FPL

Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

AUG 28 2003
L-2003-215
10 CFR 50.90

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

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
RAI Response for Addition of
Spent Fuel Pool Cask Pit Rack Amendment

By letter L-2002-187 dated October 23, 2002, Florida Power & Light (FPL) submitted a proposed license amendment to revise the Technical Specifications to include the design of a new cask pit rack for each unit and increase each unit's spent fuel storage capacity by combining the new cask pit rack and existing spent fuel pool storage rack capacities.

By letter dated July 22, 2003, the Nuclear Regulatory Commission (NRC) staff requested additional information to support the review of the submittal. The request was discussed with FPL staff and a response date of August 31, 2003 was established. Attached is FPL's response to the RAI.

Attachment 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.

Attachment 3 provides revised pages to incorporate into Enclosure 2 (the proprietary Holtec licensing report) of L-2002-187 dated October 23, 2002, the original proposed license amendment. These are the changed pages associated with Revision 1 of the Holtec Report. 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. Attachment 3 contains information that is considered proprietary pursuant to 10 CFR 2.790. The affidavit required by 10 CFR 2.790 is provided in Attachment 2. FPL requests that Attachment 3 be withheld from public viewing.

Attachment 4 provides revised pages to incorporate into Enclosure 3 (the non-proprietary Holtec licensing report) of L-2002-187 dated October 23, 2002, the original 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 attachment contains no proprietary information.

A001

Attachment 5 provides replacement markup pages for Unit 1 Technical Specification page 5-6 and Unit 2 Technical Specification page 5-4A.

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

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
Nuclear Engineering

Attachments

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

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

28th day of August, 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

RESPONSE TO

REQUEST FOR ADDITIONAL INFORMATION

CASK PIT RACKS LICENSE AMENDMENT REQUEST

ST. LUCIE PLANT, UNITS 1 AND 2

DOCKET NOS. 50-335 AND 50-389

(1) Describe how the capability to remove fuel from the spent fuel pool (SFP) will be assured with licensed fuel storage in the new cask area fuel storage rack.

FPL Response:

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

(2) The submittal states that the proposed change in the design basis will be consistent with the Standard Review Plan (SRP), which requires a bulk spent fuel pool design temperature of 140°F to provide margin with a design basis of partial core offloads. The submittal assumes 150°F. Provide justification for the deviation.

FPL Response:

The current licensing basis for both St. Lucie units is described in the respective UFSARs. UFSAR Section 9.1.3 for both units states that a design basis for the Fuel Pool Cooling System is to limit pool water temperatures to 150°F. In addition, Unit 2 UFSAR 9.1.3.3 states:

> "A maximum fuel pool temperature of 150°F has been found acceptable in previous Safety Evaluation Reports, including [NRC SER dated May 6, 1999 for St. Lucie Unit 2 Amendment No. 101]. . . Although the aforementioned temperatures [150°F] for full core offload conditions exceed the recommendations of SRP Subsection 9.1.3, there will be no detrimental effects to fuel movements, cooling system operation, fuel and fuel assemblies, or pool structures. The fuel pool cooling ion exchanger will be manually isolated before cooling water temperature reaches 140°F."

Based on the current St. Lucie licensing basis, the 150°F limit is a reasonable and established deviation from the SRP guidance.

(3) The submittal proposes to change the design basis to be a partial core offload. It also proposes to reduce the core offload time after shutdown from 168 hours to 120 hours. A review of SRP Section 9.1.3.III.1.h indicates that spent fuel pool heat loads associated with both partial and full-core offloads are calculated based on 150 hours decay. At the reduced offload time the current licensing basis can be maintained for the partial-core offload, but may not be maintained in the case of a planned full-core offload. While not routine, full-core offloads are periodically necessary and are performed during planned outages. The current licensing basis for Unit 2 as stated in Updated Final Safety Analysis Report (UFSAR) Section 9.1.3.1, requires "outage-specific calculations to demonstrate that the spent fuel pool bulk water temperature will not exceed the St. Lucie Design-Basis temperature of 150°F with one Spent Fuel Cooling System pump and one heat exchanger in operation" for refueling evolutions that propose to utilize a full-core offload (UFSAR Section 9.1.3.1). Please explain how the design basis related to bulk pool temperature will be maintained for refueling evolutions that propose to utilize a full-core offload for Unit 1.

FPL Response:

For Unit 1, the design basis for the spent fuel pool cooling system is stated in UFSAR Section 9.1.3.1. Analyses performed for this submittal resulted in fuel pool bulk temperatures less than 150°F for partial core offloads with one train of cooling operating, and for full core offloads with both trains of cooling operating, thereby satisfying Unit 1's design basis. However, to address the potential condition of only one cooling train operating during full core offloads, new procedural controls will be developed for Unit 1. Such procedural controls will be used to assure that, prior to commencing a full core offload, an outage-specific engineering evaluation will be performed to evaluate the spent fuel pool coolant temperature while assuming a cooling capacity equivalent to a single train of the SFP cooling system. The procedure will outline the need for contingency actions such as supplemental cooling requirements or fuel offload stop-temperatures if this temperature objective cannot be met for the full core offload scenario.

Whereas the proposed license amendment describes a comparable design basis for Unit 2, comparable procedural controls will also be considered for Unit 2.

Although the above procedural control will be applied to planned full core offloads on both units, it should be noted that the proposed amendment, supplemented by these administrative controls, goes well beyond the regulatory guidance found in Section 9.1.3 of the Standard Review Plan (SRP), for the following reasons:

- The normal offload condition for the St. Lucie units is a partial core offload. The submittal demonstrates that the peak SFP bulk temperature will be well below the 150°F design basis limit for a normal partial core offload with a concurrent single active failure of the cooling system. The peak temperatures calculated for the partial core offloads are 135°F (Unit 1) and 140°F (Unit 2).

- According to the SRP (Section 9.1.3.III.1.d), the abnormal offload condition is a full core offload. For this condition, the SRP recommends that no pool boiling be allowed with full cooling (no single failure assumed). The submittal demonstrates that pool temperatures will be maintained below 150°F using both cooling system trains. This temperature limit is much more restrictive than the no-boiling limit in the SRP guidance.

- Applying a cooling system single failure during an abnormal offload condition is not discussed in the SRP guidance. The submittal demonstrates that for this condition, the St. Lucie spent fuel pools do not boil. In fact, the submitted analysis demonstrates a Unit 1 peak temperature of only 161°F for this case; only 11 degrees above 150°F and providing a large margin to boiling. Therefore, this condition exceeds the SRP guidance.

The SRP guidance also does not discuss "planned" offloads, and does not equate planned offloads with the "normal" offload condition. FPL recognizes that nuclear power plants require periodic full core offloads (e.g., for 10 year in-service inspections). However, this does not mean that a planned full core offload is the normal offload condition.

Consequently, the proposed changes to the SFP cooling system design basis in the submittal are consistent with the SRP Section 9.1.3 regulatory guidance for the normal core offload cooling case, and exceed the SRP guidance for abnormal core offloads, with or without a single failure.

(4) Explain if using a 90-day operation time is more conservative than the 36-day operation time used in the SRP, Scenario 3. If it is not conservative, provide the maximum bulk SFP temperature for an emergency full-core offload having 36 days operation time since the previous refueling outage for each unit.

FPL Response:

The St. Lucie full core offload (FCO) cooling scenario that assumes a 90-day operation time is more conservative (i.e., will yield a higher peak spent fuel pool bulk temperature) than the SRP scenario that assumes a 36-day operation time. The basis for this statement is that the 90-day offload scenario assumes a single failure of one cooling train, whereas the SRP guidance does not impose a single failure concurrent with the 36-day full core offload. By limiting the heat removal path to a single train, the transfer of decay heat from the spent fuel pool is reduced by a factor of approximately two compared to the SRP scenario, thereby resulting in higher peak pool temperatures than the SRP 36-day scenario.

A 90-day operation time was chosen for the limiting FCO-with-single-cooling-train scenario in the submittal to be consistent with a current Unit 1 licensing basis cooling evaluation in UFSAR Section 9.1.3.2. The Unit 2 UFSAR does not contain a comparable cooling evaluation. However, for this submittal, calculations were performed for both units to determine the peak pool temperature following a 90-day FCO. The calculations demonstrate that no pool boiling will occur on either unit, even with a failure of one cooling train.

For the above reason, the St. Lucie analysis of the SFP peak bulk temperatures during a 90-day operation time full-core offload with one cooling train is more conservative than the SRP 9.1.3 guidance.

(5) Provide the flow rates for the SFP make-up sources.

FPL Response:

Unit 1:

As discussed in UFSAR Section 9.1.3.4, there are several non-safety-related sources of makeup available to the fuel handling building and spent fuel pool, including the refueling water tank (RWT) via the 150 gallon per minute (gpm) fuel pool purification pump, and the primary water tank. A Seismic Category I sea water makeup source is also available from the intake cooling water system via hose connections that can provide a makeup flow rate of 150 gpm.

Unit 2:

As discussed in UFSAR Section 9.1.3.3, two non-safety-related fuel pool inventory makeup systems are provided. The fuel pool purification pump draws water from the RWT at a flow of 150 gpm. In addition, the primary water pumps, with suction from the primary water tanks, can provide makeup to the fuel pool at 100 gpm. Similar to Unit 1, a hose connection is also provided on each Seismic Category 1 intake cooling water header to provide sea water makeup for an indefinite period.

(6) The submittal states in several areas that certain crane features are in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." It also states several changes will be made, including increased crane capacity and replacement or upgrade to single failure proof cranes. Identify any upgrades or replacements necessary for this license amendment.

FPL Response:

No crane upgrades are necessary for this license amendment.

Installing the new cask pit racks does not require increasing the cask crane capacity or revising the existing operating restrictions in the plant Technical Specifications (TS). Each unit's rack weighs well below the old crane capacities and TS limits, and a postulated rack drop event is bounded by the existing cask drop analyses for both units.

However, the old cask cranes for both units have recently been dismantled, so replacing the dismantled cranes with new cranes is necessary to install the new racks. New cranes will be installed prior to the time when the cask pit racks are needed.

Existing Technical Specifications and heavy load handling procedures will be followed. Each rack weighs approximately 20 tons (including lifting equipment), and the current TS load restrictions on the cask cranes are 25 tons for Unit 1 and 100 tons for Unit 2. Not only is the rack weight much less than the Technical Specification load limits, but the rack's honeycomb construction also makes the damage potential to the pool liner much less than the potential damage from the concentrated impact of a dropped spent fuel cask.

(7) The submittal identified crane features that follow the approach of NUREG-0612. Please explain any deviations from the NUREG-0612 guidance.

FPL Response:

St. Lucie plant has a heavy loads program controlled by administrative procedures that meets the NUREG-0612 guidance for the six general areas discussed in Section 5.1.1 of the NUREG. Regarding the installation and removal of the cask pit racks using the cask handling crane on each unit, the requirements of the St. Lucie heavy loads program will be followed without exception or deviation from the plant commitments to NUREG-0612 guidance.

Specific discussion of the heavy loads considerations for handling the cask pit racks relative to safe load paths, procedures, lifting devices, and supervision and training of the operators is provided in Section 3.5 of Enclosure 2 to the submittal. As shown on Table 3.5.1 of the enclosure, no deviations from NUREG-0612 are required for handling the cask pit racks.

(8) The submittal states that safe-load paths will be established for loads, specifically the spent fuel racks in and out of the cask pits. Explain the guidelines for establishing safe-load paths. In particular, address if the load could travel over spent fuel, any safety-related equipment, or any part of the spent fuel pool (e.g., weir wall).

