietary version be withheld from public viewing.

The proposed license amendments have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91(b)(1), a copy of these proposed license amendments is being forwarded to the State Designee for the State of Florida.

Regarding the proposed schedule for this amendment, it is requested that issuance be no later than December 2, 2003. However, it is anticipated that frequent meetings with NRC staff can improve that schedule. Although physical installation of the cask area racks is not planned until June 2004, the requested amendments date is necessary to provide time for pursuing another spent fuel storage alternative, should the submittal not be approved by that date. Based on a need for additional storage at Turkey Point Unit 3 prior to the loss of full core reserve in 2007, it is important to have a definitive storage solution licensed in 2003.

A001

These proposed license amendments may require noticing with respect to the hybrid hearing procedures under 10 CFR 2.1107, because this may involve a proceeding on an application for a license amendment falling within the scope of section 134 of the Nuclear Waste Policy Act of 1982.

Should there be any questions on this request, please contact Walter Parker at 305 246-6632.

Very truly yours,

J. P. McElwain
Vice President Turkey Point Plant

YK
Enclosures

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Florida Department of Health

L-2002-214

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Addition of Cask Area Spent Fuel Storage Racks

STATE OF FLORIDA)
)ss.
COUNTY OF MIAMI-DADE)

J. P. McElwain being first duly sworn, deposes and says:

That he is Vice President, Turkey Point Plant, of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

J. P. McElwain

Subscribed and sworn to before me this

26 day of November, 2002.

Scott Cousino
Name of Notary Public (Type or Print)

J. P. McElwain is personally known to me.

SCOTT COUSINO
NOTARY PUBLIC-STATE OF FLORIDA
COMMISSION #DD030974
MY COMMISSION EXPIRES JULY 16, 2005

ENCLOSURES

Enclosure 1	Proposed License Amendments Application
	(Including Appendix 1, License Amendment Report)
Enclosure 2	No Significant Hazards Considerations
Enclosure 3	Environmental Consideration
Enclosure 4	Proposed Mark-Up of Affected Technical Specification Pages
Enclosure 5	List of Regulatory Commitments
Enclosure 6	Proposed Technical Specification Clean Pages

ENCLOSURE 1

PROPOSED LICENSE AMENDMENTS APPLICATION

PROPOSED LICENSE AMENDMENT
ADDITION OF CASK AREA SPENT FUEL STORAGE RACKS

TABLE OF CONTENTS

		Page
1.0	BACKGROUND INFORMATION	3
2.0	DESCRIPTION OF PROPOSED CHANGE	5
3.0	BASIS/JUSTIFICATION FOR PROPOSED CHANGE	
3.1	Overview	6
3.2	Criticality Considerations	7
3.3	Thermal-Hydraulic Considerations	10
3.4	Seismic and Structural Evaluation	22
3.5	Handling of Heavy Loads	28
3.6	Handling Fuel Assemblies in the Cask Area	29
3.7	Radiological Considerations	30
3.8	Other Issues	34
4.0	CONCLUSION	36

APPENDIX 1: Holtec License Amendment Report HI-2022931
Including:

Holtec Affidavit	4 pages
Proprietary version	205 pages
Non-Proprietary version	205 pages

1.0 BACKGROUND INFORMATION

All spent fuel at Turkey Point (PTN) Units 3 and 4 is stored underwater in the respective unit's spent fuel pool (SFP) located in the Auxiliary Building. The SFP is also referred to as the "spent fuel pit" in some licensing documents. The SFP for each unit is currently licensed to store a total of 1404 fuel assemblies in high-density racks using Boraflex™ neutron absorbing panels. Based on this capacity, current spent fuel inventory, and projected discharges, the Unit 3 full core reserve (FCR) will expire in the year 2007. Similarly, Unit 4 will lose FCR capability by 2009.

To extend FCR capability beyond the above dates, these proposed license amendments (PLA) intend to install a freestanding spent fuel storage rack module in the cask area of each unit's SFP. The cask area is a vacant section of the SFP located near the center of the east pool wall. The floor elevation of the cask area and surrounding SFP are the same, and the new rack will be at the same elevation as the existing SFP storage racks.

The additional storage capacity provided by each unit's cask area rack will be used to temporarily store spent fuel during refueling outage fuel offloads and during non-outage fuel shuffles. Because the cask areas will eventually be needed for loading fuel into transfer casks, the cask area racks will be removed, cleaned, and stored at an alternate location in the radiation controlled area prior to any spent fuel cask loading operations.

The new racks will use Boral™ as the neutron absorbing poison. The racks are designed for Region I storage of 131 assemblies of either fresh fuel or spent fuel, regardless of its burnup history, bringing the total storage capacity of each unit up to 1535 assemblies. Each rack is configured in an 11x12 cell array, with one corner cell eliminated to provide a storage location for the fuel handling tool. The additional storage capacity provided by the cask area racks is expected to extend FCR capability by at least two fuel cycles for each unit.

Shared Thermal-Hydraulic Analysis with another license amendment request

The thermal-hydraulic (T-H) analysis described in Section 3.3 of this PLA is also included in another PLA recently submitted under FPL letter L-2002-151, dated October 21, 2002. The recently submitted PLA proposes to reduce the minimum time required for reactor subcriticality prior to removing irradiated fuel from the reactor vessel from 100 hours to 72 hours. The same T-H analysis applies to both PLAs because the analysis evaluates the additional decay heat load imposed on the SFP by the new cask area rack storage, as well as the SFP decay heat load impact of reducing the core offload start time to 72 hours after shutdown. In all other respects, the two amendment requests are independent of one another.

Precedent Licensing Actions

Similar license amendments at other plants have increased spent fuel storage by adding storage racks in confined spent fuel pool areas. In July 1998, Waterford 3 received a license amendment to increase SFP storage capacity by adding storage racks in the cask storage area and refueling canal. Similarly, Kewaunee received approval in January 2001 to increase allowable spent fuel storage capacity by adding a new storage rack in a fuel transfer canal pool. Both plants installed new racks manufactured by the same vendor as the Turkey Point racks.

In addition, a similar license amendment has recently been submitted in FPL letter L-2002-187, dated October 23, 2002 for the St. Lucie plant to add a cask pit rack to each unit. The new St. Lucie racks perform a similar function to the Turkey Point racks proposed herein.

2.0 DESCRIPTION OF PROPOSED CHANGES

FPL proposes to modify Turkey Point Units 3 and 4 Technical Specification Section 5.6, Design Features - Fuel Storage. Section 5.6 will be revised to include the new cask area rack design and reflect the increased spent fuel storage capacity when the cask area rack and existing SFP rack capacities are combined. A markup of the proposed changes is shown in Enclosure 4.

The following Technical Specification changes are proposed (added words in **bold**):

a. Section 5.6.1.1a is revised to read: "k_{eff} equivalent to less than 1.0 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in ~~WCAP-14416-P~~ **UFSAR Appendix 14D**."

b. Section 5.6.1.1b is revised to read: "A k_{eff} equivalent to less than or equal to 0.95 when flooded with **water** borated to 650 ppm ~~water~~, which includes a conservative allowance for uncertainties as described in ~~WCAP-14416-P~~ **UFSAR Appendix 14D**."

c. Section 5.6.1.1c is revised to read: "A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for **the** two region **spent** fuel **pool** storage racks. **A nominal 10.1 inch center-to-center distance in the east-west direction and a nominal 10.7 inch center-to-center distance in the north-south direction for the Region I cask area storage rack.**"

d. Section 5.6.3, Capacity, is revised to read: "The spent fuel ~~storage~~ pool **storage racks** ~~is~~ **are** designed and shall be maintained with a storage capacity limited to no more than 1404 **fuel assemblies** in two region storage racks, **and the cask area storage rack is designed and shall be maintained with a storage capacity limited to no more than 131 fuel assemblies. The total spent fuel pool storage capacity is limited to no more than 1535 fuel assemblies.**"

Regarding items a and b above, a description of the new cask area rack criticality analysis and associated uncertainties will be included in UFSAR Appendix 14D, and will provide a similar level-of-detail as the criticality analysis discussion for the existing SFP racks.

