CRITICALITY SAFETY EVALUATION OF THE POOL A SPENT FUEL STORAGE RACKS IN CRYSTAL RIVER UNIT 3 WITH FUEL OF 5.0% ENRICHMENT

Prepared for the

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1.0 INTRODUCTION

The present study is part of an evaluation of the fuel storage facilities at Crystal River Unit 3 in order to qualify the facilities for fuel of 5.0% average initial enrichment. This report addresses the Pool A spent fuel pool while a companion report evaluates the new-fuel vault (HI-931110).

The CR3 Pool A storage racks are designed to accommodate spent fuel which has attained a minimum average burnup that is dependent on the average initial enrichment of the fuel assembly. These racks use a B_4C matrix absorber and were previously qualified[1] for fuel of 4.5% enrichment burned to 7.0 MWD/KgU and employ a water flux-trap between storage cells as a means of augmenting reactivity control. In the present study, the previous curve of limiting burnups is extended to encompass 5.0% enriched fuel. The effect of the axial distribution in burnup has also been considered as specified by Regulatory Guide 1.13 (draft, Rev.2).

Calculations were made with both the CASMO-3 program and the NITAWL-KENO-5a code package. CASMO-3 was used for burnup and restart calculations and to define an equivalent enrichment for use in the KENO-5a calculations. Both normal and accident conditions are assessed. Credit for the soluble poison normally present in the pool water is allowable under accident conditions (double contingency principle).

To assure the criticality safety under all conditions and to conform to the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling", the definitive criteria contained in the April 14, 1978 USNRC letter and draft Regulatory Guide 1.13 (Rev. 2) are applicable.

Calculations were also made to demonstrate the acceptability of a checkerboard loading pattern of fresh un-burned fuel of 5.0% enrichment.

2.0 SUMMARY AND CONCLUSIONS

2.1 Normal Storage Conditions

The spent fuel storage racks in Pool A use stainless steel boxes to define the storage cells with a B_4C matrix neutron absorber of 0.015 gms B-10/cm² areal density. A water-gap between the absorber panels affords a flux-trap to augment reactivity control. Calculations were made for fuel burned to 10.5 MWD/KgU for fuel of 5.0% initial enrichment.

Table 1 summarizes the criticality safety analysis for 5.0% enriched fuel at a burnup of 10.5 MWD/KgU. The maximum k_{eff} is 0.9435, including uncertainties, which is within the Regulatory guideline (k_{eff} of 0.95) and is therefore acceptable. The previous burnup limit curve [1] was extended to include 5.0% enriched fuel and the updated curve is shown in Figure 1. The limiting burnup curve in Figure 1 is well described by a linear fit as shown below, and may be used to calculate the minimum burnup for any initial enrichment, E, up to 5.0%.

Acceptable Burnup in MWD/KgU

= 7.0 * E - 24.5

This fit is the same as previously determined, extended to include 5.0% enriched fuel.

Based upon the calculations reported here (see Table 1 and Figure 1), it is concluded that fuel of 5.0% initial enrichment is acceptable for storage in Pool A of the Crystal River Unit 3 spent fuel storage facilities, provided the fuel has attained a minimum burnup of 10.5 MWD/KgU. Minimum burnup specifications are shown in

Figure 1 for other enrichments (from the previous evaluation) with assurance that the maximum reactivity is within the regulatory limit.

Calculations were also made for checkerboard arrangements of fresh 5.0% enriched fuel. These calculations show that a checkerboard arrangement with empty cells (i.e. filled only with water or non-fissile bearing material) is acceptable with a maximum k_{eff} of 0.833.

2.2 Abnormal/Accident Conditions

The reactivity consequences of abnormal/accident conditions were considered in the previous analysis[1] and found to be within acceptable bounds. However, with the higher enrichment fuel (5.0%), the consequence of a mis-placed fuel assembly could differ from that previously evaluated. Calculations with a mis-placed fuel assembly (fresh assembly of 5.0% enrichment accidentally loaded into a Pool A cell) resulted in a maximum k_{eff} of 0.946 (including uncertainties) with all other cells filled with fuel of the maximum permissible reactivity. This is within the Regulatory guideline even without the allowable credit for soluble boron.

