Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408-0420

FPL

L-2003-213
10 CFR 50.90
SEP - 8 2003

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
RAI Response for Addition of
Spent Fuel Pool Cask Area Rack Amendment

By letter L-2002-214 dated November 26, 2002, Florida Power & Light (FPL) submitted a proposed license amendment to add a spent fuel storage rack to each unit's spent fuel pool cask area. By letter dated July 18, 2003, the Nuclear Regulatory Commission (NRC) staff made a request for additional information (RAI) to support the review of the submittal. The request was discussed with FPL staff and a response date of September 12, 2003 was established. The responses were discussed with NRC Staff in telephone conferences June 12, 2003 and July 14, 2003 and comments incorporated into the response. Attached is FPL's response to the RAI.

Enclosure 1 contains the FPL response. The original No Significant Hazards Determination bounds the information provided in the RAI response. In accordance with 10 CFR 50.91(b)(1), a copy of the RAI response is being forwarded to the State Designee for the State of Florida.

Enclosure 3 provides revised proprietary pages to incorporate into Appendix 1 (the Holtec report) of Enclosure 1 to the original proposed license amendment submitted by FPL via letter L-2002-214 dated November 26, 2002. These pages were revised to address RAI comments and other corrections made by FPL; none of which have bearing on the conclusions of the submittal. The affidavit required by 10 CFR 2.790 covering these changes was previously submitted in the original proposed license amendment. FPL requests that Enclosure 3 be withheld from public viewing.

Enclosure 4 provides revised pages to incorporate into the non-proprietary version of the Holtec licensing report in the proposed license amendment. These pages were revised to address RAI comments and other corrections made by FPL; none of which have bearing on the conclusions of the submittal. This enclosure contains no proprietary information.

Enclosures 5, 6, and 7 relate to the response to RAI Question 20 regarding the reactivity effect of Turkey Point fuel manufacturing tolerances. Enclosure 6 is a Westinghouse letter providing fuel tolerance information that is considered proprietary pursuant to 10 CFR 2.790. The affidavit required by 10 CFR 2.790 is provided in Enclosure 5. FPL requests that Enclosure 6 be withheld from public viewing. Enclosure 7 is a non-proprietary version of the fuel tolerance information.

APOI

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
Nuclear Engineering

Enclosures

cc: Mr. W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
RAI Response for Addition of
Spent Fuel Pool Cask Area Rack Amendment

STATE OF FLORIDA)
)ss.
COUNTY OF PALM BEACH)

Rajiv S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, Nuclear Engineering, 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.

Rajiv S. Kundalkar

Rajiv S. Kundalkar

Subscribed and sworn to before me this

8th day of September, 2003.

Judith Ann Creasman

Name of Notary Public (Type or Print)

Rajiv S. Kundalkar is personally known to me.

OFFICIAL NOTARY SEAL
JUDITH ANN CREASMAN
NOTARY PUBLIC STATE OF FLORIDA
COMMISSION NO. CC980677
MY COMMISSION EXP. DEC. 5,2004

ENCLOSURES

Enclosure 1	RAI Response
Enclosure 2	Holtec Affidavit
Enclosure 3	Replacement pages for Holtec License Amendment Report (Proprietary)
Enclosure 4	Replacement pages for Holtec License Amendment Report (Non-Proprietary)
Enclosure 5	Westinghouse Affidavit
Enclosure 6	Westinghouse Letter NF-FP-03-310 dated July 25, 2003 (Proprietary)
Enclosure 7	Westinghouse Letter NF-FP-03-310 dated July 25, 2003 (Non-Proprietary)

RAI Response

RESPONSE TO

REQUEST FOR ADDITIONAL INFORMATION

ADDITION OF SPENT FUEL POOL CASK AREA RACK AMENDMENT

TURKEY POINT PLANT, UNITS 3&4

DOCKET NOS. 50-250 AND 50-251

1. The Florida Power and Light Company (FPL) submittal indicates that materials containing boron will be part of the design of the spent fuel storage racks that will be installed in the cask area.

   Provide the quantity of additional tritium that is expected to be produced and released. Discuss the significance of any estimated increase.

   Response:

   In the spent fuel pool (SFP), tritium is produced from neutron interaction (capture) with boron-10 found in (1) the pool water and (2) the neutron-absorber materials found in the spent fuel racks. The predominant reactions[1] are:

   $$B^{10}(n, 2\alpha) \rightarrow H^3 \quad \text{- and -} \quad B^{10}(n, \alpha) \rightarrow Li^7(n, n+\alpha) \rightarrow H^3$$

   FPL's response to Question 33 states that the combined number of fuel assemblies that can be stored in the spent fuel storage racks and the cask area rack will be limited to no more than the capacity of the spent fuel storage racks alone at all times except during a reactor offload/refuel condition. This means that the neutron population from fuel assemblies stored long-term in the spent fuel pool should remain essentially the same as now exists under the current license condition. Therefore, tritium production in the spent fuel pool will be essentially the same with or without a cask area rack installed.

   When the cask area rack is installed, the only period when neutron emissions from spent fuel assemblies may contribute to tritium production above current licensed conditions would be during refueling. The residence time (typically under 14 days) of offloaded fuel assemblies in the SFP during a refueling outage is very small compared to the long-term storage period. Although not quantified, the additional SFP neutron and tritium production during these outage periods will be insignificant.

[1] "Tritium Activation in Borated Water", pg. 192, Basic Nuclear Engineering, Foster & Wright, 1973

2. The additional stored spent fuel will increase the amount of heat being removed from the water in the spent fuel pool (SFP) and cask area.

Describe the amount of additional heat that may be released to the cooling canal. Discuss the significance of any estimated increase.

Response:

Two factors limit the additional heat load imposed on the spent fuel pool cooling system as the result of adding a rack to the SFP cask area.

First, any additional heat load attributed to the installation of a cask area rack only occurs during refueling outages. The FPL response to Question 33 states that the combined number of fuel assemblies that can be stored in the spent fuel storage racks and the cask area rack will be limited to no more than the capacity of the spent fuel storage racks alone at all times except during a reactor offload/refuel condition. This means that during non-outage periods, the maximum number of fuel assemblies that can be stored long-term in the SFP will be the same regardless of whether or not the cask area rack is installed, resulting in no additional heat load imposed on the SFP cooling system during non-refueling periods.

Second, during refueling outages with the cask area rack providing the temporary capacity for offloading fuel, the additional heat load will be from the oldest spent fuel that is allowed to remain in the pool longer because of the new rack. For both units, this added heat load would be from the 131 oldest fuel assemblies in the pool. When this additional heat load comes into play, the oldest fuel stored in the Turkey Point spent fuel pools will have a cooling history of approximately 34 years on Unit 3 and approximately 32 years on Unit 4.

Based on the steady-state decay heat data file used to calculate the bulk temperature response of the spent fuel pool with the new cask area racks installed, the projected heat loads from the oldest fuel assemblies stored in both units are estimated at:

Unit	Cask Area Rack Capacity	Projected age of oldest fuel in SFP	Estimated decay heat load from oldest fuel	Peak SFP decay heat load during refueling	% increase over SFP refueling peak heat load
3	131 assemblies	34 years	1.2 E5 Btu/hr	~ 3 E7 Btu/hr	~ 0.4%
4	131 assemblies	32 years	1.5 E5 Btu/hr	~ 3 E7 Btu/hr	~ 0.5%

As shown, the additional decay heat load imposed on the SFP cooling system during the final two refueling outages with the cask area rack installed represents less than one percent increase above the peak refueling decay heat load. This small amount of additional heat is considered insignificant when compared to the total heat load rejected by the plant during either outage conditions or normal power operation.

3. According to Section 9.6 of the submittal, all spent fuel and spent fuel storage racks will be removed from the cask area before a cask is brought into the area. Discuss how this restriction will be formally controlled.

<u>Response</u>:

Turkey Point Technical Specification 3.9.7 restricts the weight of any load carried over fuel assemblies in the storage pool to 2000 pounds. Because the cask area is an integral part of the spent fuel pool, this restriction applies to loads over fuel in the cask area rack as well as over fuel in the remainder of the spent fuel pool. Therefore, TS 3.9.7 alone prevents a cask from being lifted over fuel in the cask area rack. In addition to TS 3.9.7, the following administrative restrictions and design features ensure that a cask or other heavy load will not be handled over a loaded cask area rack:

1. The normally closed Spent Fuel Building sliding (L-shaped) door is administratively controlled and mechanically locked to preclude inadvertent, accidental, or other inappropriate hoisting of heavy loads in proximity to the spent fuel pool. The Nuclear Plant Supervisor (NPS) must grant permission and Plant Security must be informed prior to opening the door. The door is normally secured with four mechanical locks.

2. Operation of the Cask Crane is administratively controlled by a key-operated selector switch, and permission to operate the crane must be granted by the NPS.

Whereas Turkey Point has not initiated dry cask loading operations, the plant has not developed dry cask handling procedures. When developed, these procedures will ensure that the cask area is empty of fuel, racks, and debris prior to introducing a cask to the building.

From a practical standpoint, there is no reason for handling a cask over the cask area while the cask area rack is in place because there would be nowhere to set the cask down with the rack installed.

4. Describe the extent of station health physics technician (HPT) involvement and required direct coverage (continuous or intermittent) during the following evolutions (phases) of the project: (1) pre-job planning/briefings, (2) cask area pool-bottom vacuuming/cleaning, (3) rack installation, and (4) rack removal, decontamination, and storage.

Response:

HPTs involved with the project will attend the pre-job briefing as part of the rack installation/removal team and will provide radiological input during the briefing. Health Physics shift supervisors will lead and conduct the radiological briefings.

The job evolutions of pool vacuuming and cleaning, rack installation, and rack removal/ decontamination/storage involving the removal of material/equipment from the pool and Spent Fuel Building will have continuous HPT job coverage to assess present and potential radiological hazards. These controls include:

- Dose rate determinations, including underwater dose rate surveys
- Hot particle controls
- Contamination controls
- Radioactive material controls
- Air sampling evaluations/controls
- Foreign material exclusion controls
- Housekeeping controls

If contract personnel are involved in this work, FPL HP will oversee the contractors (including HP contractors, if any).

5. Section 9.4, page 9-2 of Holtec Report HI-2022931, "Spent Fuel Storage Expansion at Turkey Point Nuclear Plant for Florida Power and Light" (the Holtec report), appears to take credit for installed air monitoring equipment for identifying unexpected increases in airborne radioactivity during the rack project. In general, the NRC staff believes that these installed process monitors/systems are for providing appropriate radiation alarms, building ventilation isolations, quantifying radioactive effluents, etc., but are not appropriate for meeting the Title 10 of the Code of Federal Regulations (10 CFR) Part 20 survey requirements for monitoring occupational worker intakes of radioactive materials (installed air monitors are too slow in responding and do not provide representative sampling of local work areas).