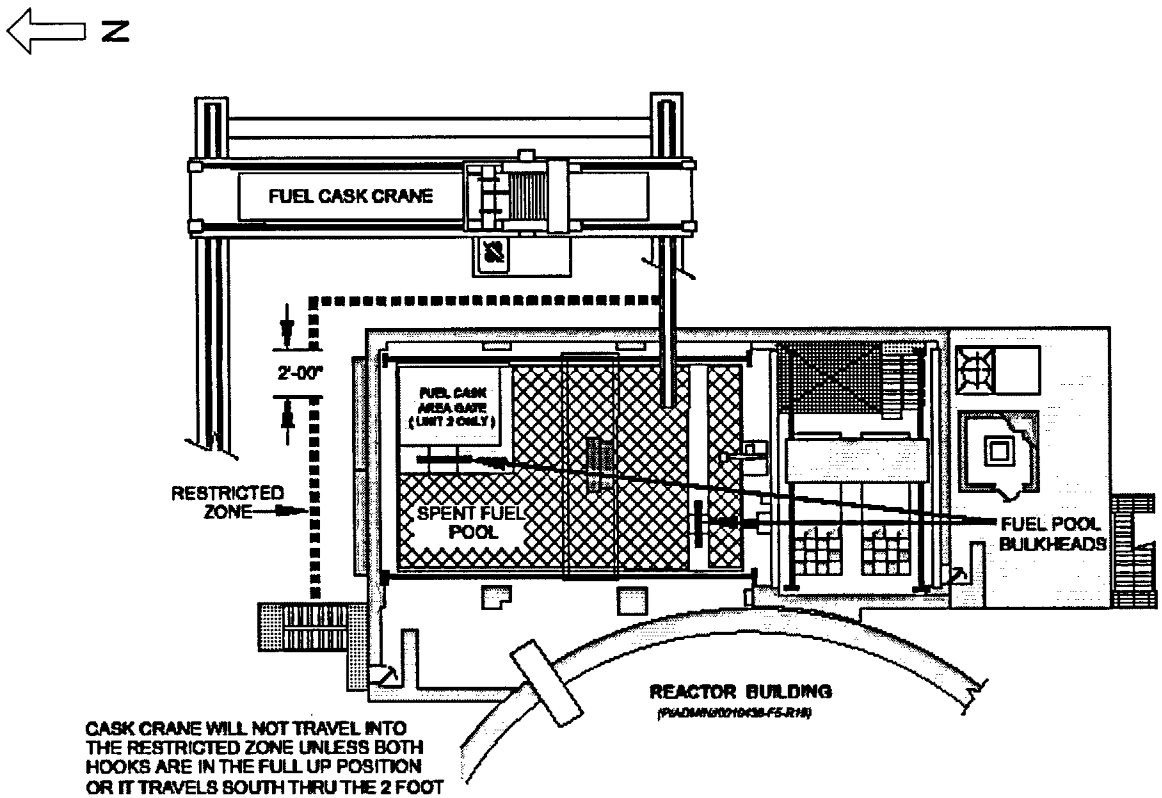
FPL Response:

The rack installations will follow the heavy load handling guidelines of St. Lucie Administrative Procedure 0010438, including the safe load paths established for the fuel handling areas depicted in Figure 5 of that procedure (a copy of which is attached). The Figure 5 safe load path for the cask crane on both units excludes the cross-hatched area over the entire spent fuel pool. As shown on Figure 5 and Unit 1 UFSAR Figure 9.1-15b, this load path restricts the crane main hook travel into the Fuel Handling Building to a two-foot-wide north/south corridor along the centerline of the L-shaped door opening into the building by the use of crane limit switches. The corridor terminates inside the fuel handling building at the approximate center of the cask pit. The objective of this restricted path is to allow crane access to the cask pit area while preventing the load from travel directly over spent fuel stored in the pool or the walls that separate the cask pit from the spent fuel pool. Because the cask pit rack is slightly smaller than the cask pit itself, the rack will be carefully lowered during installation to avoid contact with the cask pit walls.

When the cask pit rack is installed and contains fuel, the cask crane will be administratively prevented (blocked) from operating over the cask pit by use of a restricted area lockout keyswitch under the control of plant Operations. This lockout will protect any fuel stored in the cask pit rack from heavy loads and will prevent violating Technical Specification 3.9.7 weight restrictions over irradiated fuel.

ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010438, REVISION 38
CONTROL OF HEAVY LOADS

FIGURE 5
FUEL AREA SAFE LOAD PATH



CASK CRANE WILL NOT TRAVEL INTO THE RESTRICTED ZONE UNLESS BOTH HOOKS ARE IN THE FULL UP POSITION OR IT TRAVELS SOUTH THRU THE 2 FOOT ACCESS IN THE CENTER OF THE L-SHAPE DOOR

SAFE LOAD PATH IS OUTSIDE CROSS-HATCHED AREAS

DO NOT OPERATE THIS CRANE UNLESS YOU ARE QUALIFIED IN ACCORDANCE WITH ANSI B-30.2

DO NOT MOVE THE CASK CRANE LOAD BLOCKS OR LOADS IN EXCESS OF 1380 POUNDS OVER THE CROSS-HATCHED AREAS WITHOUT A FRG APPROVED PROCEDURE

(9) The proposed amendment described the methodology used to calculate the maximum effective multiplication factor (k_{eff}). The staff has outlined two acceptable methodologies to perform spent fuel pool criticality analyses in a letter 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 proposed amendment is unclear on which methodology was used. Identify which methodology was employed to calculate the maximum k_{eff}.

FPL Response:

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 10 below.

(10) The submittal indicates that maximum effective multiplication factors were calculated by statistically combining all of the tolerances and uncertainties for each of the St. Lucie cask pits with the new racks present and loaded. However, the submittal does not contain the equations used to calculate these values. Please provide the equations used to perform the k_{eff} calculations and a detailed quantitative example demonstrating how each of the tolerances and uncertainties were accounted for in the calculation. The response should include a detailed description of the statistical methods employed and the values used in the calculation of any statistical uncertainties.

<u>FPL 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

$$= [tolerance_{(1)}^2 + ... + uncertainty_{(1)}^2 + ...]^{1/2}$$

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. It should be noted that no correction for axial distribution in burnup is needed since the effect is included explicitly in k(calc); that is, a full 3-D analysis was utilized. In addition, a specific term for the temperature bias was utilized in the calculations performed for the license submittal.

For this example, the reactivity calculation for the Unit 1 cask pit rack is used. The Unit 1 rack is a Region 1 design with flux trap water gaps between the cells.

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 2 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 provides a 95% probability at a 95% confidence level result.

3) Fuel density — CASMO-4 was used to evaluate the reactivity effect of the fuel density tolerance.

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

5) Combined effect of inter-related tolerances for the flux trap water gap, rack cell ID, and lattice pitch — All three tolerances are independently controlled. CASMO-4 determined that the largest reactivity effect from these combined tolerances was for a +0.04" cell box ID tolerance, a – 0.08" water gap thickness tolerance, and a – 0.04" lattice pitch tolerance.

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

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

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

9) Sheathing thickness — CASMO-4 was used to evaluate the reactivity impact of the sheathing thickness tolerance.

10) Fuel assembly dimensional tolerances — CASMO-4 calculations were made to evaluate the reactivity impact of fuel assembly geometry tolerances for fuel rod pitch, fuel pellet OD, fuel clad thickness, guide tube OD, and guide tube wall thickness. The combined effect of statistically summing these tolerances is listed below.

11) Eccentric position of fuel assemblies — A calculation was made to evaluate the reactivity effect of four fuel assemblies at the position of closest approach compared to centered fuel assemblies in four adjacent cells.

Table of Limiting Tolerance Values

Parameter	Tolerance Amount	Reactivity Value
MCNP bias statistics (95/95)	N/A	0.0011
MCNP statistics (95/95)	N/A	0.0009
Fuel density tolerance	+/- 5%	0.0055
Fuel enrichment tolerance	+/- 0.05%	0.0018
Combined tolerance of water gap, rack cell ID, and lattice pitch	+/- tolerances for these three parameters are interrelated	0.0095
Rack wall thickness tolerance	+/- 0.007"	0.0009
Boral™ poison loading	+/- 0.002 g/cm^2	0.0022
Boral™ width tolerance	+/- 0.0625"	0.0009
Sheathing thickness	+/- 0.003"	0.0002

Fuel Assy dimensional tolerances: • Fuel rod pitch • Fuel pellet OD • Fuel clad thickness • Guide tube OD • Guide tube wall thickness	• 14x14 fuel rod array • +/- 0.001" • maximum used • +/- 0.003" • +/- 0.003"	0.0048
Eccentric position of fuel assemblies	N/A	0.0000

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.

$$\delta k(\text{uncert}) = [0.0011^2 + 0.0009^2 + 0.0055^2 + 0.0018^2 + 0.0095^2 + 0.0009^2 + 0.0022^2 + 0.0009^2 + 0.0002^2 + 0.0048^2]^{1/2}$$

$$\delta k(\text{uncert}) = 0.0125$$

The final k_{eff} for this case in Enclosure 2 of the submittal is:

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

$$= 0.8918 + 0.0009 + 0.0009 + 0.0125$$

$$= 0.9061$$

(11) The submittal identifies the worst possible moderation condition as a spent fuel pool temperature of 50°F (10°C) due to a negative moderator temperature coefficient. The maximum density of water occurs at 39.2°F (4°C). Provide justification for selection of a higher temperature (i.e., lower density) as the worst possible moderation condition.

FPL Response:

As stated in Sections 4.1.6.4.1 and 4.2.6.4 of Enclosure 2 to the submittal, the moderator temperature coefficient of reactivity is negative for each unit's cask pit rack.

FPL has determined that 50°F (10°C) is an appropriate minimum SFP temperature value for analysis of fuel pool reactivity at the St. Lucie site based on local climatology, the presence of a substantial inventory of irradiated fuel continuously adding heat to each unit's fuel pool, and the design of the fuel pool cooling systems. Specifically, the fuel pool cooling system at each St. Lucie unit uses the Atlantic Ocean as an ultimate heat sink; so fuel pool bulk water temperature is constrained to be greater than or equal to the local seawater temperature. Values of monthly average sea surface temperature in the vicinity of the St. Lucie site are presented in Unit 2 Updated FSAR Table 2.3-10a, Occurrence of Sea Breezes. This table notes a minimum annual seawater surface temperature of 65 °F is experienced during February.

The fuel pool cooling system rejects heat to the component cooling water system, which in turn rejects heat to intake cooling water. SFP heat transfer (other than by evaporation) can only occur when the pool water temperature exceeds the temperature of intake cooling water (i.e. seawater). Therefore, fuel pool temperatures below 50°F are not realistic at St. Lucie.

(12) Section 4.2.4.3 of the supporting Holtec report provided a table identifying the principal core operating parameters used to analyze the burn-up calculations for the St. Lucie, Unit 2, spent fuel assemblies. The submittal did not identify how these values represented the most limiting conditions (resulting in the highest residual reactivity) of the spent fuel to be stored in the cask pit racks. Please provide a detailed justification for each of the values provided, demonstrating how they result in the most limiting reactivity conditions of the spent fuel.

FPL Response:

In general, the core operating parameter values used in the analysis of the burnup calculations for Unit 2 spent fuel assemblies to be stored in the cask pit rack were chosen to maximize the production of plutonium (Pu) during fuel depletion. Maximizing Pu production increases the reactivity of spent fuel assemblies when subsequently analyzed for reactivity at fuel pool storage conditions. Plutonium production is favored by increasing the average energy at which fissions occur (i.e., "hardening" the neutron spectrum). The assumed reactor operating conditions in Section 4.2.4.3 were therefore chosen at temperatures and a boron concentration which yield a harder neutron spectrum during fuel depletion than normal core operating conditions.

As discussed below, the core operating parameter values listed in the Holtec report Section 4.2.4.3 table are a conservative interpretation of values experienced during St. Lucie Unit 2 plant operation.

Average Fuel Pellet Temperature

Prior to performing cask pit rack reactivity calculations, fuel assemblies were assumed to be depleted at an average fuel pellet temperature of 1604°F. For comparison, the calculated core average fuel pellet temperature at full power beginning of cycle (BOC) conditions is approximately 1125°F for the upcoming St. Lucie Unit 2 cycle. The fuel temperature is highest at BOC conditions. At end of cycle (EOC) conditions, the core average fuel temperature at full power decreases by approximately 60°F. Therefore, selecting a higher-than-actual fuel pellet temperature value for the fuel depletion calculation bounds the actual core operating conditions to favor Pu production and increased fuel reactivity.

Hot Leg Moderator Temperature

Fuel assembly depletion calculations were performed using a moderator temperature of 606°F. This value exceeds the St. Lucie Unit 2 core average exit temperature of approximately 601°F. The core average moderator temperature is approximately 575°F at full power conditions. The use of a value close to the core average moderator temperature would be appropriate for generation of isotopics; however, a value above the core exit temperature was conservatively chosen to increase the fuel reactivity for a given depletion.