3.0 CRITICALITY SAFETY ANALYSES

3.1 Fuel Assembly Specifications

The fuel assemblies used in the analyses is the Babcock & Wilcox 15 x 15 fuel assembly, the same as that used in the previous analyses[1]. Table 2 attached lists the design specifications for the fuel used in the analyses.

3.2 Storage Rack Specifications

The storage rack cell design, illustrated in Figure 2, is composed of B_4C absorber material sandwiched between two 0.060 inch thick stainless steel boxes of 8.9375 inch inside dimension. The cells are arranged on a 10.50 inch lattice spacing with a 1.173 inch water gap between the storage cells. The stainless steel tabs connecting the storage boxes have a slightly negative reactivity effect and were neglected in the calculations. The B_4C absorber has a thickness of 0.075 inches and a B-10 loading of 0.015 ± 0.003 gms B-10/cm^2.

3.3 Manufacturing Tolerances and Uncertainties

The small reactivity increments associated with manufacturing tolerances obtained in the previous evaluation[1] (± 0.0097) were assumed to remain applicable. Combined with the uncertainty in bias (± 0.0024, Appendix A) results in a total uncertainty of ± 0.0100. Fuel of 5.0% enrichment also requires an increase in the allowance for uncertainty in the depletion calculations. As in the original evaluation for 4.5% fuel, the depletion uncertainty was assumed to be 5.0% of the reactivity decrement from beginning-of-life to the burnup of 10.5 MWD/KgU. This allowance amounts to 0.0035 Δk which is conservatively treated as an additiive term rather than being statistically combined with the other uncertainties.

Previous calculations have demonstrated a continuous reduction in reactivity with storage time (after Xe decay) primarily due to Pu-241 decay and Am-241 growth. No credit is taken for this reduction in reactivity except to acknowledge an additional level of conservatism in the calculations.

3.4 Calculational Methodology

3.4.1 Computer Codes

The principal methods of analysis were CASMO-3[2], a two-dimensional multigroup transport theory code for fuel assemblies and NITAWL - KENO-5a[3], a three dimensional Monte Carlo code package, using the 27-group SCALE* cross-section library. The calculational methods used for the present evaluation are comparable to those used in the original calculations, differing only in that updated versions of the codes were used, ie, CASMO-3 rather than CASMO-2E, and KENO-5a rather than KENO-4. Results of these codes are not significantly different from those of the earlier versions, and benchmarking of the updated codes resulted in a bias of 0.0000 ± 0.0024 for CASMO-3 and 0.0103 ± 0.0018 for NITAWL - KENO-5a (95% probability, 95% confidence level[4]). A summary of the detailed bench-marking analyses is included in Appendix A.

CASMO-3 was also used both for burnup calculations and for restart calculations in the rack geometry. Since KENO-5a cannot perform burnup analyses, CASMO-3 is used to define an equivalent enrichment, ie, the U-235 enrichment that yields the same reactivity in the racks as the burned fuel. It was found that an enrichment of 3.4% yields the same reactivity in the storage racks as 5.0% fuel burned to 10.5 MWD/KgU. Independent check calculation

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* SCALE is an acronym for Standardized Computer Analyses for Licensing Evaluation, developed for the USNRC by the Oak Ridge National Laboratory.

for the reference case with NITAWL-KENO-5a (3.4% equivalent enrichment) gave a bias corrected k_{eff} of 0.9280 ± 0.0010 (without uncertainties) which confirms the CASMO-3 calculation (k_{eff} of 0.9300).

In the geometric model used in the calculations, each fuel rod and its cladding were described explicitly in both the CASMO-3 and KENO-5a models. Reflecting boundary conditions (zero neutron current) were used in the radial direction which has the effect of creating an infinite array of storage cells in X-Y directions. In the KENO-5a model, the actual fuel assembly length was used in the axial direction, assuming thick (30 cm) water reflectors top and bottom.

Monte Carlo (KENO-5a) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the KENO-calculated reactivity, a minimum of 500,000 neutron histories in 1000 generations of 500 neutrons each, were accumulated in each calculation.

3.4.2 Axial Distribution in Burnup

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. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of lower reactivity worth due to neutron leakage. Consequently, it would be expected that over most of the burnup history, distributed burnup fuel assemblies would exhibit a slightly lower reactivity than that calculated for the average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

Among others, Turner[5] has provided generic analyses of the axial burnup effect based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnups at values less than about 30,000 Mwd/MtU. Because a burnup of only 10.5 MWD/KgU is necessary for 5.0% fuel, the reactivity consequence of the axial distribution in burnup will be slightly negative.