3.0 BASIS/JUSTIFICATION FOR PROPOSED CHANGES

3.1 OVERVIEW

The basis for requesting the proposed changes to Section 5.6 of the Turkey Point Technical Specifications (TS) is to extend the full core reserve capability of each Turkey Point unit by at least two fuel cycles, by increasing the spent fuel wet storage capacity. Extending FCR capability will provide FPL additional time to evaluate future spent fuel storage strategies.

The physical layout of the Unit 3 and Unit 4 cask area racks and the nominal gaps relative to other SFP racks and the east SFP wall are shown in Figures 1.1.1 and 1.1.2 of Appendix 1, respectively. Appendix 1 Section 2 describes the physical design of the rack module and individual rack cells. Appendix 1 Section 8 describes the structural arrangement of each unit's spent fuel pool.

The following sections provide technical justification for installing a cask area rack on both units. Areas evaluated include:

- criticality
- thermal-hydraulics
- rack and pool structural integrity
- handling of heavy loads
- handling of fuel assemblies
- radiological considerations
- other issues

The basis for changing TS 5.6.1.1 to revise the location where reactivity uncertainties are described to UFSAR Appendix 14D is to make the location consistent with the Westinghouse Improved Standard Technical Specifications (NUREG-1431 Section 4.3.1), which references an appropriate section of the UFSAR. A further discussion of this editorial change is included under Other Issues in Section 3.8.

3.2 CRITICALITY CONSIDERATIONS

This section summarizes the cask area rack criticality analysis performed by the rack vendor. Criticality analyses for the existing SFP racks are not impacted by the addition of the new rack and were therefore not re-analyzed, although reactivity interaction between the new rack and the existing racks was evaluated, as discussed below. A more detailed discussion of the analysis methodology, assumptions, and results is included in Section 4 of the attached License Amendment Report (Appendix 1). Because the cask area racks are essentially identical and Turkey Point fuel is of common design, a single criticality analysis was performed covering both units.

The criticality analysis demonstrates that the Unit 3 and Unit 4 cask area rack designs maintain subcriticality with margin during both normal and abnormal conditions. In compliance with Turkey Point Technical Specification 5.6.1.1 and 10 CFR 50.68, the analyses achieved an effective neutron multiplication factor (k_{eff}) less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with 200 ppm borated water at a temperature corresponding to the highest reactivity. In addition, the analyses achieved a k_{eff} value less than 1.0 assuming unborated water in the spent fuel pool.

The rack moderator temperature coefficient of reactivity is negative. Therefore, the temperature corresponding to the greatest water density at 4°C [39.2°F] was assumed for the calculations, which assured that the true reactivity would always be lower over the expected range of SFP bulk temperatures.

The soluble boron concentrations assumed for the rack criticality analysis (200 ppm and 624 ppm) are less than the minimum SFP boron concentration of 1950 ppm required by Technical Specification 3.9.14, and also less than the 650 ppm soluble boron level required by TS 5.6.1.1b for safe subcritical storage of fuel assemblies in the existing spent fuel storage racks. Therefore, the existing boron dilution analyses (i.e., time to dilute from 1950 ppm to 650 ppm) are bounding and need not be revised. A fully loaded cask area rack was found to displace a water volume comparable to a spent fuel cask, so that the boron dilution analysis SFP volume assumptions are not adversely impacted by the cask area rack.
The maximum calculated k_{eff} includes consideration of abnormal fuel drop and loading scenarios, and includes margin for biases and uncertainties in the reactivity calculations, including manufacturing tolerances. As permitted in the USNRC OT Position for Review

and Acceptance of Spent Fuel Storage and Handling Applications, independent uncertainties are statistically combined, such that the final k_{eff} will satisfy the required subcriticality limit with a 95% probability at a 95% confidence level.

The cask area racks are designed as Region I fuel storage racks, capable of storing either fresh fuel assemblies of 4.50 ± 0.05 weight percent (w/o) maximum U-235 enrichment, or spent fuel assemblies regardless of their burnup history. Fresh fuel is inherently more reactive than spent fuel, and provides the limiting storage case for the cask area rack criticality analysis.

For reactivity control, the rack cells employ Boral™ neutron absorber panels mounted on the outside faces of stainless steel boxes of nominal 8.75 inch square inside dimension (except cells on the rack periphery facing the east SFP wall, which contain no Boral panel on the outer face) in conjunction with water gaps between adjacent cells. The Boral panels are nominally 7.5 inches wide and 147 inches long, and are held in place and protected against damage by stainless steel sheathing. The storage cells are assembled into an 11x12 cell array with a nominal lattice center-to-center spacing (pitch) of 10.1 inches in the east-west direction and 10.7 inches in the north-south direction, using welded connector bars. This cell spacing forms a nominal flux-trap water gap between adjacent cells of approximately one inch in the E-W direction and approximately 1.6 inches in the N-S direction. For neutron leakage, the analysis conservatively assumes an infinite radial array of storage cells, and a 12 inch (30 cm) water reflector is conservatively assumed in the axial direction.

The criticality analysis uses the three-dimensional MCNP4a Monte Carlo code developed by the Los Alamos National Laboratory as the primary methodology for the reactivity calculations. The CASMO4 code was used to determine the reactivity effects of manufacturing tolerances. The fresh fuel assembly types expected to be stored in the Region I cask area rack were evaluated to determine the most reactive fuel assembly (Westinghouse 15x15 Optimized Fuel Assembly/Debris Resistant Fuel Assembly [OFA/DRFA]).

As shown in Figures 1.1.1 and 1.1.2 of Appendix 1, the cask area rack will be in close proximity (2 to 3 inches) to surrounding Region I and Region II storage racks in the spent fuel pool. Three sides of the cask area rack will face other racks in the south, west, and north directions. As discussed in Section 4.5.4 of Appendix 1, the criticality analysis included an assessment of reactivity interaction between the new rack and the existing SFP racks, to determine if the reactivity due to rack proximity exceeded the highest reactivity for the individual racks evaluated. In all cases, the analysis showed that the resultant reactivity from interaction was less than the highest rack reactivity when the racks are analyzed separately. Therefore, the interaction does not invalidate the reactivity analyses of either the cask area rack or the adjacent SFP racks. Because the interaction analysis assumed a minimum two inch gap between the racks, the actual gap dimension will be verified to meet or exceed the minimum gap during installation of the cask area rack.

In addition to calculating the reactivity with fresh fuel stored in the rack, a fuel mispositioning event and two dropped fuel assembly events were also evaluated. A worst-case deep drop of a fresh fuel assembly into an open cell was found to cause less than two inches of local deformation of the base plate. The reactivity consequence of this deformation was calculated, conservatively assuming an infinite array of assemblies in this displaced condition, and found to be negligible. Similarly, a dropped assembly that comes to rest horizontally on top of the rack will be separated from the stored fuel by more than 12 inches, which also causes a negligible reactivity impact.

Regarding the fuel mispositioning event, each cask area rack is designed with one corner cell eliminated to provide a storage location for the fuel handling tool. The mispositioning of a fresh fuel assembly in this space was evaluated, to determine the reactivity impact on an adjacent Region I storage rack. It was found that a mispositioned fresh fuel assembly in close proximity to the adjacent Region I rack requires at least 624 ppm of soluble boron to maintain k_{eff} less than 0.95. This boron requirement is bounded by the 650 ppm minimum SFP boron level required by TS 5.6.1.1b, and is therefore acceptable.

The criticality analysis described in Section 4 of Appendix 1 demonstrates that the maximum cask area rack reactivity of 0.9544 when flooded with unborated water provides significant margin from the analysis acceptance criterion of less than 1.0 with no credit for borated water. Further, the maximum rack reactivity will remain below 0.95 for a fuel mispositioning event when credit is taken for 624 ppm soluble boron and will remain below

0.95 for all other normal and abnormal conditions when credit is taken for 200 ppm soluble boron. Therefore, the cask area rack criticality analysis results are acceptable.

3.3 THERMAL-HYDRAULIC CONSIDERATIONS

A thermal-hydraulic analysis has been performed that includes the spent fuel pool (SFP) decay heat load impact caused by adding a new cask area rack. As discussed in Section 1.0, this analysis is also based upon reducing the core offload start time to 72 hours after reactor shutdown, and has been submitted previously under a separate proposed license amendment. For completeness, the previously submitted thermal-hydraulic analysis is repeated below. The analysis determines the peak SFP bulk temperature, the maximum local water and fuel clad temperatures in an SFP storage rack, and the minimum SFP time-to-boil on a loss of forced cooling condition.