   Describe how 10 CFR Part 20 air sampling requirements will be met and when representative samples of the workers' breathing zones will be taken. For example, will air samples be taken during out-of-the-pool decontamination of the rack (in preparation for interim storage)?

   Response:

   Health physics technicians (HPTs) will communicate any changes in radiological conditions to the work crews in the area. HPTs have the responsibility and authority to stop any work activity if it would result in the violation of radiological protection standards or would otherwise endanger the safety of personnel.

   The air sampling requirement for work in and around the spent fuel pool is achieved through the use of a continuous air monitoring system (AMS-4). An AMS-4 continuous air monitor is placed in the area to continually monitor and display the area air concentrations, with preset alarm set points to warn the workers.

   Based on the rack vendor's experience with similar new rack and re-racking projects, installation of the new rack should not generate any airborne release, because the rack is clean and is slowly lowered into the pool without disturbing activity in the pool water. Likewise, rack removal is not expected to create any significant airborne contamination because rack pressure cleaning will be conducted underwater and the rack will be monitored incrementally before being exposed to the air. Nevertheless, portable air samplers will be used in the vicinity of the cask area to complement the fixed monitors already in place, and HP will swipe and sample at regular intervals to make sure no contamination is becoming loose or airborne that could threaten workers. Air samples representative of workers' breathing zones will be taken during the rack decontamination process, in preparation for interim storage.

6. Describe all the types of radiation surveys performed by HPTs and when they will be performed. For example, will the HPT: (1) check external radiation levels of, and contamination on, materials or equipment removed from the pool, and (2) survey equipment as it breaks the surface of the pool to detect unexpected sources of high radiation?

Response:

HPTs perform the following radiation surveys:

- Pre-job and each shift radiation/contamination surveys

- Any item being removed from the pool will have a radiation survey performed as it breaks the water surface. Airborne concentrations in the area and external contamination levels are also checked when any item is removed from the pool.

- HP technicians will perform a tacky roller or masslin hot particle survey of the area each shift when work is in progress.

- HP technicians will perform RO-2 (or equivalent) hot particle surveys directly on the working individuals at approximately 2 hour intervals while in a hot particle area and upon exit from the hot particle area.

- Prior to releasing hot particle controlled areas, HP will perform a hot particle survey to ensure that there are no hot particles in the area.

Based on previous practice of the rack vendor, HP will perform the following radiation surveys during rack removal:

1) With the unloaded cask area rack sitting on the SFP floor, the inside surface of each rack cell will be pressure-washed with an extended wand to dislodge loose surface contamination. Then, an underwater survey with a cell probe will be performed to determine if any cells have rad levels significantly above 'background'. If any are found, the pressure washing will be repeated to attempt to reduce the radiation level of the affected cell region.

2) An underwater survey around the rack perimeter will be conducted before the rack breaks the water surface. Further pressure washing may be conducted if a problem area is found.

3) As the rack breaks the water surface, the rack will be hosed off with low pressure water and the rack perimeter will be surveyed at each 4' increment as the rack is raised above the pool surface. If a high survey reading is obtained, the rack may be re-submerged for additional pressure-washing to attempt to remove loose contamination.

4) An above-water survey will be conducted around and under the rack after the rack exits the pool, drains into the pool, and drip-dries over the cask area.

5) At this stage, rack outer surface contamination will also be evaluated using smears.

These surveys are suitable to detect sources of high radiation and contamination. In addition, rack lifting rigs and other handling equipment will be comparably surveyed. Once surveyed, any equipment that cannot be "free-released" will be dried and packaged for storage under appropriate radiological control. Based on the inaccessibility of some internal areas of the storage rack, the rack will not be suitable for free-release after it is used to store spent fuel. Once surveyed, a plastic bag (diaper) will be installed under and around the rack to catch any incidental drainage and remain with the rack during storage (to keep the storage container clean).

7. a. After completion of the rack installation project, does the licensee plan to store miscellaneous irradiated radioactive materials (MIRM) atop the rack? Examples of MIRM include activated portions of incore detectors/cabling, neutron start-up sources, or any other irradiated material that is usually stored underwater due to their high external radiation levels (e.g., greater than 5 rem/hour at 30 cm in air).

   b. If MIRM storage is allowed atop the fuel storage racks, describe the controls that would be established to limit the materials height above the fuel racks and the resultant external radiation level increases above and around the pool in the event of an inadvertent loss of pool water level (shielding).

<u>Response</u>:

FPL has no plans to store MIRM atop the cask area rack. However, if design changes were made to store MIRM on a platform above the rack, those changes would necessarily be subject to 10 CFR 50.59 evaluation prior to implementation to ensure that the consequences of all previously evaluated accidents are not increased more than a minimal amount and that no new accidents are created. The changes would also be subject to review under the ALARA program to ensure occupational doses are kept as low as reasonably achievable. As a baseline objective, the height of such a platform would be established to limit the spent fuel pool surface radiation levels to 15 millirem per hour as described in FSAR Table 11.2-5, "Refueling Shield Design Parameters".

8. Section 9.6, page 9-4 of the Holtec report describes the process of removing the rack and preparing it for storage. The decontaminating techniques discussed include rinsing with clean water, drip drying, and manually wiping the external surfaces.

Discuss the criteria (smearable contamination and/or external radiation levels) in place that would require more robust forms of decontamination (e.g., high-pressure hydro-lazing) to maintain the rack at manageable levels of external radiation/contamination.

<u>Response</u>:

Equipment that is found to be contaminated (> 50,000 dpm/100cm^2) is evaluated by Health Physics supervision for application of a more robust decontamination method. Any item being removed from the pool reading ≥ 1 R/hr on contact will be re-submerged and HP supervision notified. Efforts such as further underwater pressure washing will be utilized to reduce contamination and radiation levels. The goal is that removed items will have contamination levels less than 10,000 dpm/100 cm^2 and 100 mr/hr on contact, such that a high radiation area (>100 mr/hr) will not exist in the pool area when the rack is suspended above the water or during rack storage.

FPL does not intend to use hydro-lazing techniques because the rack vendor has achieved satisfactory results and more control using a high-pressure washer to flush the rack surfaces while the rack is still submerged. This pressure-washer provides a 3600 psi spray of water from a long pole that reaches inside the individual cells. Based on this underwater washing technique and the design of the Turkey Point cask area rack, which does not have open-ended flux traps to capture hot particles, the rack vendor does not expect the rack to develop pockets of contamination or hot particles lodged inside the rack structure.

The RWP and decontamination plan will establish reasonable objectives for contamination and radiation levels for the storage rack, subject to the ALARA program. FPL does not expect that the rack will be certified for free-release following this decontamination, based on the inaccessibility of some rack surfaces. Necessarily, the rack will be stored in a radiologically-controlled area, with appropriate protections and postings.

9. Discrete hot particles (fuel and/or activated corrosion and wear products) of sufficient activity to cause significant shallow-dose equivalent and whole body, deep dose exposures, can be present in SFPs (e.g., on fuel racks).

   a. Describe the survey program for identifying hot particles, minimizing their potential spread and, the measures that may be employed to ensure that workers decontaminating (wiping down) the rack for packaging and storage are protected from unexpected hot particle doses.

   b. Describe 10 CFR Part 19 worker training provided specific to rack installation including lessons learned by the contractor relative to past experience in SFP racking. Discuss whether this training will include the extremity dose hazards of improperly handling (e.g., picking up by hand) potential highly activated debris from the pool or during removal and preparation of the rack for storage. (For previous incidents of mishandling debris, see NRC Information Notice No. 90-47: "Unplanned Radiation Exposures to Personnel Extremities Due to Improper Handling of Potentially Highly Radioactive Sources.")

<u>Response</u>:

a. Hot particle survey methods are described in the response to Question 6. The extent of contaminated rack handling and the potential for exposure to hot particles is limited by two facts: (1) FPL does not expect to perform a rack removal until cask handling operations are necessary, which should not occur for several years, and (2) as discussed in the response to Questions 6 and 8 above, the rack surfaces and cell internals will be washed and monitored for contamination by HP before workers are allowed to get close to the rack for further decontamination and wrapping. The rack will be lifted slowly from the pool and monitored at lift increments to evaluate the decontamination process, and re-submerged for additional cleaning if necessary. Appropriate whole-body and extremity dosimetry will be provided to radiation workers during this process.

b. In conformance with 10 CFR 19, FPL will provide worker training specific to rack installation and removal, including a videotape program prepared by the rack vendor and lessons learned from other rack handling projects. The training will include the potential dose hazards of improperly handling activated debris from the pool or materials during removal and preparation of the rack for storage. Proper radiological work practices are discussed in the Turkey Point Radiation Protection Manual, and work controls in hot particle areas are specifically discussed in HP procedure 0-HPS-027.1.

10. a. The submittal notes that use of divers is not anticipated during the proposed rack installation. However, in the event that divers are needed, describe the procedural controls to be implemented to ensure that divers maintain a safe distance from any high and very high radiation sources in the pool. Guidance regarding procedural controls is provided in Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," Appendix A, "Procedures for Diving Operations in High and Very High Radiation Areas."

b. Describe pre-pool-entry radiation surveys of the dive area and how FPL plans to monitor the divers' doses (use of whole body and extremity dosimetry, remote readout (telemetry) radiation detectors, etc.).

Response:

a. It is FPL policy that the use of divers is a last resort that must be justified from an ALARA perspective. If divers are necessary, the operation will be administered under a plant procedure titled "Radiological Controls for Diving Operations". This procedure provides the controls to ensure that divers remain a safe distance from any high radiation sources, including performing pre-dive radiation surveys, establishing diver radiological stay times, and determining if high radiation area barricades or other warning devices will be needed to restrict diver access into high radiation areas.

If the radiation survey indicates the presence of a Very High Radiation Area, the length of the diver's safety lines, along with stay time, will be controlled as directed by HP Supervision to keep the radiation exposure as low as reasonably achievable. To further reduce the dose rate and risk of high exposure, consideration will be given to vacating adjacent rack areas of stored fuel to reduce the radiological sources in the dive vicinity.

b. If diving is necessary, plant procedures require a comprehensive pre-job radiation survey of the diving area, including surveying the following: (1) entry/exit area, (2) floors and walls of the work/travel path(s), (3) areas around or near physical barriers, and (4) any components the diver may encounter. In addition, a dive water sample is analyzed for gamma isotropic and tritium, and the diver(s) perform verification surveys of the work area with an underwater survey probe read remotely by HPTs at pool side.

Diver dose will be monitored via telemetry. This gives the HP technicians real time data of the dive and allows them to give proper and informed directions to the diver via the dive radio. The diver will also have a survey meter with him with a direct readout to the surface. The diver will do a survey of an underwater area prior to entry and the HP technicians on the surface will be able to read the results of the survey and give the diver proper direction. Additionally, the diver will wear multiple TLDs on his body to support official monitoring criteria for the dive.

