Florida Power
CORPORATION

December 10, 1985
3F1285-05

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Technical Specification Change Request No. 140

Dear Sir:

Florida Power Corporation (FPC) hereby submits the enclosed three (3) originals and forty (40) copies of Technical Specification Change Request No. 140 requesting amendment to Appendix A of Operating License No. DPR-72. As part of this request, the proposed replacement pages for Appendix A are enclosed.

This submittal requests an amendment to the "design features" in the Technical Specifications to allow an increase in the allowable fuel enrichment to 4% in fuel pool B and the dry fuel storage rack. The two analyses attached provide technical justification for this increase. The analysis for the dry storage rack assumes the fuel is loaded in three 6 x 3 arrays such that every fourth row in the 6 x 11 rack is vacant. FPC will cover the rows assumed to be vacant to prevent fuel insertion in these rows. FPC would appreciate dialogue with the appropriate NRC reviewer about the content of this request during the week of January 27-31, 1986.

A001
3/40

ADD: PWR - B/RC's TECH SUPPORT
AD - D. CRUTCHFIELD (Ltr only)
EB (W. JOHNSTON)
RSB (THOMAS)
EICSB (PARR)
FOB (W. REGAN)

8512160140 851210
PDR ADOCK 05000302
P PDR

Rec'd w/check $150.00
#750540

GENERAL OFFICE 3201 Thirty-fourth Street South • P.O. Box 14042, St. Petersburg, Florida 33733 • 813—866-5151

An amendment application fee, check number 750540, of one hundred fifty dollars ($150), as required by 10 CFR 170, has been included with this Change Request.

Sincerely,

E C Simpson

E. C. Simpson
Director, Nuclear Operations
Engineering & Licensing

PGH/feb

Enclosures

xc: Dr. J. Nelson Grace
Regional Administrator, Region II
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, GA 30323

STATE OF FLORIDA

COUNTY OF PINELLAS

E. C. Simpson states that he is the Director, Nuclear Operations Engineering and Licensing for Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

E C Simpson

E. C. Simpson
Director, Nuclear Operations
Engineering and Licensing

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 10th day of December, 1985.

Sheryl D. Roberson

Notary Public

Notary Public, State of Florida at Large,

My Commission Expires: July 22, 1989

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF)
) DOCKET No. 50-302
FLORIDA POWER CORPORATION)

CERTIFICATE OF SERVICE

E. C. Simpson deposes and says that the following has been served on the Designated State Representative and the Chief Executive of Citrus County, Florida, by deposit in the United States mail, addressed as follows:

Chairman,
Board of County Commissioners
of Citrus County
Citrus County Courthouse
Inverness, FL 32650

Administrator
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, FL 32301

A copy of Technical Specification Change Request No. 140 requesting amendment to Appendix A of Operating Licensing No. DPR-72.

FLORIDA POWER CORPORATION

E C Simpson

E. C. Simpson
Director, Nuclear Operations
Engineering and Licensing

SWORN TO AND SUBSCRIBED BEFORE ME THIS 10th DAY OF DECEMBER 1985.

Sheryl D. Roberson

Notary Public

Notary Public, State of Florida at Large
My Commission Expires: July 22, 1989

(NOTARIAL SEAL)

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72
REQUEST NO. 140, REVISION 0
FUEL ENRICHMENT INCREASE

LICENSE DOCUMENT INVOLVED: TECHNICAL SPECIFICATION

PORTIONS: 5.3.1 FUEL ASSEMBLIES
5.6.1 CRITICALITY

DESCRIPTION OF REQUEST:

This submittal requests an increase in the allowable nominal fuel enrichment up to 4.0 weight percent U-235 for the dry fuel storage racks, Storage Pool B and the reload fuel assemblies.

REASON FOR REQUEST:

Florida Power Corporation plans to use approximately 4.0% enriched fuel during Cycle 7 and possibly subsequent cycles. Our current fuel storage analyses and Technical Specifications reflect a 3.5% enrichment for all storage areas. This change supports an enrichment increase to 4.0% for Storage Pool B and the dry fuel storage racks only.