Average Core Soluble Boron Concentration

A soluble boron concentration of 750 ppm was used in cask pit rack fuel depletion calculations. The typical BOC soluble boron concentration for an 18-month cycle core at St. Lucie Unit 2 is approximately 1100 ppm. This value is representative of unrodded equilibrium xenon conditions. A typical Middle of Cycle (MOC) boron concentration for full power unrodded conditions is 620 ppm and the typical EOC boron concentration is near zero ppm. Therefore, a 750 ppm value is slightly higher than the average boron concentration for full power operation throughout a typical operating cycle. Because boron-10 has a large thermal neutron absorption cross-section, a higher-than-average soluble boron concentration will tend to harden the neutron spectrum to favor Pu production.

(13) The submittal described a limitation of the Monte Carlo N-Particle Transport Code (MCNP) calculations that prevented modeling some fission product nuclides in the criticality analyses, and described a process to calculate an equivalent amount of boron that provides nearly the same reactivity in MCNP as the CASMO4 results. The submittal stated that this process would compensate for the inability to model these nuclides. Please provide detailed technical information demonstrating that this alternate methodology is conservative or provides bounded results.

FPL Response:

The use of the boron equivalent methodology produces results that show equivalent or higher reactivity, and therefore typically produces bounding results. This methodology is designed to address the need to accurately model burned fuel in the MCNP model. To ensure that the reactivity calculated by MCNP is accurate, the isotopes that are not utilized in MCNP are removed from CASMO-4 and a B^{10} number density placed into CASMO-4. The number density of B^{10} is determined such that the calculated reactivity is equivalent or slightly higher than the CASMO-4 calculation with the explicit isotopes. These calculations are performed in the CASMO-4 fuel rack model at conditions consistent with the spent fuel pool temperature to ensure that the most accurate equivalencing occurs.

The boron equivalent methodology approach is needed when performing criticality analyses that credit burnup with MCNP due to the methodology utilized by the depletion codes. It should be noted that the same limitation exists with the KENO code system.

The main isotopes that are required to be evaluated in this manner are the lumped fission products, designated LFP1 and LFP2 as calculated by CASMO-4. These lumped fission products are not designated as any particular isotope; they represent the accumulated effect of fission products which are not explicitly evaluated. In addition, the Pm-148M isotope is calculated in CASMO-4 but is not available in MCNP and is also treated by use of the boron equivalent method.

(14) The submittal identified the most limiting postulated accident condition as the misplacement of a fresh fuel assembly into a location intended for storage of a spent fuel assembly. The description of the analysis only includes a discussion of the maximum effective multiplication factor obtained. Please provide a detailed description of the following: (1) the assumptions used in the analysis, (2) how each assumption represents the most bounding or limiting condition, (3) the biases and uncertainties included in the analysis, (4) how each bias or uncertainty was accounted for, (5) how the data was evaluated, and (6) how this analysis varies from the currently licensed worst-case misplacement accident in the spent fuel pool.

FPL Response:

This question only applies to the Unit 2 cask pit rack. A Unit 1 cask pit rack mispositioning event is not evaluated because the Unit 1 rack is designed for storage of either fresh or spent fuel assemblies, and no adverse consequences are associated with fresh fuel assembly placement in the rack. The Unit 2 misloading accident analysis uses assumptions and methods comparable to the normal design basis cases (described in Sections 4.2.2 and 4.2.5, respectively, of Enclosure 2 to the submittal) except as noted below:

(1) Assumptions used in the analysis. The supporting criticality analysis for the misloading includes the following key assumptions: (a) the most reactive (fresh) fuel assembly is loaded into an internal cask pit rack cell, (b) fresh fuel is enriched to 4.50 ± 0.05 weight percent U-235, and (c) the evaluated array was modeled as a 7x7 array of spent fuel assemblies of maximum permissible reactivities with a fresh fuel assembly of 4.5 weight percent enrichment in the center cell. The reflecting boundary condition at the outer boundary of the model effectively replicated the misplaced fresh fuel assembly every forty nine (49) assemblies in an infinite array.

(2) How each assumption represents the most bounding or limiting condition. The assumption of a fresh, unshimmed fuel assembly of maximum permissible enrichment as the misplaced fuel assembly bounds all other fuel assembly assumptions. The reflecting boundary conditions, replicating the accident every 49 fuel assemblies throughout the assumed infinite array, bounds all other accident configurations.

(3) Biases and uncertainties included in the analysis. Biases and uncertainties in the analysis are the same as in the reference non-accident configuration, as described in the response to Question 10. The bias and uncertainty values are not influenced by the mispositioning event.

(4) How each bias or uncertainty is accounted for. Each bias and uncertainty is accounted for in the same manner as in the reference calculations similar to the example shown in the Question 10 response. The uncertainties are combined statistically using a square root sum of the squares approach.

(5) How the data was evaluated. Similar to the sample calculation shown in the response to Question 10, the data was evaluated by recording the k_{eff} in the MCNP calculation and adding to it the biases and uncertainties necessary to determine the maximum k_{eff}.

(6) How this analysis varies from the currently licensed worst-case misplacement accident in the spent fuel pool. The worst-case Unit 2 misloading event is described in License Amendment 101 and UFSAR Section 9.1.2.3. The analysis involved the misplacement of an unrodded and unshimmed 4.5 w/o fresh fuel assembly. Three misload positions were evaluated for this assembly: (1) a misload into a position reserved for a 4.5 w/o fresh fuel assembly containing a control element assembly (CEA); (2) a misload into a position designated for a highly burned (1.3 w/o equivalent) fuel assembly; and (3) a misload into selected water cell locations. The largest delta-k observed for any of the postulated assembly misloads was for the third condition, and is not applicable to the cask pit rack analysis because no cells are reserved as water cells (flux traps). The resulting soluble boron requirement in the current licensing basis to compensate for this misload is 746 parts per million (ppm). By comparison, the cask pit rack requires no credit for soluble boron (0 ppm) to meet criticality criteria for the misloading event. This demonstrates that the cask pit rack misloading event yields more conservative results than the existing licensing basis SFP misloading event.

(15) A 5 percent uncertainty in fuel density was assumed when performing the Unit 1 cask pit criticality analysis; however, only 1 percent uncertainty was assumed in the Unit 2 analysis. The reduced uncertainty will result in lower residual reactivities in the spent fuel. Please provide a detailed justification for the values assumed in each of the analyses and the basis for the differences.

FPL Response:

The difference between the fuel density uncertainty values in the Unit 1 and Unit 2 criticality analyses is due to different nominal fuel density values that were provided by the two fuel vendors, as follows.

The Unit 1 criticality analysis was based on Framatome 14 x 14 fuel and was performed with a nominal fuel density value of 10.30 g/cm^3 that accounts for fuel pellet dishing and chamfering. Accounting for dishing and chamfers lowers the nominal stack density value compared to pellets without these features, and allows the application of a density tolerance of ± 5% without exceeding the theoretical UO_2 density limit of 10.96 g/cm^3. Adjusting the nominal density for the 5% tolerance yields the maximum density of 10.815 g/cm^3 used in the Unit 1 analysis.

The Unit 2 criticality analysis was based on CE 16 x 16 fuel and was performed with a nominal fuel density value of 10.522 g/cm^3, which is 96% of the theoretical UO_2 density. This density value does not consider the effects of fuel pellet dishing and chamfering. Ignoring these effects results in a larger nominal density value than if the effects are considered. A tolerance of ± 1% was applied to this nominal value, yielding the maximum fuel density of 10.627 g/cm^3 used in the analysis.[1] If a 5% tolerance had been applied instead, the resulting maximum stack density would have exceeded the theoretical UO_2 density of 10.96 g/cm^3.

[1] Subsequent to performing the criticality analysis, Westinghouse-CE advised that the nominal fuel density value for CE 16 x 16 fuel was 10.32 g/cm^3 with a tolerance of ± 2.5%, yielding a maximum density of 10.578 g/cm^3. The revised maximum value is less than the maximum density of 10.627 g/cm^3 used in the Unit 2 criticality analysis. Therefore, the original maximum density was retained in the analysis as bounding.

(16) The references ([1], [2],...) located within the licensee's submittal Sections 4.1 and 4.2 do not correlate to the references listed in section 4.3. Provide a revised list of references to correctly identify the appropriate documents.

FPL Response:

The pages containing the incorrect reference numbers have been revised with the correct numbers. The revised pages are attached to the letter transmitting this response.

(17) The current St. Lucie, Unit 2, spent fuel racks have a nominal 8.96-inch center-to-center distance between the fuel assemblies; however, the proposed amendment calls for a nominal 8.80-inch center-to-center distance between fuel assemblies placed in the cask pit storage rack. Please describe the basis for the reduced spacing and discuss how the change was accounted for in the reactivity calculations.

FPL Response:

The Unit 2 cask pit storage rack cell pitch is less than the pitch of the current Unit 2 spent fuel racks because the new rack employs integral neutron absorbing material in the cell walls. The use of Boral panels between adjacent cells in the cask pit rack reduces the calculated k_{eff} for a given center-to-center spacing compared to the existing SFP racks.

The existing Unit 2 spent fuel storage racks do not contain either Boral or Boraflex neutron absorbing material. Reactivity control is achieved by use of (1) stainless steel L-inserts which provide some neutron absorption, (2) blocked-open cells (water gaps), and (3) the larger cell pitch (center-to-center spacing) identified in the Technical Specifications.

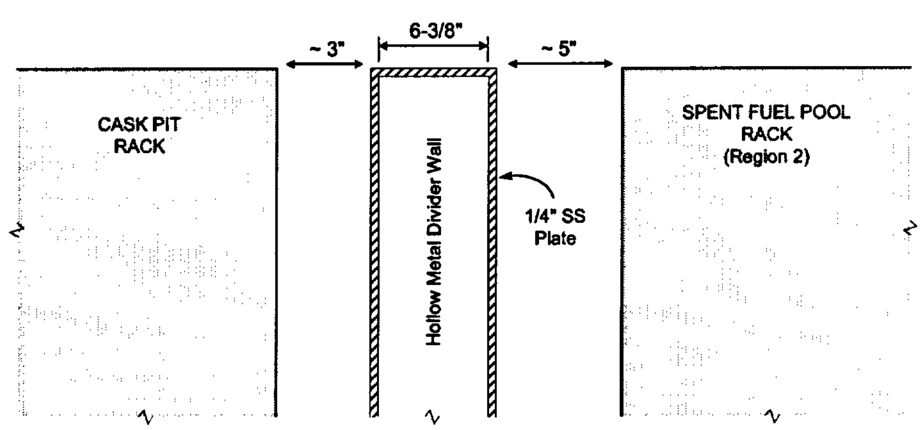
The Unit 2 cask pit rack reactivity calculation was based on the presence of Boral panels, an 8.80 inch pitch, and the related cell geometry depicted in Figure 4.2.2 of Enclosure 2 to the submittal.

(18) Section 3.2 of the amendment request states, "...the Unit 1 rack cells employ Boral neutron absorber panels mounted on the outside faces of stainless steel boxes... (except cells on the rack periphery which contain no Boral panel on the outer face)...." The licensee's criticality analysis assumed an infinite array of storage cells. This array assumed the presence of two Boral panels between adjacent assemblies. The lack of a Boral panel on the outside periphery of the rack may result in greater neutron coupling between the cask pit racks and adjacent spent fuel pool racks. Please provide a detailed description, explaining why greater neutron coupling will not occur with racks adjacent to the cask pit racks, or submit a criticality analysis to evaluate this condition. Additionally, please perform the same evaluation and provide the same requested information for the Unit 2 spent fuel pool.

FPL Response:

For the Unit 1 Cask Pit Rack

The cask pit area adjacent to the Unit 1 spent fuel pool faces only Region 2 storage racks across a submerged divider wall whose height extends above the top of the fuel assemblies in the storage racks. From a reactivity viewpoint, fuel placed in the Region 2 storage locations will be de-coupled from fuel stored in the Region 1 cask pit rack. These storage locations are de-coupled because their separation distance is sufficiently large to ensure that the neutron flux from each region is negligible before fissile material in the other region is encountered. The following cross-section figure and discussion describe the separation distance between the cask pit rack and the closest Region 2 storage rack in the spent fuel pool.