3. Checkerboard Configuration

Fuel up to 5.0% initial enrichment may also be stored in a checkerboard pattern, alternating with cells filled with only water or non-fissile material. For this case, the maximum reactivity, including uncertainties, was calculated to be 0.8331, which is well below the USNRC guideline.

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Table 1

SUMMARY OF CRITICALITY SAFETY ANALYSES
POOL A STORAGE RACKS

Fuel Enrichment, wt% U-235	5.00
Design Burnup, MWD/KgU	10.50
Reference Temperature, °F	68
Reference k_∞ (CASMO-3)	0.9300
Calculational Bias, Δk[1]	0.0000
Axial Burnup Distribution	Negative
Δk Uncertainties[2]	±0.0100
Total	0.9300 ±0.0100
Δk allowance for depletion calculations[2]	±0.00035
Maximum Reactivity (k_∞)	0.9435
Regulatory Limit (k_{eff}	0.95

(1) Appendix A

(2) Section 3.3

TABLE 2

FUEL ASSEMBLY SPECIFICATIONS

Fuel Rod Data	
Outside dimension, in.	0.430
Cladding ID, in.	0.377
Cladding thickness, in.	0.0265
Cladding material	Zr-4
Pellet diameter, in.	0.369
UO_2 density, g/cm^2	10.420 ± 0.166
Enrichment, wt.% U-235	5.0 ± 0.02
Fuel Assembly Data	
Number of fuel rods	208 (15x15 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.	0.02
Material	Zr-4