Description of Spent Fuel Pool Cooling Systems

There are two SFPs at Turkey Point, one for each unit. Each storage pool is provided with a dedicated cooling system. Each SFP cooling system consists of a pump, heat exchanger, filter, demineralizer, piping and associated valves and instrumentation. The pump draws water from the pool, circulates it through the heat exchanger, and returns it to the pool. Component cooling water (CCW) cools the heat exchanger. A 100-percent-capacity spare pump is also permanently piped into the SFP cooling system. Both SFP cooling pumps are powered from the same breaker via a transfer switch. Thus, this spare pump is capable of operating in place of the main pump, but not in parallel with it.

The SFP cooling systems at Turkey Point are not safety grade systems. However, they are seismically qualified and will remain functional during and after a safe shutdown earthquake. Essential SFP cooling equipment is also enclosed within a reinforced concrete structure. The doorway into the SFP cooling building is secured with heavy metal grating to restrict personnel access.

The water level in the SFPs is maintained in accordance with TS 3/4.9.11, "Refueling Operations, Water Level – Storage Pool." Makeup to the SFPs to maintain this level can be provided from a variety of sources. The credited makeup source for the SFPs is 100 gpm from the demineralized water system. In the event that the demineralized water system is not available, alternate makeup can be provided via the seismic Category I refueling water

storage tank, or via temporary (non-Category I) connections from the fire water system or primary water storage tank.

Reducing the decay time from 100 hours to 72 hours and adding the cask area rack will each increase the decay heat load that must be removed by the SFP cooling system. The following sections discuss the result of these changes on SFP bulk temperature, the local rack water and fuel clad temperatures, and the time-to-boil analysis for a loss of forced cooling.

SFP Bulk Heat-Up Analysis and Administrative Controls

SFP Bulk Temperature Analysis

The current decay heat calculations of record are described in Turkey Point Updated Final Safety Analysis Report (UFSAR) Appendix 14D, Section 3.2. Originally prepared to describe the supporting analysis for installation of high-density storage racks, the analysis was updated to reflect thermal power uprate and 24-month fuel cycle assumptions. Subsequently revised pursuant to 10 CFR 50.59 requirements, the UFSAR currently reflects analyses supporting full core offload and moving irradiated fuel in the reactor vessel as early as 108 hours after reactor shutdown for a typical 18-month refueling cycle.

The proposed amendments to allow moving irradiated fuel 72 hours after shutdown require updated supporting decay heat calculations. The UFSAR currently addresses four different SFP heat-up cases derived from the Standard Review Plan (SRP). These cases address maximum *normal* and maximum *abnormal* heat load conditions. The maximum normal heat load is based on a 1/2 core offload consistent with FPL plans at the time for a 24-month refueling cycle. The maximum abnormal cases are based on a full core offload that occurs 36 days after a previous normal refueling. In support of the requested amendments for a reduced decay time, FPL has redefined the SRP cases to reflect the planned refueling practice of full core offloads. The abnormal case is now interpreted to be an unplanned or emergency offload case. The SRP assumption of a 36 day period between shutdown is retained in the updated analysis scenarios.

In keeping with the above, a planned refueling would offload the entire core (157 fuel assemblies) beginning at 72 hours. The postulated unplanned, forced shutdown scenario would also offload the entire core beginning at 72 hours. The forced offload is assumed to begin 36 days after a previous reactor shutdown for a planned refueling. The analysis assumes offload capacity in the SFP includes an added spent fuel storage rack in the cask loading area of the pool and that all other storage cells are filled with previously discharged fuel, including the 1/3 core recently offloaded. The analysis for these cases and results are described below.

Case 1: Planned Refueling
Full core offload initiated at 72 hours after shutdown

Case 2: Planned Operation
1/3 core offload with full capacity inventory at 36 days after shutdown

Case 3: Unplanned Operation with Spent Fuel Pool Cooling
Full core offload at 72 hours following a forced shutdown with 1/3 core recently offloaded (36 days after a planned refueling shutdown)

Case 4: Unplanned Operation without Spent Fuel Pool Cooling
Full core offload at 72 hours following a forced shutdown with 1/3 core recently offloaded (36 days after a planned refueling shutdown) with loss of SFP cooling at bulk peak pool water temperature (time to boiling begins at bulk peak pool water temperature)

The planned refueling (Case 1) is evaluated at two different CCW temperatures. A low CCW temperature of 85 °F is analyzed as Case 1a with a high fuel transfer rate of 8 fuel assemblies per hour. A second case is analyzed with CCW at its maximum temperature of 105 °F. This latter case is designated Case 1b below and is analyzed with a fuel transfer rate of 6 fuel assemblies per hour. These two cases demonstrate the capability of the SFP cooling system at various CCW temperatures.

The following input parameters were used in the analysis:

Input Parameter	Value
Full Core Decay Heat Load	30.5 MBtu/hr at 72 hours 12.3 MBtu/hr at 36 days
Full Capacity (past refuelings) SFP Heat Load	3.82 MBtu/hr
CCW Inlet Temperature Case 1a Cases 1b, 2, 3, and 4	 85 °F 105 °F
CCW Flow Rate	2800 gpm (minimum)
SFP Cooling Flow Rate	2200 gpm (minimum)
Heat Exchanger Fouling	0.000075 hr-ft^2-°F/Btu
Heat Exchanger Tube Plugging Allowance	0%
Fuel Assembly Transfer Rate Case 1a Case 1b	 8 per hour 6 per hour
SFP Water Inventory	2,033,099 lbm
SFP Water Make-Up Rate	100 gpm
Capacity of Existing Racks	1404 Fuel Assemblies
Allowance for Potential Future Rack Addition	132 Fuel Assemblies

The analysis uses the same methodology and assumptions for heat exchanger performance as those used to support thermal power uprate (performed in 1996 under license Amendments 191 and 185 for Units 3 and 4, respectively).

Heat exchanger effectiveness was quantified in 1996 to support the SFP cooling analyses at uprated conditions. Heat exchanger effectiveness was calculated using plant data obtained from the 1993 and 1994 Unit 4 refueling outages, and an empirically derived fouling factor of 0.000075 hr-ft^2-°F/Btu. The use of this fouling factor (in lieu of the design fouling factor used by the heat exchanger manufacturer for sizing purposes) is justified by the fact that tube side SFP water is continuously purified and slightly acidic and the shell side water is treated CCW. Data collected during the recent 2002 Unit 4 refueling outage confirmed that there has been no observable change in heat exchanger performance compared to 1993/1994 data.

Minimum tube and shell side flow rates have been assumed in the analysis to conservatively model SFP heat exchanger performance. The assumed flow rates are 10% lower than the operating flowrates during a refueling outage. This provides additional conservatism to account for potential future heat exchanger degradation, (e.g., fouling, tube plugging). No tube plugging is assumed in the heat-up analysis since no tubes are currently plugged (after 30 years of heat exchanger operation).

The decay heat values in the updated analysis were determined from the ORIGEN-2 computer code using historic and projected burnup schedules for Units 3 and 4. The 1/3 core is assumed to be 64 assemblies to bound one reload batch.

Acceptance criteria for the SFP bulk heat-up analysis:

a. The bulk maximum SFP temperature shall remain below 150 °F from a full core offload during a planned refueling.

b. The bulk maximum SFP temperature shall remain below 212 °F during an unplanned offload evolution.

The 150 °F acceptance criterion specified above for planned refuelings was established for the SFP cooling systems as part of the thermal power uprate. The 150 °F value was based on a review of other plants' licensing requirements, and the first re-racking at Turkey Point (performed in 1977 under license Amendments 23 and 22 for Units 3 and 4, respectively). It was applied during the analysis of the Turkey Point Unit 4, Cycle 16 (pre-uprate) full core offload. Accordingly, the 150 °F temperature limitation represents a reasonable criterion for both partial and full core offloads for both Turkey Point SFPs.

The 212 °F acceptance criterion specified for unplanned offloads is representative of bulk SFP boiling conditions.

Results of the SFP bulk heat-up analysis:

Case 1: Planned Refueling

1a. The maximum expected SFP bulk temperature for a full core offload at 72 hours after shutdown is 147 °F with a CCW inlet temperature of 85 °F, and a transfer rate of 8 fuel assemblies per hour.