11. The submittal described a methodology used to calculate the maximum effective multiplication factor (k_{eff}). The U.S. Nuclear Regulatory Commission (NRC) staff has outlined two acceptable methodologies to perform SFP criticality analyses in a memorandum entitled "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," from L. Kopp to T. Collins dated August 19, 1998. The two methodologies are: (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of the tolerance variations. The licensee's amendment is unclear on which methodology was used.

Identify the methodology that was employed to calculate the maximum k_{eff}.

<u>Response</u>:

As allowed in the referenced "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," the methodology employed to calculate the maximum k_{eff} combined both the worst-case bounding value and sensitivity study approaches. A discussion of how the reactivity effects of mechanical and material tolerances and uncertainties were combined is provided in the response to Question 12.

12. The licensee calculated maximum effective multiplication factors by statistically combining all of the reactivity effects due to tolerances and uncertainties for the Turkey Point SFPs. However, the submittal does not contain the equations used to calculate these values.

Provide the equations used to perform the maximum k_{eff} calculations and a detailed quantitative example demonstrating how the reactivity effects of each tolerance and uncertainty were calculated. The example should clearly and numerically demonstrate the methodology used to calculate the reactivity associated with each uncertainty or tolerance. Additionally, calculate the values presented in one of the reference cases of the amendment as the example. A detailed description of the statistical methods employed and the values used in the calculation of any statistical uncertainties should be included.

<u>Response</u>:

The following equation was used to perform the k_{eff} calculations:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(temp) + \delta k(uncert)$$

where

k(calc) = nominal conditions k_{eff}

δk(bias) = method bias determined from benchmark critical comparisons

δk(temp) = temperature bias

δk(uncert) = statistical summation of tolerance and uncertainty components

$$= [\text{tolerance}_{(1)}{}^2 + \text{tolerance}_{(2)}{}^2 + \text{uncertainty}_{(1)}{}^2 + \ldots]^{1/2}$$

As stated in the NBS-Handbook 91, the tolerances are defined as maximum permissible variations. Each parameter was investigated independently, the impact on k_{eff} from nominal was determined for each tolerance, and the results are presented below. This approach follows the format documented in "Guidance of the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," from L. Kopp to T. Collins dated August 19, 1998. Note that no burnup effect term is considered in this calculation because fresh (unburned) fuel is considered in the rack (i.e., no burnup credit).

For the requested numerical example demonstrating how the reactivity effects of each tolerance and uncertainty were combined, the criticality analysis case was chosen for the cask area rack reactivity without soluble boron credit (i.e., 0 ppm boron). The details of the reactivity effects for each tolerance and uncertainty and how they were calculated are provided below. A table following the discussion of parameters lists the reactivity value for each tolerance and uncertainty.

For each of the tolerance values, a case representing the nominal condition was first performed in CASMO-4. Then a specific calculation with a variation in the parameter of interest was performed in CASMO-4 to determine the reactivity effect of the tolerance. For the parameters associated with a tolerance, no statistical distribution was assumed. Conservatively, the full tolerance value was utilized to determine the maximum reactivity effect.

1) MCNP4a Bias Statistics — The MCNP4a Bias statistics represent the uncertainty or standard error associated with the bias in the form $K\sigma_{kaverage}$. The K value represents the one-sided statistical tolerance limits for 95 % probability at the 95 % confidence level for 56 critical experiments. The K value used for MCNP4a is 2.04 for this application. Appendix 4A, pages 2 and 3 of Enclosure 1 to the license amendment discusses this parameter in detail. Note that the MCNP4a Bias itself is applied directly to the calculated k_{eff} as the δk(bias) term.

2) MCNP4a Statistics — This value represents two times the standard deviation of the calculated k(calc). The standard deviation value is determined directly from the MCNP4a calculation. The 2σ value[2] provides a 95% probability at a 95 percent confidence level result.

3) Fuel density tolerance — CASMO-4 evaluations were performed with the nominal density and with the density increased to the tolerance limit.

4) Enrichment — CASMO-4 was used to evaluate the maximum enrichment tolerance of 0.05 w/o.

5) Fuel Rack Cell Inner Diameter — CASMO-4 was used to evaluate the impact of the fuel rack cell inner diameter tolerance.

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[2] Use of a 2σ value is conservative. The K multiplier is 1.84 for a one-sided statistical tolerance with 95% probability at the 95% confidence level corresponding to a sample size of 200.

6) Fuel Rack Wall Thickness — CASMO-4 was used to evaluate the impact of the rack wall thickness tolerance.

7) Flux Trap Water Gap — CASMO-4 was used to evaluate the reactivity impact of the rack cell water gap tolerance.

8) Boral™ poison loading — CASMO-4 was used to evaluate the reactivity impact of the Boral™ poison loading tolerance.

9) Boral™ width — CASMO-4 was used to evaluate the reactivity impact of the Boral™ width tolerance.

Table of Limiting Tolerance Values

Parameter	Tolerance Amount	Reactivity Value
MCNP Bias Statistics (95/95)	N/A	0.0011
MCNP Statistics (95/95, 2σ)	N/A	0.0016
Fuel density tolerance	± 2%	0.0022
Fuel enrichment tolerance	± 0.05%	0.0019
Rack Cell ID tolerance	± 0.04"	0.0008
Rack Wall Thickness tolerance	± 0.007"	0.0004
Water gap	± 0.08"	0.0096
Boral™ Poison Loading	± 8%	0.0025
Boral™ Width tolerance	± 0.0625"	0.0010

The δk(uncert) term is then calculated using a square root of the sum of the squares (SRSS) approach to statistically combine these reactivity values. These values may be combined using SRSS since they are independent (±) variables.

$$\delta k(\text{uncert}) = [0.0011^2 + 0.0016^2 + 0.0022^2 + 0.0019^2 + 0.0008^2 + 0.0004^2 + 0.0096^2 + 0.0025^2 + 0.0010^2]^{1/2}$$

$$\delta k(\text{uncert}) = 0.0106$$

The final k_{eff} for the unborated water case is:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(temp) + \delta k(uncert)$$

$$= 0.9414 + 0.0009 + 0.0033 + 0.0106$$

$$= 0.9562$$

13. The NRC staff has performed an initial review of the submittal and has concerns regarding the current regulatory licensing basis for the Turkey Point SFPs. After reviewing recent amendments and the Updated Final Safety Analysis Report, the currently described licensing basis is unclear whether the design of the SFP complies with 10 CFR 50.68 or 10 CFR 70.24.

Identify the current regulations and regulatory guidance that FPL considers its licensing basis for the SFPs. Additionally, describe how the proposed amendments will affect compliance with the regulations as described in 10 CFR 50.68 or 10 CFR 70.24. Finally, state how compliance with the regulations will continue if the proposed changes are approved.

Response:

The current licensing basis for the spent fuel pool is 10 CFR 70.24; including criticality accident monitoring and routine criticality drill requirements. Whereas 10 CFR 70.24 does not prescribe criticality limits, previous design basis criticality analyses have used the commonly-accepted criticality limits of 10 CFR 50.68(b) with credit for soluble boron. No previous commitment has been made to comply with 10 CFR 50.68(b), which explains why the FSAR has not previously been revised per 10 CFR 50.68(b)(8).

Upon implementation of the proposed amendment, the spent fuel pool licensing basis will be 10 CFR 50.68(b). As allowed by regulation, compliance with 10 CFR 50.68 exempts licensees from compliance with the criticality accident requirements of 10 CFR 70.24. The following discusses how Turkey Point will comply with each of the eight requirements in 10 CFR 50.68(b).

(1) *"Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water."*

Verbatim compliance with existing plant procedures will ensure safe subcritical conditions when handling and storing fuel assemblies, even under the most adverse moderation conditions feasible by unborated water. In the spent fuel pool, procedures specify handling a fuel assembly with the Spent Fuel Pool Bridge Crane or the New Fuel Elevator (or other transfer equipment); equipment that is limited in capacity to only one fuel assembly. No procedure allows simultaneous handling of more than one assembly with any particular handling device. The Spent Fuel Bridge Crane hooks are interlocked to prevent both hooks from being simultaneously loaded. Furthermore, procedures require that any fuel assembly moved in the spent fuel pool for storage must be placed in a storage location that complies with Technical Specifications. Thereby, procedures and design features prohibit fuel assembly configurations that may result in unsafe or critical conditions in the spent fuel pool.

Procedures specify lifting a fresh fuel assembly from its shipping container with the New Fuel Bridge Crane in preparation for placement in the New Fuel Storage Room. Procedures also specify moving a fuel assembly from the New Fuel Storage Room to the New Fuel Elevator using the New Fuel Monorail Hoist. No procedure allows simultaneous handling of more than one assembly with any particular handling device. Furthermore, Technical Specification Design Feature 5.6.1.2 ensures that any fuel assemblies placed in the New Fuel Storage Room racks will remain safely subcritical even under optimum moderation conditions. Thereby, procedures prohibit fuel assembly configurations that may result in unsafe or critical conditions in the New Fuel Room.

(2) *"The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used."*

Turkey Point Technical Specification 5.6.1.2 currently embodies the criticality criteria prescribed for 50.68(b)(2). In the unborated water (fully flooded) condition, TS 5.6.1.2.a requires a k_{eff} less than or equal to 0.95. To demonstrate conformance to this Technical Specification, a criticality analysis of the fresh fuel racks at Turkey Point was performed in 1999. The analysis showed that under a full density water flooding accident, the 95/95 basis k_{eff} of the fresh fuel was 0.92392. This value of k_{eff} meets the fresh fuel requirement to remain at or below 0.95 when flooded with unborated water.

(3) *"If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls*

and/or design features prevent such moderation or if fresh fuel storage racks are not used."

Turkey Point Technical Specification 5.6.1.2 currently embodies the criticality criterion prescribed for 50.68(b)(3). In the optimum moderation condition, TS 5.6.1.2.a requires a k_{eff} less than or equal to 0.98. To demonstrate conformance to this Technical Specification, a criticality analysis of the fresh fuel racks at Turkey Point was performed in 1999. For low density optimum moderation (water content which gives the highest reactivity of the storage array), the 1999 fresh fuel criticality analysis determined that the 95/95 basis k_{eff} was 0.84188. This value of k_{eff} meets the fresh fuel requirement to remain at or below 0.98 with low density optimum moderation.

(4) *"If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.*

"If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at 95 percent probability, 95 percent confidence level, if flooded with unborated water."

Turkey Point credits soluble boron in the spent fuel pool, so the latter criterion is adopted. The criticality analyses for the existing spent fuel storage racks are summarized in FSAR Appendix 14D Section 3.1.3. The analyses for Region I and Region 2 racks each require a k_{eff} less than 1.0 when flooded with unborated water, and a k_{eff} less than or equal to 0.95 when flooded with borated water. Turkey Point Technical Specification 5.6.1.1 requires that these k_{eff} limits be met. The minimum boron concentration required in the spent fuel pool water to meet this design criterion is 650 ppm.

