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March 13, 1984

Dr. Robert M. Lazo, Chairman
Administrative Judge
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dr. Richard F. Cole
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Dr. Emmeth A. Luebke
Administrative Judge
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Re: In the Matter of FLORIDA POWER AND LIGHT COMPANY (Turkey Point Plant, Unit Nos. 3 and 4) - Docket Nos. 50-250, 50-251 OLA

Dear Administrative Judges:

Page 11 of "Licensee's Response to Petitioners' Brief" (Licensee's Response), dated March 14, 1984, states that "No approval to utilize fuel of a higher enrichment has been sought, however, and licensee has not even made a decision as to whether to seek such approval." The purpose of this letter is to inform the Board and the parties that, under letter dated April 4, 1984, Florida Power & Light Company (FPL) requested a set of operating license amendments to accommodate the possible use of higher fuel enrichments at Turkey Point in future cycles.

A copy of the application is enclosed. As stated at page 12 of Licensee's Response, FPL views this separate amendment application to be outside of the scope of this proceeding. However,

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this notice is provided anyway to keep the Board and the parties fully and completely apprised.

Sincerely,

Michael A. Bauser

Michael A. Bauser

DOCKETED
USNRC

FPL

FLORIDA POWER & LIGHT COMPANY

'84 APR 16 A11:18

April 4, 1984
L-84-92

Office of Nuclear Reactor Regulation
Attention: Darrell G. Eisenhut
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 and 50-251
Proposed Amendment to
Facility Operating License DPR-31 and 41
Fuel Storage U-235 Linear Loading Increase

Dear Mr. Eisenhut:

In accordance with 10 CFR 50.90, Florida Power & Light Company (FPL) submits herewith three signed originals and forty copies of a request to amend Facility Operating Licenses DPR-31 and 41.

FPL proposes to modify the existing Turkey Point Units 3 and 4 U-235 linear loading and delete the reactor core U-235 enrichment specification to accommodate storage of higher enrichments for possible use in future fuel cycles.

The proposed modification will involve Technical Specification changes as described below and as shown on the accompanying Technical Specification pages. (Attachment 1).

Technical Specification Page 5.2.1

The reactor core description specification is modified to reflect the deletion of the enrichment restriction.

Technical Specification Page 5.4.1

The fuel storage specification is modified to reflect the proposed increase in fuel storage U-235 linear loading and increase the limiting reactivity in the new fuel storage area.

The proposed facility modification has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board. FPL has concluded that operation of the Turkey Point Plants under the proposed license amendments would not:

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

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

Attachment 2 provides an evaluation of the proposed action in light of three standards contained in 10 CFR 50.92 (listed above, regarding the issue of no significant hazards consideration.

In summary, FPL submits that the activities associated with the proposed amendments do not constitute a significatnt hazard to the public health and safety or to the environment and, therefore, that these amendments do not involve a significant hazards consideration. FPL respectfully requests therefore, that this application be processed pursuant to 10 CFR 50.91 and 50.92, as involving no significant hazards consideration.

Approval of these technical specification changes is desired by August 1, 1984 in order to meet our enrichment notification schedules to allow use of enrichments greater than 3.5 w/o in Turkey Point Unit 4, Cycle 11.

In accordance with 10 CFR 170.22, this proposed license amendment has been determined to represent a Class I and III amendment. Accordingly, a check for $4,400.00 will be forwarded under separate cover.

Very truly yours,

JWWilliams Jr

J. W. Williams, Jr.
Vice President
Nuclear Energy

JWW/ERK/cab

cc: Mr. J. P. O. Reilly
Mr. Harold Reis, Esquire

Lyle E. Jerrett, Administrator
Radiological Health & Rehabilitative Services
1323 Winewood Boulevard
Tallahassee, Florida 32301

Attachments

1. Proposed Technical Specification (pages 5.2-1, 5.4-1)
2. No Significant Hazards Consideration
3. Safety Analysis Report

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

J. W. Williams, Jr. , being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & 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 tne document on behalf of said Licensee.

J. W. Williams, Jr.

Subscribed and sworn to before me this

4 day of APRIL , 1984.

NOTARY PUBLIC, in and for the County of Dade, State of Florida.

My commission expires: 2-14-1988

PROPOSED TECHNICAL SPECIFICATION
Turkey Point 3 and 4

5.2 REACTOR

Reactor Core

1. The reactor core contains approximately 71 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy - 4 tubing to form fuel rods. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods.

2. The average enrichment of the initial core is a nominal 2.50 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.10 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core.

4. Burnable poison rods are in the form of rod clusters, which are located in vacant rod cluster control guide tubes, are used for reactivity and/or power distribution control.

5. There are 45 full length RCC assemblies and 8 partial length* RCC assemblies in the reactor core. The full

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*Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

PROPOSED TECHNICAL SPECIFICATION
Turkey Point Units 3 and 4

5.4 FUEL STORAGE

1. The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class 1 structures. Each spent fuel pit has a stainless steel liner to ensure against leakage.

2. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The fuel in the spent fuel pit is stored vertically in an array with sufficient center-to-center distance between assemblies to assure K_{eff} equal to or less than 0.95 with new fuel containing not more than 52.4 grams of U-235 per axial centimeter of fuel assembly even if boron was not added to the pit water.

   The fuel in the new fuel storage racks is stored vertically in an array with sufficient center-to-center distance between assemblies to assure K_{eff} equal to or less than 0.98 with new fuel containing not more than 57.7 grams of U-235 per axial centimeter of fuel assembly.

3. The boron concentration in the spent fuel pit is that used in the reactor cavity and refueling canal during refueling operations, whenever there is fuel in the pit, except for initial new fuel storage.

ATTACHMENT 2

No Significant Hazards Consideration

Florida Power & Light Company (FPL) presents this evaluation of the hazards considerations involved with the proposed amendment, focusing on the three standards set forth in 10CFR 50.92(c) as quoted below:

> "The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards considerations, unless it finds that operation of the facility in accordance with the proposed amendment would:
>
> 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> 3. Involve a significant reduction in a margin of safety."

FPL submits that the activities associated with this amendment request do not meet any of the significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified. In support of this determination, necessary background information is first provided, followed by a discussion of each significant safety hazards consideration factors with respect to the proposed amendments.

Background

The Turkey Point Plants were designed and constructed with two new fuel storage racks and two spent fuel storage pools, one of each associated with Unit 3 and one with Unit 4. The new fuel storage racks have a capacity of 54 new fuel assemblies. The spent fuel storage pools had a capacity for 217 spent fuel assemblies (equivalent to 1-1/3 cores).

The Turkey Point Units 3 and 4 Final Safety Analysis Report addressed the safety implications of these facilities and included relevant parameters associated with criticality, structural integrity, and cooling. The Turkey Point Units 3 and 4 Safety Evaluation Report (Docket No.'s 50-250 and 50-251) found the environmental and safety impacts of storage in these facilities to be acceptable.