1b. The maximum expected SFP bulk temperature for a full core offload at 72 hours after shutdown is 165 °F with a CCW inlet temperature of 105 °F, and a transfer rate of 6 fuel assemblies per hour.

Case 2: Planned Operation

The maximum expected SFP bulk temperature for a 1/3 core offload with capacity inventory at 36 days after shutdown is 121 °F.

Case 3: Unplanned Operation with Spent Fuel Pool Cooling

The maximum expected SFP bulk temperature for a full core offload at 72 hours following a forced shutdown (36 days after a planned refueling shutdown) with 1/3 core recently offloaded is 183 °F. This temperature assumes that the entire core is offloaded as a complete unit at 72 hours. The time to reach this maximum steady-state temperature with SFP cooling is 25 hours after offload.

Case 4: Unplanned Operation without Spent Fuel Pool Cooling

The maximum expected SFP bulk temperature for a full core offload at 72 hours following a forced shutdown (36 days after a planned refueling shutdown) with 1/3 core recently offloaded, with a subsequent loss of cooling, is 212 °F. If SFP cooling were lost at the time of the peak pool temperature (183 °F), the pool would reach boiling conditions in 1.5 hours.

Decay heat analysis results for the above cases are similar to, or bounded by, those currently described in UFSAR Appendix 14D, Section 3.2.

A comparison between the current analysis and the new analysis for a planned refueling is provided below.

Current and New Heat-up Analysis Results – Planned Refueling

Analysis	CCW Temperature	Peak SFP Temperature
Current UFSAR Case	100 °F	155 °F
UFSAR Case at Elevated CCW Temperature	105 °F	160 °F
72-Hour Offload Case 1a	85°F	147 °F
72-Hour Offload Case 1b	105 °F	165 °F

As shown above, the current analysis for a planned refueling predicts peak SFP temperatures > 150 °F for a full core offload. Administrative controls are currently credited to maintain pool peak temperature below 150 °F. The new analysis similarly predicts that the bulk SFP temperature would overshoot 150 °F under some offload scenarios (e.g., Case 1b) such that administrative controls will be relied upon to maintain pool temperature below that value.

Administrative Controls

Administrative controls are proceduralized to suspend offload activities at a lower SFP temperature to maintain pool peak temperature below 150 °F.

The administrative controls described in this submittal will be implemented by the TS Bases document and various plant procedures as illustrated below:

TS Bases Document

The TS Bases document will prohibit fuel movement in the reactor vessel from occurring before the 72-hour decay time has elapsed. This operating restriction, coupled with the inherent delay associated with completing the required preparatory steps for moving fuel in the reactor vessel will ensure that the proposed 72-hour decay time will be met for each refueling outage.

Procedural Controls

Plant procedures will control the allowable offload start time, fuel assembly offload rate and administrative SFP bulk temperature limit required to maintain pool temperature below 150 °F. As indicated previously, Turkey Point currently uses administrative controls to ensure SFP bulk water temperature does not exceed 150 °F during planned refuelings. The requisite controls include minimum offload start time, maximum SFP bulk temperature and maximum fuel assembly transfer rate. These controls are already incorporated into the plant procedures that govern reactor refueling so plant operators are already accustomed to the practice.

The proposed offload schedule evaluated herein is also affected by CCW temperature. This variable, along with offload start time and fuel assembly transfer rate, affects the SFP heat-up rate and the peak water temperature. To address this process variable, the plant operating procedure controlling minimum start time and maximum fuel assembly transfer rate will be revised to relate these parameters to CCW temperature required to maintain SFP temperature below the 150 °F limit. An administrative bulk pool temperature limit will continue to be imposed to ensure that the 150 °F limit is not exceeded after completion of offload activities due to the inherent lag in SFP heat-up. The specified administrative limit will maintain SFP temperature below 150 °F without intervening operator action. Bounding values will be provided in the procedures with an option to obtain cycle-specific values from engineering prior to commencing offload activities.

Due to the many variables that can have an impact on SFP temperature, FPL may elect to use a cycle-specific offload start time and fuel assembly offload rate in lieu of the bounding restrictions. Consideration will be given to the actual core power history, scheduled off load start time, actual CCW temperature, predicted SFP heat exchanger performance, and planned fuel assembly off load rate in the establishment of the specific control values.

Regardless of whether cycle-specific or bounding offload parameters are used for a particular refueling, plant procedures will require that fuel transfer to the SFP be suspended if the administrative temperature limit is reached during the offload. Resumption of offload activities would occur when the bulk temperature decreases below the administrative limit. Note that FPL may elect to perform a partial in-core shuffle in conjunction with administrative temperature controls, to complete a planned offload without interruption and maintain bulk SFP water temperature below 150 °F.

Various methods are available to monitor the bulk SFP pool water temperature during an offload. It is monitored locally by Operators in the SFP. An annunciator panel alarm also exists and will alert Operators in the control room of pending high SFP water temperature conditions.

SFP Local Thermal-Hydraulic Analysis

The current thermal-hydraulic analysis of record is described in Turkey Point UFSAR Appendix 14D, Section 3.3. This analysis was performed in support of the currently installed high-density storage racks. A new analysis was performed to determine if the water in the storage racks will remain subcooled given the increased decay heat associated with 72-hour offload conditions.

Acceptance criteria for the SFP local thermal-hydraulic analysis:

a. The local maximum SFP temperature shall remain below the local saturation temperature of the water.

b. The maximum fuel cladding temperature in the SFP shall remain below the local saturation temperature of the water. If the maximum fuel cladding temperature exceeds the local saturation temperature of the water, a departure from nucleate boiling shall not occur.

In the SFP storage rack cells, decay heat from the fuel induces a natural circulation of water upward through the fuel assembly. Cooler water is supplied to the bottom of the rack cells through various flow holes. Water gaps or plenums between the racks and the SFP floor and walls allow water from the area above the rack to flow to the inlet of the rack cells.

Fluid flows and temperatures within a rack cell loaded with fuel having a 72-hour decay time were determined by rigorous computational fluid dynamics (CFD) analysis. The CFD analysis was performed using the FLUENT™ fluid flow and heat transfer modeling program. A single bounding case was evaluated that includes the highest bulk SFP temperature (150 °F) and decay heat load, and conservative hydraulic parameters.

Key assumptions of the SFP local thermal-hydraulic analysis include:

- No downcomer flow exists between the individual storage rack modules.
- The hydraulic resistance of every rack cell in the SFP includes the hydraulic resistance that would result from a dropped fuel assembly lying across the top of the rack.
- A fouling factor of 0.0005 hr-ft^2-°F/Btu is imposed on the outside of the fuel rods to account for any crud layer.
- The maximum local water temperature (at the fuel rack exit) and peak fuel assembly heat flux (typically near the mid-height of the active fuel) occur coincidentally.
- The radial peaking factor is applied to the hottest batch decay heat generation rate to account for variations in heat emission within the batch.
- The rack cell inlet temperature is equal to the SFP bulk temperature of 150 °F.

Results of the SFP local thermal-hydraulic analysis:

Results Parameter	Value
Peak Local Water Temperature	192 °F
Peak Fuel Cladding Temperature	236 °F

The saturation temperature of water in the SFP increases with increasing depth (pressure). The critical location for localized boiling in the fuel racks is at the top of the active fuel height. The minimum depth of water at the top of the active fuel height is 25.75 feet. At this water depth, the saturation temperature of water is 241 °F. Both the calculated peak local water temperature and the peak fuel cladding temperature are below the local saturation temperature.

Time-To-Boil Analysis

The current time-to-boil analysis of record is described in Turkey Point UFSAR Appendix 14D, Section 3.2. That analysis is based on a full core offload at 150 hours following a forced shutdown with 1/2 core recently offloaded to the pool (36 days after a normal refueling shutdown) – consistent with 24-month fuel cycle assumptions. A time-to-boil of approximately 1 hour was calculated for that case based on the assumption that SFP cooling is lost at the time of peak pool temperature (194.5 °F). The maximum boil-off (make-up) rate at 212 °F for that case was 76.3 gpm.

An updated time-to-boil analysis is required to support 72-hour offload conditions. The acceptance criteria for this analysis is:

a. The time to heat the SFP to 212 °F after loss of SFP cooling during an unplanned offload evolution shall be sufficient to permit alignment of available make-up sources.

b. The required make-up rate to replace water due to boiling shall be less than the existing rate of 100 gpm.