As discussed in Section 3.2 of Enclosure 1 to the submittal, the criticality analysis for the new cask area racks is based on acceptance criteria of a $k_{eff} \leq 0.95$ when flooded with 200 ppm borated water and < 1.0 when flooded with unborated water.

(5) *"The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass."*

Turkey Point has a limited inventory of SNM other than nuclear fuel. The most significant constituent of this kind of SNM would be the movable incore detectors. However, common handling practices for these devices and engineering judgement indicate that, in their manufactured form, movable incore detectors will not comprise a critical mass.

(6) *"Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions."*

FSAR Section 11.2.3 describes area radiation monitors that are provided in the New Fuel Building and Spent Fuel Building at each unit to detect excessive radiation levels in the storage and handling areas for new and spent fuel. A high radiation level actuates a horn and a red flashing light locally to notify personnel and initiate appropriate safety actions.

(7) *"The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight."*

Turkey Point Technical Specification 5.6.1.1 limits the maximum enrichment of fuel assemblies to 4.5 weight percent of U-235.

(8) *"The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b)."*

Upon implementation of the proposed amendment, the Turkey Point FSAR will be amended to indicate that the licensee has chosen to comply with 10 CFR 50.68(b).

14. The licensee's amendment identifies Westinghouse 15 x 15 Optimized Fuel Assembly (OFA), Debris Resistant Fuel Assembly (DRFA) and low parasitic LOPAR spent and fresh fuel assemblies as the fuel types the new cask area racks are designed to accommodate. Therefore, only these fuel types were considered in the criticality analysis. The licensee stated that the Westinghouse 15 x 15 OFA and DRFA (referred to as the Westinghouse 15 x 15 OFA/DRFA assembly in the amendment) assemblies provided the most limiting reactivity conditions and were used in the licensing basis criticality analyses.

Specify whether any other fuel types (other than Westinghouse 15 x 15 LOPAR) are currently stored in either of the Turkey Point SFPs. If additional fuel types are stored in the pools, demonstrate quantitatively that the Westinghouse 15 x 15 OFA/DRFA assemblies provide the most conservative criticality analyses.

<u>Response</u>:

No other fuel types other than the three listed in the proposed amendment are stored in either Turkey Point spent fuel pool. Turkey Point has the following fuel types stored in the spent fuel pool:

- Westinghouse LOPAR fuel assemblies, also known as "standard fuel"
- Westinghouse OFA (optimized fuel assembly)
- Westinghouse OFA/DRFA (Debris Resistant Fuel Assembly)

The OFA/DRFA is the same as the OFA with the exception of containing a longer end plug for debris mitigation and the presence of axial blankets.

15. The results of the criticality analysis appear to apply to only the fuel types currently stored in the Turkey Point SFP. How will new fuel types be incorporated into the existing analysis, or will a new analysis be required?

<u>Response</u>:

The use of any new fuel type in the reactor core and the spent fuel pool will be evaluated as part of FPL's core reload design process. Procedurally, all core reloads are treated as design modifications and are subject to appropriate engineering reviews and 10 CFR 50.59 evaluation. If the 10 CFR 50.59 evaluation concludes that the new fuel type can be implemented without prior NRC approval, the change will be implemented and the FSAR will be revised pursuant to 10 CFR 50.71(e). Otherwise, the introduction of a new fuel type will be submitted under 10 CFR 50.90 for NRC approval.

16. The licensee's criticality analysis has determined that the misloading of a fresh fuel assembly into the corner cell intended to be used to store the fuel handling tool requires 624 parts per million of soluble boron to assure the maximum k_{eff} does not exceed 0.95. The licensee stated that this misloading event provided the bounding criticality accident condition because the cell does not contain Boral panel inserts and was not intended to contain a fuel assembly.

Identify the controls in place or planned to prevent misloading of a fresh fuel assembly into the corner cell.

Response:

To prevent misloading a fresh fuel assembly into the corner cell, administrative controls on fuel handling operations provide a defense-in-depth, including independent verification during the design stage and independent verification during the loading operation. Prior to fuel movement, Nuclear Engineering prepares a specific move-sheet using computer programs that identify restricted cells. Several cells in each pool are already identified as restricted in these computer programs. When the cask area rack is included in the computer programs, the corner cell in the cask area rack will be identified as another restricted cell. Once prepared, the move-sheet is independently verified by another qualified engineer. Fuel movement is then performed by a qualified crane operator under the direction of a Senior Reactor Operator using the specific move-sheets. Each fuel assembly and target location are identified and independently verified prior to placing the fuel assembly.

In addition to the defense-in-depth, operator awareness of this corner cell will help prevent misloading into the cell. The corner cell will be the normal storage location for the fuel handling tool. A cantilevered tool storage bracket will be located above the corner cell on the pool wall. At the onset of fuel handling operations during a refueling, the operator will first grapple the fuel handling tool from its storage bracket and recognize that the tool will have to be returned to that location upon completion of operations. The presence of the bracket over the corner cell will provide a continuous visual reminder to the operator that this location is reserved for the fuel handling tool, and not intended for fuel assemblies. Therefore, as a practical matter, a cognizant operator will not put a fresh fuel assembly in the corner cell; recognizing that the space is reserved for the fuel handling tool.

17. Section 4.1 of the Holtec report states that an infinite radial array of fuel assemblies was assumed in the analysis "except for . . . certain abnormal/accident conditions where neutron leakage is inherent."

Provide a table of all abnormal/accident events analyzed. The table should identify whether an infinite radial array was assumed for each event. Additionally, for events where an infinite radial array was not assumed, provide a justification for why it was not assumed, and what conservative assumptions, with accompanying justification, were made instead.

Response:

An infinite radial array was used for all analyses with the exception of the evaluation of a misloading of a fresh fuel assembly in the vacant corner rack cell intended to be used to store the fuel handling tool. For this analysis, the cells surrounding the corner cell were conservatively assumed to contain fresh fuel assemblies in both the cask area rack and the adjacent Region I rack. Consistent with the geometry of the cell for this condition, the evaluation took credit for the water gap between the rack and the wall. This is acceptable since it represents an actual physical configuration.

The abnormal/accident events analyzed are listed in the table below, together with the characterization of the radial modeling assumption for each case. The conservative assumptions used in the analyses are the same as those listed in Section 4.1 of Enclosure 1 Appendix 1 to the submittal.

Abnormal/Accident Conditions	Radial Model
Temperature Increase	Infinite
Void (boiling)	Infinite
Assembly Drop	Infinite
Lateral Rack Movement	Infinite
Mislocation of a Fresh Fuel Assembly outside Cask Area Rack	Leakage considered

18. Section 4.5.4 of the Holtec report described the modeling of the inter-rack gap between cask area racks and Region 2 racks. The report stated, "These calculations are also valid for the rack-to-rack interaction between the cask area rack and the Region 1 racks as the Region 1 racks are licensed to the same regulatory limits as the Region 2 racks." Although the NRC staff agrees that the racks are licensed to the same regulatory limits, the licensee is permitted to store higher reactivity (i.e., fresh) fuel in the Region 1 racks. Therefore, it is reasonable to assume there may be greater interaction between Region 1 racks and the cask area rack than between the Region 2 racks and the cask area rack.

Either provide a discussion regarding the interaction between the Region 2 racks and the cask area rack as the limiting interface condition or reanalyze the pool to consider the Region 1 interaction with the cask area rack.

Response:

Figures 1.1.1 and 1.1.2 provided in Enclosure 1 Appendix 1 to the submittal show the arrangement of storage racks in each unit's spent fuel pool. The new cask area rack faces Region 1 racks to the north and south in both units, and Region 2 racks to the west. In the figures, the Region 1 racks are characterized by the larger rack pitch and a water gap between adjacent cells, whereas the Region 2 racks have a smaller pitch and no water gaps between cells.

The adjacent region with the highest k_{eff} was chosen to evaluate the cask area rack's interaction with these adjacent racks. While Region 1 can store fresh fuel assemblies with enrichments up to 4.5%, its overall k_{eff} is less than the Region 2 racks loaded with fuel assemblies that meet the Technical Specification Table 3.9-1 requirements of initial enrichment and burnup. The k_{eff} of the Region 1 racks is lower because of their larger nominal pitch of 10.6" and four Boraflex panels per cell, and also because the boron-10 areal density in the Region 1 panels is greater than the density in the Region 2 panels (0.020 g/cm^2 and 0.012 g/cm^2, respectively). The Region 2 rack pitch is a nominal 9.0" with only two equivalent Boraflex panels per cell. Since the Region 2 k_{eff} is higher than Region 1, the reactivity interface between Region 2 and the cask area rack is the bounding rack-to-rack interaction case.

19. Section 4.5.4 of the Holtec report described how the analysis of the inter-rack gap was performed. The report stated, "The reactivity of the inter-rack gap calculation was bounded by the maximum of the two infinite array calculations."

Provide a table listing the results of all the calculations performed to support this conclusion. Additionally, include a more detailed description of how the analysis was performed, specifying any assumptions used in the calculations, how the calculations were compared, and how the most limiting condition was identified.

Response:

The basis for selecting Region 2 as the limiting rack for the inter-rack gap calculation is described in the response to Question 18.

For the inter-rack gap evaluation, the k_{eff} of the Region 2 rack was determined for different fuel enrichments. Spent fuel in Region 2 is represented by low enrichment fresh fuel. A similar calculation was performed for the cask area rack. Then a k_{eff} calculation for the rack interface was performed by assuming a Region 2 rack with no Boraflex on the east outside face, a 2" water gap, and the cask area rack with Boral on its west outside face. The interface is acceptable provided the k_{eff} of the entire system is statistically the same or lower than the maximum of either Region 2 or the cask area rack by itself. The following table presents a summary of the results for both 0 ppm and 200 ppm boron cases:

Case Description (0 ppm)	keff	Standard Deviation	Δk margin
Cask Area Rack with 4.5% Fuel	0.9414	0.0008	
Existing Region 2 Rack with 1.4% Fuel	0.8873	0.0006	
Cask Area Rack & Region 2 with 2" water gap	0.9369	0.0009	- 0.0045
Cask Area Rack with 4.5% Fuel	0.9414	0.0008	
Existing Region 2 Rack with 1.6% Fuel	0.9315	0.0006	
Cask Area Rack & Region 2 with 2" water gap	0.9361	0.0007	- 0.0053
Cask Area Rack with 4.5% Fuel	0.9414	0.0008	
Existing Region 2 Rack with 1.8% Fuel	0.9706	0.0007	
Cask Area Rack & Region 2 with 2" water gap	0.9656	0.0006	- 0.0050

Case Description (200 ppm)	keff	Standard Deviation	Δk margin
Cask Area Rack with 4.5% Fuel	0.9165	0.0008	
Existing Region 2 Rack with 1.4% Fuel	0.8445	0.0005	
Cask Area Rack & Region 2 with 2" water gap	0.9148	0.0008	- 0.0017
Cask Area Rack with 4.5% Fuel	0.9165	0.0008	
Existing Region 2 Rack with 1.6% Fuel	0.8912	0.0006	
Cask Area Rack & Region 2 with 2" water gap	0.9136	0.0008	- 0.0029
Cask Area Rack with 4.5% Fuel	0.9165	0.0008	
Existing Region 2 Rack with 1.8% Fuel	0.9283	0.0007	
Cask Area Rack & Region 2 with 2" water gap	0.9264	0.0007	- 0.0019

Note that the table includes Region 2 equivalent-initial enrichments (i.e., no burnup) up to 1.8%, which is greater than that allowed by the current Technical Specifications. Technical Specification 3.9.14 and Table 3.9-1 limit the initial enrichment of unburned fuel stored in Region 2 to 1.6%. Analysis of 1.8% fuel was done to investigate the reactivity effects at bounding higher enrichments, demonstrating that the reactivity acceptance criteria are still met at an enrichment above the Technical Specification limit.