In 1976, a request to amend the Turkey Point operating licenses for increased spent fuel storage was submitted by FPL. By letter dated March 17, 1977, the Commission approved Amendments 23 and 22 to facility operating licenses DPR-31 and DPR-41, respec-

tively, for modification to Turkey Point Units 3 and 4 spent fuel storage facilities. These modifications consisted of reracking the Unit 3 and 4 spent fuel pools with high density fuel storage racks which increased the storage capacity from 217 fuel assemblies to 621 fuel assemblies. Approval of the amendments included a detailed review and analysis of all relevant storage parameters and potential accidents. The analyses resulted in a finding that environmental and safety impacts were negligible.

The safety evaluation performed in support of the request to amend the Turkey Point operating licenses to allow reracking of the Unit 3 and 4 fuel pools addressed the following:

1. Structural and Seismic Analysis

2. Nuclear Criticality Analysis

3. Thermal-Hydraulic

4. Accident Analyses

5. Radiation Exposures

6. Spent Fuel Cask Drop Accident

It was determined that the proposed modifications to the Unit 3 and 4 spent fuel pools would be acceptable because: (1) the rack structural design would withstand conditions during normal operation combined with the maximum earthquake, (2) the rack design would preclude criticality for any moderating condition, (3) the existing spent fuel cooling system was determined to adequately cool the increased heat load and a redundant 100% capacity spare pump would be installed, (4) the increased radiation doses, both onsite and offsite, would be negligible, and (5) spent fuel cask handling operations would not change from the original design.

The current spent fuel storage capacity at Turkey Point consists of 621 storage locations in each spent fuel pool.

With this application, FPL is requesting approval to increase the U-235 linear loading in all fuel storage areas and delete the reactor core reload fuel U-235 enrichment specification, as set forth in the attached Safety Analysis Report.

Evaluation

The following evaluation demonstrates (by reference to the analysis contained in the attached Safety Analysis Report) that the proposed amendment to increase the fuel storage U-235 linear loading does not exceed any of the three significant hazards consideration standards. The analysis of this proposed increase in fuel storage enrichment has been accomplished using current accepted codes and standards as specified in Section 2.1 of the attached Safety Analysis Report. The results of the analysis meet the specified acceptance criteria in these standards as

presented in the Safety Analysis Report. The basis of the proposed deletion of the reactor core reload fuel enrichment specification is that this specification is unnecessary and superfluous in that there are other provisions in the Technical Specifications which determine safe operating and fuel storage limits related to fuel enrichment. These other safe operating limits include dynamic parameters, rod worths and peaking factors. In other words, specification of reload fuel enrichment has no bearing on the safe operation of the reactor core provided that existing safety limits and limiting conditions for operation (LCOs) are satisfied.

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

In the course of the analysis, FPL has identified the following potential accident scenarios:

1. A fuel assembly drop in the spent fuel pool.

2. Loss of spent fuel pool cooling system flow.

3. A spent fuel cask drop.

For 1, "A fuel assembly drop in the spent fuel pool", the criticality acceptance criterion is not violated as identified in Section 3.0 of the Safety Analysis Report. The radiological consequences of this type of accident in the spent fuel pool are bounded by the cask drop accident. Thus the consequences of this type accident will not be significantly increased from previously evaluated fuel assembly drops.

The consequences of 2, "Loss of spent fuel cooling system flow" will not be effected since this application is not intended to qualify the fuel for extended burnup operation. The use of a higher U-235 linear loading by itself will not affect the decay heat characteristics of the fuel assembly or the previous evaluation of the loss of spent fuel cooling system flow. The proposed amendment to increase the fuel storage U-235 linear loading specification will not result in an increase in the probability or consequences of an accident previously evaluated for loss of spent fuel cooling system flow.

The consequences of 3, "A spent fuel cask drop", as previously evaluated will not be affected by an increase in fuel assembly U-235 linear loading since this application is not intended to qualify the fuel for extended burnup nor does this amendment alter the configuration of the storage racks.

The proposed amendment to increase the fuel storage U-235 linear loading will not result in an increase in the probability or consequences of an accident previously evaluated for a spent fuel cask drop.

Thus, it is concluded that the proposed amendment to increase the fuel storage U-235 linear loading and deletion of the reactor core enrichment specification will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

FPL has evaluated the proposed technical specification changes in accordance with the guidance of the NRC position paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", appropriate NRC Regulatory Guides, appropriate NRC Standard Review Plans, and appropriate Industry Codes and Standards as listed in Section 2.1 of the attached Safety Analysis Report. As a result of this evaluation, FPL finds that the proposed technical specification changes do not, in any way, create the possibility of a new or different kind of accident from any accident previously evaluated for the Turkey Point Fuel Storage Facilities.

(3) Involve a significant reduction in a margin of safety.

The NRC Staff Safety Evaluation review process has established that the issue of margin of safety, when applied to modification, will need to address the area of nuclear criticality considerations.

The established acceptance criteria for criticality is that the neutron multiplication factor, including all uncertainties, under all conditions:

(a) shall be less than or equal to 0.98 for the new fuel storage facility; and

(b) shall be less than or equal to 0.95 for the spent fuel pool.

This margin or safety has been adhered to in the criticality analysis methods for the spent fuel and new fuel storage, as discussed in Section 3.0 and 4.0 of the attached Safety Analysis Report.

The methods to be used in the criticality analysis conform with applicable codes, standards, or pertinent sections thereof, as referenced in Section 2.1 of the Safety Analysis Report.

In meeting the acceptance criteria for criticality in the Turkey Point Unit 3 and Unit 4 fuel storage facilities such that:

(a) K_{eff} is always less than 0.98, including uncertainties at a 95/95 probability confidence level in the new fuel storage facility.

(b) K_{eff} is always less than 0.95, including all uncertainties at a 95/95 probability confidence level in the spent fuel pool.

Increasing the limiting K_{eff} in the new fuel storage facility to 0.98 is solely administrative and consistent with the value established in USNRC Standard Review Plan, NUREG-0800, Section 9.1.1.

The proposed amendment to increase the fuel storage U-235 linear loading and increase the limiting K_{eff} in the new fuel storage area will not involve a significant reduction in the margin of safety for nuclear criticality.

In summation, it has been shown that the proposed increase in the fuel storage facility U-235 linear loading, increasing the limiting K_{eff} in the new fuel storage facility, and deletion of the reactor core enrichment specification does not:

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

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

FPL has determined and submits that the proposed amendments described do not involve a significant safety hazard and that the standards in 10 CFR 50.92 have been met.

CRITICALITY ANALYSIS OF THE TURKEY POINT PLANTS UNITS 3 & 4 STORAGE RACKS WITH INCREASED ENRICHMENT

Prepared for the
Florida Power & Light Co.

by
S. E. Turner, Ph.D., P.E.
M. K. Gurley

February 1984

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1.0 INTRODUCTION AND SUMMARY

Both the new fuel and spent fuel storage racks in the Turkey Point Plant, Units 3 & 4, are currently licensed to store fuel of 43.9 grams U-235 per axial centimeter of fuel assembly corresponding to 3.5 wt.% U-235 initial enrichment. The previous criticality analysis, submitted in support of the current Technical Specification limit on fuel enrichment, documented a neutron multiplication factor substantially below the NRC limiting reactivity value of 0.95 including all uncertainties. The evaluation reported here was prepared to justify the criticality safety of an increase in the Technical Specification limit on fuel enrichment in the existing storage racks for both new and spent fuel.