SAFETY EVALUATION OF REQUEST:

The purpose of limiting allowable fuel enrichment of assemblies stored in the wet and dry racks is to assure sufficient safety margin exists to prevent inadvertent criticality. This is done by assuring a Keff 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 attached analysis performed by Southern Science for Florida Power indicates that storage of 4.0% (nominal) enriched fuel in Pool B will not cause Keff to exceed 0.95 under the conditions above. A second analysis for the dry storage racks is also attached. The racks are assumed to be loaded in three 6 x 3 arrays such that every fourth row in the 6 x 11 rack is vacant. The analyses include margins for uncertainty in reactivity calculations and in mechanical tolerances.

SHOLLY EVALUATION OF REQUEST:

Using the standards in 10 CFR 50.92, Florida Power Corporation concludes this amendment will not involve a significant hazards consideration for the following reasons:

1. This amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   An increase in fuel enrichment will not by itself affect the mixture of fission product nuclides. A change in fuel cycle design which makes use of an increased enrichment may result in fuel burnup consisting of a somewhat different mixture of nuclides. The effect in this instance is insignificant for the following reasons.

a) The isotopic mixture of the irradiated assembly is relatively insensitive to the assembly's initial enrichment.

b) Because most accident doses are such a small fraction of 10 CFR 100 limits, a large margin exists before any change becomes significant.

c) The change in Pu content which would result from an increase in burnup would produce more of some fission product nuclides and less of other nuclides. Small increases in some doses are offset by reductions in other doses. The radiological consequences of accidents are not significantly changes.

2. This amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

As indicated in the enclosed analyses, an unplanned criticality event will not occur as Keff will not exceed .95 even with Pool 'B' fully loaded with the highest enrichment fuel and flooded with cold unborated water or dry storage racks immersed in a water mist of 7.5% moderator density. Criticality is possible for a mist environment only if the higher enriched fuel occupies all of the locations in the dry storage racks including those which are required to be vacant. To prevent this occurrence, Florida Power commits to establish controls to preclude improper fuel storage.

3. This amendment will not involve a significant reduction in a margin of safety.

While the increased enrichment in Pool 'B' and the dry storage racks may lessen the margin to criticality, this reduction is not significant because the overall safety margin is within NRC criteria of Keff $\leq$.95 (NRC Standard Review Plan, Section 9.1.2).

Therefore, this amendment request satisfies the criteria specified in 10 CFR 50.92 for amendments which do not involve a significant hazards consideration.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The Reactor Containment building is designed and shall be maintained for a maximum internal pressure of 55 psig and a temperature of 281°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy - 4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2253 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.0 (nominal) weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 60 safety and regulating and 8 axial power shaping (APSR) rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The APSRs shall contain a nominal 63 inches of absorber material at their lower ends. The absorber material for the APSRs shall be 100% Iconel.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line, which is 670°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,180 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEORLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The dry storage racks and the spent fuel storage racks in pool "B" are designed and shall be maintained with a nominal 21-1/8 inch center-to-center distance between fuel assemblies placed in the storage racks. The high density spent fuel storage racks in pool "A" are designed and shall be maintained with a nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks. All of these rack designs ensure a keff equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The keff of $\leq$ 0.95 includes a conservative allowance of >1% $\Delta K/K$ for uncertainties. In addition, fuel stored in pool "A" shall have a U-235 loading of $\leq$ 46.14 grams of U-235 per axial centimeter of fuel assembly ($\leq$ an enrichment of 3.5 weight percent U-235). Fuel stored in the dry storage racks and pool "B" shall have a U-235 loading of $\leq$ 52.73 (nominal) grams of U-235 per axial centimeter of fuel assembly ($\leq$ an enrichment of 4.0 (nominal) weight percent U-235).

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 138 feet 4 inches.

CRITICALITY SAFETY ANALYSIS

OF

THE NEW-FUEL STORAGE VAULT

WITH

FUEL OF 4.5% ENRICHMENT

1.0 INTRODUCTION

In a companion evaluation (SSA-160), the results of the criticality safety analysis of the pool "B" standard spent fuel storage racks were presented in support of an increase in the Technical Specification limit on fuel enrichment. The evaluation reported here is intended to supplement that analysis and to qualify the new-fuel storage facility to receive fuel of up to 4.5% enrichment.

To assure the criticality safety of the new-fuel storage vault and to conform to the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," two separate criteria must be satisfied as defined in NUREG-0800, Standard Review Plan 9.1.1, "New Fuel Storage." These criteria are as follows.