Separation Distance between Unit 1 Cask Pit Rack
and Spent Fuel Pool Rack

The divider wall separating the cask pit area from adjacent Region 2 racks consists of two parallel stainless steel plates, each 0.25 inches thick. These plates are welded to other steel plate comprising the floor and walls of the fuel pool and are secured together with stiffeners that separate the plates by a 6-⅜ inch gap. Conservatively, this inter-plate gap is assumed to be dry. As shown in Figure 1.1.1 of Enclosure 2 to the submittal, the gap between the cask pit rack and the divider wall exceeds three inches. Similarly, as shown in FSAR Figure 9.1-21, a water gap of approximately five inches separates the outer edge of Region 2 storage racks from the divider wall. Thus, Region 2 fuel and cask pit rack fuel are separated by a total water gap distance of more than 8 inches, as well as 0.5 inches of stainless steel plate and a 6-⅜ inch air gap. This separation distance is sufficient to de-couple the adjacent storage regions from the effects of neutron exchange.

<u>For the Unit 2 Cask Pit Rack</u>

Unlike the Unit 1 cask pit area, the Unit 2 cask pit is isolated from the spent fuel pool by a thick (5½-foot) concrete wall, except for a three-foot wide slot in the west wall that extends underwater and allows fuel transfers between the pool and pit areas. The bottom of the transfer slot is above the top of the fuel assemblies seated in the fuel storage racks, so the full concrete wall is between the SFP racks and the cask pit rack. The slot is described in Section 3.3 of the submittal under the thermal-hydraulic considerations for the Unit 2 cask pit rack local temperatures.

Based on the concrete wall thickness separating the Unit 2 cask pit from the spent fuel pool, the separation distance between fuel stored in the two areas is sufficiently large to ensure that the neutron flux from each region is negligible before fissile material in the other region is encountered.

(19) The criticality analysis assumed a minimum Boral density as an uncertainty of the analysis. Please provide a detailed description of the surveillance program that will be used to monitor the installed Boral panels to verify they will continue to meet this limit in the future. Additionally, demonstrate that the surveillance program schedule will be sufficient to identify the depletion of the Boral panels before the boron density decreases below the value assumed in the criticality analysis.

FPL Response:

At this time, FPL does not intend to undertake a formal Boral surveillance program to monitor the material's boron-10 density and neutron absorption properties. Boral panels installed in the new cask pit racks are not accessible to visual surveillance, because they are enclosed in stainless steel sheathing that is spot-welded to the cell walls. To FPL's knowledge, no Boral degradation mechanisms have been identified that require surveillance of the material properties. Unlike Boraflex material, Boral does not contain organic compounds that are subject to gamma-induced degradation, and Boral is in use at many other plants without formal surveillance programs.

Based upon a 2/15/95 letter from Laurence I. Kopp (NRC) to Holtec International, the NRC has no current requirement for in-service surveillance of Boral in spent fuel pool storage racks, although the letter acknowledged that several plants do have surveillance coupons.

FPL will continue to monitor industry operating experience feedback and 10 CFR 21 notices regarding Boral. If Boral employed in spent fuel racks at other plants is found to exhibit degradation in the future, FPL may initiate a surveillance program.

(20) The application 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. The application does not address the potential increase in tritium that might be produced by neutron capture of boron-10 and released in liquid effluent pathways from the plant. Identify how much additional tritium is expected to be produced and released. Describe the significance of any estimated increase.

FPL 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[2] are:

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

FPL's response to Question 1 states that the combined number of fuel assemblies that can be stored in the spent fuel pool storage racks and the cask pit rack will be limited to no more than the capacity of the spent fuel pool 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 and cask pit 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 pit rack installed.

When the cask pit 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 the additional two or three refueling outages that are possible because the rack is installed. 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.

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

(21) The additional stored spent fuel will increase the amount of heat being removed from the water in the spent fuel pool and cask area. Specify how much additional heat will be released to the cooling canal. Explain the significance of any estimated increase.

FPL 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 cask pit area on each unit.

First, any additional heat load attributed to the installation of a cask pit rack only occurs during the additional two or three refueling outages that will be possible when the rack is installed. The FPL response to Question 1 states that the combined number of fuel assemblies that can be stored in the spent fuel pool storage racks and the cask pit rack will be limited to no more than the capacity of the spent fuel pool 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 allowed to be stored long-term in the SFP will be the same regardless of whether or not the cask pit rack is installed, resulting in no additional heat load imposed on the environment during non-refueling periods.

Second, during the final two or three additional refueling outages made possible by the cask pit 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 Unit 1, the added heat load would be from the 143 oldest fuel assemblies that will have an average cooling history of approximately 31 years. For Unit 2, the added heat load would be from the 225 oldest fuel assemblies that will have an average cooling history of approximately 30 years.

Based on the steady-state decay heat data file that was used to calculate the bulk temperature response of the spent fuel pool with the new cask pit racks installed, the projected heat loads for the oldest fuel assemblies stored in both units are estimated below.

Unit	Cask Pit 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
1	143 assemblies	31 years	8.7 E4 Btu/hr	~ 3 E7 Btu/hr	< 0.3%
2	225 assemblies	30 years	1.7 E5 Btu/hr	~ 3 E7 Btu/hr	< 0.6%

As shown, the additional decay heat load imposed on the SFP cooling system during the additional two or three refueling outages with the cask pit rack installed represents less than a one percent increase above the peak refueling decay heat load. Therefore, the additional decay heat load rejected to the environment during the additional refueling outages with a cask pit rack installed is insignificant when compared to the total heat load rejected by the plant during outage conditions and during normal power operation.

(22) According to Section 9.6 of the application, all spent fuel and spent fuel storage racks will be removed from the cask pit before a cask is brought into the pit. State whether this restriction is formalized in an administrative control. If so, describe how will this restriction be formalized.

FPL Response:

An operational restriction in Technical Specification 3.9.7 already limits the weight of any load carried over irradiated fuel to 2000 pounds on Unit 1 and 1600 pounds on Unit 2, and the St. Lucie fuel handling accident (FHA) restricts the plant configuration and plant operations such that the consequences of an FHA are limited to releasing the fission products from no more than one fuel assembly. [Note that FPL has submitted a license amendment request in our letter L-2002-111 dated July 18, 2002 to relocate the TS 3.9.7 operational restriction to the respective unit's Updated Final Safety Analysis Report.]

The plant procedures for the handling of spent fuel casks will clearly stipulate that no cask handling in the vicinity of the Fuel Handling Building will be permitted while the cask pit rack is installed. However, to preclude the remote possibility of handling a cask over the cask pit area while the storage rack is installed, the control system for the cask crane on each unit will include a restricted area lockout keyswitch that will prevent any hook movement over the cask pit area when there is fuel stored in the cask pit. The keyswitch will be under administrative control per plant procedures 1/2-0010123.

From a practical standpoint, there is no logical reason for handling a cask over the cask pit while the cask pit rack is in place because there would be nowhere to set the cask down inside the Fuel Handling Building.

(23) Section 3.7,"Radiological Considerations," indicates that the radiological consequences of a fuel-handling accident are discussed in the Unit 1 and 2 UFSAR fuel-handling accident analyses. In Table 15.4.1-7 of the Unit 1 UFSAR, the control room thyroid dose resulting from the fuel-handling accident is different from the dose stated in the analyses supporting the requests for amendment dated October 30, 2000 (Amendment 172), and May 23, 2002 (Amendment 184). Please clarify.

FPL Response:

The control room thyroid dose value in Unit 1 UFSAR Table 15.4.1-7 is a typographical error; the correct value is 8.48 Rem (thyroid). An UFSAR change has been initiated to correct this error in the next UFSAR update.

AFFIDAVIT PURSUANT TO 10 CFR 2.790

5 total pages including this
cover page

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 St. Lucie Units 1 and 2," Holtec Report HI-2022882, revision 2. The proprietary material in this document is delineated by proprietary designation (i.e., shaded text) on pages 3-15, 4-4, 4-6, 4-7, 4-8, 4-28, 4-34, 4-36, 5-6, 5-7, 6-23, 6-24, 6-29, 7-3, and 7-4.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

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

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

Executed at Marlton, New Jersey, this 26th day of March, 2003.

Scott H. Pellet

Mr. Scott H. Pellet
Holtec International

Subscribed and sworn before me this 26th day of March, 2003.

Maria C. Pepe

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

Revised pages to Holtec License Amendment Report (Non-Proprietary)

Replacement Pages

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attached to the bottom edge of the boxes. The baseplate is a 0.75 inch thick austenitic stainless steel plate stock which has 5 inch diameter holes (except 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 supports (shown in Figure 2.6.6) are welded to the underside of the baseplate. The adjustable legs provide a ± 1/2-inch vertical height adjustment at each leg location.

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.

2.6.2 Rack Module for Region 2

Region 2 storage cell locations have a single poison panel between adjacent cell boxes on the wall surfaces separating them. The significant components (discussed below) of the Unit 2, Region 2 rack are: (1) the storage box subassembly (2) the baseplate, (3) the neutron absorber material, (4) the sheathing, and (5) the support legs.

1. Storage cell box subassembly: As described for Region 1, the boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 2.6.1 shows the box.

   Each box has two lateral holes punched near its bottom edge to provide auxiliary flow holes. As shown in Figure 2.6.3, sheathing is attached to each side of the box with the poison material installed in the sheathing cavity. The edges of the sheathing and the box are welded together to form a smooth edge. The box, with integrally connected sheathing, is referred to as the "composite box".

   The composite boxes are arranged in a checkerboard array to form an assemblage of storage cell locations (Figure 2.6.7). Filler panels and corner angles are welded to the edges of boxes at the outside boundary of the rack to make the peripheral formed cells. The inter-box welding and pitch adjustment are accomplished by small longitudinal connectors. This assemblage of box assemblies is welded edge-to-edge as shown in Figure 2.6.7, resulting in a honeycomb structure with axial, flexural and torsional rigidity depending on the extent of intercell welding provided. It can be seen from Figure 2.6.7 that two edges of each interior box are connected to the contiguous boxes resulting in a well-defined path for "shear flow".

2. Baseplate: The baseplate provides a continuous horizontal surface for supporting the fuel assemblies. The baseplate has a 5 inch diameter hole (except lift locations which are

Table 2.5.1

MODULE DATA FOR UNIT 1 REGION 1 CASK PIT RACK †

Storage cell inside nominal dimension	8.58in.
Cell pitch	10.3in.
Storage cell height (above the plate)	180.0 in.
Baseplate hole size (except for lift location)	5.0 in.
Baseplate thickness	0.75 in.
Support pedestal height	4.25 in.
Support pedestal type	Remotely adjustable pedestals
Number of support pedestals per rack	4
Number of cell walls containing 1" diameter flow holes at base of cell wall	All Four Cell Walls
Remote lifting and handling provisions	Yes
Poison material	Boral
Poison length	140 in.
Poison width	7.25 in.

† All dimensions indicate nominal values

Table 2.5.2 MODULE DATA FOR UNIT 2 REGION 2 CASK PIT RACK †	
Storage cell inside nominal dimension	8.58 in.
Cell pitch	8.80 in.
Storage cell height (above the plate)	180.0 in.
Baseplate hole size (except for lift location)	5.0 in.
Baseplate thickness	0.75 in.
Support pedestal height	4.25 in.
Support pedestal type	Remotely adjustable pedestals
Number of support pedestals per rack	4
Minimum number of cell walls containing 1" diameter supplemental flow holes at base of each cell located away from pedestals	2
Number of cell walls containing 1" diameter flow holes at base of each cell located above a pedestal	4
Remote lifting and handling provisions	Yes
Poison material	Boral
Poison length	140 in.
Poison width	7.25 in.

† All dimensions indicate nominal values

4.1.3 Acceptance Criteria

The primary acceptance criterion for analysis of the Cask Pit rack is that, under a hypothetical condition of 0 ppm soluble boron in the cask pit, the maximum k_{eff} shall be less than or equal to 0.95, including calculational uncertainties and effects of mechanical tolerances. Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- Code of Federal Regulation 10CFR50.68, Criticality Accident Requirements
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications
- ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum, L. Kopp to Timothy Collins, August 19, 1998.

4.1.4 Design and Input Data

4.1.4.1 Fuel Assembly Design Specifications

Two different fuel assembly designs were considered in the analyses; the CE 14x14 lattice and the Framatome 14x14 lattice. Table 4.1.4.1 provides the design details for the fuel assemblies.