Results of the present evaluation confirm that the maximum reactivity of the spent fuel storage racks will be less than 0.95, including all uncertainties, with the racks fully loaded with fuel containing 52.40 grams U-235 per axial centimeter of fuel assembly and flooded with unborated water at a temperature corresponding to the highest reactivity, provided the UO_2 stack density is no less than 10.08 g/cm^3 (93% of theoretical density). The limiting axial U-235 loading includes tolerances on fuel density and enrichment and corresponds to a nominal enrichment of 4.085 wt.% U-235 at a UO_2 pellet density of 97% of theoretical.

Criticality safety of the new fuel storage rack (a separate facility located near, but independent of, the spent fuel pool) was evaluated for fuel of 57.72 grams U-235 per axial centimeter, corresponding to a nominal enrichment of 4.5% at a UO_2 pellet density of 97% of theoretical. Transport theory calculations confirm that the neutron multiplication factor under optimum moderating conditions (e.g., fog, spray, or foam) is substantially less than the limiting value of 0.98 specified in SRP 9.1.1, "New Fuel Storage."

Other than criticality safety, increasing the enrichment capability of the spent fuel storage pool and new fuel storage racks does not introduce any significant new or unreviewed safety considerations. On the basis of the analyses and evaluations presented herein, it is concluded that the spent fuel

pool and the new fuel storage racks can safely accommodate fuel of 52.40 and 57.72 grams U-235 per axial centimeter of fuel assembly respectively with no significant new or unreviewed hazard considerations under the guidelines of 10CFR50.92(c).

2.0 CRITERIA AND METHODOLOGY FOR CRITICALITY ANALYSES

2.1 Design Bases

The objective in the spent fuel storage racks for the Turkey Point Plant is to assure that a neutron multiplication factor (k_{eff}) equal to or less than 0.95 is maintained with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations and in mechanical tolerances, statistically combined, such that the true k_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level.

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

- General Design Criterion 62 - Prevention of Criticality in Fuel Storage and Handling.
- NRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.1, New Fuel Storage, and Section 9.1.2, Spent Fuel Storage.
- Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis (proposed), December 1981.
- Regulatory Guide 3.41, Validation of Calculational Method for Nuclear Criticality Safety (and related ANSI N16.9-1975).
- ANSI N210-1976, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.
- ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made.

- Moderator is pure, unborated water at a temperature corresponding to the highest reactivity.

- Lattice of storage racks is infinite in all directions; i.e., no credit ~~is~~ taken for axial or radial neutron leakage (except in the consideration of certain abnormal/accident conditions).

- Neutron absorption in minor structural members is neglected; i.e., spacer grids are replaced by water.

- Pure zirconium is used for cladding, control rod guide tubes, and instrument thimbles; i.e., higher neutron absorption of alloying materials in Zircaloy is neglected.

2.2 Reference Fuel Assembly

The reference design fuel assembly, illustrated in Fig. 1, is a 15 x 15 array of fuel rods (Westinghouse design), with 21 rods replaced by 20 control rod guide tubes and one instrument thimble. Two alternative fuel assembly designs have been used in the Turkey Point reactors: an optimized fuel assembly design with Zircaloy grids and an earlier design using inconel grids. The optimized fuel assembly is more reactive and has been used as the reference in the fuel rack criticality analyses. Table 1 summarizes the optimized fuel assembly design specifications and expected range of significant fuel tolerances.

2.3 Reference Fuel Storage Cell

The nominal spent fuel storage cell model used for the criticality analyses is shown in Fig. 1. The rack is composed of 0.25-in. stainless-steel boxes of 8.790-in. inside dimension. The fuel assemblies are centrally located in each storage cell on a nominal lattice spacing of 13.659 in. The outer water space constitutes a flux-trap between the two steel plates. For two-dimensional X-Y analysis, a zero current (white albedo) boundary condition was applied in the axial direction and at the centerline through the outer water space (flux-trap) on all four sides of the cell, effectively creating an infinite array of storage cells.

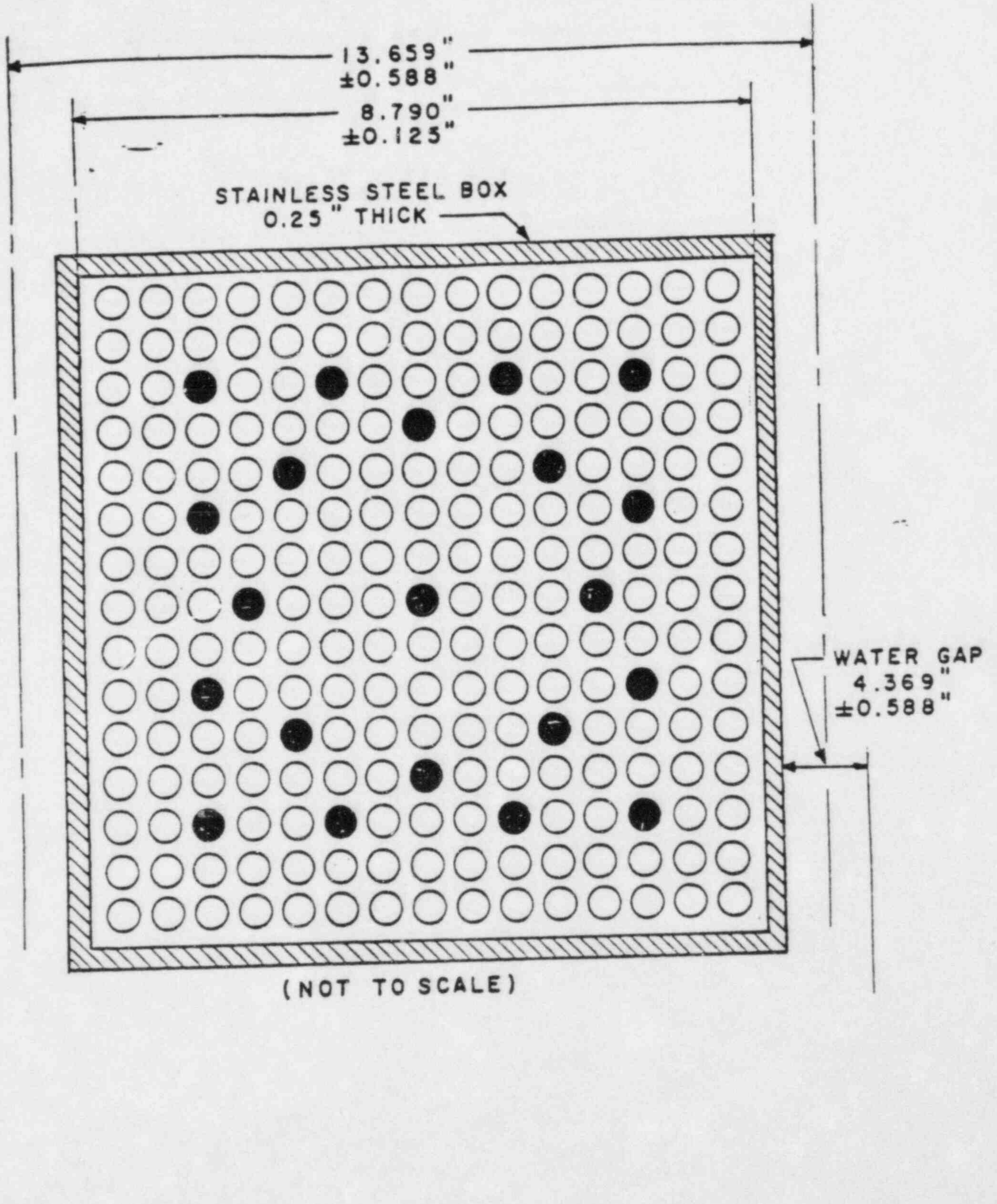

Fig. 1 Reference fuel assembly and configuration of spent fuel storage rack for the Turkey Point Plant.