20. Table 4.5.1 in the Holtec report presented the reactivity effects of the manufacturing tolerances considered in the criticality analysis.

Does Table 4.5.1 comprise the complete list of all tolerances considered? If so, justify why tolerances on other parameters, such as those in Table 4.1.1, were not included. Provide detailed quantitative information to support the exclusion of any parameters from the calculation of the maximum effective multiplication factor. If exclusion of these parameters results in a nonconservative maximum effective multiplication factor, provide additional information describing the net maximum reactivity effect, how this effect was quantified, and how these parameters are either physically or administratively controlled to prevent changes in their reactivity effect in the future.

If not, discuss whether the table should be amended to include all tolerances analyzed. This discussion should include a complete list of the tolerances.

Response:

Table 4.5.1 contains the list of all tolerances considered in the submitted analysis. Other manufacturing tolerances not considered were fuel assembly parameters for rod pitch, pellet diameter, cladding thickness, and guide tube thickness. These tolerances were not considered because their combined reactivity effect is negligible (i.e., < 0.001) compared to the tolerances that were considered.

To determine the combined effect of the above manufacturing tolerances, a sensitivity calculation was performed using the submitted analysis (for the unborated case). The results are summarized in Enclosure 6 to the transmittal letter for this response. When the reactivity effect of the above tolerances is statistically combined with the tolerance effects that were considered, the overall change to reactivity has been calculated to be + 0.0002. This demonstrates that the net effect of these other tolerances on reactivity is negligible, and may therefore be neglected in the analysis.[3]

[3] The combined effect of the manufacturing tolerances considered in the criticality analysis is 0.0104. This value is derived from a statistical sum of the parameter tolerances listed in Table 4.5.1 of the Holtec Report. This value also corresponds to the combined effect of the tolerances listed in the Question 12 response, when the two MCNP tolerances are not considered.

21. Section 3.2 of the submittal presented a summary of the criticality analyses performed. The licensee stated "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."

Provide a table summarizing the differences between the cask area racks, SFPs designs, currently installed spent fuel storage racks, and any other factors that will affect the criticality analysis. Additionally, for each difference identified, describe which condition was used in the criticality analyses, including a detailed justification for why it represented the most limiting condition.

Response:

As shown in Figures 1.1.1 and 1.1.2 in Enclosure 1 Appendix 1 to the submittal, the Turkey Point spent fuel pool layouts are symmetrical and are therefore considered identical with respect to the criticality analysis for the cask area rack. From a reactivity standpoint, the only physical differences between the units are the length of the Boraflex panels in the existing racks and the nominal gap between the cask area rack and the east wall or adjacent racks. The criticality analysis used the smaller Unit 3 Boraflex panel length, and a conservative two-inch gap size surrounding the rack on all sides, which is smaller than the nominal gaps shown below.

	Unit 3	Unit 4
Configuration	Mirror image	Mirror image
Boraflex panel length	139.4 inches	141.4 inches
Nominal gap between Cask Pit Rack and wall or adjacent racks	2.4 inches (E & W) 3.3 inches (N & S)	2.5 inches in all four directions

22. Section 3.2 of the submittal states "Because the interaction analysis assumed a minimum 2-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." Figure 1.1.1 "Unit 3 Spent Fuel Pit Layout" shows a nominal rack spacing of 2.4 inches on the western edge of the new cask area rack. Additionally, Figure 1.1.2 "Unit 4 Spent Fuel Pit Layout" shows nominal rack spacings of 2.5 inches on the northern, southern, and western edges of the new cask area rack. The submittal also stated that the baseplates extend 1/4-inch beyond the rack module periphery wall and "act to center the rack in the cask area and establish the required minimum separation between the rack and the surrounding racks or wall." This 1/4-inch spacing will not provide the 2-inch gap assumed in the analysis.

As the criticality analysis contains a limited gap margin (less than 0.5 inches) on multiple interfaces, describe all controls that will be used to ensure that the 2-inch margin assumed will be provided. If physical properties of the racks will provide the 2-inch gap, provide a figure depicting their location and how they will function to ensure the proper spacing.

Response:

During rack installation, the minimum 2-inch spacing between the cask area rack and each adjacent rack will be verified using a 2-inch go/no-go gauge. The rack baseplate protrusion will not be relied on for this minimum spacing. The 2-inch go/no-go gauge measurement will be an independent verification established as a Quality Control (QC) hold-point in the written installation procedure.

The submittal statement regarding the rack baseplate acting to "center the rack" in the cask area is incomplete. The baseplate will only provide a minimal stand-off, and it is the go/no-go gauge that will centrally position the rack by establishing the required minimum separation distance between the rack and the surrounding racks. Additionally, rack guides that protrude two inches are installed near the top corners of the rack. Though not specifically credited for ensuring the minimum two-inch spacing, the guides do provide additional assurance that the minimum gap will be maintained.

Whereas the rack faces will be separated by the minimum 2 inches, the structural members may be provided less separation. For example, the baseplate may not be separated from the adjacent rack by 2 inches due to the 1/4-inch baseplate extension. However, this baseplate protrusion will not affect the criticality analysis because the baseplate is made of non-fissionable material and protrudes into an area below the active fuel region of the host fuel.

23. Section 3.2 of the submittal states, "the rack cells employ Boral neutron absorber panels mounted on the outside faces of stainless steel boxes... (except cells on the rack periphery facing the east SFP wall, which contain no Boral panel on the outer face) ...." Since the cask area rack is not symmetrical with respect to neutron absorption properties, the proper orientation in the pool becomes crucial. The NRC staff has identified this event as a potential new accident for which the licensee has not performed a criticality analysis.

As improper orientation of the rack could result in a higher accident k_{eff} than misloading a single fuel assembly, perform a criticality analysis of the effects of improperly orienting the cask pit rack within the SFP and then, subsequently, fully loading it with fresh fuel. Additionally, identify all rigorous controls that will be implemented to reduce the likelihood this accident will occur.

Response:

From a safety perspective, the improper orientation and subsequent misloading of the cask area rack during rack installation and fuel loading operations is unlikely and nearly impossible for the following reasons.

- The cask area rack design includes an excluded corner cell designated for the storage of the fuel handling tool, making the rack asymmetric. A portion of the corner cell's two outer faces has been removed to accommodate storage of the tool. This feature establishes a unique orientation for the cask area rack that is easily verified through visual observation prior to lowering the rack into the pool. Orienting the rack in any configuration without the vacant corner cell visible facing the pool wall would be apparent to the installation crew and a violation of the installation procedure.

- The cask area rack has a rectangular footprint. Because of the close tolerances between the cask area dimensions and the rack dimensions, the only possible misorientation of the rack is by 180 degrees (i.e., backwards). This would result in the excluded corner cell being installed in the center of the spent fuel pool as opposed to along the east wall. Again, this misorientation would be apparent to the installation crew and a violation of the installation procedure.

- The rack installation procedure will require a Quality Control holdpoint to verify proper rack orientation before immersion into the pool. This holdpoint provides an independent verification that the rack is properly oriented. The procedural controls during rack installation and the independent verification of proper rack orientation provide a reasonable defense-in-depth to ensure that a rack misorientation will not occur.

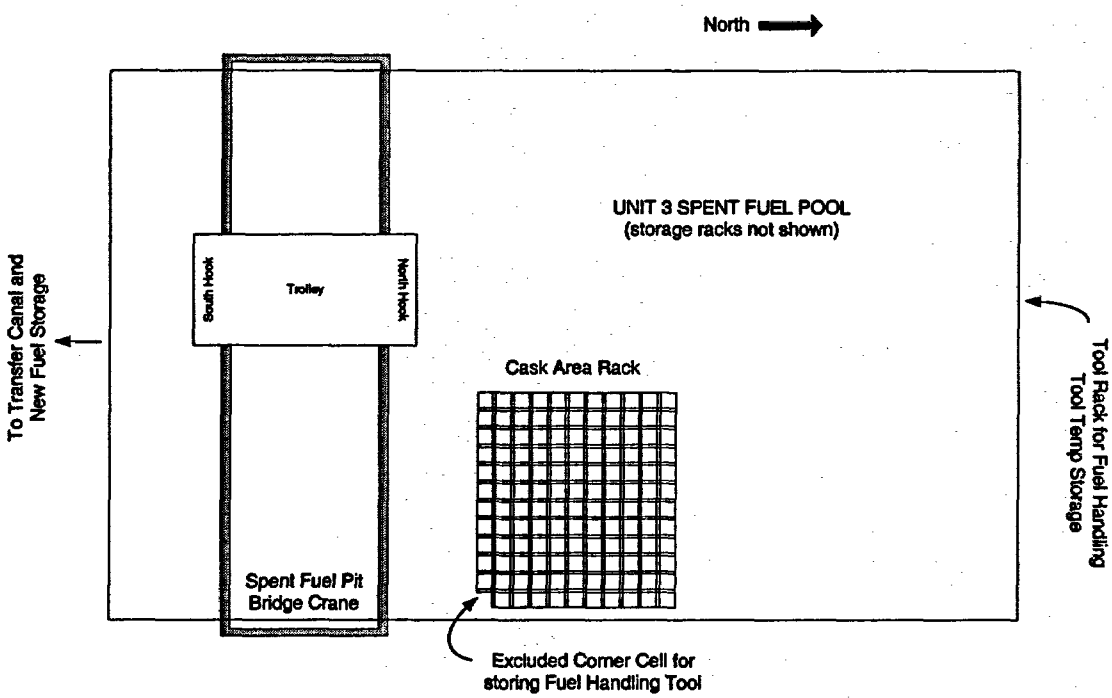
- Even assuming that the cask area rack was installed backwards, the misloading of fresh fuel into the rack would be nearly impossible due to the storage location of the fuel handling tool with respect to the fuel transfer canal (where the new fuel would originate) and the physical limitations of the spent fuel bridge crane. Due to limitations in bridge travel, the hoist on the tool tree side of the bridge cannot access the fuel transfer canal and the hoist on the transfer canal side cannot access the tool tree. Transfer of the fuel handling tool from one hoist to the other requires the use of the handling tool bracket mounted along the east wall by the cask area (corresponding to the excluded cell). The bracket provides a temporary location for the tool so that the tool can be released from one hoist and then attached to the other.