4.1.4.2 Fuel Storage Cells

The nominal Cask Pit rack storage cell used for the criticality analyses is shown in Figure 4.1.1. The cell is composed of each box face of an 8.58 inch square (inside dimension) stainless steel box that has a wall thickness of 0.075 inches with Boral absorber material mounted on the outside. The fuel assemblies are assumed to be centrally located in each storage cell on a nominal lattice spacing of 10.30 inches. This forms a water flux-trap between Boral absorber panels of adjacent cells of [ ] inches. The Boral absorber has a thickness of [ ] inches and a nominal B-10 areal density of [ ] g/cm^2 ([ ] g/cm^2 minimum). The outer stainless steel sheath is [ ] inches thick.

4.1.5 Methodology

The primary criticality analyses were performed with the three-dimensional MCNP Monte Carlo code [1] developed by the Los Alamos National Laboratory. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009±0.0011 (95%/95%) [2].

KENO5a [3], a 3-dimensional multi-group Monte Carlo code developed by the Oak Ridge National Laboratory, was used as an independent check and to determine the reactivity-effect of eccentric fuel assembly positioning. In these calculations, the 238-group SCALE cross-section library was used, together with the Norderm integral treatment for U-238 resonance shielding effects. Benchmark calculations (Appendix A) showed a calculational bias of 0.0030±0.0012.

CASMO4, a two-dimensional deterministic code [4] using transmission probabilities, was used to evaluate the small (differential) reactivity effects of manufacturing tolerances. Validity of the CASMO4 code was established by comparison with results of the MCNP calculations for comparable cases.

4.2.5 Methodology

The primary criticality analyses were performed with the three-dimensional MCNP Monte Carlo code [1] developed by the Los Alamos National Laboratory. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009±0.0011 (95%/95%) [2].

CASMO4, a two-dimensional deterministic code [3] using transmission probabilities, was used to evaluate the small (differential) reactivity effects of manufacturing tolerances. Validity of the CASMO4 code was established by comparison with results of the MCNP calculations for comparable cases.

In the geometric model used in the calculations, each fuel rod and each fuel assembly were explicitly described. Reflecting boundary conditions effectively defined an infinite radial array of storage cells. In the axial direction, a 30-cm water reflector was used to conservatively describe axial neutron leakage. Each stainless steel box and water within the box was explicitly described in the calculational model.

Monte Carlo (MCNP) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the calculated reactivities, a minimum of 3 million neutron histories was accumulated in each calculation. MCNP cannot perform depletion calculations, and depletion calculations were performed with CASMO4. Explicit description of the fission product nuclide concentrations in the spent fuel was determined from the CASMO4 calculations and used in the MCNP calculations. To compensate for those few fission product nuclides that cannot be described in MCNP, an equivalent amount of boron-10 in the fuel was determined which produced very nearly the same reactivity in MCNP as the CASMO4 result. This methodology explicitly incorporates approximately 40 of the most important fission products, accounting for all but about 1% in k. The remaining ~1 % in k is included by the equivalent B-10 concentration in the fuel.

4.2.6 Evaluation of Uncertainties

4.2.6.1 Uncertainty in Manufacturing Tolerances

CASMO4 calculations were made to determine the uncertainties in reactivity associated with tolerances in the rack's dimensions, fuel density and fuel enrichments. The reactivity effects of each independent tolerance were combined statistically. The rack dimensions and tolerances are shown in Figure 4.2.2.

For estimating the reactivity uncertainties associated with tolerances in fuel enrichment and density, tolerances of ± 0.05% in enrichment and ±1% in UO_2 density were assumed. The reactivity associated with the fuel density tolerance is listed in Table 4.2.6.1. The reactivity effects of the tolerances in the rack dimensions are also listed in Table 4.2.6.1. The reactivity effects for the tolerance in fuel enrichment are listed in Table 4.2.6.2.

4.2.6.2 Uncertainty in Depletion Calculations

The uncertainty in depletion calculations is part of the methodology uncertainty and was taken as 5% of the reactivity decrement from beginning-of-life to the burnup of concern for the spent fuel [5]. This methodology uncertainty is included in the calculations of the final k_{eff} in Table 4.2.6.4.

4.2.6.3 Eccentric Location of Fuel Assemblies

The fuel assemblies are nominally stored in the center of the storage cells. Eccentric positioning of fuel assemblies in the cells normally results in a negligible effect or a reduction in reactivity for poisoned racks. Calculations have been made confirming negative reactivity effect of the eccentric positioning of four fuel assemblies at the position of closest approach. These calculations gave a small reduction in k_{eff} (-0.0013) confirming that eccentric positioning of fuel has a negligible effect.

4.2.6.4 Temperature and Void Effects

Temperature effects were also evaluated, using CASMO4, in the temperature range from 10 °C to 120 °C and the results are listed in Table 4.2.6.3. These results show that the temperature coefficient of reactivity is negative. The void coefficient of reactivity (boiling conditions) was also found to be negative for the St. Lucie Unit 2 cask pit rack. The reference temperature is 20 °C. The reactivity effects of pool water temperatures below 20 °C to 10 °C are calculated using CASMO (Table 4.2.6.3). These data were interpolated for water temperature to 10 °C (50 °F) and the resulting reactivity increment is added to the calculated k_{eff} at 20 °C.

4.2.6.5 Reactivity Effect of the Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends. The more reactive fuel near the ends of the fuel assembly (less than average burnup) has reactivities slightly above that of the assembly average. Axial burnup penalty calculations based upon a conservative burnup distribution [4], gave a positive reactivity effect of the axial burnup distribution of 0.0071Δk for spent fuel of 36 MWD/KgU. These calculations are based on 10 zone axial calculations, using specific (CASMO) concentrations of actinides and fission products in each zone. Calculations for 4% fuel at 30 MWD/KgU gave a correction of 0.0007Δk and the correction becomes negative (neglected) below a burnup of ~ 29 MWD/KgU.

4.2.7 Accident Conditions And Soluble Boron Requirements

The accident scenarios considered in this analysis are summarized below:

- A dropped fuel assembly coming to rest horizontally across the top of the storage cell.
- A dropped fuel assembly, which enters the storage cell vertically and impacts the base plate.
- An extraneous assembly positioned outside and immediately adjacent to the storage rack

Table 4.1.4.1 Design Basis Fuel Assembly Specifications

PARAMETER	CE 14X14	Framatome 14X14
Rod Array Size	14x14	14x14
Rod Pitch (inches)	0.580±0.015	0.580±0.015
Active Fuel Length (inches)	136.7±0.50	136.7±0.50
Stack Density (gm/cm^3)	10.05 ± 5%	10.30 ± 5%
Total Number of Fueled Rods	176	176
Fuel Rod Outer Diameter (inches)	0.440±0.0055	0.440±0.002
Cladding Thickness (inches)	0.026 - 0.028 ±0.002	0.028 - 0.031 ±0.003
Cladding Material	Zr–4	Zr–4
Pellet Diameter (inches)	0.3805±0.001	0.3770±0.001
Number of Guide/Instrument Tubes	5	5
Guide/Instrument Outer Diameter (inches)	1.115±0.003	1.115±0.003
Guide/Instrument Wall Thickness (inches)	0.040±0.004	0.040±0.004
Material	Zr-4	Zr-4

APPENDIX 4A: BENCHMARK CALCULATIONS

4A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the rack designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [4A.1] is a continuous energy Monte Carlo code and KENO5a [4A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [4A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors† (trends) that have been reported (e.g., [4A.3 through 4A.5]) for calculations with collapsed cross section sets.

In rack designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the ^{10}B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of rack and fuel designs, have a smaller effect, but are also included in the analyses.

Table 4A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission" (EALF). In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the EALF determined (post-processing).

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† Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

Figures 4A.1 and 4A.2 show the calculated k_{eff} for the benchmark critical experiments as a function of the EALF for MCNP4a and KENO5a, respectively (UO_2 fuel only). The scatter in the data (even for comparatively minor variation in critical parameters) represents experimental error† in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 4A.1 and 4A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a k_{eff} of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a	
MCNP4a	0.0009±0.0011
KENO5a	0.0030±0.0012

The bias and standard error of the bias were derived directly from the calculated k_{eff} values in Table 4A.1 using the following equations††, with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [4A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n} \sum_{i}^{n} k_i \qquad (4A.1)$$

† A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

†† These equations may be found in any standard text on statistics, for example, reference [4A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma_{\bar{k}}^2 = \frac{\sum_{i=1}^{n} k_i^2 - (\sum_{i=1}^{n} k_i)^2 / n}{n\,(n-1)} \qquad (4A.2)$$

$$Bias = (1 - \bar{k}) \pm K\,\sigma_{\bar{k}} \qquad (4A.3)$$

where k_i are the calculated reactivities of n critical experiments; $\sigma_{\bar{k}}$ is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); K is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [4A.18]).

Formula 4.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 4.A-2. The first portion of the equation, ($1 - \bar{k}$), is the actual bias which is added to the MCNP4a and KENO5a results. The second term, $K\sigma_{\bar{k}}$, is the uncertainty or standard error associated with the bias. The K values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual K values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The bias values are used to evaluate the maximum k_{eff} values for the rack designs. KENO5a has a slightly larger systematic error than MCNP4a, but both result in greater precision than published data [4A.3 through 4A.5] would indicate for collapsed cross section sets in KENO5a (SCALE) calculations.

4A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46 w/o to 5.74 w/o and therefore span the enrichment range for rack designs. Figures 4A.3 and 4A.4 show the calculated k_{eff} values (Table 4A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, a typical configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 4A.2 and Figure 4A.5, confirm no significant difference in the calculated values of k_{eff} for the two independent codes as evidenced by the 45° slope of the curve. Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

4A.3 Effect of ^{10}B Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the rack designs. Of these critical experiments, those performed by B&W are the most representative of the rack designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 4A.3 lists the subset of experiments using thin neutron absorbers (from Table 4A.1) and shows the reactivity worth (Δk) of the absorber.†

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 4A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with ^{10}B concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 4A.6 and Table 4A.4 for a typical geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

† The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental (Δk) change in reactivity due to the absorber.

4A.4 Miscellaneous and Minor Parameters

4A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.† Analysis of these critical experiments are listed in Table 4A.5 (subset of data in Table 4A.1). There appears to be a small tendency toward overprediction of k_{eff} at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the rack calculations may be slightly more conservative than otherwise.

4A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches, and lattice spacings from 0.476 to 1.00 inches. In the rack designs, the fuel pellet diameters range from 0.303 to 0.3805 inches O.D. (0.496 to 0.580 inch lattice spacing) for PWR fuel and from 0.3224 to 0.494 inches O.D. (0.488 to 0.740 inch lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of power reactor fuel. Based on the data in Table 4A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments applicable to rack designs.

4A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 4A.6. Analyses of the very high boron concentration experiments (> 1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the racks with higher soluble boron concentrations could be slightly conservative.

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† Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec rack design.

4A.5 MOX Fuel

The number of critical experiments with PuO_2 bearing fuel (MOX) is more limited than for UO_2 fuel. However, a number of MOX critical experiments have been analyzed and the results are shown in Table 4A.7. Results of these analyses are generally above a k_{eff} of 1.00, indicating that when Pu is present, both MCNP4a and KENO5a overpredict the reactivity. This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggesting that a small trend may exist with KENO5a. It is also possible that the overprediction in k_{eff} for both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated k_{eff} over a wide range of the spectral index (energy of the average lethargy causing fission).