Table 1 OPTIMIZED FUEL ASSEMBLY DESIGN SPECIFICATIONS

Fuel Rod Data	
Outside dimension, in.	0.422
Cladding thickness, in.	0.0243
Cladding material	Zr-4
Pellet diameter, in.	0.3659
Axial dishing factor	0.988
UO_2 density, g/cm^3	10.08 - 10.514
% T.D.	93 - 97
Fuel Assembly Data	
Number of fuel rods	204 (15 x 15 array)
Fuel rod pitch, in.	0.563
Active fuel height, in.	144
Control rod guide tube	
Number	20
O.D., in.	0.533
Thickness, in.	0.017
Material	Zr-4
Instrument thimble	
Number	1
O.D., in.	0.533
Thickness, in.	0.017
Material	Zr-4

2.4 Analytical Methods

2.4.1 Calculational Models

Four different methods of calculation were used to enhance the credibility of the analysis and to provide assurance that the true reactivity will be less than the limiting value of 0.95 including uncertainties. These methods of calculation include the following.

- CASMO - a two-dimensional multigroup transport theory code[1] (based upon capture probabilities), which provides the capability for a detailed geometric description of the storage cell and each fuel rod.
- AMPX-KENO - a multigroup Monte Carlo code package,[3,4] using the 123-group GAM-THERMOS cross-section set developed by ORNL, and the NITAWL subroutine for U-238 resonance shielding effects (Nordheim integral treatment). AMPX-KENO has been benchmarked against a number of critical experiments (Refs. 5, 6, and 7) with generally good agreement for most critical experiments analyzed, although both Refs. 6 and 7 indicate an underprediction in reactivity in arrangements with large water gaps between fuel assemblies.
- AMPX-KENO - the Monte Carlo technique described above, but using the more recent 27-group SCALE cross-section set[8] developed by ORNL for criticality safety analysis. Benchmark calculations[9,10] indicate that the 27-group SCALE cross-section set consistently underpredicts reactivity by ~0.012 Δk.
- Diffusion/Blackness Theory - a calculational technique based upon the multigroup cell homogenization code, NULIF,[11] to calculate diffusion theory constants for input to PDQØ7.[12] A small correction, based upon blackness theory, was applied to the macroscopic absorption cross-section calculated by NULIF for stainless steel.

For investigation of reactivity effects due to uncertainties (e.g., mechanical and fabrication tolerances), the CASMO code was used to calculate the small incremental reactivity changes. Diffusion/blackness theory calculations were used to estimate the reactivity effects of abnormal/accident conditions.

2.4.2 Calculational Bias and Uncertainty

The infinite multiplication factor (k_∞) for the Turkey Point spent fuel storage rack is based upon CASMO calculations with fuel of the highest anticipated reactivity. Supporting calculations with both the 123-group and 27-group AMPX-KENO code package confirm that the CASMO calculations are conservative. A diffusion/blackness theory calculation (NULIF-PDQ07) provides further confirmation of the conservatism in the reference CASMO calculation. To illustrate the inherent conservatism of CASMO calculations, one case (57.72 grams U-235 per axial centimeter with the minimum cell box inside dimension) was selected for intercomparison. Results of criticality calculations with the four independent methods of analysis for this case are as follows:

Method	Calculated k_∞	Bias	Corrected k_∞
CASMO	0.9404	-	0.9404
123-gp AMPX-KENO (150,000 histories)	0.9052 ± 0.0037*	0.0315	0.9367 ± 0.0037* (95%/95%)
27-gp AMPX-KENO (50,000 histories)	0.9095 ± 0.0063*	0.012	0.9215 ± 0.0063* (95%/95%)
Diffusion/Blackness Theory	0.9340	-	0.9340

The 123-group AMPX-KENO calculational model has been benchmarked[7] against critical experiments as nearly representative as possible of the Turkey Point fuel racks. These benchmark calculations indicate a nominal bias of 0.000 ± 0.003 (95%/95%) plus a correction for the water gap thickness between storage cells. Linear extrapolation of the trend identified in reference 7 to the 4.7-in. water gap of the Turkey Point spent fuel rack results in a bias correction of 0.0315 Δk, although linear extrapolation probably overestimates the water gap correction. Similar benchmark calculations of the 27-group AMPX-

*With one-sided tolerance factor for 95% probability at 95% confidence level.

KENO calculational model, reported in references 9 and 10 (and confirmed by independent calculations), indicate a bias of 0.012 Δk, with no apparent trend with water gap thickness observed at least for the largest water gap (2.58-in.) measured in the critical experiments. Thus, the storage rack k_∞ described above, as calculated by the Monte Carlo technique, probably lies between 0.9215 and 0.9367. Both values confirm the conservatism of the CASMO calculation (k_∞ = 0.9404).

Diffusion/blackness theory calculations would normally be expected to provide a reliable estimate of the rack k_∞, since the stainless-steel wall of the storage cells is a relatively weak absorbing medium. The value calculated by diffusion/blackness theory (k_∞ of 0.9340) tends to further confirm the conservatism of the CASMO calculated value (0.9404). CASMO benchmark calculations on critical experiments representative of the Turkey Point racks (reference 2, Section 2.1.3) suggest an overprediction of 0.006 Δk which, if applied to the reference CASMO calculation, would indicate good agreement with the diffusion/blackness theory value. Nevertheless, for conservatism and to assure the true reactivity is less than the calculated value, the higher k_∞, as calculated by CASMO, was used as the reference value for the Turkey Point spent fuel storage rack. All three of the alternate methods of analysis indicate a lower value and therefore confirm that the reference CASMO value is conservative.

3.0 CRITICALITY ANALYSIS OF SPENT FUEL STORGE RACKS

3.1 Summary of Criticality Analyses

Criticality analyses for the Turkey Point spent fuel storage racks were performed for several fuel densities and U-235 loadings, (in grams per axial centimeter of fuel assembly) and plotted as shown in Fig. 2. These data show that, for a given U-235 axial loading, the higher reactivity results for the lower assumed UO_2 stack density of 10.08 grams per axial centimeter. To this latter curve was added the total uncertainty in k_∞ associated with manufacturing tolerances, 0.0243 Δk, as summarized in Table 2, to generate the upper curve in Fig. 2 identified as the "maximum with uncertainties." This curve was then read to determine the limiting U-235 loading of 52.40 grams per axial centimeter corresponding to the maximum allowble k_∞ of 0.95.