- During rack installation, the fuel handling tool would reside on the tool tree on the opposite side of the spent fuel bridge from the fuel transfer canal. Upon completion of rack installation, the fuel handling tool bracket is aligned and installed. As a note, misorientation of the rack would be obvious during alignment of the bracket. In order to retrieve a fresh fuel assembly, the fuel handling tool must pass from the tool tree side to the transfer canal side of the spent fuel bridge crane via the bracket. With the rack misoriented, the lowering of the fuel handling tool onto the bracket becomes very difficult if not impossible due to the presence of a complete boxed cell. The misorientation of the rack would be obvious and new fuel movement would therefore not occur.

- Since the hanger support for the fuel handling tool on the east wall would be blocked by the misorientation of the new rack, relocating the handling tool to its hanger (which is necessary prior to handling new fuel) would not be possible. In addition, operator awareness and administrative controls (see description of the move-sheet process in the reply to Question 16) would further preclude loading a misoriented Cask Area Rack. Accordingly, the rack misorientation and subsequent fresh fuel handling are independent and unlikely events. Therefore, the described accident scenario would be excluded by the Double Contingency Principle and is outside the licensing basis.

Based on the above reasons, the improper orientation of the cask area rack is not considered credible and additional criticality analyses for rack misloading events are not warranted.

Please refer to the figure and discussion below for a further explanation of the movements of the fuel handling tool necessary to first install the rack and then to load the rack with fuel.



Unit 3 Configuration (Unit 4 is a mirror image)

A tool rack is located in the north end of the Spent Fuel Pool (SFP). This tool rack has a bracket for the fuel handling tool and other tools. Another storage bracket for the fuel handling tool is provided along the east wall, in the cask area. This second location is provided to allow transferring the fuel handling tool between hooks on the SFP Bridge Crane, as discussed below.

The SFP Bridge Crane has two hooks; a north hook and a south hook. A single hook cannot reach all locations in the pool because of the physical width of the trolley and the travel limits at the end of the bridge rails. Fuel can only be introduced into the pool through the new fuel elevator or from the containment (fuel transfer cart / upender), both located in the transfer canal at the south end of the pool. To move assemblies from the fuel transfer canal to the spent fuel pool / cask area rack, the fuel handling tool needs to be attached to the crane's south hook.

Because the fuel handling tool will be stored on the tool rack on the north end of the pool during cask area rack installation, it cannot be immediately attached to the south hook. First, the tool must be attached to the north hook, then moved to the bracket in the cask area (east wall). The bridge is then moved north, and the tool is attached to the south hook. At this point, the south hook can access new fuel in the transfer canal and access most of the spent fuel pool, including the cask area.

24. Experience shows that upon initial installation (i.e. contact with water) Boral™ releases hydrogen gas. The buildup of hydrogen gas has been known to cause bulging and deformation of the cells that form the fuel storage rack.

What design features are in place on the proposed cask area spent fuel storage racks to liberate hydrogen gas from the rack cells?

Response:

Boral™ panels are held in place on the external face of cells in the new racks by stainless steel sheathing covers. The new Region I racks contain a water gap between adjacent cells. As such, these panels will either face the water gap between adjacent cells or be attached to the outer rack surface. The stainless steel sheathing that covers the panels is attached to the cell boxes using spot and/or intermittent welds. The gaps between these welds allow any hydrogen gas produced to escape freely from the Boral pocket. Released hydrogen will enter the water gaps or enter water surrounding the rack. Therefore, no bulging or cell deformation is expected. If bulging were to occur, the protrusion would only affect the gaps between cells and would not interfere with the placement of fuel into the storage cells. The metal matrix would remain fundamentally intact and bulging would not affect the Boral neutron poison capability to any significant extent.

25. Enclosure 1, Section 3.5 of the submittal states that:

> Section 3.5 in Appendix 1 details the defense-in-depth approach taken to ensure that the handling of racks by the cask handling crane will comply with the NUREG-0612 guidance.

However, the discussion in Section 3.5 in Appendix 1 is limited to the general guidance provided in Section 5.1.1 of NUREG-0612. Section 5.1.2 of NUREG-0612, "Spent Fuel Area – PWR [Pressurized-Water Reactor]," recommends that, in addition to satisfying the general guidelines of Section 5.1.1, one of the four criteria outlined in Section 5.1.2 should be met.

Describe how the Spent Fuel Cask Handling Crane meets the guidelines of Section 5.1.2 of NUREG-0612.

<u>Response</u>:

Of the four methods described in NUREG-0612 Section 5.1.2 to provide assurance that the guidelines of Section 5.1.1 are met for heavy loads lifted by the cask handling crane in the spent fuel pool area, Turkey Point analyzes the effects of dropping a heavy load (the fourth method). The cask drop analysis is discussed in UFSAR Section 14.2.1.3 and Appendix 14D Section 5.3.1.2.2.

The cask crane is not single-failure-proof (eliminating the first method), and the location of the cask area within each unit's spent fuel pool precludes meeting the horizontal separation distances between the cask travel path and spent fuel in the pool that are recommended in the second and third methods.

In reviewing the Turkey Point response to NUREG-0612 and the corresponding NRC SER, an explicit commitment to meet the fourth method [Section 5.1.2(4)] was not found. However, the fourth method is satisfied by the existence of a cask drop analysis that pre-dates NUREG-0612 in the plant licensing basis. The drop analysis found in UFSAR Section 14 addresses the structural and radiological effects of a postulated drop, including assumed damage to 157 fuel assemblies in the spent fuel pool (see UFSAR Section 14.2.1.3).

26. Enclosure 1, Section 3.5 of the submittal states the following:

> 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.

Describe the criteria that were used to determine the cells to be vacated. Include whether this vacating of cells will ensure that the movement of the racks over or within 25 feet horizontal of the "hot" spent fuel will be prevented as recommended in Section 5.1.2(3)(b) of NUREG-0612.

<u>Response</u>:

FPL expects to vacate one row of cells in the adjacent Region 1 fuel racks in the north and south directions and also one row in the adjacent Region 2 racks to the west. Vacating one row surrounding the cask area will create a horizontal separation distance ("shadow") of approximately 12 inches between the fuel assemblies stored in the pool and the projected vertical lift envelope of the cask area rack. This distance will provide a reasonable margin to ensure that a postulated rack drop, if it occurred, will not impact stored nuclear fuel. This "shadow" is comparable to that routinely provided for temporary cranes used for re-racking projects, with due consideration that the Turkey Point cask crane is permanently installed and will be constrained by limit switches and the physical limits of the roof hatch.

The minimum 25-foot horizontal separation distance recommended in NUREG-0612 Section 5.1.2(3)(b) does not apply to Turkey Point. As discussed in the NUREG, the recommendation is intended for pools that are large enough to maintain wide separation between a heavy load and "hot" spent fuel, in conjunction with cranes that lift the heavy load no more than 6 inches above the operating floor to preclude rolling. Neither of these restrictions can be met by the Turkey Point pool and crane designs. The new rack will be brought into the unit Spent Fuel Building through a roof hatch and lowered into the cask area that is part of the spent fuel pool. It is not possible to provide the 25 foot separation distance within the pool.

A cask drop analysis that predated NUREG-0612 satisfies the guidance in Section

5.1.2(4). The cask drop analysis bounds the consequences of a rack drop, because the rack weighs much less than a cask, and if the rack were to drop, the rack's honeycomb structure would absorb the impact, causing less damage than dropping a solid cask. The consequences of a cask drop and a rack drop are discussed in response to Question 27. With a vacated row of cells in the surrounding racks, a cask area rack dropping vertically would not be expected to cause any damage to fuel assemblies stored in the pool.

27. Enclosure 2, page 3 of the submittal states:

> The probability and consequences of a heavy load drop of the cask area rack are bounded by the existing cask drop analyses. The consequences are not adversely affected because a fuel transfer cask is much heavier than the empty rack.

a. Discuss whether the dropping of the cask on the SFP liner was analyzed. If so, describe the results, and explain how you plan to limit consequences if the perforation of the liner occurs. If not, explain why this is not a credible event at Turkey Point.

b. Discuss the effect of dropping the cask on the pool structure.

<u>Response</u>:

a. The dropping of the 25-ton spent fuel cask on the spent fuel pool liner has been previously analyzed and the consequences were found to be acceptable. Turkey Point UFSAR Appendix 14D, Section 3.4.3 states:

> "In the event that a rack should drop on the floor, the potential for loss of pool cooling could be postulated. An analysis has previously been submitted and accepted by the NRC [Reference letter from G. Lear, NRC, to R. E. Uhrig, FPL, dated July 9, 1976] for dropping of the spent fuel cask. The results of this analysis demonstrated that the pool floor would remain elastic during impact and that a crack would not develop. This cask weighs substantially more than a single rack assembly and has a smaller cross sectional area for load distribution. The loss of pool water inventory from a rack drop is bounded by this previous analysis for loss of pool water inventory from a cask drop. Therefore, loss of spent fuel cooling from loss of pool water inventory will not occur as a result of a rack drop."

For the reasons described above, spent fuel pool integrity will be maintained if a cask area rack drop occurred.

b. The effect of dropping the 25-ton spent fuel cask on the pool structure would be limited to the local crushing of concrete underlying the stainless steel liner plate at the point of impact. As described in a. above, the floor will absorb the impact elastically and will not crack.

28. Enclosure 1, Section 3.5 of the submittal states that:

> 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 .... .

Provide details regarding this temporary hoist. Explain how the hoist will be used, and what industrial standards it meets.

Response:

The hoist proposed for rack installation is an electric hook-to-hook style chain hoist rated for 37.5 metric tons (>82,500 lbs). The hoist is suspended from the cask crane main hook, such that when the rack is placed in the spent fuel pool, the lower block and chain of the temporary hoist will be submerged rather than submerging the cask crane main hook and wire rope. The hoist is suspended from the cask crane main hook by rigging that is sized to provide a safety factor compliant with NUREG-0612 (10:1 or 5:1 with redundancy) based on the weight of the lifted load. The hoist itself also meets the safety factor recommendation of NUREG-0612 for this particular rack lift based on the rack weight (approximately 35,800 lbs) as compared to the 82,500 lbs hoist rating and its design safety factor of 5:1. NUREG-0612 Section 5.1.6 stipulates that if a single lifting device is used, it should have twice the safety factor of redundant (dual) lifting devices. Because the hoist is rated for greater than twice the lifted load and has a design safety factor of 5:1, the safety factor of the single hoist relative to the cask area rack load is greater than 10:1.

The hoist is built to ASME B30.16 and inspected to ASME B30.2 during its life.