- o When fully loaded with fuel of the highest anticipated reactivity and flooded with clean unborated water, the maximum reactivity, including uncertainties, shall not exceed a k_{eff} of 0.95.
- o With fuel of the highest anticipated reactivity in place and assuming the optimum hypothetical low density moderation (i.e., fog or foam), the maximum reactivity shall not exceed a k_{eff} of 0.98.

Results of the present evaluation confirm that, with assemblies of up to 4.5% enrichment, the new-fuel storage vault at the Crystal River nuclear station satisfies both criteria cited above, provided administrative procedures are employed to preclude the use of certain specified storage locations as defined herein.

2.0 SUMMARY

The new-fuel storage vault at the Crystal River nuclear plant normally provides a 6 x 11 array of storage locations arranged on a 21-1/8-inch-square lattice spacing, as shown in Fig. 1. Results of the criticality safety analysis of this storage rack with fuel of 4.5% enrichment are summarized in Table 1 for the two design criteria cases. In both cases--with one restriction--the maximum reactivity (k_{eff}) including all known uncertainties is less than the corresponding reactivity limit, confirming that the new-fuel storage vault can safely accommodate fuel assemblies of 4.5% enrichment. The single restriction is that administrative procedures are necessary to preclude the use of six locations in each of two rows, as indicated by the shaded bars in Fig. 1.

The limiting reactivity condition occurs for the low density moderator case in which optimum moderation (maximum k_{eff}) occurs at a hypothetical moderator density of ~7.5%. With the 12 indicated storage locations vacant, there are 54 usable storage locations arranged in three groups of 3 x 6 storage locations. With all 54 usable locations filled with fuel of 4.5% enrichment, the maximum reactivity at 7.5% moderator density is 0.952 including uncertainties, which is below the limiting value of 0.98.

In the case of flooding with clean unborated water, the resulting storage cell configuration is identical to that of a pool "B" standard fuel storage cell and no restrictions are necessary. Under these conditions, and with fuel of 4.5% enrichment, the maximum reactivity including all known uncertainties is 0.944 (actually k_{∞}, since an infinite array was assumed in the analysis). This maximum reactivity is less than the limiting value of 0.95.