Table 2 SUMMARY OF UNCERTAINTIES IN k_∞ DUE TO TOLERANCES

	k_∞	Reference
Uncertainties		
Fuel enrichment	±0.0008	Section 3.2.1
Fuel density	±0.0006	Section 3.2.1
Cell inside dimension	±0.0020	Section 3.2.2
Lattice spacing	±0.0230	Section 3.2.3
SS tolerance	±0.0015	Section 3.2.4
Statistical combination	±0.0232	
Eccentric positioning	+0.0011	Section 3.2.5
Maximum uncertainty	+0.0243	

The design basis temperature for these calculations was 65.6°C (150°F) which, as shown in Fig. 3 is the coolant temperature of the highest neutron multiplication factor. Lower pool temperatures expected for normal operations, as well as neutron leakage from the finite size racks, provide additional margin in k_{eff} below the limiting value of 0.95 for k_∞ used in the analysis.

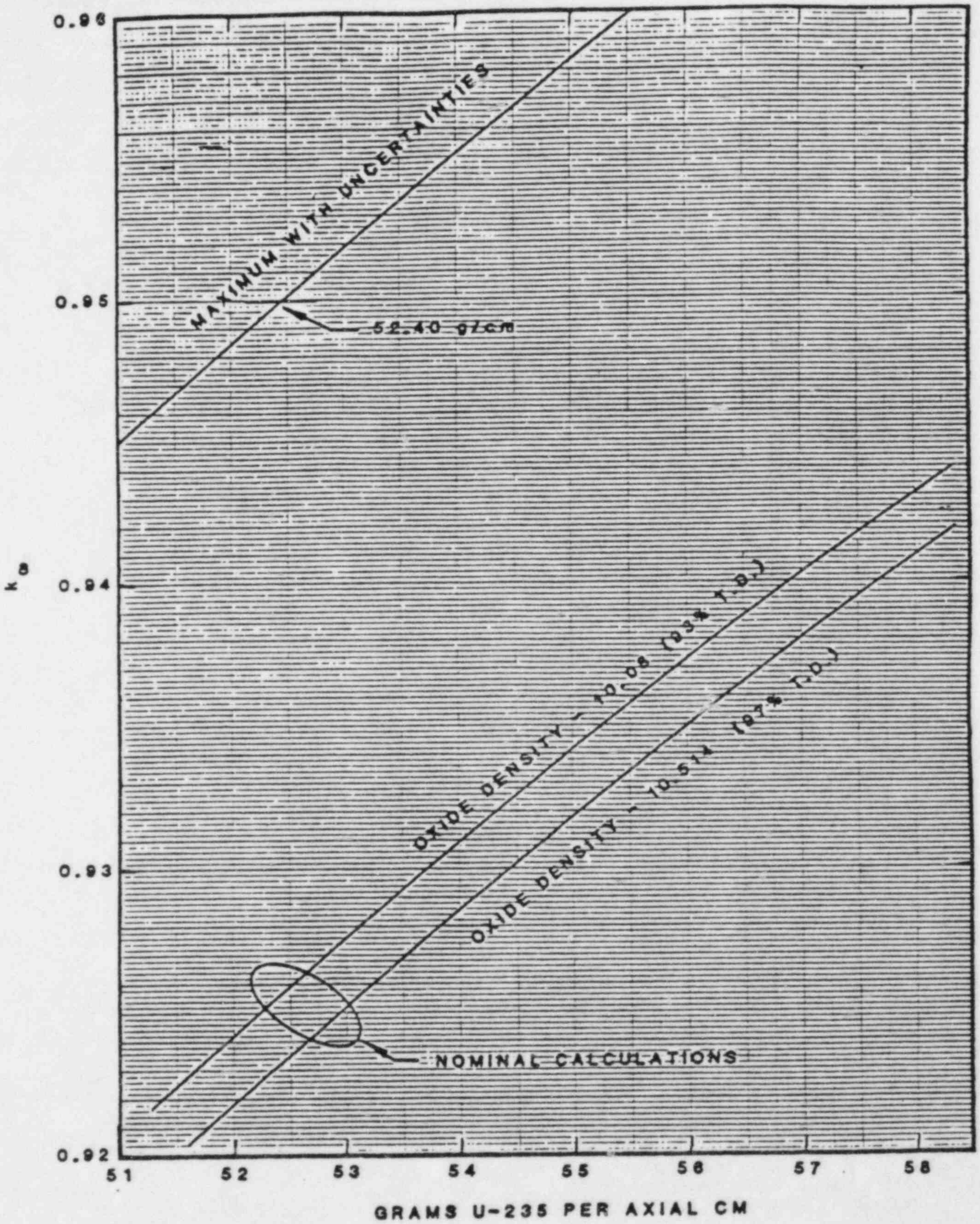

Fig. 2 Infinite multiplication factor of spent fuel storage rack for various U-235 loadings in fuel assembly.