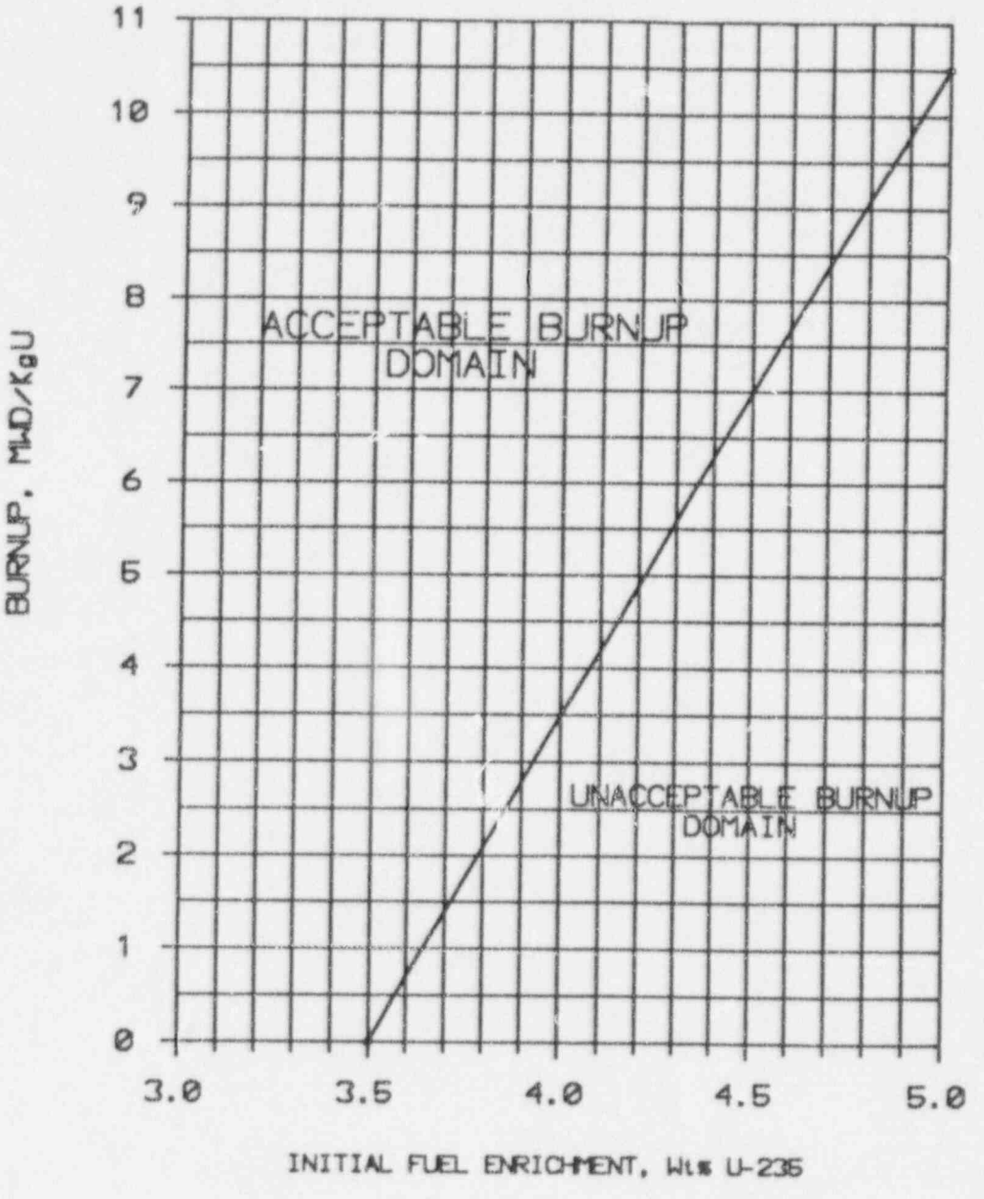

Fig. 1 Acceptable Burnup Domain in Pool A

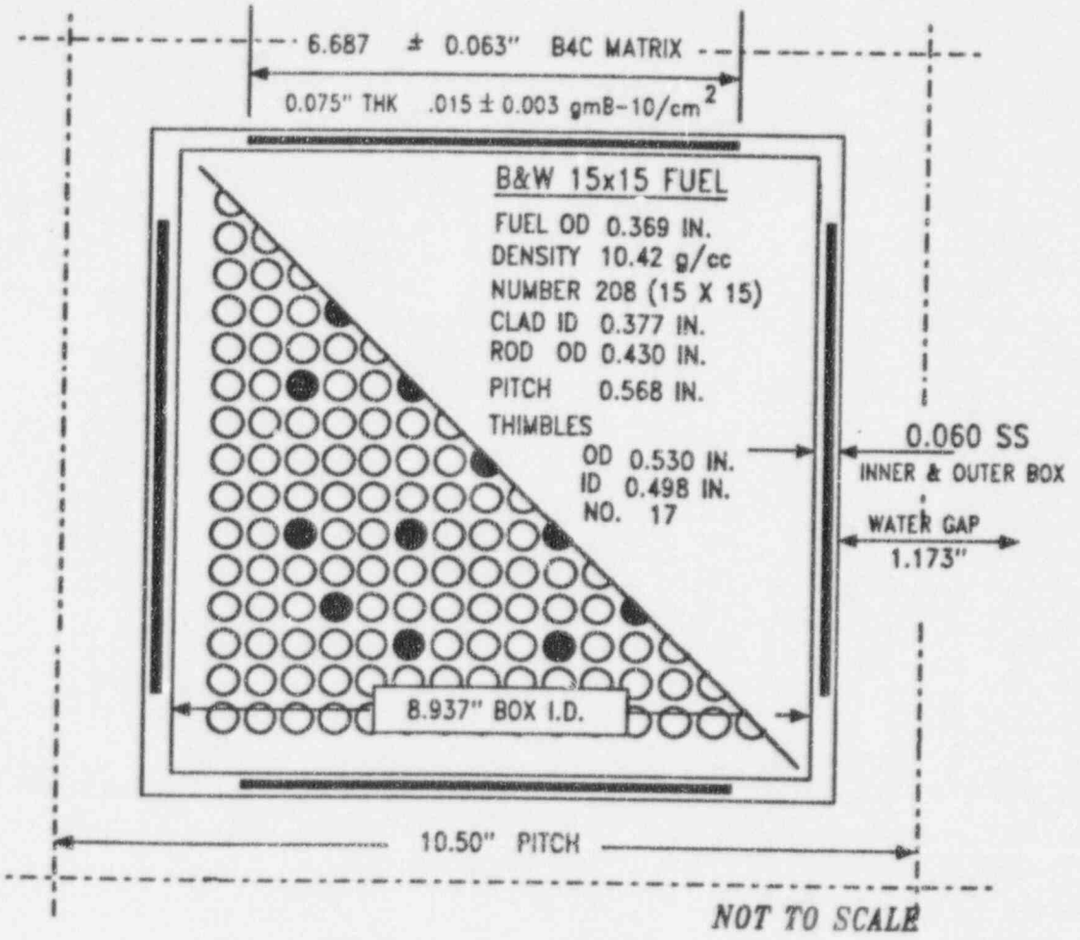

FIG. 2 CR3 POOL A FUEL STORAGE CELL

APPENDIX A

BENCHMARK CALCULATIONS

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1.0 INTRODUCTION AND SUMMARY