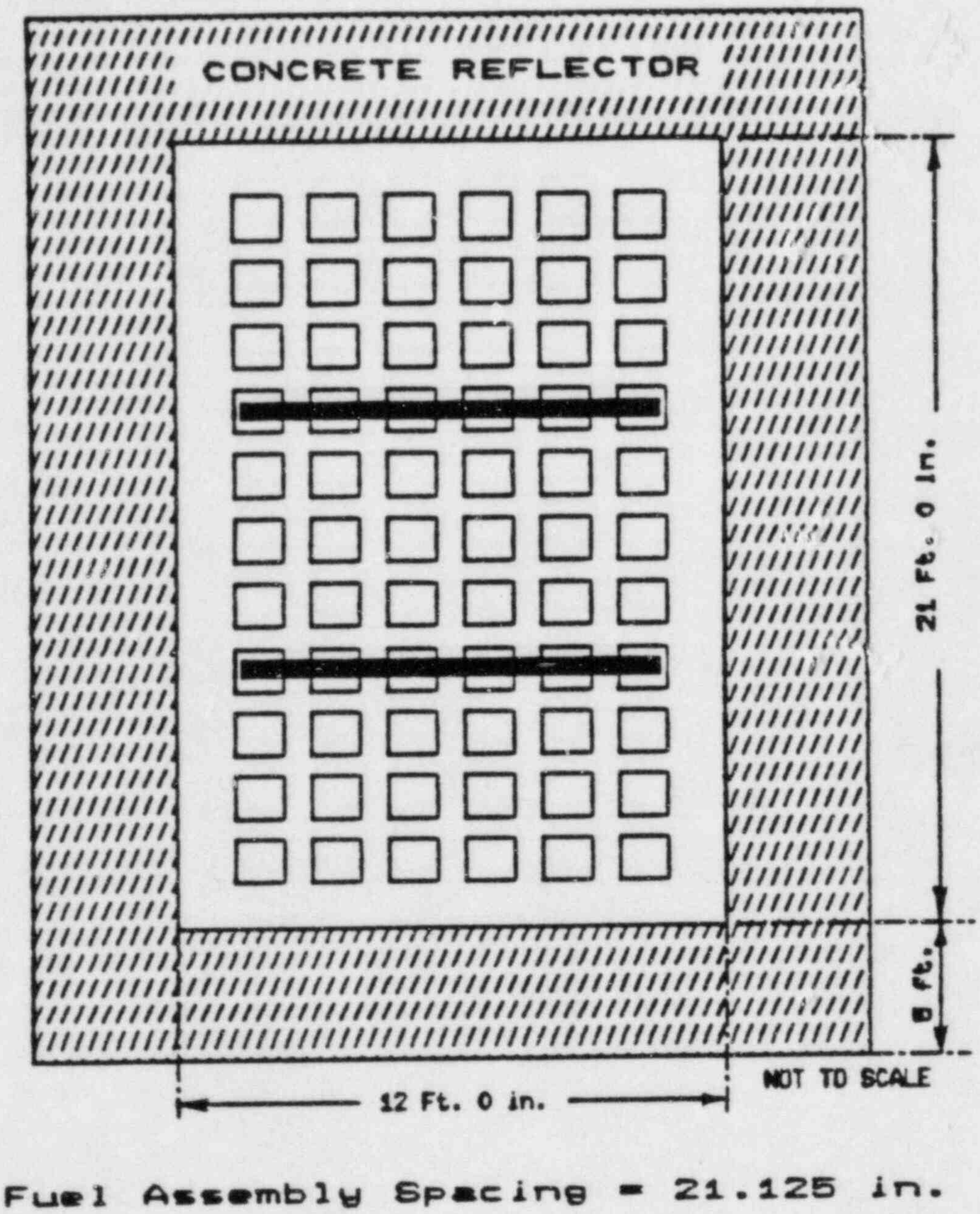

Fig. 1 Crystal River nuclear plant new-fuel storage vault.

Table 1 CRITICALITY SAFETY ANALYSIS OF THE NEW-FUEL STORAGE VAULT WITH FUEL OF 4.5% ENRICHMENT

	Flooded	Optimum Moderation
1. Moderator density, %	100	7.5
2. Nominal k_{eff}	0.9393	0.9412[1]
3. Calculational bias, Δk	0.0013	0.0024
4. Uncertainty in bias, Δk	0.0018	0.0030
5. Statistical variation, Δk	NA	0.0079
6. Fuel enrichment & density, Δk	0.0024	0.0024
7. Total	0.9406 ± 0.0030[2]	0.9436 ± 0.0088[2]
8. Maximum k_{eff}	0.9436	0.9524
9. Analytical methodology	CASMO-2E	123 Gp. AMPX-KENO
10. Reactivity limit	0.95	0.98

(1) Assumes certain specified positions are vacant.
(2) Includes statistical combination of Items 4, 5 and 6.

3.0 CRITICALITY ANALYSES

3.1 Reference Fuel Assembly

The reference design fuel assembly is a standard Babcock & Wilcox 15 x 15 array of fuel rods, with 17 rods replaced by 16 control rod guide tubes and one instrument thimble. Table 2 summarizes the fuel assembly design specifications and expected range of significant fuel tolerances.

3.2 Flooded Conditions

When flooded with clean unborated water, the array of fuel assemblies in the new-fuel vault is identical to that of the pool "B" storage cells evaluated in SSA-160.[1] For this case, calculations were made for an infinite array of fuel assemblies on a 21.125-inch lattice spacing, using the CASMO-2E[2] program, documented in SSA-160 as being conservative. In the CASMO-2E calculational model, each fuel pin is represented explicitly and the fuel assemblies are assumed to be separated only by clean unborated water at 20°C. At higher temperatures, reactivity is further reduced as documented in SSA-160.

With fuel of 4.5% enrichment, the CASMO-2E calculated infinite multiplication factor is 0.9393, and, assuming the uncertainties evaluated in SSA-160, the maximum reactivity is 0.9436. This maximum reactivity (k_∞) is less than the limiting k_{eff} value of 0.95 and confirms that the new-fuel storage vault satisfies the criticality safety requirement for fuel of 4.5% enrichment in the flooded condition.