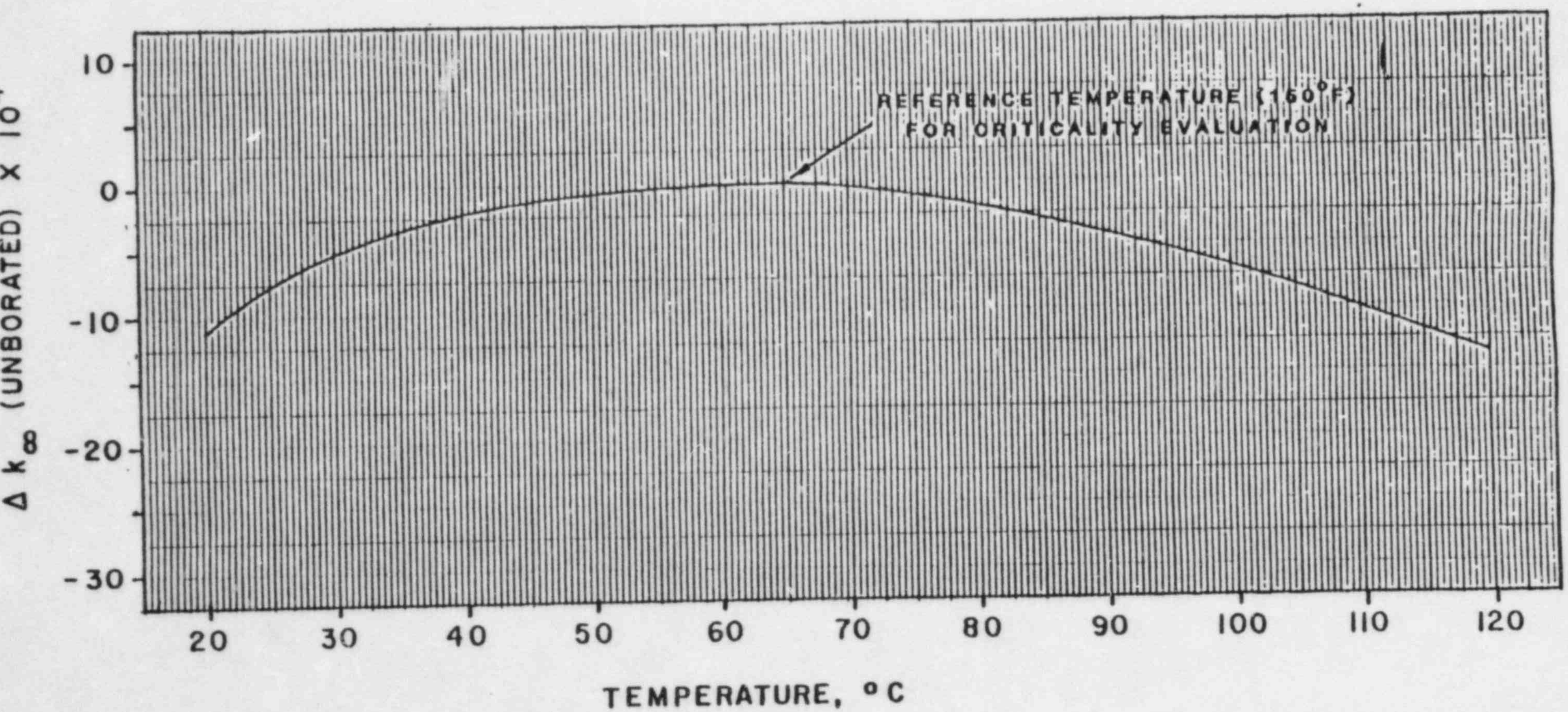

Fig. 3 Effect of coolant temperature on reactivity of spent fuel storage racks.

Thus, to assure a maximum k_{eff} including uncertainties of less than 0.95, a U-235 axial loading of 52.40 grams per axial centimeter is the maximum which may be accommodated in the Turkey Point spent fuel storage racks. This loading may be realized by fuel of 4.261% U-235 enrichment at 10.08 grams per axial centimeter UO_2 stack density (93% T.D.) or lower enrichments at higher UO_2 densities (e.g 4.085% enrichment at a UO_2 stack density of 10.514 grams per axial centimeter, 97% T.D.).

Credible abnormal or accident conditions will not result in exceeding the limiting k_{eff} of 0.95, with credit for the presence of soluble poison (nominally 1950 ppm boron).

3.2 Uncertainties Due to Manufacturing Tolerances

3.2.1 Fuel Enrichment and Density Variations

The maximum loading of 52.40 grams U-235 per axial centimeter can be realized over a range of enrichments and UO_2 stack densities. Table 3 identifies the range considered and gives the k_∞ values for three combinations of enrichment and density.

Table 3 INFINITE MULTIPLICATION FACTORS OVER ANTICIPATED RANGE OF FUEL ENRICHMENTS AND DENSITIES

UO_2 Density % T.D.	g/cm^3	Enrichment, % U-235	Axial Loading g/cm	k_∞ (CASMO)
97	10.514	4.085	52.40	0.9232
95	10.297	4.173	52.40	0.9244
93	10.080	4.261	52.40	0.9256

These data show that the highest k_∞ occurs at a UO_2 density of 93% of theoretical and an enrichment of 4.261 wt.% U-235. Thus, the low density case has been assumed as the design basis for the criticality safety evaluation. Higher UO_2 densities, at the same axial loading in grams U-235 per axial centimeter, will always yield a lower k_∞ .

In addition, there is a certain level of confidence to which the fuel enrichment and density are known. To evaluate the reactivity uncertainty, it is assumed that the UO_2 density is known to ±0.05 g/cm^3 and enrichment to ±0.02 wt.% U-235. Evaluating the uncertainty for these tolerance limits (by differential CASMO calculations) yields an uncertainty of ±0.0006 Δk for density and ±0.0008 Δk for enrichment.

3.2.2 Inside Cell Dimensional Tolerance

The stainless-steel inner box dimension, 8.790 ± 0.125 in., defines the inner water thickness between the fuel and the inside box. For the tolerance of ±0.125 in. on the box inside dimension, the calculated uncertainty in k_∞ is ±0.0020 Δk, with k_∞ increasing as the inner stainless-steel box dimension increases.

3.2.3 Storage Cell Lattice Spacing Variation

The storage cell lattice spacing between fuel assemblies is nominally 13.659 in., positioned by a lattice of support grids intended to provide a nominal water gap between cells of 4.369 in. Receipt inspection of the racks confirmed that the water gap between adjacent storage cells is greater than 3.781 in. for all locations, indicating a toleranace of ±0.588 in. in lattice spacing. Calculations with this minimum spacing between cells resulted in an uncertainty in k_∞ of ±0.0230 Δk due to the tolerance in lattice spacing.

3.2.4 Stainless-Steel Thickness Variations

The nominal stainless-steel box thickness is 0.25 in. The maximum positive effect on k_∞ of the expected stainless-steel thickness tolerance variation (±0.01 in.) was calculated to be ∓0.0015 Δk.

3.2.5 Eccentric Positioning of Fuel Assembly within Storage Cell

The fuel assembly is normally located in the center of the storage cell. Nevertheless, calculations were made with adjacent fuel assemblies

moved into the corner of the storage cell (four-assembly cluster at closest approach), resulting in a small positive effect on k_∞ (0.0011 Δk). Fuel assembly bowing will produce a smaller positive reactivity effect locally. The calculated reactivity increment due to eccentric positioning is considered an additive allowance, although eccentric positioning (if any) would normally be expected to be randomly distributed throughout the storage rack.

3.3 Abnormal and Accident Conditions

3.3.1 Temperature and Water Density Effects

Increasing or decreasing temperature from the nominal temperature of 150°F (65.56°C) is calculated to decrease k_∞ in unborated water as indicated in Fig. 3 (reactivity effects calculated by CASMO). At 120°C (248°F), introducing voids in the water internal to the storage cell (to simulate boiling) further reduced k_∞ indicating a negative void coefficient of reactivity at the boiling temperature. Voids due to boiling will not occur in the outer (flux-trap) water region.

3.3.2 Fuel Assembly Abnormally Located Outside Storage Rack

To investigate the possible effect of a fuel assembly abnormally located outside the rack, diffusion calculations were made for unpoisoned assemblies separated only by water. Figure 4 shows the results of these calculations. From these data, the infinite multiplication factor will be less than 0.95 for any fuel assembly spacing greater than ~15 in. in the absence of any soluble poison or neutron-absorbing material other than water between assemblies.

For a drop on top of the rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation greater than 15 in. (~24 in.). Consequently, fuel assembly drop accidents will not result in an increase in reactivity above that calculated for the infinite nominal design storage rack.

An extraneous fuel assembly cannot physically be positioned outside the rack between the rack and pool wall or between rack modules. However, it is possible, although not likely, to position an extra fuel assembly adjacent to the rack in ~~the~~ region of the cask area. Two-dimensional PDQ calculations show that a fresh fuel assembly positioned adjacent to the storage rack can increase the reactivity to 0.957 in the absence of soluble poison. However, soluble boron of ~1950 ppm is normally present in the spent fuel pool (for which credit is permitted under accident conditions)* and would reduce the maximum k to substantially less than 0.95. Consequently, it is concluded that the postulated accident conditions will not adversely affect the criticality safety of the Turkey Point spent fuel storage racks.