The supporting analysis determines the time-to-boil for a full core offload at 72 hours following forced shutdown (36 days after a planned refueling shutdown) with 1/3 core recently offloaded – consistent with the current 18-month fuel cycles. The time-to-boil is 1.5 hours assuming that SFP cooling is lost at the time of the peak pool temperature (183 °F). The maximum boil-off (make-up) at 212 °F for this condition is 81 gpm. This make-up rate is within the 100 gpm acceptance criteria established for the SFP bulk heat-up analysis.

Makeup rates from available unborated water sources to the SFP have previously been submitted to the NRC in Attachment 6 to FPL letter L-99-176 in support of Amendments 206 and 200 for Turkey Point Units 3 and 4, respectively. As documented in L-99-176, the following makeup sources satisfy the 100 gpm acceptance criterion.

Estimated SFP Makeup Rates from Unborated Water Sources

Makeup Source	Rate (gpm)
Demineralized Water System	174
Primary Water System	
- Direct Connection	415
- Local Hose Station	500
Fire Hose Station Outside SFP	100

Additionally, the SFP makeup rate from the borated RWST is 100 gpm.

Several factors account for the increase in time-to-boil calculated for the 72-hour offload condition. These factors include lower bulk peak temperature (183 °F versus 194.5 °F), use of representative refueling practices (1/3 core discharged versus 1/2 core), and heat exchanger performance derived from actual plant data (consistent with that assumed in the SFP bulk heat-up analysis). When the current UFSAR case (a full core offload at 150 hours following a forced shutdown with 1/2 core recently offloaded to the pool 36 days after a normal refueling shutdown), is analyzed with equivalent heat exchanger performance assumptions, the time-to-boil is increased from 1 hour to approximately 2 hours. When this supplemental time-to-boil value is compared to that determined for 72-hour offload conditions (1/3 core and decay heat based on ORIGEN-2 code), the time to boil will decrease to 1.5 hours. The 1.5 hour time-to-boil calculated for the 72 hour offload conditions provides sufficient time to establish makeup to the SFP.

The time-to-boil analysis assumptions are sufficiently conservative such that actual times to reach boiling conditions in the SFP will be longer than those documented herein. For example, the analysis assumes that the entire core is off loaded to the SFP in one complete step at 72 hours after shutdown. No credit is taken for the time dependent nature of the offload activities which can span 26 hours for a full core offload at a rate of 6 assemblies per hour. The CCW system is also assumed to be at its maximum temperature of 105 °F.

3.4 SEISMIC AND STRUCTURAL EVALUATION

This section summarizes the structural analyses performed for the new rack and supporting structures. A more detailed discussion of the structural analysis methodology, assumptions, and results is included in Sections 6 and 7 (for the racks) and in Section 8 (for the supporting structures) of Appendix 1.

The analyses performed to demonstrate structural adequacy include:

1) Rack structural evaluation during seismic events
2) Rack structural evaluation during fuel assembly drop events
3) Spent fuel pool (Auxiliary Building) east wall and slab structural evaluation

The plan view dimensions of the cask area racks and the nominal gaps between the rack and the surrounding east SFP wall and SFP racks are shown in Appendix 1, Figures 1.1.1 and 1.1.2, for Units 3 and 4, respectively. A structural model of each unit's spent fuel pool east wall is shown in Appendix 1, Figures 8.1 through 8-4.

In summary, an evaluation of the mechanical and civil structures has been performed to address the structural issues resulting from the installation of a storage rack in the cask area of each unit. The analysis considered the loads from seismic, thermal, and mechanical forces to determine the margin of safety in the structural integrity of the new storage rack, the spent fuel pool and liner, and the affected portions of the Auxiliary Building. The loads, load combinations, and acceptance criteria for the storage rack and liner were based on ASME Section III, Subsection NF, and on NUREG-0800, Standard Review Plan (SRP), Section 3.8.4, Appendix D. Load combinations and structural assessment of the Auxiliary Building concrete followed the requirements of the UFSAR and the American Concrete Institute, ACI 318-63.

Cask Area Rack Structural Evaluation during Seismic Events

The analyzed configuration consists of one free-standing, self-supporting rack module in the cask area of each unit. The cask area rack is effectively isolated from the surrounding racks in the SFP, because the gap between racks is sufficient to prevent impact interactions. Nevertheless, possible interaction between the racks is accounted for by performing simulations considering in-phase and out-of-phase dynamic motion between the existing racks and the new rack in each unit. The seismic analysis was based on the simulation of

the Safe Shutdown Earthquake (SSE). The SSE simulation results were compared against normal condition stress allowables. These evaluations were performed in accordance with SRP 3.7.1 requirements.

The cask area racks were modeled as fully loaded with fuel assemblies weighing 1608 lbs each. This fuel weight conservatively represents each fuel assembly stored with a rod cluster control assembly (RCCA). Other analysis assumptions and details of the dynamic model for the rack structure are discussed in Section 6.5 of Appendix 1.

The seismic analysis results indicate that the maximum seismic displacement of the new rack does not result in any impacts with either the adjacent east SFP wall or with adjacent racks. The resultant member and weld stresses in the rack are all below the allowable stresses, as shown in Table 6.9.1 of Appendix 1. This maximum calculated stress factor is 0.124 associated with the cell membrane material. The maximum stress factor for the pedestal supports is 0.068. Therefore, the rack will remain functional during and after all postulated load conditions, including the SSE.

As shown in Section 6.8.1 of Appendix 1, a maximum rack lateral displacement of 0.11" was found to occur under SSE conditions. Therefore, rack tipover would not occur during a seismic event, even without the proximity of the east SFP wall or surrounding racks. Comparing half the distance between the rack pedestals to the maximum lateral displacement yields a tipover safety factor of approximately 389, which far exceeds the acceptance criterion of 1.5. Therefore, a support or restraint system to limit rack lateral movement is not necessary.

Local cell wall integrity was evaluated for peak impact loads from stored fuel assemblies. As shown in Table 6.9.1 of Appendix 1, the allowable impact load is much greater than the highest calculated impact load from any rack analysis. Therefore, fuel impacts during seismic events do not result in rack cell deformation.

The rack structural evaluation determined the resulting stress factors for each rack pedestal, and for the entire rack cellular cross-section just above the bottom casting. These locations are the most heavily loaded sections of the structure, so that satisfaction of the stress factor criteria at these locations ensures that the overall structural criteria are met. The maximum pedestal stress factor is 0.068 and the maximum cell wall stress factor is 0.124. An

evaluation of the stress factors for all of the simulations performed demonstrates that all stress factors are less than the 1.0 limit.

The largest computed thread stress for each pedestal under SSE conditions was calculated to be 3,161 psi. For conservatism, the actual stress for the SSE condition was compared against the allowable stress for the OBE condition, which is 8,520 psi for the female pedestal threads. The allowable stress for the male pedestal threads is much larger due to the higher material strength. Therefore, both the female and male pedestal thread stresses are acceptable.

As discussed in Appendix 1 Section 6.9.5, weld locations at the bottom of the rack (i.e., the baseplate-to-cell connection, the pedestal-to-baseplate connection, and cell-to-cell connection) are subjected to significant seismic loading. The calculated stress value at each of these weld locations was found to be below the allowable stress value.

As discussed in Appendix 1, Section 6.12, evaluations were performed on cell-to-cell welded joints and on potential cell wall buckling due a thermal gradient between an isolated hot cell and surrounding cold cells. The maximum compressive stress in the cell wall was found to be significantly below the critical stress calculated using the classical plate buckling method, demonstrating that buckling is not a concern. The maximum shear stress in cell-to-cell welded joints from a hot cell-to-cold cell thermal gradient is below faulted conditions and is therefore acceptable.

Results of a fatigue analysis for seismic-induced motion on the cask area racks is summarized in Appendix 1, Section 6.9.4. Based on the cumulative effects of twenty one SSE events, the safety factor is greater than 2,000 for fatigue within the rack components.

Cask Area Rack Structural Evaluation during Fuel Assembly Drop Events

The NRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications" specifies that spent fuel rack designs must ensure the functional integrity of racks under all credible fuel assembly drop events. An evaluation of the consequences of fuel assembly drops onto the cask area racks was conducted to demonstrate that the racks continue to safely store nuclear fuel following the drop. Although Turkey Point Technical Specification 3.9.7 restricts loads over spent fuel storage

to 2000 pounds, the drop analyses conservatively evaluated a 3000 pounds load drop. Two categories of accidental drop events were considered.