3.3 Low-density Optimum Moderation

For the hypothetical conditions of low density moderation, criticality analyses were performed with the AMPX[3]-KENO[4] computer package (Monte Carlo) using the 123-group GAM-THERMOS cross-section library and the NITAWL routine for U-238 resonance shielding (Nordheim integral treatment). This method of analysis has been benchmarked[5] against critical experiments and found to have

Table 2 FUEL ASSEMBLY DESIGN SPECIFICATIONS

Fuel Rod Data	
Outside dimension, in.	0.430
Cladding thickness, in.	0.0265
Cladding material	Zr-4
Pellet diameter, in.	0.369
UO_2 density, g/cm^3	10.420 ± 0.166
Enrichment, wt.% U-235	4.50 ± 0.02
Fuel Assembly Data	
Number of fuel rods	208 (15 x 15 array)
Fuel rod pitch, in.	0.568
Control rod guide tube	
Number	16
O.D., in.	0.530
Thickness, in.	0.016
Material	Zr-4
Instrument thimble	
Number	1
O.D., in.	0.493
Thickness, in.	0.026
Material	Zr-4

a bias of 0.000 ± 0.003 (95% probability at a 95% confidence level) plus a small correction for the water gap between fuel assemblies (+0.0024 Δk at 7.5% density). Although no critical experiments are available for the hypothetical low density moderator, Napolitano et al.[6] have compared the 123-group AMPX-KENO model with continuous-energy SAM-CE calculations with good agreement. These results provide additional confidence in the calculated k_{eff} value for the new-fuel storage vault.

Preliminary calculations indicated that, with the new-fuel vault filled with fuel in all locations, the calculated reactivity would not provide an adequate subcriticality margin at optimum low density moderation. By trial and error, it was determined that by leaving 12 storage locations vacant in two symmetric rows of six locations each (as illustrated in Figs. 1 and 2), the reactivity is reduced sufficiently to provide an adequate subcriticality margin.

The geometric model used in the AMPX-KENO calculations of the new-fuel storage vault is illustrated in Fig. 2. Homogenized compositions were used for the fuel region with U-238 resonance shielding effects calculated in the NITAWL routine of AMPX. (Independent sensitivity studies with KENO showed that k_{eff} values calculated with homogenized compositions and with explicit pin descriptions were indistinguishable at the low moderator densities.)

Results of the AMPX-KENO calculations at various moderator densities are illustrated in Fig. 3. These data show that the optimum moderation (maximum k_{eff}) occurs at ~7.5% density (0.0749 g/cc water). In making these calculations, 3,000 neutron histories were used to identify the optimum density and 30,000 histories used for the optimum moderation calculation. At optimum moderation, with fuel of 4.5% enrichment, the calculated k_{eff} is 0.9412 ± 0.0079 (with a one-sided tolerance factor[7] for 95% probability at a 95% confidence level). Assuming the same uncertainty for fuel enrichment and density (0.0024 Δk) as in the flooded case, the maximum reactivity is 0.9524. This maximum reactivity is less than the limiting k_{eff} value of 0.98 for optimum low-density moderation and confirms that the new-fuel storage vault satisfies the criticality safety requirement for fuel of 4.5% enrichment. Administrative procedures are necessary, however, to preclude the use of the 12 storage locations identified in Figs. 1 and 2.