*An implementing Technical Specification for 1950 ppm soluble boron has been submitted via L-84-71, dated March 14, 1984.

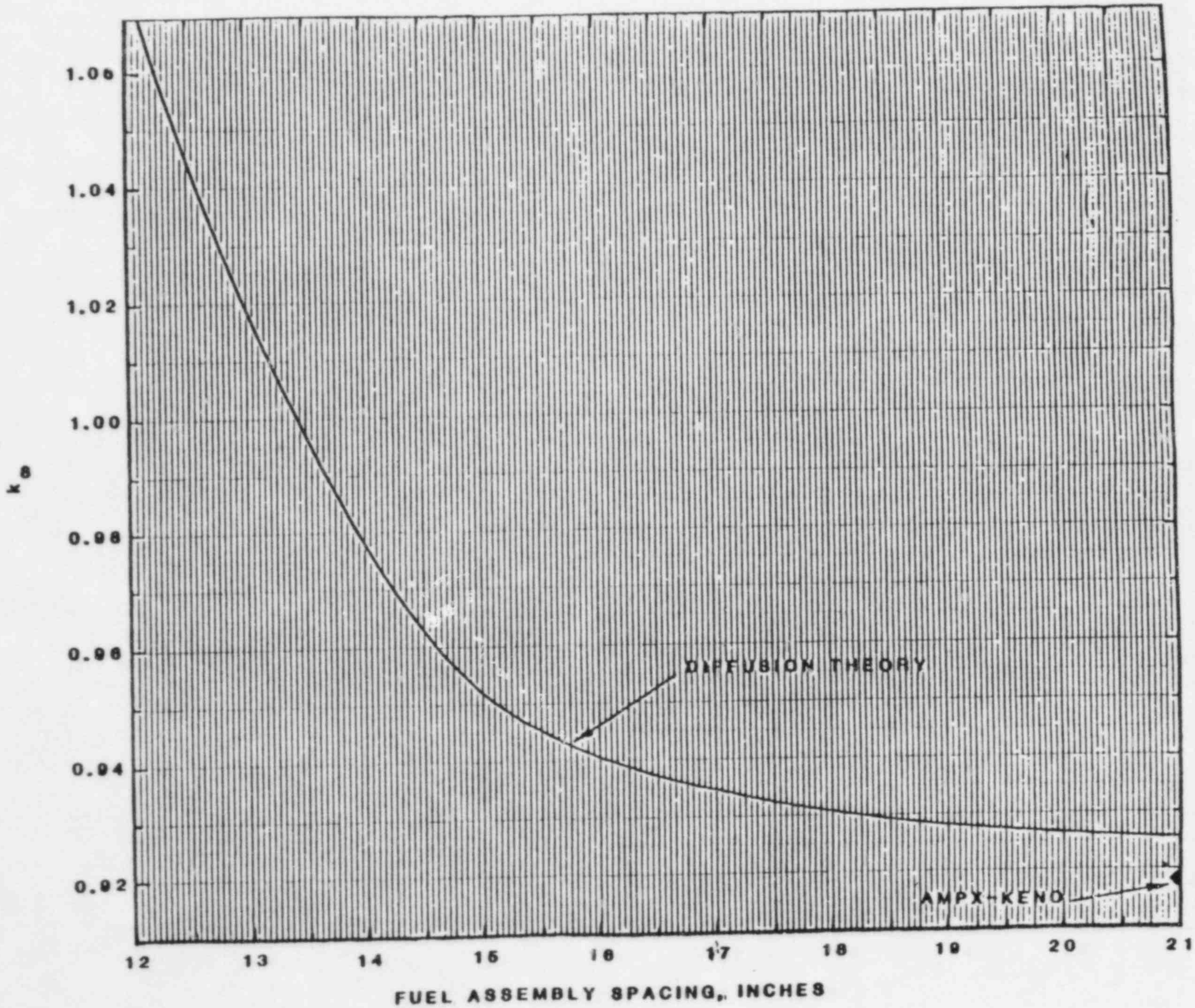

Fig. 4 Variation in k_∞ with fuel assembly spacing (infinite lattice without stainless steel).

4.0 FRESH FUEL STORAGE RACKS

The fresh fuel storage racks for the Turkey Point Plant consist of an "L"-shaped array of storage cells containing 54 cells on a 21-in. lattice spacing. Figure 5 illustrates the fresh fuel storage cell arrangement and shows the geometry used in the criticality analysis. Although fuel is normally stored in the dry condition, the criticality analysis considered flooding with clean, unborated water ranging in density from 1.0 to very low hypothetical values (e.g., fog, mist, or foam). Preliminary survey calculations with diffusion theory suggested a second maximum in reactivity peaking at a hypothetical water density of ~10%-15%.

The criticality safety of the new fuel storage racks was evaluated for fuel of 57.72 grams U-235 per axial centimeter of assembly (~ 4.5% enrichment). Since diffusion theory is known to be inadequate in very dry lattices, three-dimensional AMPX-KENO calculations were used in the low-moderator-density region to define the maximum k_{eff} under optimum moderating conditions. For these calculations, the array of fuel storage cells, as indicated in Fig. 5, was assumed to be reflected by full-density water on the outer boundaries and on both top and bottom of the array. Low-density water was used within the storage boxes and between the array of storage cells. The XSDRNPM routine in the AMPX code package was used to homogenize the fuel assemblies for each moderator density calculated and to generate the weighted cross-section set for use in KENO.

Figure 6 shows the calculated k_{eff} values as a function of moderator density within and between the storage cells. These calculations indicate a low-density maximum reactivity of ~0.925 occurring at a water density of 0.10 g/cm^3. This low-density maximum reactivity is approximately the same as that for the fully flooded condition. In either event, the reactivity is substantially less than the limiting reactivity of 0.98 specified in SRP 9.1.1 under optimum moderating conditions. Hence, it is concluded that unirradiated fuel with a loading of 57.72 grams U-235 per axial centimeter (4.5% enrichment) may be safely stored in the new fuel racks of the Turkey Point Plant.