The objective of this benchmarking study is to verify both the NITAWL-KENO5a[1,2] methodology with the 27-group SCALE cross-section library and the CASMO3 code[3] for use in criticality safety calculations of high density spent fuel storage racks. Both calculational methods are based upon transport theory and have been benchmarked against critical experiments that simulate typical spent fuel storage rack designs as realistically as possible. Results of these benchmark calculations with both methodologies are consistent with corresponding calculations reported in the literature.

Results of the benchmark calculations show that the 27-group (SCALE) NITAWL-KENO5a calculations consistently underpredict the critical eigenvalue by 0.0103 ± 0.0018 δk (with a 95% probability at a 95% confidence level) for critical experiments[4] that are as representative as possible of realistic spent fuel storage rack configurations and poison worths.

Extensive benchmarking calculations of critical experiments with CASMO3 have also been reported[5], giving a mean k_{eff} of 1.0004 ± 0.0011 for 37 cases. With a K-factor of 2.14[6] for 95% probability at a 95% confidence level, and conservatively neglecting the small overprediction, the CASMO3 bias then becomes 0.0000 ± 0.0024. CASMO3 and NITAWL-KENO5a intercomparison calculations of infinite arrays of poisoned cell configurations (representative of typical spent fuel storage rack designs) show very good agreement, confirming that 0.0000 ± 0.0024 is a reasonable bias and uncertainty for CASMO3 calculations. Reference 5 also documents good agreement of heavy nuclide concentrations for the Yankee core isotopics, agreeing with the measured values within experimental error.

The benchmark calculations reported here confirm that either the 27-group (SCALE) NITAWL-KENO5a or CASMO3 calculations are acceptable for criticality analysis of high-density spent fuel storage racks. Where possible, reference calculations for storage rack designs should be performed with both code packages to provide independent verification. CASMO3, however, is not reliable when large water gaps (> 2 or 3 inches) are present.

2.0 NITAWL-KENO5a BENCHMARK CALCULATIONS

Analysis of a series of Babcock & Wilcox critical experiments[4], including some with absorber panels typical of a poisoned spent fuel rack, is summarized in Table 1, as calculated with NITAWL-KENO5a using the 27-group SCALE cross-section library and the Nordheim resonance integral treatment in NITAWL. Dancoff factors for input to NITAWL were calculated with the Oak Ridge SUPERDAN routine (from the SCALE[2] system of codes). The mean for these calculations is 0.9899 ± 0.0028 (1 σ standard deviation of the population). With a one-sided tolerance factor corresponding to 95% probability at a 95% confidence level[6], the calculational bias is + 0.0103 with an uncertainty of the mean of ± 0.0018 for the sixteen critical experiments analyzed.

Similar calculational deviations have been reported by ORNL[7] for some 54 critical experiments (mostly clean criticals without strong absorbers), obtaining a mean bias of 0.0100 ± 0.0013 (95%/95%). These published results are in good agreement with the results obtained in the present analysis and lend further credence to the validity of the 27-group NITAWL-KENO5a calculational model for use in criticality analysis of high density spent fuel storage racks. No abnormal deviations in k_{eff} with intra-assembly water gap, with absorber panel reactivity worth, with enrichment or with poison concentration were identified with the 27 group SCALE library, comparable to those previously observed[8] with the 123-group GAM-THERMOS cross-section library.

Additional benchmarking calculations were also made for a series of French critical experiments[9] at 4.75% enrichment and for several of the BNWL criticals with 4.26% enriched fuel. Analysis of the French criticals (Table 2) showed a tendency to overpredict the reactivity, a result also obtained by ORNL[10]. The calculated k_{eff} values showed a trend toward higher values with decreasing core size. In the absence of a significant enrichment effect (see Section 3 below), this trend and the overprediction is attributed to a small inadequacy in NITAWL-KENO5a in calculating neutron leakage from very small assemblies.