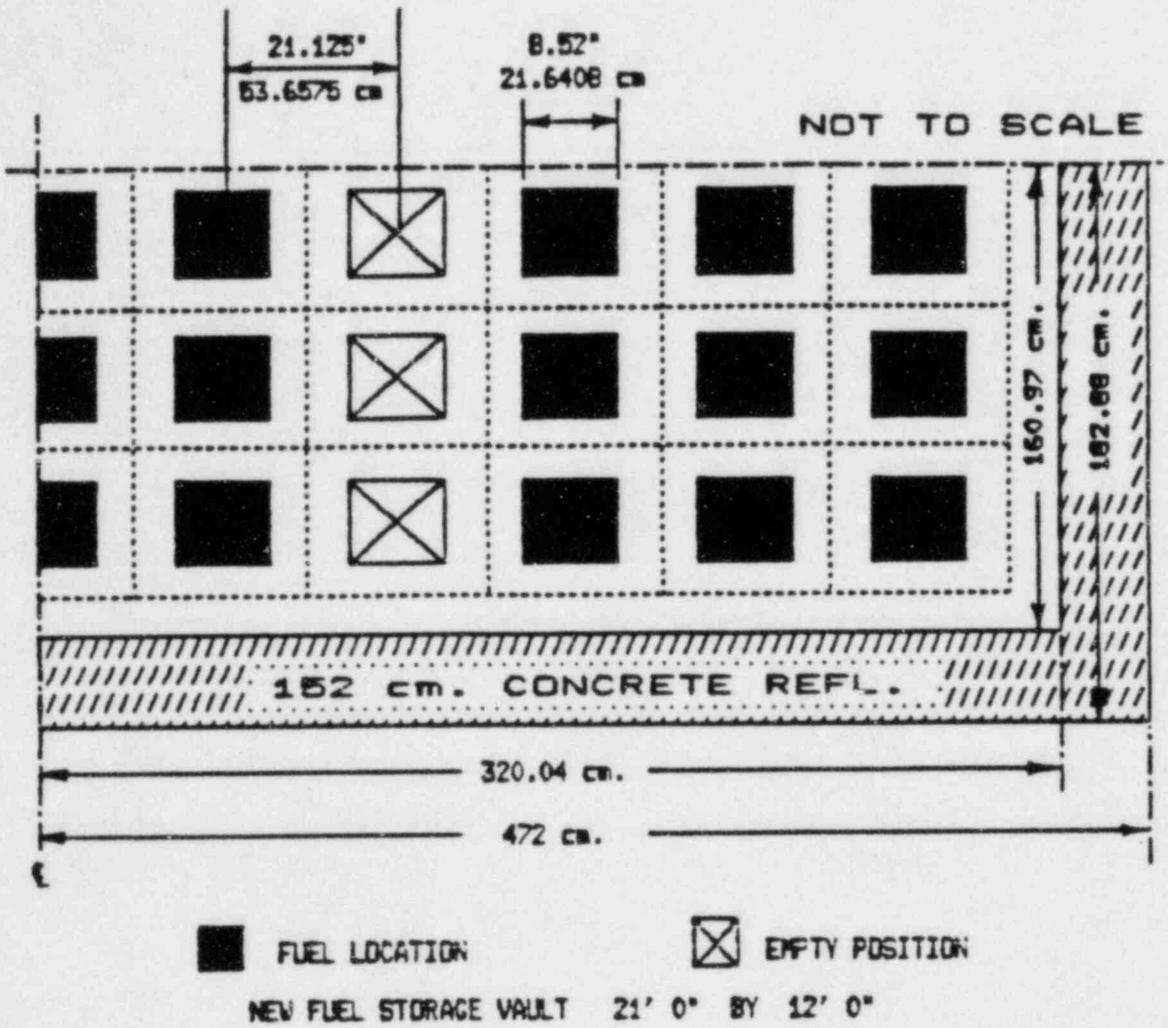

Fig. 2 Geometric model for new-fuel storage rack calculations with AMPX-KENO.