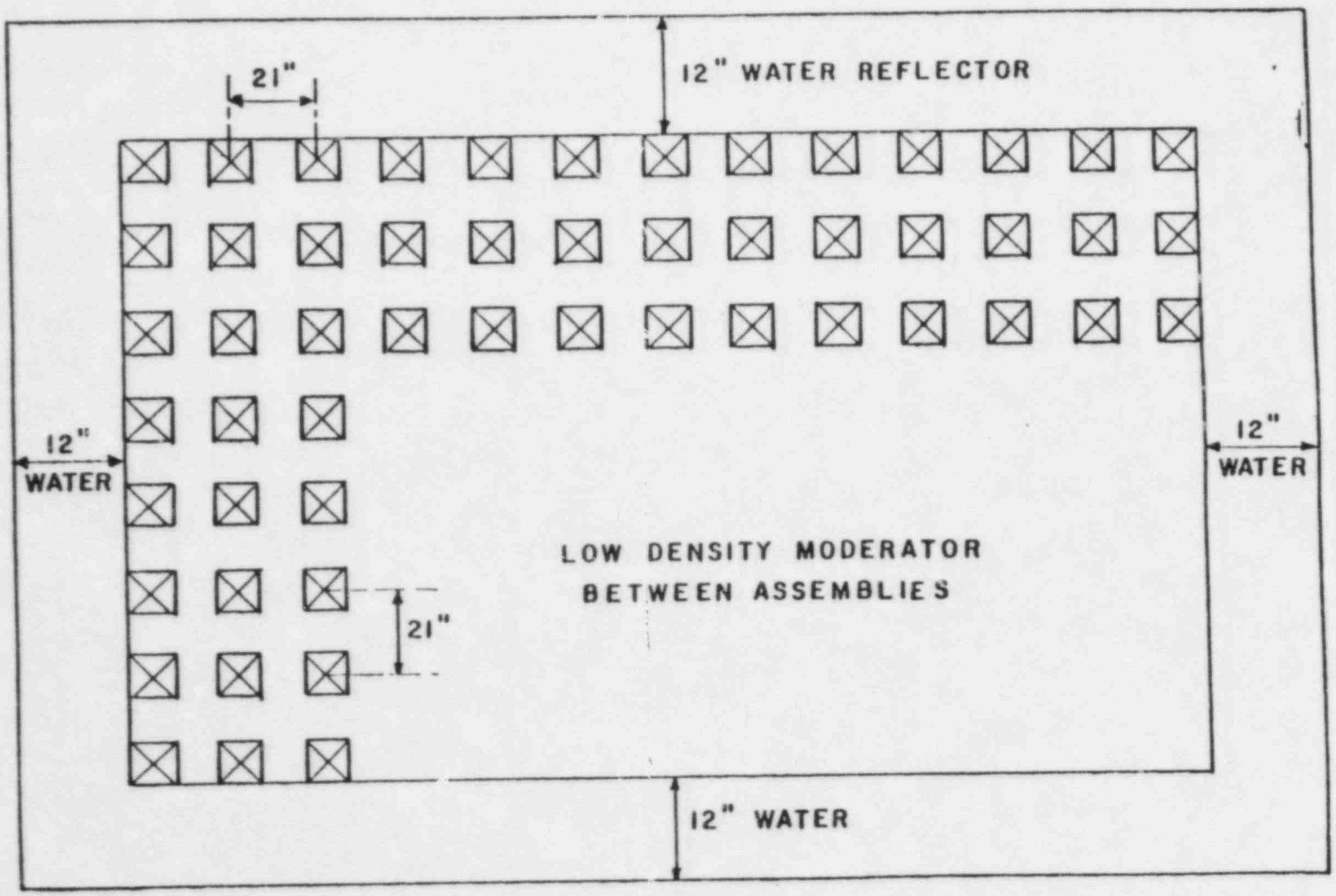

Fig. 5 Fresh fuel storage rack configuration and analytical model.

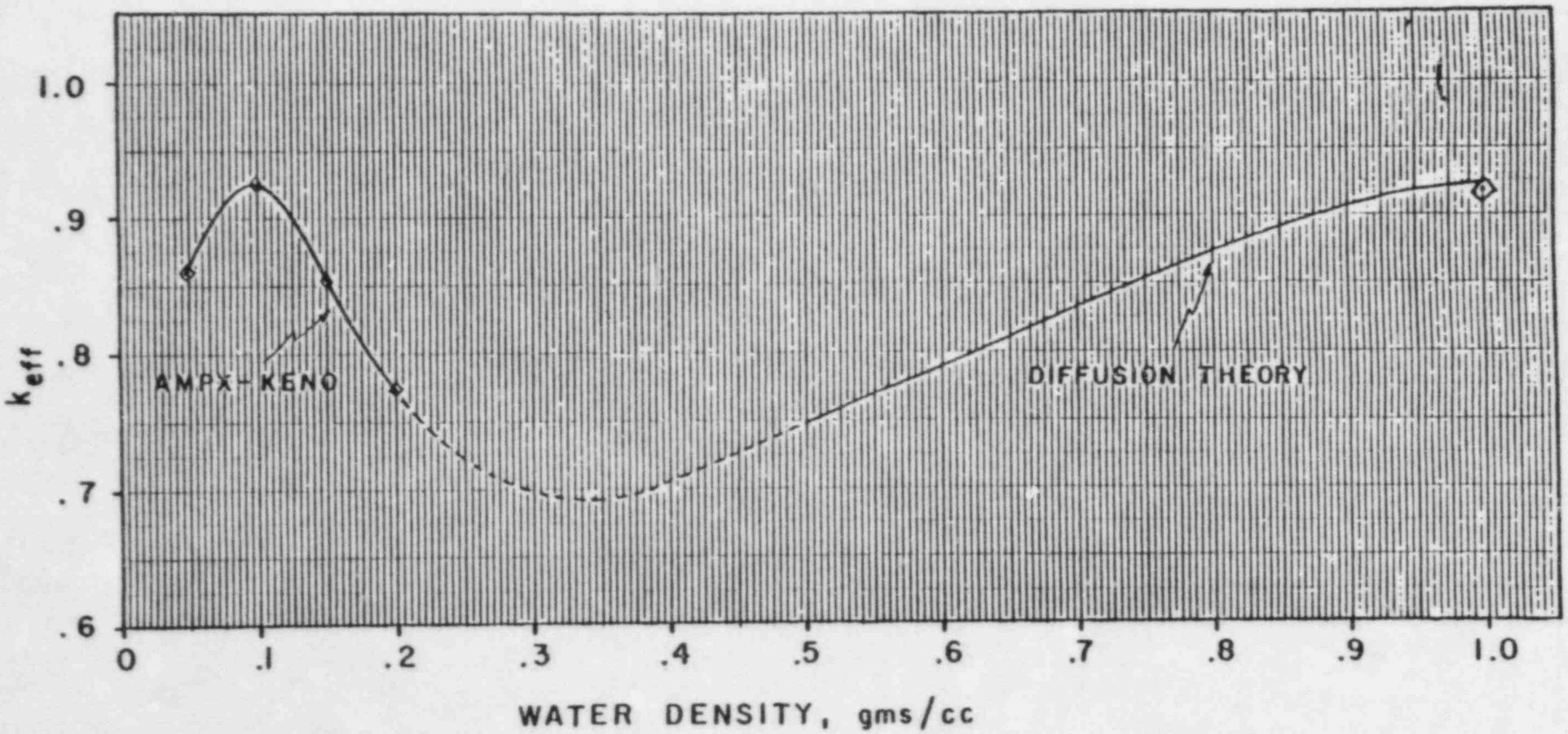

Fig. 6 Reactivity effect of low-density moderator in fresh fuel storage rack. with fuel of 4.5% enrichment.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

'84 APR 16 A11:18

In the Matter of	)	
	)	OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH
FLORIDA POWER & LIGHT COMPANY	)	Docket Nos. 50-250 OLA
	)	50-251 OLA
(Turkey Point Nuclear Generating Units 3 and 4	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of a letter to the Members of the Licensing Board in the above-captioned proceeding, dated April 13, 1984, and attachments thereto were served on the following by deposit in the United States mail, first class, properly stamped and addressed, on the date shown below.

Dr. Robert M. Lazo, Chairman
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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Chief, Docketing and Service Section
(original plus two copies)

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