Similar results were observed for the BNWL series of critical experiments[11], which are also small assemblies (although significantly larger than the French criticals). In this case (Table 2), the calculated mean k_{eff} was 0.9959 ± 0.0013 (1 σ population standard deviation). Because of the small size of the BNWL critical experiments (compared to the B&W criticals used to determine the KENO5a bias) and the absence of any significant enrichment effect, the results also suggest a small inadequacy of NITAWL-KENO5a in treating large neutron leakage from very small assemblies.

Since the analysis of high-density spent fuel storage racks generally does not entail neutron leakage, the observed inadequacy of NITAWL-KENO5a is not significant. Furthermore, omitting results of the French and BNWL critical experiment analyses from the determination of bias is conservative since any leakage that might enter into the analysis would tend to result in overprediction of the reactivity.

3.0 CASMO3 BENCHMARK CALCULATIONS

The CASMO3 code is a multigroup transport theory code utilizing transmission probabilities to accomplish two-dimensional calculations of reactivity and depletion for BWR and PWR fuel assemblies. As such, CASMO3 is well-suited to the criticality analysis of spent fuel storage racks, since general practice is to treat the racks as an infinite medium of storage cells, neglecting leakage effects.

CASMO3 is a modification of the CASMO-2E code and has been extensively benchmarked against both mixed oxide and hot and cold critical experiments by Studsvik Energiteknik[5]. Reported analyses[5] of 37 critical experiments indicate a mean k_{eff} of 1.0004 ± 0.0011 (1σ). To independently confirm the validity of CASMO3 (and to investigate any effect of enrichment), a series of calculations were made with CASMO3 and with NITAWL-KENO5a on identical poisoned storage cells representative of high-density spent fuel storage racks. Results of these intercomparison calculations* (shown in Table 3 and in Figure 1) are within the normal statistical variation of KENO5a calculations and confirm the bias of 0.0000 ± 0.0024 (95%/95%) for CASMO3.

Since two independent methods of analysis would not be expected to have the same error function with enrichment, results of the intercomparison analyses (Table 3) indicate that there is no significant effect of fuel enrichment over the range of enrichments involved in power reactor fuel.

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* Intercomparison between analytical methods is a technique endorsed by Reg. Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety".

A second series of CASMO3 and KENO5a intercomparison calculations consisting of five cases from the BAW critical experiments were analyzed for the central cell only. The calculated results, also shown in Table 3, indicate a mean difference within the 95% confidence limit of the KENO5a calculations. This lends further credence to the recommended bias for CASMO3.

4.0 WORKER ROUTINE

The WORKER routine was obtained from ORNL and is intended to interpolate the hydrogen scattering matrices for temperature in order to correct for the deficiency noted in NRC Information Notice 91-66 (October 18, 1991). Benchmark calculations were made against CASMO3, based on the assumption that two independent methods of analysis would not exhibit the same error. Results of these calculations, shown in Table 4, confirm that the trend with temperature obtained by both codes are comparable. This agreement establishes the validity of the WORKER routine, in conjunction with NITAWL-KENO5a, in calculating reactivities at temperatures between 20°C and 120°C.

The deficiency in the NITAWL hydrogen scattering matrix at temperatures above 20 °C does not appear except in the presence of a large water gap where the scattering matrix is important. However, the absolute value of the k_∞ from CASMO3 is not reliable in the presence of a large water gap, although the relative values should be accurate. In the calculations shown in Table 4 and in Figure 2, the absolute reactivity values differ somewhat but the trends with temperature are sufficiently in agreement to lend credibility to the WORKER routine over the temperature range from 20°C to 120°C.

5.0 CLOSE-PACKED ARRAYS

The BAW close-packed series of critical experiments[12] intended to simulate consolidated fuel, were analyzed with NITAWL-KENO5a. Results of these analyses, shown in Table 5, suggest a slightly higher bias than that for fuel with normal lattice spacings. Similar results were obtained by ORNL[13]. Because there are so few cases available for analysis, the maximum bias for close-packed lattices may be taken as 0.0155, including uncertainty. This would conservatively encompass all but one of the cases measured.