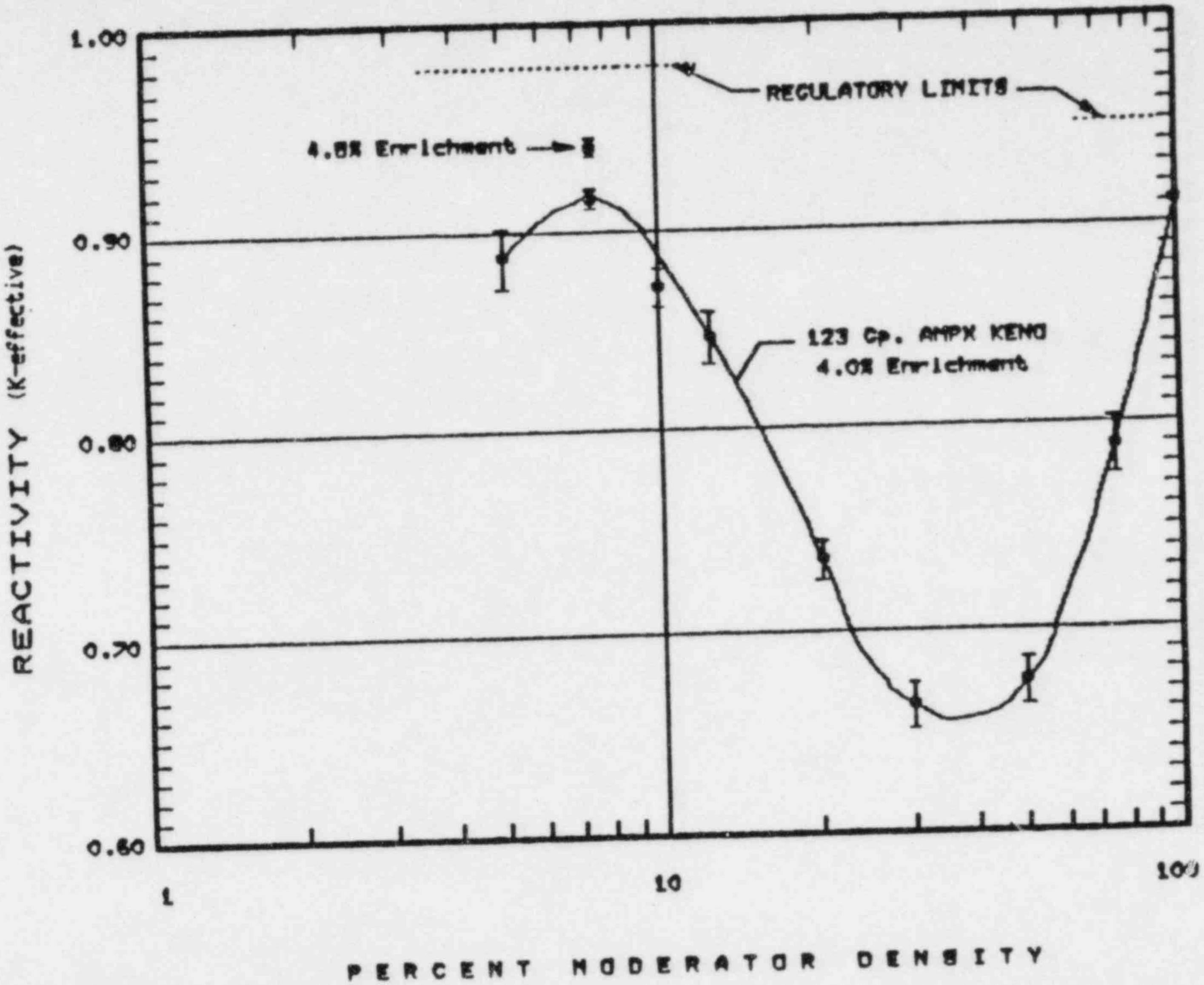

Fig. 3 Calculated reactivity (k_{eff}) values for the new-fuel storage vault at various assumed moderator densities (1σ error bars).

4.0 ABNORMAL AND ACCIDENT CONDITIONS

Since the new-fuel storage vault is normally dry, the two limiting criteria constitute the accident conditions affected by an increase in authorized enrichment to 4.5%. No other safety concerns are affected, and no new or unreviewed safety considerations are introduced by the increase in enrichment. Under the double contingency principle of ANSI N16.1-1975, it is not necessary to consider the simultaneous occurrence of a failure in administrative procedures concurrent with an accident resulting in optimum moderation. Nevertheless, the new-fuel storage facility would remain subcritical even in the event of the simultaneous occurrence of the two independent accident conditions.

REFERENCES

1. S. E. Turner, "Criticality Safety Analysis of the Crystal River Pool "B" Fuel Storage Rack with Fuel of 4% Enrichment," SSA-160, Southern Science Office of Black & Veatch, Sept. 1985.

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

2.b A. Ahlin and M. Edenius, "CASMO - A Fast Transport Theory Depletion Code for LWR Analysis," Trans. Am. Nucl. Soc. 26, 604, 1977.

2.c M. Edenius et al., "CASMO Benchmark Report," Studsvik/RF-78-6293, Aktiebolaget Atomenergi, March 1978.

3. Green, Lucious, Petrie, Ford, White, Wright, "PSR-63/AMPX-1 (code package), AMPX Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL-TM-3706, Oak Ridge National Laboratory, March 1976.

4. L. M. Petrie and N. F. Cross, "KENO-IV, An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory, November 1975.

5. S. E. Turner and M. K. Gurley, "Evaluation of AMPX-KENO Benchmark Calculations for High Density Spent Fuel Storage Racks," Nuclear Science and Engineering, 80(2): 230-237, February 1982.

6. D. G. Napolitano et al., "Validation of the NITAWL-KENO Methodology in Modeling New Fuel Storage Criticality," Trans. Am. Nucl. Soc. 44, 291, 1983.

7. M. G. Natrella, Experimental Statistics National Bureau of Standards, Handbook 91, August 1963.