Shallow Drop Scenario

A "shallow drop" of a fuel assembly is assumed to strike the top of the rack and damage the honeycomb structure, but not enter an open cell or land directly on an already-stored assembly. The structural acceptance criterion for this event is that the damage to the rack structure must be limited to the portion of the cell(s) above the top of the active fuel region for stored assemblies, which is approximately 15.7 inches below the top surface of the rack. The assumed free-fall height for this event is 36 inches above the rack, and the assumed conservative weight of the dropped assembly plus its handling tool is 3000 lbs. Figure 7.5.1 in Appendix 1 shows the maximum plastic strain and deformation resulting from a shallow drop on an outside corner of the cask area rack.

Based on the design of the rack honeycomb structure, the limiting shallow drop scenario that would cause the maximum cell wall deformation occurs at a cell on the rack periphery, rather than at an internal cell. For this limiting case, the dynamic analysis shows that the top of the impacted peripheral cell undergoes plastic deformation to a maximum depth of 14.25 inches, which is less than the 15.7 inch distance required to reach the top of active fuel in the cask area rack. Therefore, the functional integrity of the cask area rack is not compromised by a shallow drop event.

The radiological concern with a Fuel Handling Accident was also evaluated to ensure that the cask area rack structure can withstand the limiting drop of 2000 pounds without causing more damage than that assumed in the FHA; i.e., release from all fuel rods in one fuel assembly. As discussed in the UFSAR accident analyses, an end-to-end drop of one assembly onto another will not result in yielding any fuel cladding. However, to create a bounding analysis, the limiting failure to all rods in one assembly was assumed to occur non-mechanistically. Inherently, any fuel assembly drop that first strikes a portion of the cask area rack structure before hitting a stored assembly will result in a lower impact energy to the affected assembly or assemblies. The impact energy will be reduced by that amount absorbed by the rack and distributed between any adjoining cells and assemblies. In any case, the imparted energy to an individual assembly cannot exceed the end-to-end drop, and like that analysis, will not result in any damage to fuel assemblies. Therefore, drops on the cask area rack structure will not change the radiological results of the Fuel Handling Accident.

Deep Drop Scenario

A "deep drop" of a fuel assembly occurs when the dropped assembly enters an empty storage cell and impacts the rack baseplate. A sufficiently large impact force could threaten the structural integrity of the baseplate. Two deep drop locations were evaluated: (1) a drop in a cell located directly above a rack pedestal, and (2) a drop in an interior cell away from a pedestal where the baseplate is more flexible. The structural acceptance criteria for a deep drop event are that the baseplate must remain intact and any deformation of the baseplate from the impact must be acceptable both from a structural and a criticality standpoint. In addition, the high impact load from a deep drop onto a rack pedestal must not tear the spent fuel pool liner plate when the force is transmitted into the structure.

The analysis shows that a fuel assembly deep drop through an interior cell away from a pedestal causes a maximum local baseplate deformation of 1.75 inches, which is less than the 6.0 inch distance from the baseplate to the liner. As discussed earlier in Section 3.2, the reactivity increase caused by baseplate deformation from a deep drop is negligible. A deep drop above a pedestal was found to produce a maximum stress below the yield stress of the cask area liner material, but may cause some localized damage to the underlying concrete. The maximum compressive stress applied to the concrete area floor under this drop scenario is 9,644 psi, which exceeds the concrete compressive strength of 3,000 psi. This will result in local damage (crushing) of the concrete below the liner, but does not represent any threat to the overall integrity of the pool concrete floor. Therefore, the liner plate and concrete slab will remain intact without loss of water from the cask area on any deep drop.

Spent Fuel Pool (Auxiliary Building) Structural Evaluation

A structural evaluation was performed for the effect that the fully-loaded cask area rack in each unit will have on the Auxiliary Building structure. The evaluation is limited to the pool floor and the portion of the east SFP wall adjacent to the cask area. The evaluation is described in detail in Section 8.0 of Appendix 1. Refer to Figures 8.2 through 8.4 in Appendix 1 for diagrams of the cask area wall section and finite element models.

Loads applied to the structural analysis and structural capacity assessments followed the requirements of the UFSAR and ACI 318-63. The load definitions and combinations are given in Section 8.4 of Appendix 1, and include static, seismic, and tornado-induced loads. The cask area load included the dead weight of a cask area rack fully loaded with fuel assemblies.

The thermal loading on the east SFP wall caused by the temperature gradient between the pool water and the outside air temperature was also considered. A steady-state thermal gradient of 150°F was based on a 180°F pool water temperature and 30°F outside air temperature. A transient thermal gradient of 182°F was based on 212°F pool water and a 30°F outside air temperature. The analysis is consistent with the current UFSAR structural analysis, using the same load combinations, load factors, and temperature gradients across the SFP exterior walls, and demonstrated acceptable results.

Sections of the 4½ foot thick floor slab in the vicinity of the east wall were evaluated for the additional loading of a fully-loaded cask area rack. Bending of the slab in the east-west direction due to the new rack was qualified by using output from the east wall model. The acceptable results of the wall and slab reinforced concrete analyses are given in Section 8.5 of Appendix 1.

Section 8.6 of Appendix 1 discusses the results of the pool liner evaluation for the effects of adding the cask area rack. The liner was evaluated for stress due to lateral loads during a seismic event, including fatigue assessment considering twenty-one SSE events. The friction loads due to a seismic event will not tear the liner plate or cause the liner seam welds to rupture.

The structural evaluation concluded that the Auxiliary Building regions affected by the load of a new cask area rack have adequate safety margins under the required loading combinations. The evaluation also concluded that the local loading on the spent fuel pool liner plate from a fully loaded cask area rack, seismic motion, and thermal loading does not compromise liner integrity or exceed concrete bearing strength limits.

3.5 HANDLING OF HEAVY LOADS

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", defines a heavy load as a load whose weight is greater than the combined weight of a single fuel assembly and its handling tool. Dry weights for the empty cask area racks are approximately 16 tons. Therefore, installation of each rack into the unit's flooded cask area (and the rack's eventual removal from the cask area) will involve handling heavy loads in the vicinity of the spent fuel pool. To ensure compliance with Technical Specification 3.9.7, spent nuclear fuel stored in existing racks adjacent to the cask area will be relocated prior to installing and removing the cask area rack. A physical survey of the respective cask area in relation to its door opening and cask crane travel path will determine which storage cells will be vacated of spent nuclear fuel. This information will be incorporated into the rack installation/removal procedure. Vacating the appropriate rows of stored fuel will ensure that no heavy load will be carried directly over spent nuclear fuel.

The cask area is located along the approximate center of the eastern wall of the respective unit SFP. An L-shaped door in the Auxiliary Building roof designed for ingress and egress of spent fuel casks is located directly over the cask area. The common spent fuel cask handling crane, which is shared between both units and travels outside and above the east end of the Auxiliary Building, will be used to lower the rack vertically through the L-shaped door into the cask area.

The cask handling crane is designed with a main hook whose load capacity is 105 tons. However, Technical Specification Limiting Condition For Operation (LCO) 3.9.12 limits the load that may be carried over the spent fuel pool to that of a single element cask (about 25 tons), to ensure pool integrity for an uncontrolled cask descent. The weight of an empty cask area rack (16 tons) is well below the crane hook capacity and the Technical Specification LCO.

To prevent submerging the crane's main hook during rack installation and removal, a temporary hoist with the appropriate capacity will be attached to the main hook, and a vendor-designed rack lifting rig will be used. With FPL oversight, vendor personnel will perform the initial rack installation process and will train FPL personnel in the installation and removal procedure. The rack lifting rig is similar to rigs used for handling Holtec racks at other plants such as Hope Creek, Millstone 1, James A. FitzPatrick, and Three Mile Island (TMI-1). The rig consists of four independently-loaded traction rods that lock into

four locations in the rack baseplate. Each rod has a safety factor greater than 10. If one of the rods should break, the rack will still be supported by at least two rods, with a safety factor of more than 5. This arrangement meets the duality criteria for lifting rigs called for in Section 5.1.6(3) of NUREG-0612.