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Table 1

RESULTS OF 27-GROUP (SCALE) NITAWL-KENO5a CALCULATIONS
OF B&W CRITICAL EXPERIMENTS

Experiment Number	Calculated k_{eff}	σ
I	0.9922	± 0.0006
II	0.9917	± 0.0005
III	0.9931	± 0.0005
IX	0.9915	± 0.0006
X	0.9903	± 0.0006
XI	0.9919	± 0.0005
XII	0.9915	± 0.0006
XIII	0.9945	± 0.0006
XIV	0.9902	± 0.0006
XV	0.9836	± 0.0006
XVI	0.9863	± 0.0006
XVII	0.9875	± 0.0006
XVIII	0.9880	± 0.0006
XIX	0.9882	± 0.0005
XX	0.9885	± 0.0006
XXI	0.9862	± 0.0006
Mean	0.9897	± 0.0007[1]
Bias (95%/95%)	0.0103	± 0.0018

(1) Standard Deviation of the Mean, calculated from the k_{eff} values.

Table 2

RESULTS OF 27-GROUP (SCALE) NITAWL-KENO5a CALCULATIONS OF FRENCH and BNWL CRITICAL EXPERIMENTS

French Experiments		
Separation Distance, cm	Critical Height, cm	Calculated k_{eff}
0	23.8	1.0302 ± 0.0008
2.5	24.48	1.0278 ± 0.0007
5.0	31.47	1.0168 ± 0.0007
10.0	64.34	0.9998 ± 0.0007

BNWL Experiments		
Case	Expt. No.	Calculated k_{eff}
No Absorber	004/032	0.9942 ± 0.0007
SS Plates (1.05 B)	009	0.9946 ± 0.0007
SS Plates (1.62 B)	011	0.9979 ± 0.0007
SS Plates (1.62 B)	012	0.9968 ± 0.0007
SS Plates	013	0.9956 ± 0.0007
SS Plates	014	0.9967 ± 0.0007
Zr Plates	030	0.9955 ± 0.0007
Mean		0.9959 ± 0.0013

Table 3

RESULTS OF CASMO3 AND NITAWL-KENO5a
BENCHMARK (INTERCOMPARISON) CALCULATIONS

Enrichment[1] Wt. % U-235	NITAWL-KENO5a[2]	k_∞ CASMO3	$\lvert\delta k\rvert$
2.5	0.8376 ± 0.0010	0.8386	0.0010
3.0	0.8773 ± 0.0010	0.8783	0.0010
3.5	0.9106 ± 0.0010	0.9097	0.0009
4.0	0.9367 ± 0.0011	0.9352	0.0015
4.5	0.9563 ± 0.0011	0.9565	0.0002
5.0	0.9744 ± 0.0011	0.9746	0.0002
		Mean	0.0008
Expt. No.[3]			
XIII	1.1021 ± 0.0009	1.1008	0.0013
XIV	1.0997 ± 0.0008	1.1011	0.0014
XV	1.1086 ± 0.0008	1.1087	0.0001
XVII	1.1158 ± 0.0007	1.1168	0.0010
XIX	1.1215 ± 0.0007	1.1237	0.0022
		Mean	0.0012

(1) Infinite array of assemblies typical of high-density spent fuel storage racks.

(2) k_∞ from NITAWL-KENO5a corrected for bias.

(3) Central Cell from BAW Critical Experiments

Table 4

Intercomparison of WORKER-NITAWL-KENO5a
and CASMO3 Calculations at Various Temperatures

Temperature	CASMO3	W-N-KENO5a(*)
4°C	1.2276	1.2345 ± 0.0014
17.5°C	1.2322	1.2328 ± 0.0015
25°C	1.2347	1.2360 ± 0.0013
50°C	1.2432	1.2475 ± 0.0014
75°C	1.2519	1.2569 ± 0.0015
120°C	1.2701	1.2746 ± 0.0014