4A.6 References

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Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF [†] (eV) MCNP4a	EALF [†] (eV) KENO5a
1	B&W-1484 (4A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898± 0.0006	0.1759	0.1753
2	B&W-1484 (4A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (4A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (4A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (4A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (4A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (4A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (4A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (4A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (4A.7)	Core XV [††]	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (4A.7)	Core XVI [††]	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (4A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021
13	B&W-1484 (4A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF† (eV) MCNP4a	EALF† (eV) KENO5a
14	B&W-1484 (4A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (4A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (4A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (4A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (4A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (4A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (4A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (4A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (4A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (4A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (4A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC
25	French (4A.10)	Water Moderator 10 cm gap	4.75	0.9979 ± 0.0010	NC	0.1736	NC
26	PNL-3602 (4A.11)	Steel Reflector, 0 separation	2.35	NC	1.0004 ± 0.0006	NC	0.1018

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF † (eV) MCNP4a	EALF † (eV) KENO5a
27	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	2.35	0.9980 ± 0.0009	0.9992 ± 0.0006	0.1000	0.0909
28	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn	2.35	0.9968 ± 0.0009	0.9964 ± 0.0006	0.0981	0.0975
29	PNL-3602 (4A.11)	Steel Reflector, 3.912 cm sepn.	2.35	0.9974 ± 0.0010	0.9980 ± 0.0006	0.0976	0.0970
30	PNL-3602 (4A.11)	Steel Reflector, infinite sepn.	2.35	0.9962 ± 0.0008	0.9939 ± 0.0006	0.0973	0.0968
31	PNL-3602 (4A.11)	Steel Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3282
32	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	4.306	0.9997 ± 0.0010	1.0012 ± 0.0007	0.3016	0.3039
33	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn.	4.306	0.9994 ± 0.0012	0.9974 ± 0.0007	0.2911	0.2927
34	PNL-3602 (4A.11)	Steel Reflector, 5.405 cm sepn.	4.306	0.9969 ± 0.0011	0.9951 ± 0.0007	0.2828	0.2860
35	PNL-3602 (4A.11)	Steel Reflector, Infinite sepn. ††	4.306	0.9910 ± 0.0020	0.9947 ± 0.0007	0.2851	0.2864
36	PNL-3602 (4A.11)	Steel Reflector, with Boral Sheets	4.306	0.9941 ± 0.0011	0.9970 ± 0.0007	0.3135	0.3150
37	PNL-3926 (4A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159
38	PNL-3926 (4A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044
39	PNL-3926 (4A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[†] (eV) MCNP4a	EALF[†] (eV) KENO5a
40	PNL-3926 (4A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854
41	PNL-2615 (4A.13)	Experiment 004/032 - no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159
42	PNL-2615 (4A.13)	Experiment 030 - Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154
43	PNL-2615 (4A.13)	Experiment 013 - Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164
44	PNL-2615 (4A.13)	Experiment 014 - Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164
45	PNL-2615 (4A.13)	Exp. 009 1.05% Boron-Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162
46	PNL-2615 (4A.13)	Exp. 012 1.62% Boron-Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173
47	PNL-2615 (4A.13)	Exp. 031 - Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171
48	PNL-7167 (4A.14)	Experiment 214R - with flux trap	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812
49	PNL-7167 (4A.14)	Experiment 214V3 - with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (4A.15)	Case 173 - 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (4A.15)	Case 177 - 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF† (eV) MCNP4a	EALF† (eV) KENO5a
53	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1139	0.1165
56	WCAP-3385 (4A.17)	Saxton Case 52 PuO2 0.52" pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (4A.17)	Saxton Case 52 U 0.52" pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (4A.17)	Saxton Case 56 PuO2 0.56" pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (4A.17)	Saxton Case 56 borated PuO2	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (4A.17)	Saxton Case 56 U 0.56" pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954
61	WCAP-3385 (4A.17)	Saxton Case 79 PuO2 0.79" pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555
62	WCAP-3385 (4A.17)	Saxton Case 79 U 0.79" pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047

Notes: NC stands for not calculated.

† EALF is the energy of the average lethargy causing fission.

†† These experimental results appear to be statistical outliers (>3σ) suggesting the possibility of unusually large experimental error. Although they could justifiably be excluded, for conservatism, they were retained in determining the calculational basis.

Table 4A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES[†]
FOR VARIOUS ENRICHMENTS

Enrichment	Calculated $k_{eff} \pm 1\sigma$	
	MCNP4a	**KENO5a**
3.0	0.8465 ± 0.0011	0.8478 ± 0.0004
3.5	0.8820 ± 0.0011	0.8841 ± 0.0004
3.75	0.9019 ± 0.0011	0.8987 ± 0.0004
4.0	0.9132 ± 0.0010	0.9140 ± 0.0004
4.2	0.9276 ± 0.0011	0.9237 ± 0.0004
4.5	0.9400 ± 0.0011	0.9388 ± 0.0004

† Based on the GE 8x8R fuel assembly.

Table 4A.3

MCNP4a CALCULATED REACTIVITIES FOR
CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS

Ref.	Experiment		Δk Worth of Absorber	MCNP4a Calculated k_{eff}	EALF † (eV)
4A.13	PNL-2615	Boral Sheet	0.0139	0.9994±0.0012	0.1165
4A.7	B&W-1484	Core XX	0.0165	1.0008±0.0011	0.1724
4A.13	PNL-2615	1.62% Boron-steel	0.0165	0.9996±0.0012	0.1161
4A.7	B&W-1484	Core XIX	0.0202	0.9961±0.0012	0.2103
4A.7	B&W-1484	Core XXI	0.0243	0.9994±0.0010	0.1544
4A.7	B&W-1484	Core XVII	0.0519	0.9962±0.0012	0.2083
4A.11	PNL-3602	Boral Sheet	0.0708	0.9941±0.0011	0.3135
4A.7	B&W-1484	Core XV	0.0786	0.9910±0.0011	0.2092
4A.7	B&W-1484	Core XVI	0.0845	0.9935±0.0010	0.1757
4A.7	B&W-1484	Core XIV	0.1575	0.9953±0.0011	0.2022
4A.7	B&W-1484	Core XIII	0.1738	1.0020±0.0011	0.1988
4A.14	PNL-7167	Expt 214R flux trap	0.1931	0.9991±0.0011	0.3722

†EALF is the energy of the average lethargy causing fission.

Table 4A.4

COMPARISON OF MCNP4a AND KENO5a
CALCULATED REACTIVITIES† FOR VARIOUS ^{10}B LOADINGS

^{10}B, g/cm^2	Calculated k_{eff} ± 1σ	
	MCNP4a	KENO5a
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ± 0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

† Based on a 4.5% enriched GE 8x8R fuel assembly.

Table 4A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH THICK LEAD AND STEEL REFLECTORS†

Ref.	Case	E, wt%	Separation, cm	MCNP4a k_{eff}	KENO5a k_{eff}
4A.11	Steel Reflector	2.35	1.321	0.9980±0.0009	0.9992±0.0006
		2.35	2.616	0.9968±0.0009	0.9964±0.0006
		2.35	3.912	0.9974±0.0010	0.9980±0.0006
		2.35	∞	0.9962±0.0008	0.9939±0.0006
4A.11	Steel Reflector	4.306	1.321	0.9997±0.0010	1.0012±0.0007
		4.306	2.616	0.9994±0.0012	0.9974±0.0007
		4.306	3.405	0.9969±0.0011	0.9951±0.0007
		4.306	∞	0.9910±0.0020	0.9947±0.0007
4A.12	Lead Reflector	4.306	0.55	1.0025±0.0011	0.9997±0.0007
		4.306	1.956	1.0000±0.0012	0.9985±0.0007
		4.306	5.405	0.9971±0.0012	0.9946±0.0007

† Arranged in order of increasing reflector-fuel spacing.

Table 4A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE BORON CONCENTRATIONS

Reference	Experiment	Boron Concentration, ppm	Calculated k_{eff}	
			MCNP4a	KENO5a
4A.15	PNL-4267	0	0.9974 ± 0.0012	-
4A.8	B&W-1645	886	0.9970 ± 0.0010	0.9924 ± 0.0006
4A.9	B&W-1810	1337	1.0023 ± 0.0010	-
4A.9	B&W-1810	1899	1.0060 ± 0.0009	-
4A.15	PNL-4267	2550	1.0057 ± 0.0010	-

Table 4A.7

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case[†]	MCNP4a		KENO5a	
		k_{eff}	EALF[††]	k_{eff}	EALF[††]
PNL-5803 [4A.16]	MOX Fuel - Exp. No. 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel - Exp. No. 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel - Exp. No. 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel - Exp. No. 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP-3385-54 [4A.17]	Saxton @ 0.52" pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56" pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56" pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79" pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

Note: NC stands for not calculated

† Arranged in order of increasing lattice spacing.

†† EALF is the energy of the average lethargy causing fission.

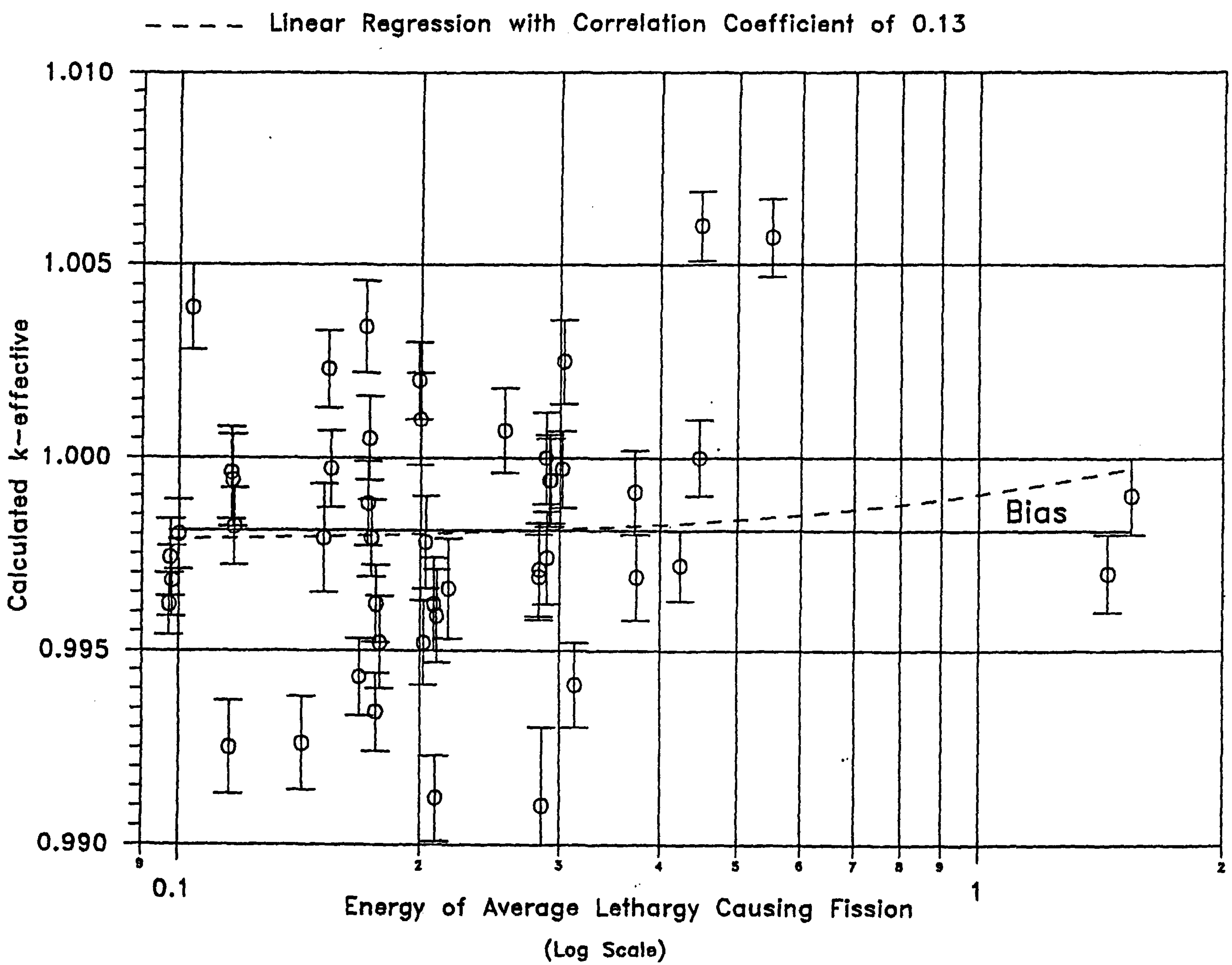

FIGURE 4A.1 MCNP CALCULATED k-eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