29. Standard Review Plan (SRP), Section 9.1.2.III.e states:

> Conventionally the plant's Technical Specification states that the weight of all loads being handled above stored spent fuel shall not exceed that of one fuel assembly and its associated handling tool. This weight and its normal carrying height above the storage racks establishes what was considered the upper bound on the potential energy available to damage the stored spent fuel if a load drop occurs. It has been subsequently noted that lighter loads handled at greater drop heights may have greater amounts of potential energy.

a. Explain whether the potential energy associated with the weight of loads being handled above stored fuel has been considered, given the likely occurrence of greater damage should such a load be dropped from a higher height than established.

b. Describe the presence of any control measures that would prevent this type of occurrence.

<u>Response</u>:

Turkey Point Technical Specification 3.9.7 prohibits loads in excess of 2000 pounds from travel over fuel assemblies in the storage pool. This specification will apply to any loads carried over the cask area rack (when fuel is in the rack), because the cask area is an integral part of the spent fuel pool.

The potential energy of non-fuel weights that could be handled above stored fuel has not been specifically evaluated in the Fuel Handling Accident analysis found in FSAR Section 14.2.1. The analysis considered the dropping of a single fuel assembly onto a flat surface, another assembly, or a sharp object. Although none of these drop scenarios are predicted to cause a fuel cladding integrity failure either to the dropped assembly or to any other stored assembly, the radiological consequences of non-mechanistically damaging one row of 15 rods as well as all the fuel rods in one assembly were evaluated for conservatism.

In a 1978 FPL letter[4] to NRC, a list of objects moved over the spent fuel storage pool was provided, including their approximate carrying height. For the objects listed, the carrying height above the spent fuel racks ranged from 5 feet for spent fuel assemblies to 20 feet

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[4] FPL letter L-78-324 to NRC dated 10/4/78, Control of Heavy Loads, Response to Question 2

for fuel handling tools with an approximate weight of 800 pounds. The 25-ton spent fuel shipping cask was also listed at 26 feet, but TS 3.9.7 prohibits this type of load over fuel assemblies.

The movement of a non-fuel weight over the spent fuel pool at any height that would produce greater fuel damage than the current analysis of the fuel handling accident is not postulated. The spent fuel pit bridge crane is designed to limit maximum lift height to maintain a safe shielding depth while handling fuel assemblies. By procedure, any object other than a fuel assembly or its handling tool that is to be carried over spent fuel must be logged and verified to be less than the 2000 pound limit in TS 3.9.7. In addition, dropping any object into the SFP that could exceed the fuel damage assumed for a fuel handling accident is not postulated because the handling of any non-fuel load not previously evaluated would be subject to 10 CFR 50.59 criterion (c)(2)(iii) such that the consequences of the accident could not be increased more than a minimal amount without prior NRC approval.

Non-fuel loads that are currently handled by the spent fuel pit bridge crane over irradiated fuel assemblies are fuel handling tools (such as funnel tools and cameras), L-inserts, and baskets for trash and surveillance capsules. None of these non-fuel load weights are comparable to the weight of a fuel assembly. Test weights exceeding 2000 pounds that are used for calibrating the spent fuel pit bridge crane overload circuitry are only handled in non-fuel areas. These test weights are not allowed to be carried over irradiated fuel.

In summary, the combination of crane operation that limits the load height, 10 CFR 50.59 evaluation requirements for non-fuel loads that could potentially cause fuel damage, and procedural controls to verify that the weight of any non-fuel load is within the plant Technical Specifications means that non-fuel load drop scenarios that could cause fuel damage exceeding a fuel handling accident are not postulated to occur over irradiated fuel.

30. a. Describe the controls to prevent the inadvertent draining of the SFP water level below a height of approximately 10 feet above the top of active fuel in the event of a failure of inlets, outlets, piping, or drains (SRP Section 9.1.3).

b. Explain how, for all planned offloads, the SFP water level is maintained. Assume a worst-case active component failure for SFP cooling.

<u>Response</u>:

a. As described on UFSAR page 9.3-8, the fuel pool cooling pump suction line penetrates the spent fuel pool wall above the height of the stored fuel assemblies. This penetration location prevents the loss of water resulting from a potential suction line rupture. Whereas the elevation may not satisfy the guideline of SRP Section 9.1.3, an NRC Safety Evaluation Report dated March 15, 1972 found that the piping is arranged so that the failure of any pipe would not drain the pool below a level six (6) feet above the tops of the fuel elements; and is therefore acceptable. The spent fuel pool cooling return line has a one-half inch hole in the pipe near the normal level that serves as a siphon break.

b. Technical Specification LCO 3.9.11 requires that SFP water level shall be maintained greater than or equal to elevation 56'-10" whenever irradiated fuel assemblies are in the storage pool. The water level is verified at least once per seven days. If level were to fall below this minimum elevation, normal makeup water sources would be used to restore the level above the TS-required minimum level.

As evaluated in license amendments 223/218, the worst-case single failure to the spent fuel cooling system could cause bulk boiling of the spent fuel pool during limiting full core offload conditions. The resulting boil-off rate of 81 gallons per minute (gpm) was found acceptable because it did not exceed the available makeup rate of 100 gpm. Otherwise, an active failure within the SFP cooling system would not reduce the SFP level. Active failures involve loss of those components that perform their function through mechanical motion, such as pumps and valves. An active failure will not cause the loss of cooling system pressure integrity and therefore, no loss of SFP water will occur due to an active failure of any single cooling system component.

31. a. Explain the means provided for mixing to produce a uniform SFP water temperature throughout the pool (SRP Section 9.1.3)

b. Describe any local heat-up in the cask area.

c. Discuss the adequacy of the thermal-hydraulic interaction between the SFP water and cask area.

<u>Response</u>:

At Turkey Point, the cask area is an integral portion of each unit's spent fuel pool, with no walls or barriers separating the cask area from the remainder of the pool. As shown in Figures 1.1.1 and 1.1.2 in Enclosure 1 Appendix 1 to the submittal, the cask area is located along the east wall in the middle of the spent fuel pool. Therefore, there is free exchange of water between the cask area and the surrounding areas of the spent fuel pool to the north, west, and south. Thermal-hydraulic mixing within the cask area, as well as within the remainder of the pool, is provided by the spent fuel pool cooling system. The local heat-up effect in the cask area is the same as elsewhere within the spent fuel pool and is analyzed in the SFP Local Thermal-Hydraulic Analysis (Section 5.6 in the Holtec report).

32. Page 13 of the submittal provides the fuel assembly transfer rates for analysis involving two offload cases. The fuel assembly transfer rates are eight per hour and six per hour for Cases 1a and 1b, respectively.

Explain the difference in these transfer rates and how the transfer rates will not be exceeded during actual offload operations.

<u>Response</u>:

The Case 1a and 1b fuel transfer rates are different because different initial conditions were chosen for the two analysis cases to determine the fuel pool temperature sensitivity to offload rate. Both assumed offload rates exceed the actual transfer rate historically achieved at Turkey Point.

Case 1 consists of a planned refueling with a full core offload initiated at 72 hours after shutdown. Cases 1a and 1b provide two variations of this case emphasizing the sensitivity of the maximum bulk pool temperature to varying initial conditions including fuel transfer rates. A core offload rate of 8 assemblies per hour results in a higher transient decay heat load on the spent fuel pool than a slower offload rate of 6 assemblies per hour. The current maximum offload rate permitted at Turkey Point is 6 assemblies per hour, as discussed below.

The results of Case 1a demonstrated that acceptable bulk pool temperatures (i.e., less than 150 °F) are maintained even with a currently unachievable high fuel transfer rate of 8 assemblies per hour if realistic component cooling water (CCW) temperatures are assumed.

The results of Case 1b demonstrated that with a realistic bounding fuel transfer rate of 6 assemblies per hour and design CCW temperature, bulk pool temperatures would exceed 150 °F; however, administrative controls on spent fuel pool bulk temperature would suspend fuel offload activities at a temperature that would limit the maximum spent fuel pool bulk temperature to less than the 150 °F criterion.

As such, the assembly transfer rates of 6 per hour and 8 per hour were selected to reflect a currently bounding transfer rate and maximum future transfer rate, respectively. Actual fuel transfer rates during refueling outages have not exceeded 5 assemblies per hour. The fuel transfer rate is administratively controlled by plant procedures and is presently limited to a maximum of 6 fuel assemblies per hour.

33. Describe how the capability to remove fuel from the SFP will be assured with licensed fuel storage in the new cask area fuel storage rack.

<u>Response</u>:

When the cask area racks are installed, a procedural restriction will be placed on the combined number of fuel assemblies that can be stored in the spent fuel storage racks and cask area rack. The combined number will be limited to no more than the capacity of the spent fuel storage racks alone at all times except during a reactor offload/refueling condition. This restriction will assure the capability to unload and remove the cask area rack when cask loading operations are necessary.

On the basis that the fuel offload capability is an important economic consideration in sustaining plant operation and a capability that FPL intends to preserve, FPL suggests that the following license condition may be appropriate:

> "The licensee shall restrict the combined number of fuel assemblies for each unit loaded in the spent fuel pool storage racks and cask area rack to no more than the capacity of the spent fuel storage racks at all times except during a reactor offload/refueling condition. This restriction will assure the capability to unload and remove the cask area rack when cask loading operations are necessary."

Holtec Affidavit

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

AFFIDAVIT PURSUANT TO 10CFR2.790

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

Replacement pages for Holtec License Amendment Report
(Non- Proprietary)

page 2-11
page 3-9
page 4-18
Figures 6.12.1 and 6.12.2

clamps to minimize distortion due to welding heat input. Figure 2.6.1 shows the box. The minimum weld seam penetration is 80% of the box metal gage, which is 0.075 inch (14 gage).

A die is used to flare out one end of the box to provide the tapered lead-in (Figure 2.6.2). 1 ¼ inch diameter holes are punched on all four sides near the other end of the box to provide the redundant flow holes.

Each box constitutes a storage location. Each external box side is equipped with a stainless steel sheathing, which holds one integral Boral sheet (poison material) on each side, except the boxes on the east periphery of the rack, which only have Boral on the interior sides. The design objective calls for attaching Boral tightly on the box surface. This is accomplished by die forming the box sheathings, as shown in Figure 2.6.3. The flanges of the sheathing are attached to the box using skip welds and spot welds. The sheathings serve to locate and position the poison sheet accurately, and to preclude its movement under seismic conditions.

Having fabricated the required number of composite box assemblies, they are joined together in a fixture using connector elements in the manner shown in Figure 2.6.4. Figure 2.6.5 shows an elevation view of two storage cells of a Region 1 rack module. A representative connector element is also shown in the figure. Joining the cells by the connector elements results in a well-defined shear flow path, and essentially makes the box assemblage into a multi-flanged beam-type structure. The "baseplate" is attached to the bottom edge of the boxes. The baseplate is a 0.75 inch thick austenitic stainless steel plate stock which has 5-1/4 inch diameter holes (except at four lift locations, which are rectangular) cut out in a pitch identical to the box pitch. The baseplate is attached to the cell assemblage by fillet welding the box edge to the plate.