Other guidelines of NUREG-0612 regarding the safe handling of heavy loads will also be followed, including proper procedures, operator training, supervision by qualified individuals, crane inspection, maintenance, and testing. Section 3.5 in Appendix 1 details the defense-in-depth approach taken to ensure that the handling of the racks by the cask handling crane will comply with the NUREG-0612 guidance.

3.6 HANDLING FUEL ASSEMBLIES IN THE CASK AREA

Fuel assembly movement into the new cask area racks will take essentially the same path as fuel movement required to load a spent fuel transfer cask. Therefore, no new fuel movement pathways are created by the addition of a cask area rack. Each unit's spent fuel handling crane inside the Auxiliary Building will be used to handle spent fuel assemblies in the cask area rack. Because the cask areas were not originally considered for spent fuel storage, some peripheral areas of each cask area rack may not be accessible to the spent fuel handling crane as it is now configured. FPL will determine whether either unit's spent fuel handling crane requires modifications to access all of the cask area rack cells, and if needed, may modify the crane under 10 CFR 50.59. The Turkey Point fuel handling procedures will be modified to include the cask area racks when the racks are installed.

The radiological consequences of a Fuel Handling Accident (FHA) occurring in the cask area were reviewed, to determine if the consequences are bounded by the existing analysis in UFSAR Section 14.2.1. The existing analysis determines the consequences of a single fuel assembly drop inside the Auxiliary Building, which includes the SFP area. The cask area rack will be installed at the same water depth as the existing SFP racks, providing the same iodine decontamination factors assumed in the FHA analysis, and the source term will be unchanged from the current analysis. Therefore, the radioactivity released from a fuel assembly dropped over the new cask area rack would have essentially the same consequences as a fuel assembly drop on the existing storage racks in the remainder of the spent fuel pool. Because there is no damage mechanism to increase the radiological consequences of a FHA occurring in the cask area beyond the consequences of a FHA elsewhere in the SFP and Auxiliary Building, the existing FHA analysis is bounding.

As discussed previously in Section 3.2, the criticality analysis concluded that dropping a fuel assembly onto the cask area rack or into an open rack cell will not cause an unacceptable reactivity excursion. From a rack structural standpoint, Section 3.4 discussed that a fuel assembly drop will not threaten the structural integrity of the stored fuel assemblies or the integrity of the storage rack. Therefore, the reactivity and structural integrity consequences of a FHA occurring in the cask area are acceptable.

The probability of a FHA occurring will not be increased by the addition of the cask area racks. As stated in Section 1.0, the cask area racks will provide temporary storage of nuclear fuel during refueling outage core offloads and during non-outage handling of irradiated fuel in the spent fuel pool. The number or frequency of fuel assembly movements should not be significantly increased by addition of the cask area racks. With the new racks installed, full core reserve capability will be extended by at least two fuel cycles, allowing spent fuel assemblies created during that period to be discharged directly to storage racks. Without the racks installed, spent fuel assemblies would have to be moved from existing racks to transfer casks to create room for offloaded fuel. Therefore, the probability of a FHA with a cask area rack installed on either unit is not expected to be increased when compared to the current FHA probability without a cask area rack installed.

3.7 RADIOLOGICAL CONSIDERATIONS

Fuel Handling Accident

The impact of the cask area racks on the probability and radiological consequences of a fuel handling accident is discussed previously in Section 3.6 above. Note that the Turkey Point fuel handling accident analysis is currently being revised for a fuel assembly 72-hour decay time by a separate PLA submitted in FPL letter L-2002-151 dated October 21, 2002. Addition of a cask area rack will not impact, nor be impacted by, this revision of the FHA analysis.

Additional Radwaste Generation

No significant increase in solid, liquid, or gaseous radwaste generation is expected to result from the installation, use, or removal of the new cask area racks. Prior to installing the cask area racks, equipment and material temporarily stored in the cask area will be removed and the area floor (liner plate) will be cleaned using an underwater vacuum to remove any accumulated silt. These activities are expected to generate a small volume of low-level solid radwaste that will be captured underwater in vacuum filter cartridges and properly disposed of using Turkey Point radwaste handling procedures.

Storing and removing fuel assemblies from the cask area racks is not expected to generate any additional solid or gaseous radwaste from either unit compared to the current practice of storing fuel assemblies in the remainder of the spent fuel pool. Because each unit's cask area rack must eventually be removed for cask handling operations, rack contamination and activation will be minimized by a preferential fuel loading process which will avoid the selection of fuel with gross defects for storage in this rack. Furthermore, the cask area rack will be primarily used to temporarily store fresh, unburned fuel and discharged fuel of known integrity, such that rack contamination will be minimized.

When a cask area rack is removed to allow cask handling operations, the rack will be visually inspected underwater to ensure all fuel assemblies and loose debris are removed, then lifted and rinsed with water over the cask area, flushing loose contamination into the spent fuel pool. While the rack is suspended over the cask area, the individual rack cells will drain through the large cooling holes in the rack baseplate. To catch residual water that might drain during movement, the bottom of the rack will be covered with a liner prior to its removal from the Auxiliary Building. The rack will then be stored in a suitable radiologically-controlled location protected from the elements and capable of containing any postulated leakage. During storage, the rack will be routinely monitored for any residual leakage. Therefore, no significant radwaste is expected to be generated from the rack removal and storage process.

Personnel Radiation Exposure

The potential for increased radiation exposure to personnel resulting from the installation of cask area racks was evaluated. The following possible sources of increased radiation exposure were considered:

- Cask area cleanup activities prior to rack installation
- Rack installation activities
- Fuel assembly movements
- Area radiation level changes due to fuel stored in the rack
- Rack removal and cleaning activities

Cask area cleanup will be performed from the refueling floor elevation. The combined exposure for the cleanup work on both units is estimated to be 0.1 person-rem, based on a conservative dose rate of 2.5 mrem/hr and 32 manhours of work. Material removed from the area will be moved underwater to another storage location. Silt on the area floor will be vacuumed using an underwater vacuum unit, and the silt will be retained in filter cartridges stored underwater. Once the area is cleaned, no significant area preparation should be necessary to install the racks, because no underwater interferences have been identified in either unit's area that would require removal or modification. No underwater diving is anticipated to be necessary for cleaning or installing the racks.

The cask area racks will not be radioactive when initially installed, and personnel exposure during initial rack installation will be governed by the time spent above the SFP and cask area. Based on the results of limited physical surveys of each cask area, the preparation for rack installation will not require divers. Combined personnel exposure from rack installation in both units is estimated to be less than 0.2 person-rem, based on a conservative dose rate above the pool area of 2.5 mrem/hr and 64 manhours of work.

Use of the cask area racks will require underwater fuel transits in the vicinity of the east wall of the SFP, similar to fuel transit paths that would be used during cask loading. During these fuel transits, the area dose rate is not expected to increase beyond that already experienced along this wall, since fuel movement already takes place adjacent to the east SFP wall to load fuel racks located to the north and south of the cask area.

The shielding effect of the 5½-foot thick east pool wall was evaluated to determine if the Region I cask area rack loaded with fuel assemblies from a 72-hour core offload would cause any significant dose rate increase in the area outside the wall. The evaluation required that the maximum dose rate on the exterior surface of the east wall not exceed 1.0 mrem/hr. In order to achieve ≤ 1.0 mrem/hr with fuel from a 72-hour core offload, it was found that the first row of fuel in the cask area rack closest to the east wall will have to consist of spent fuel with at least five years of decay time or an equivalent barrier, to provide a "barrier row" to shield the wall from the remaining rows of freshly-discharged fuel. By making this row a barrier row, the resulting area dose rate at the exterior surface of the east wall is calculated to be 0.825 mrem/hr.

It should be noted that the outdoor cask washdown area beyond the east wall of the cask area is not normally occupied on either unit. In response to Turkey Point GDC 68, UFSAR Section 11.2 allows dose rates not exceeding 2.5 mrem/hr in periodically occupied areas of the Auxiliary Building, as compared to 0.5 mrem/hr in normally occupied areas. Therefore, the calculated dose rate beyond the east SFP wall in the vicinity of the cask area rack is acceptable and the radiation zoning in accessible areas of the Auxiliary Building and outside walls will not change due to the new fuel storage.