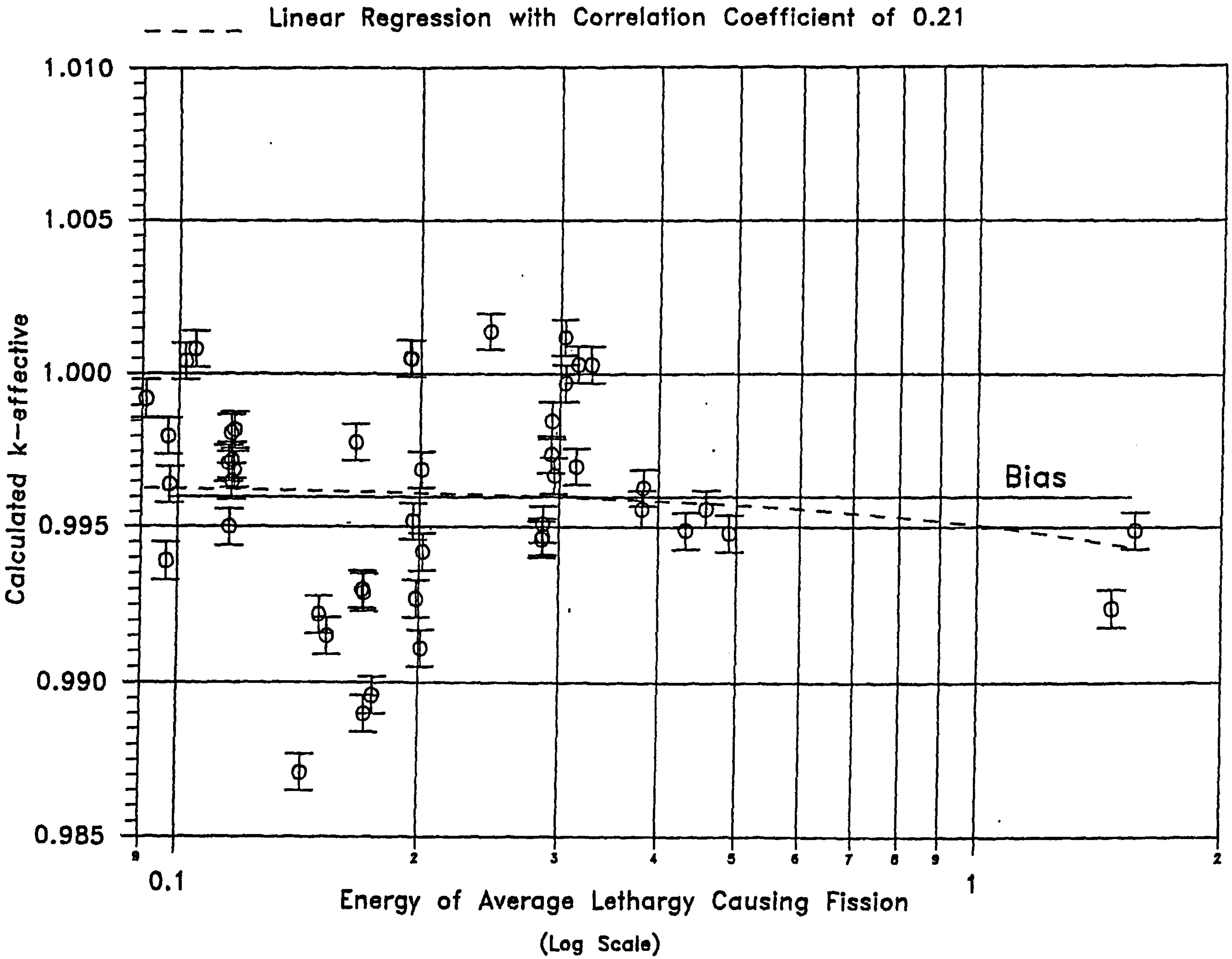

FIGURE 4A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

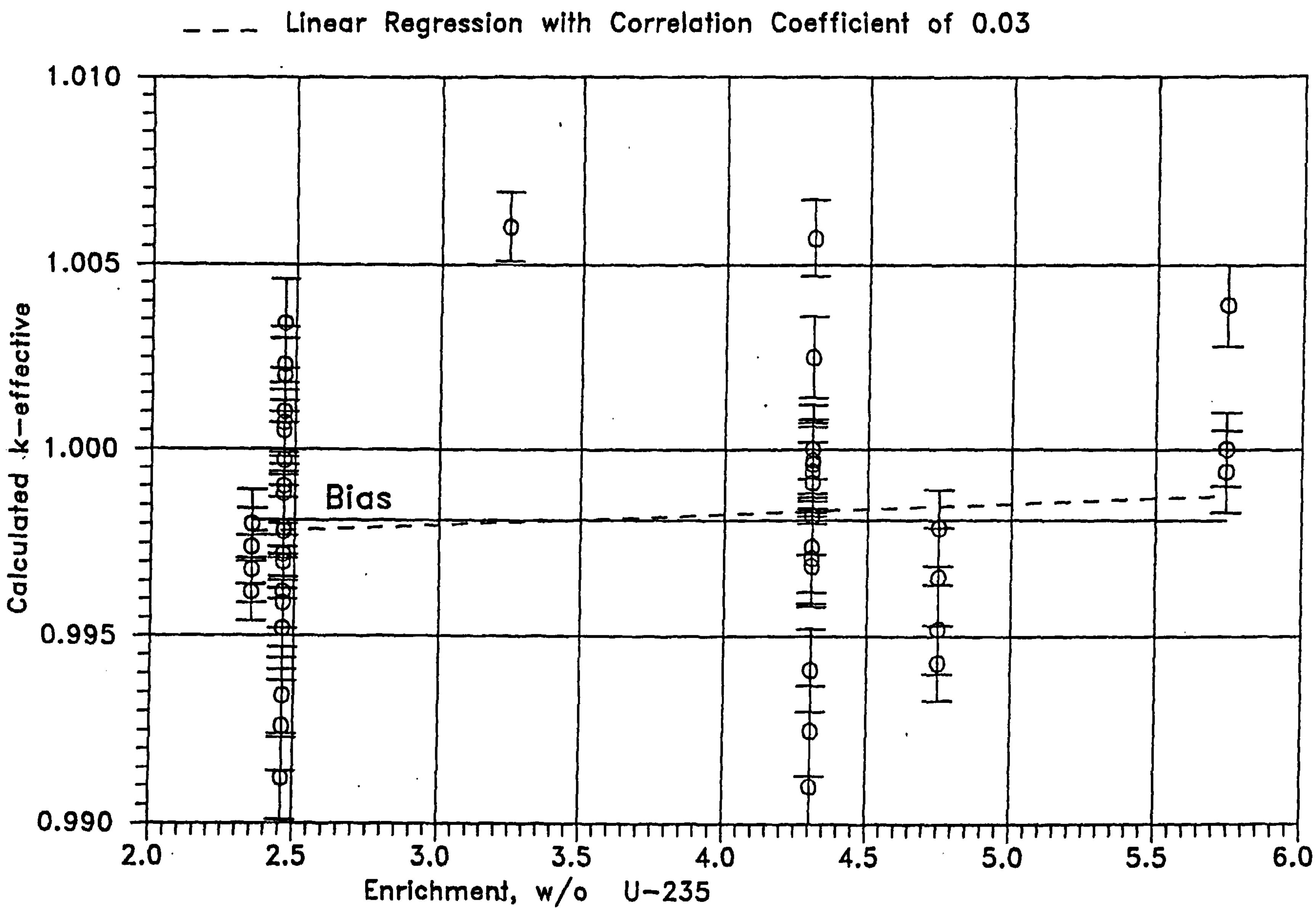

FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

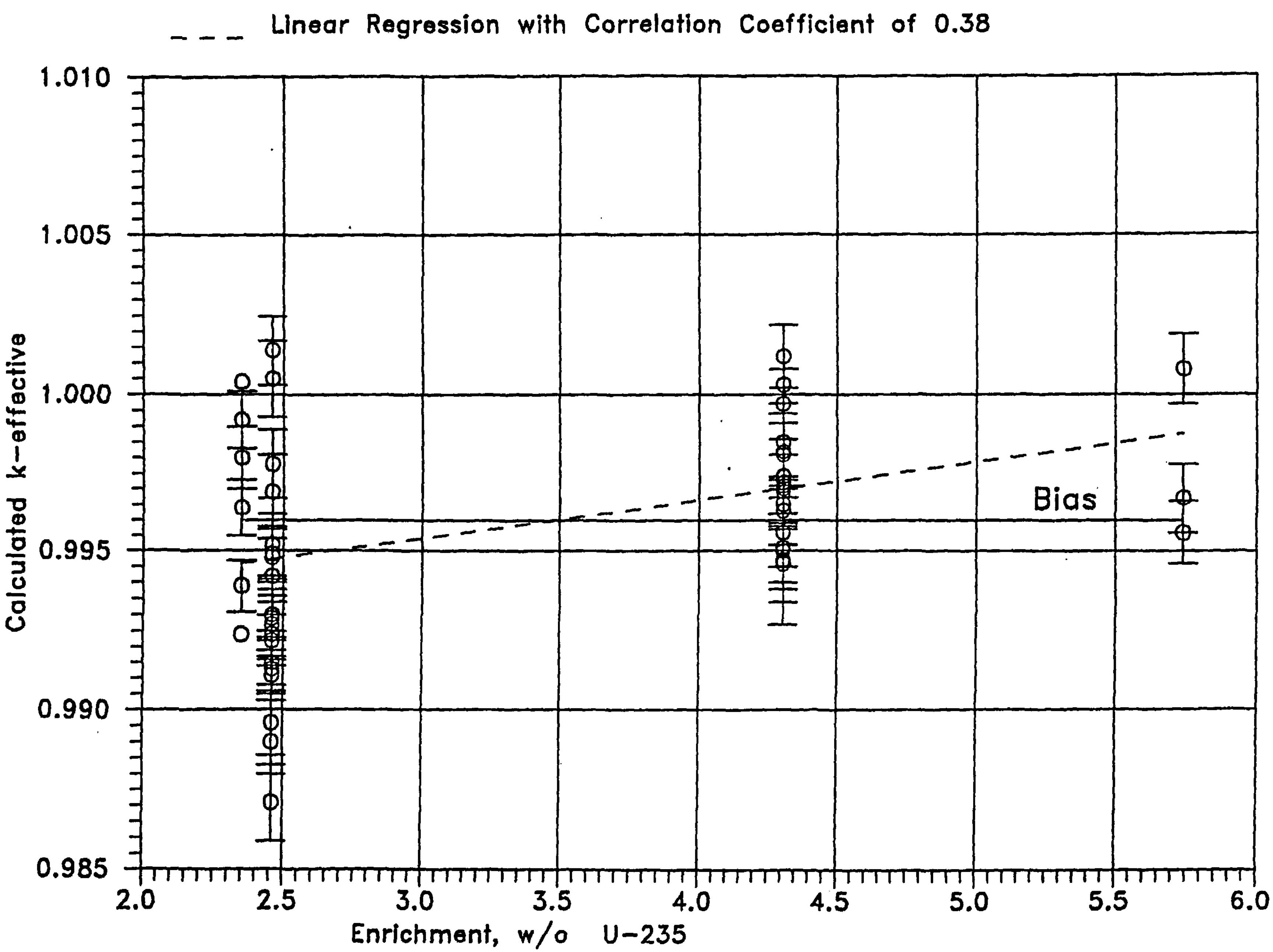

FIGURE 4A.4 KENO CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

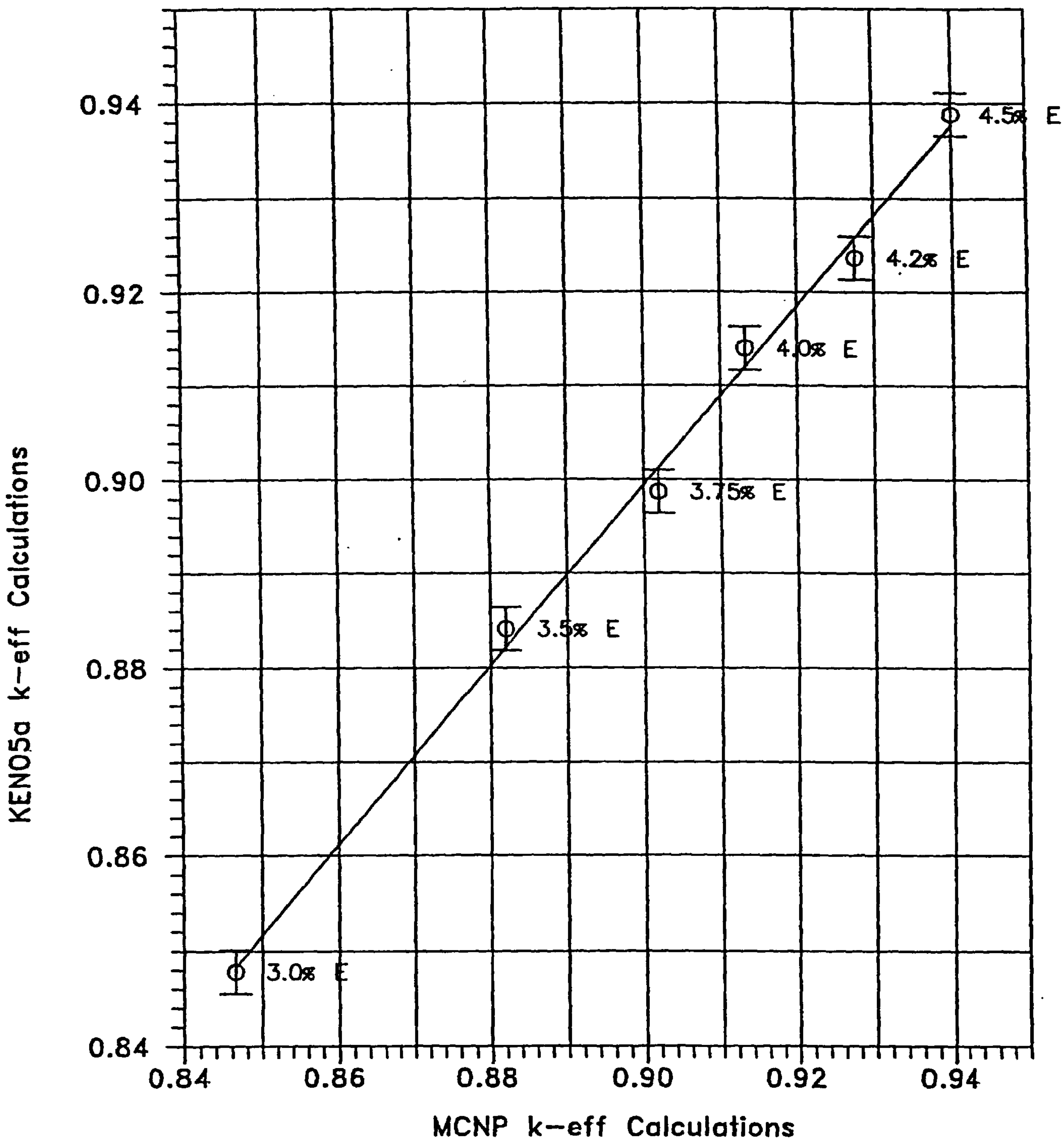

FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

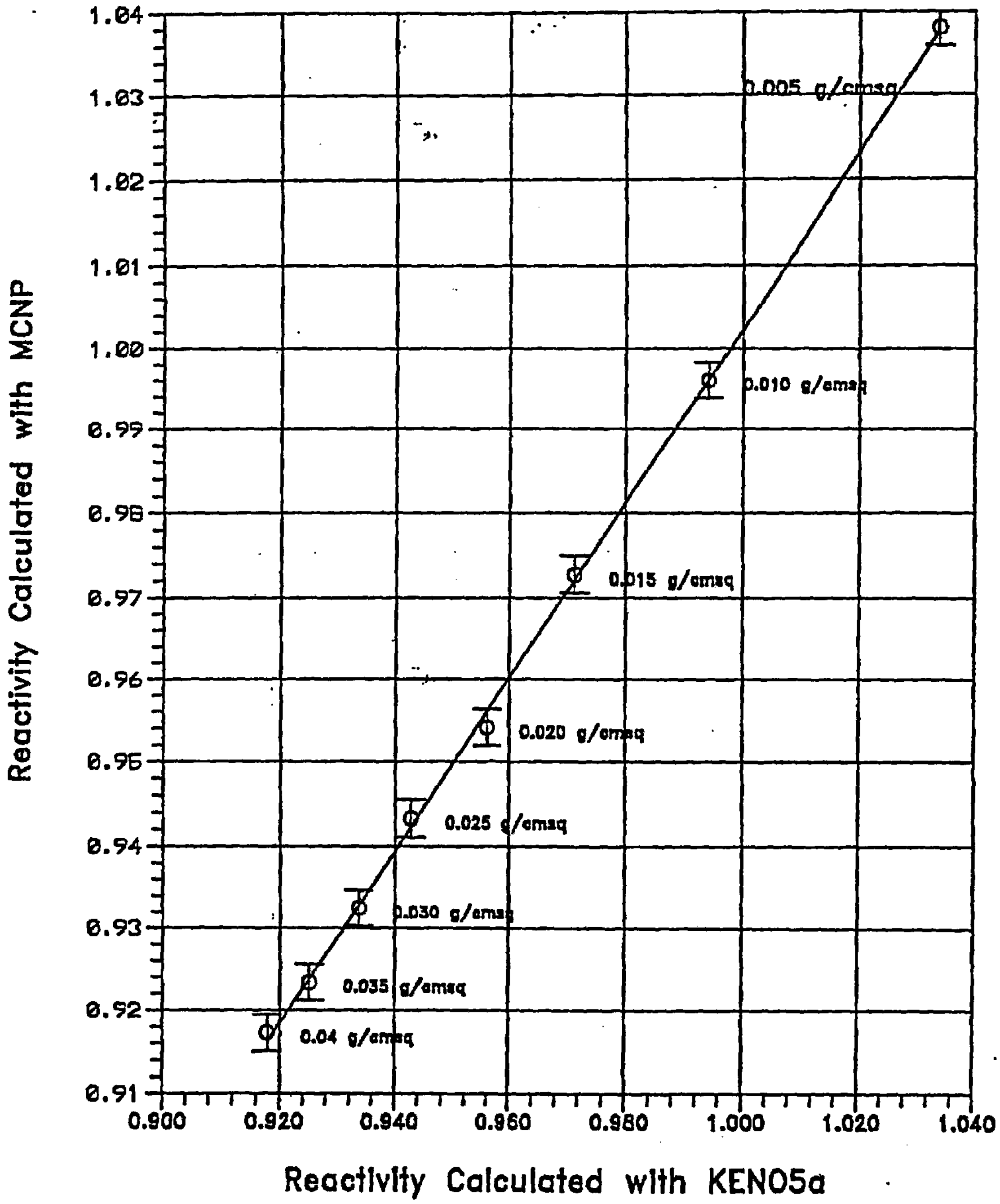

FIGURE 4A.6 COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

direction is approximately 9". The dimensions are chosen to ensure that the rack is centered within the Cask Pit. During rack installation, these dimensions will be met to the extent possible, considering rack and wall straightness and leveling tolerances. The walls and distances separating the Cask Pit Area from the Spent Fuel Pool will effectively eliminate fluid coupling between the proposed rack in each Unit and the racks in the adjacent SFP. The excitation of the proposed racks will be primarily dependent on the motion of the floor and walls of the Cask Pit. The independence of motion of the proposed racks from the racks located in the adjacent SFP allows single rack analysis to produce accurate predictions of the rack motion during dynamic simulations.

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

x = Horizontal axis along plant East
y = Horizontal axis along plant North
z = Vertical axis upward from the rack base

6.3.1 <u>Fuel Weights</u>

The maximum dry weight for a Unit 1 fuel assembly is 1336 lbs. The maximum weight considering an integrally stored Control Element Assembly (CEA) is 1437 lbs. The Unit 1 dynamic rack simulations used a dry weight of 1423 lbs for every fuel assembly to account for CEAs being stored. The difference between the maximum weight with a CEA and that used in the evaluation represents a potential increase of less than 1%. The maximum dry fuel weight for Unit 2 is 1365 lbs. However, for the Unit 2 dynamic rack simulations, a higher fuel weight value of 1384 lbs is used to account for control components being stored along with fuel assemblies. The actual maximum weight of a fuel assembly plus CEA is 1452 lbs. However, this represents an increase of less than 5% over the value used in the evaluations and the analyses conservatively consider the increased weight in the assemblies at every location. Nevertheless, even if every fuel cell contained the heaviest fuel assembly type and an integrally stored CEA the weight increase would not represent a significant impact on actual loads and stresses for either rack. The conservatisms discussed below in Section 6.5.1.1 (especially item c) more than compensate for the possibility of an additional 68 pounds in each cell. Given the small weight difference and large margins of safety in the design, restrictions on CEA storage are not warranted for either Unit.

the platforms are designed as Class 3 Linear-Type component supports and must meet applicable stress levels of ASME Section III, NF-3553 [6.6.1]. The evaluations show that all required geometry checks and stress checks are satisfied for both Level A and Level D conditions and for the lifting condition. Safety factors for bearing, tearout and gross force and moment are greater than 1.0.

8.2.4.4 <u>Static Loading (D = Dead Loads)</u>

1) Dead weight of wall

8.2.4.5 <u>Seismic (E or E') or Tornado Induced Loads (Wt)</u>

1) Vertical and Lateral seismic inertia loads acting on the wall
2) Tornado pressure load
3) Seismic and tornado loading from the overhead crane transferred at the crane support.

8.2.4.6 <u>Thermal Loading Above Elevation 62'</u>