In the final step, adjustable leg support pedestals (shown in Figure 2.6.6) are welded to the underside of the baseplate. The top (female threaded) portion is made of austenitic steel material. The bottom male threaded part is made of 17:4 Ph series stainless steel to avoid galling problems. All support legs are the adjustable type (Figure 2.6.6), which provide a ± 1/2-inch vertical height adjustment at each leg location for leveling the rack. Each support leg is equipped with a readily accessible socket to enable remote leveling of the rack after its placement in the pool.

Appropriate NDE (nondestructive examination) occurs on all welds including visual examination of sheathing welds, box longitudinal seam welds, box-to-baseplate welds, and box-to-box connection welds; and liquid penetrant examination of support leg welds, in accordance with the design drawings.

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] ANSI/ASME B30.2, "Overhead and Gantry Cranes, (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," American Society of Mechanical Engineers, 1976.

[3.5.2] ANSI B30.9, "Safety Standards for Slings," 1971.

[3.5.3] CMMA Specification 70, "Electrical Overhead Traveling Cranes," Crane Manufacturers Association of America, Inc., 2000.

[3.5.4] ANSI N14.6-1993, Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds or more for Nuclear Materials," American National Standard Institute, Inc., 1978.

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

Table 4.5.1

Reactivity Effects of Manufacturing Tolerances with Unborated and Borated Water

Tolerance	Reactivity Effect, Δk	
	Unborated	Borated – 200 ppm
Minimum Boral loading ([redacted] g/cm^2, 0.0220 g/cm^2 nominal)	+0.0025	+0.0024
Minimum Boral width ([redacted]", 7.5" nominal)[6]	+0.0010	+0.0009
Minimum Water Gap (0.907" & 1.507", 0.987 & 1.587" nominal Water Gap)[7]	+0.0096	+0.0093
Maximum box wall thickness ([redacted]", 0.075" nominal)	+0.0004	+0.0004
Maximum Box I.D. ([redacted]", 8.75" nominal)	+0.0008	+0.0007
Density tolerance ([redacted] g/cm^3, [redacted] g/cm^3 nominal)	+0.0022	+0.0025
Enrichment (4.55 wt% ^{235}U, 4.5 wt% ^{235}U nominal)	+0.0019	+0.0020
Total (statistical sum)[8]	**+0.0104**	**+0.0102**

[6] This is conservative as the specified minimum width of the Boral (including tolerances) is modeled.

[7] This is the maximum possible change in the water gap, predicated on the box I.D. and pitch being manufactured at their greatest tolerance in opposition to each other (i.e. maximum box I.D. and minimum pitch).

[8] Square root of the sum of the squares.

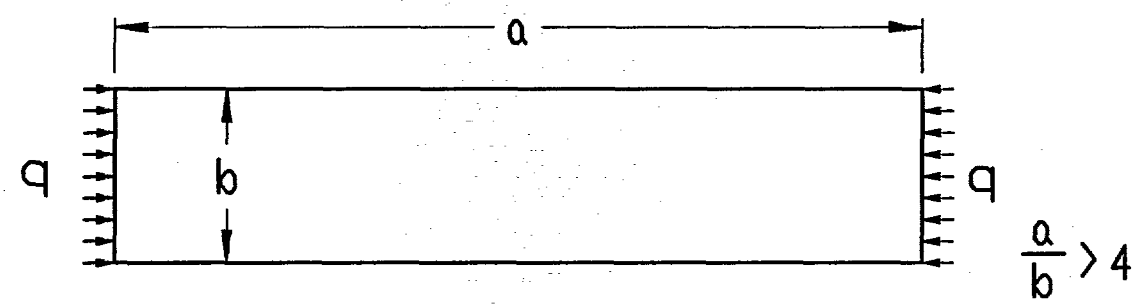

FIGURE 6.12.1; LOADING ON RACK WALL

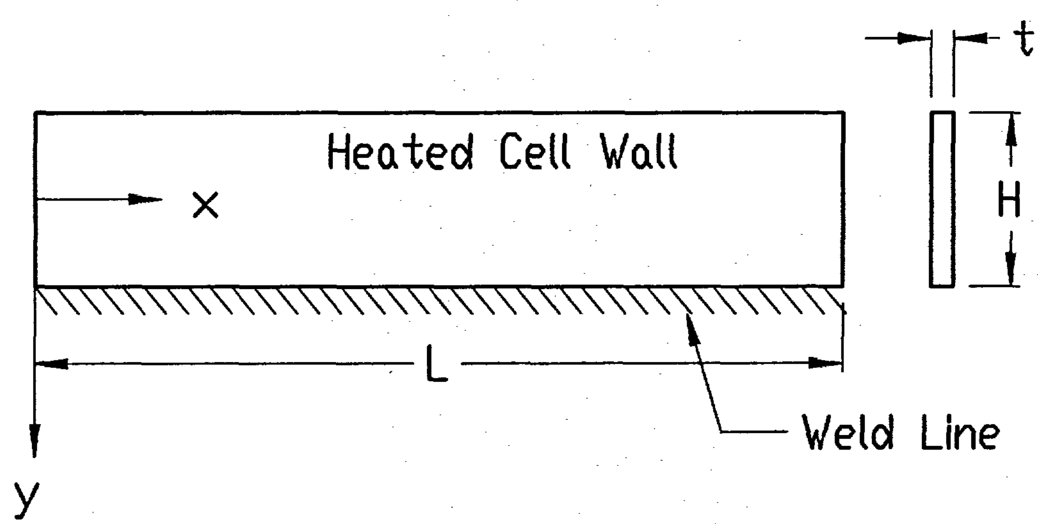

FIGURE 6.12.2; WELDED JOINT IN RACK

Westinghouse Affidavit

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 25th day
of July, 2003.

Sharon L. Fiori
Notary Public

SHARON L. FIORI
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, of the Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse") and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use 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 a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Response in support of NRC's Request for Additional Information, Turkey Point Nuclear Plant, Unit Nos. 3 and 4, "Proposed License Amendments: Addition of Cask Area Spent Fuel Storage Racks," L-2002-214, dated November 26, 2002," for information in support of NRC's Request for Additional Information (RAI # 20) to the Commission, transmitted via Florida Power & Light Company for Turkey Point Nuclear Plant, Unit Nos. 3 and 4 letter and Application for Withholding Proprietary Information from Public Disclosure, H. A. Sepp, Westinghouse, Manager Regulatory Compliance and Plant Licensing to the attention of J. S. Wermiel, Chief, Reactor Systems Branch, Division of Systems Safety and Analysis. The proprietary information provides the technical information requested in the NRC's RAI # 20.

This information is part of that which will enable Westinghouse to:

(a) Provide technical information requested in the NRC's RAI # 20.

(b) Assist customers to obtain license changes.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this information to further enhance their licensing position with their competitors.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC. In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

Westinghouse Letter NF-FP-03-310 dated July 25, 2003 (Non-Proprietary)

Westinghouse

Westinghouse Electric Company
Nuclear Fuel
Columbia Fuel Site
P.O. Drawer R
Columbia, South Carolina 29250
USA

Mr. Jimmie L. Perryman ENG-JB Room D 4466
Turkey Point Project Engineer
Florida Power & Light Company
700 Universe Boulevard
PO Box 14000
Juno Beach, Florida 33408-0420

Direct tel: 803-647-2200
Direct fax: 803-647-2027
e-mail: robinsdb@westinghouse.com

Ref: See Below
Our ref: NF-FP-03-310

July 25, 2003

Reference:

1. Nuclear Fuel Fabrication and Related Services Contract between Florida Power and Light Company and Westinghouse Electric Corporation dated May 23, 1997.
2. Design Interface Procedure between Florida Power and Light Company and Westinghouse Electric Company, LLC for Turkey Point Units 3 & 4, Amendment 3 dated May 30, 2003.
3. FPL/W Design Interface Exception Log, Revision 12, dated March 3, 2003.
4. FPL to Westinghouse letter number NF-03-162/NF-FP-03-303 dated 7/17/03. Request for Affidavit.

Dear Mr. Perryman:

FLORIDA POWER AND LIGHT COMPANY
Turkey Point Units 3 & 4
Westinghouse authorization letter, CAW-03-1677 and accompanying Affidavit

Enclosed is:

1. One copy of "Response in support of NRC's Request for Additional Information, Turkey Point Nuclear Plant, Unit Nos. 3 and 4, "Proposed License Amendments: Addition of Cask Area Spent Fuel Storage Racks," L-2002-214, dated November 26, 2002," (Proprietary), July 2003.

Also enclosed are a Westinghouse authorization letter, CAW-03-1677 and accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse"), it is supported by an affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-03-1677 and should be addressed to Henry A. Sepp, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Please let me know if you have questions or if I can be of additional assistance.

Very Truly Yours,

M. F. Muenks
for Diana Robinson
Project Engineer
U.S. Fuel Commercial Operations

cc:	J. E. Rivera	Juno Beach
	C. O'Farrill	Juno Beach
	R. Tomonto	Turkey Point
	C. Villard	Juno Beach
	M. F. Muenks	EC 4-23
	P. McDonough	EC 4-23
	D. Peck	EC-4-23
	A. DeGrasse	St. Lucie
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Westinghouse

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Our ref: CAW-03-1677

Attention: J. S. Wermiel, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

July 25, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response in support of NRC's Request for Additional Information, Turkey Point Nuclear Plant, Unit Nos. 3 and 4, "Proposed License Amendments: Addition of Cask Area Spent Fuel Storage Racks," L-2002-214, dated November 26, 2002

Dear Mr. Wermiel:

The proprietary information for which withholding is being requested in the above-referenced response is further identified in Affidavit CAW-03-1677 signed by the owner of the proprietary information, Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse"). The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorized the utilization of the accompanying Affidavit by Florida Power & Light Company for Turkey Point Nuclear Plant, Unit Nos. 3 and 4.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-03-1677, and should be addressed to the undersigned.

Very truly yours,

Henry A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Table of Fuel Manufacturing Tolerances and
Corresponding Reactivity Effects
For Westinghouse PTN Fuel

Parameter	Tolerance	delta-k
Fuel Rod Pitch	± [ ][a,b,c]	0.0012
Fuel Pellet OD	± [ ][a,b,c]	0.0003
Fuel Rod Cladding ID	± [ ][a,b,c]	0.0000
Fuel Rod Cladding OD	± [ ][a,b,c]	0.0017
Guide Tube ID	± [ ][a,b,c]	0.0003
Guide Tube OD	± [ ][a,b,c]	0.0003
	Combined Effect :	0.0022
	Effect from other Manufacturing Tolerances :	0.0104
	Total combined Effect :	0.0106
	Total change :	+ 0.0002