Based on vendor experience with rack module removal and decontamination projects, the cask area rack removal and storage process will not create significant personnel exposure. The removal and decontamination process should not result in more than 0.2 person-rem based on a pool surface dose rate of 2.5 mrem/hr, a rack surface dose rate of 20 mrem/hr and an estimated 80 manhours of work.

Therefore, the total personnel radiation exposure from activities related to the installation, fuel storage, and removal of the cask area racks is not expected to be significant and will be carefully monitored under the Turkey Point ALARA program. Radiation zoning in accessible areas is not expected to change as a result of cask area rack installation or operation.

3.8 OTHER ISSUES

a. Soluble Boron Level in the Cask Area Rack

The required TS 3/4.9.14 minimum boron concentration in the entire spent fuel pool is 1950 ppm. Because the cask area is part of the SFP, the soluble boron concentration in the cask area rack will also be maintained at or above 1950 ppm.

The cask area rack criticality analysis assumed soluble boron levels of 200 ppm and 624 ppm to demonstrate that the rack reactivity remains at or below 0.95 under normal and abnormal fuel handling conditions, when credit is taken for soluble boron. These soluble boron levels are less than the soluble boron level required for the existing spent fuel racks (650 ppm per TS 5.6.1.1b). Therefore, the existing boron dilution analyses (i.e., time to dilute from 1950 ppm to 650 ppm) are bounding and need not be revised.

The SFP water volume displaced by the cask area rack loaded with fuel assemblies was compared to the water displaced by a spent fuel transfer cask, to determine if adding a cask area rack would adversely impact the existing boron dilution analyses. The displacement volumes for the filled rack and cask were found to be comparable, and the boron dilution analysis was unaffected.

b. Foreign Material Exclusion (FME) during rack installation and removal

Fuel stored in the two unit spent fuel pools must be protected during installation and removal of the cask area racks. To ensure that the stored fuel is protected against foreign materials being dropped into the pool, foreign material exclusion methods will be closely followed in accordance with the Turkey Point foreign material exclusion procedure.

c. Minimum spent fuel decay time prior to rack installation and removal

Turkey Point Technical Specification LCO 3.9.12 prohibits the movement of a spent fuel cask into the cask area until all of the spent fuel in the SFP has decayed for a minimum of 1,525 hours (approximately 64 days), to satisfy the radiological release assumptions of the Turkey Point cask drop analysis. Although the cask area rack is not a spent fuel cask, the same 1,525 hour decay time restriction will be imposed during

the installation and removal of the rack. This will ensure that in the unlikely event a rack drop were to occur that damaged any spent fuel adjacent to the cask area, the affected fuel would meet the minimum decay time assumption of the cask drop analysis.

d. Justification for Changes to TS Design Features 5.6.1.1a and b

TS Sections 5.6.1.1a and b are being revised to stipulate that the criticality analyses include a conservative allowance for uncertainties as described in Appendix 14D of the UFSAR, rather than the current reference to Westinghouse WCAP-14416-P. The change is made for the following reasons:

1. It is not necessary to specifically name a methodology in Technical Specifications. 10 CFR Part 50 rules provide protections to ensure that the NRC-accepted methods of evaluation are used and that the NRC is afforded the opportunity for an appropriate review of any change. For example, a specific change to TS 5.6.1.1a or b requires a license amendment submittal per 10 CFR 50.90, and any changes to methods would receive the appropriate NRC review. Likewise, for any plant change that does not change the TS but changes the criticality analyses, methods would have to meet 10 CFR 50.59 evaluation criteria; thereby ensuring that NRC-accepted methods are applied.

2. The change is consistent with Westinghouse Improved Standard Technical Specifications (NUREG-1431). NUREG-1431 Design Feature 4.3.1 requires reference to the appropriate section of the UFSAR.

3. The allowance for uncertainties used in the cask area rack analyses generally followed the guidelines of the referenced WCAP. To avoid a TS revision to include complex discussion of the analysis methods with distinctions for the new cask area rack analysis, it is more appropriate to describe those analyses and any distinctions in the UFSAR.

Subsequent to issuance of the amendment, UFSAR Appendix 14D will be revised to include a description of the cask area rack criticality analysis, including the allowance for uncertainties described in Section 4.0 of Appendix 1. The description will provide a similar level-of-detail as the criticality analysis discussion for the existing SFP racks.

4.0 CONCLUSION

The addition of a new spent fuel storage rack to each unit's cask area was evaluated and found to be acceptable for criticality, thermal-hydraulic considerations, structural adequacy, handling of heavy loads, fuel handling operations, and radiological considerations. The rack design and installation comply with applicable regulatory guidance and industry standards, and are similar to spent fuel storage racks licensed in other nuclear power plants.

The TS 5.6.1.1 changes to identify UFSAR Appendix 14D as the location where reactivity calculation uncertainties are described: (1) is consistent with the Westinghouse Improved Standard Technical Specifications in NUREG-1431, and (2) will identify the appropriate location for the discussion of criticality analysis uncertainties.

APPENDIX 1

HOLTEC LICENSE AMENDMENT REPORT HI-2022931

SPENT FUEL STORAGE EXPANSION

at

TURKEY POINT NUCLEAR PLANT

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Scott H. Pellet, being duly sworn, depose and state as follows:

(1) I am the Project Manager for Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the document entitled "Spent Fuel Storage Expansion at Turkey Point Nuclear Plant," Holtec Report HI-2022931, revision 0. The proprietary material in this document is delineated by proprietary designation (i.e., shaded text) on pages 3-16, 4-5, 4-14, 4-18, 6-24, 6-29, 7-3, 7-4, and 7-5.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures

outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the

information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)

) ss:

COUNTY OF BURLINGTON)

Scott H. Pellet, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 11th day of November, 2002.

Scott H. Pellet

Mr. Scott H. Pellet
Holtec International

Subscribed and sworn before me this 11th day of November, 2002.

Maria C. Pepe

MARIA C. PEPE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005

FPL

NOV 2 6 2002

L-2002-214
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Addition of Cask Area Spent Fuel Storage Racks

In accordance with the provisions of 10 CFR 50.90, Florida Power and Light Company (FPL) requests that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4 be amended to incorporate the attached Technical Specification (TS) revisions. The proposed amendments would revise TS Section 5.6, Design Features – Fuel Storage to include the design of a new cask area spent fuel storage rack for each unit and increase each unit's spent fuel storage capacity by combining the cask area rack and existing SFP storage rack capacities.

Enclosure 1 is an evaluation of the proposed changes. Enclosure 2 is the Determination of No Significant Hazards Consideration. Enclosure 3 is a statement of Environmental Consideration. Enclosure 4 contains the affected Technical Specifications pages marked-up to show the proposed changes. Enclosure 5 contains a list of regulatory commitments resulting from the proposed change. Enclosure 6 contains the proposed Technical Specification clean pages.

The evaluation of proposed changes (Enclosure 1) contains proprietary and non-proprietary versions of an attached license amendment report for the new cask area racks prepared by the rack vendor. The affidavit required by 10 CFR 2.790 is also attached to the proprietary version of the report. FPL requests that the proprietary version be withheld from public viewing.

The proposed license amendments have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91(b)(1), a copy of these proposed license amendments is being forwarded to the State Designee for the State of Florida.

Regarding the proposed schedule for this amendment, it is requested that issuance be no later than December 2, 2003. However, it is anticipated that frequent meetings with NRC staff can improve that schedule. Although physical installation of the cask area racks is not planned until June 2004, the requested amendments date is necessary to provide time for pursuing another spent fuel storage alternative, should the submittal not be approved by that date. Based on a need for additional storage at Turkey Point Unit 3 prior to the loss of full core reserve in 2007, it is important to have a definitive storage solution licensed in 2003.

These proposed license amendments may require noticing with respect to the hybrid hearing procedures under 10 CFR 2.1107, because this may involve a proceeding on an application for a license amendment falling within the scope of section 134 of the Nuclear Waste Policy Act of 1982.

Should there be any questions on this request, please contact Walter Parker at 305 246-6632.

Very truly yours,

J. P. McElwain
Vice President Turkey Point Plant

YK
Enclosures

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Florida Department of Health

information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)

) ss:

COUNTY OF BURLINGTON)

Scott H. Pellet, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 11th day of November, 2002.

Scott H. Pellet

Mr. Scott H. Pellet
Holtec International

Subscribed and sworn before me this 11th day of November, 2002.

Marie C Pe

NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005