The east wall, above El. 62' is exposed to a maximum inside air temperature of 108°F; the inside wall temperature is approximately 78°F. This is combined with the outside air temperature of 41°F and a conservatively computed surface heat transfer coefficient to establish a lower bound outside wall temperature of 45°F.

8.2.4.7 <u>Load Combinations</u>

Results from a suite of unit load analyses are used to form appropriate load cases and then combined in accordance with the load combinations specified in Subsection 3.8.4.3.2.1 of the St. Lucie Unit 2 UFSAR [8.2.1].

The final load combinations evaluated for structural integrity are:

For "Normal and Severe Environmental Conditions" the following load combinations are:

- Load Combination No. 1 = 1.4* D +1.3* To

- Load Combination No. 2 = 1.4* D+ 1.3* To + 1.9*E

Revised markup pages to Unit 1 and Unit 2 Technical Specifications

, a nominal 10.30 inches center to center distance between fuel assemblies in the Region 1 cask pit storage rack,

DESIGN FEATURES

CRITICALITY (Continued)

spent fuel pool

2. A nominal 10.12 inches center to center distance between fuel assemblies in Region 1 of the storage racks and a nominal 8.86 inches center to center distance between fuel assemblies in Region 2 of the storage racks.

spent fuel pool

3. A boron concentration greater than or equal to 1720 ppm.

4. Neutron absorber (boraflex) installed between spent fuel assemblies in the storage racks in Region 1 and Region 2.

spent fuel pool

Neutron absorber (boral) installed between spent fuel assemblies in the Region 1 cask pit storage rack.

b. Region 1 of the spent fuel storage racks can be used to store fuel which has a U-235 enrichment less than or equal to 4.5 weight percent. Region 2 can be used to store fuel which has achieved sufficient burnup such that storage in Region 1 is not required. The initial enrichment vs. burnup requirements of Figure 5.6-1 shall be met prior to storage of fuel assemblies in Region 2. Freshly discharged fuel assemblies may be moved temporarily into Region 2 for purposes of fuel assembly inspection and/or repair, provided that the configuration is maintained in a checkerboard pattern (i.e., fuel assemblies and empty locations aligned diagonally). Following such inspection/repair activities, all such fuel assemblies shall be removed from Region 2 and the requirements of Figure 5.6-1 shall be met for fuel storage.

c. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U-235 enrichment less than or equal to 4.5 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

DRAINAGE

5.6.2 The fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

storage racks are

5.6.3 The spent fuel pool ~~is~~ designed and shall be maintained with a storage capacity limited to no more than 1706 fuel assemblies~~.~~

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirement.

, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 143 fuel assemblies. The total Unit 1 spent fuel pool and cask pit storage capacity is limited to no more than 1849 fuel assemblies.

DESIGN FEATURES (continued)

CRITICALITY (continued)

5.6.1 d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U-235 enrichment less than or equal to 4.5 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

storage racks are

5.6.3 The spent fuel ~~storage~~ pool ~~is~~ designed and shall be maintained with a storage capacity limited to no more than 1360 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

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

, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 225 fuel assemblies. The total Unit 2 spent fuel pool and cask pit storage capacity is limited to no more than 1585 fuel assemblies.