*Corrected for bias

Table 5

Reactivity Calculations for Close-Packed Critical Experiments

Calc. No.	BAW Expt. No.	Pin Pitch cm	Module Spacing cm	Boron Conc. ppm	Calculated k_{eff}
KS01	2500	Square 1.4097	1.792	1156	0.9891 ± 0.0005
KS02	2505	Square 1.4097	1.792	1068	0.9910 ± 0.0005
KS1	2485	Square Touching	1.778	886	0.9845 ± 0.0005
KS2	2491	Square Touching	1.778	746	0.9849 ± 0.0005
KT1	2452	Triang. Touching	1.86	435	0.9845 ± 0.0006
KT1A	2457	Triang. Touching	1.86	335	0.9865 ± 0.0006
KT2	2464	Triang. Touching	2.62	361	0.9827 ± 0.0006
KT3	2472	Triang. Touching	3.39	121	1.0034 ± 0.0006

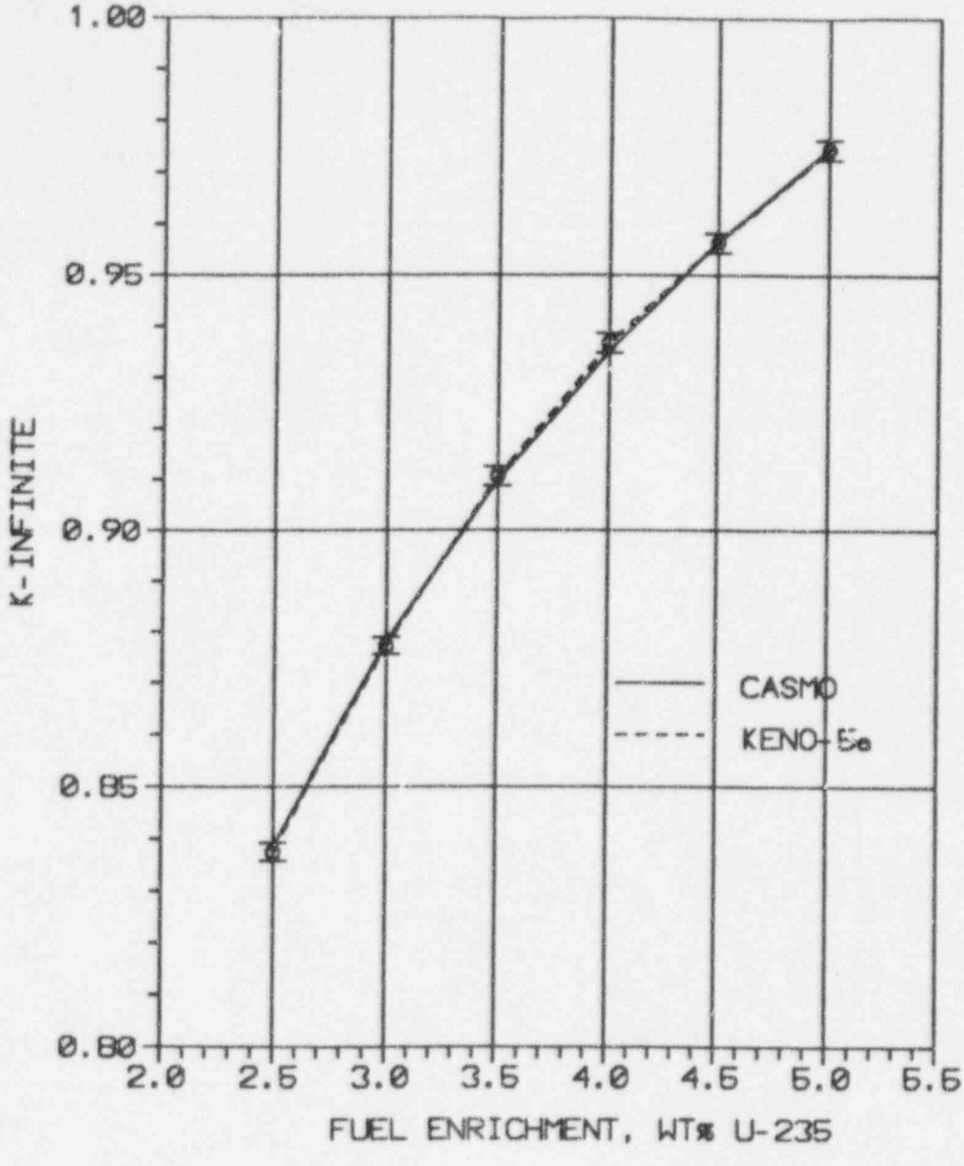

Fig. 1
COMPARISON OF CASMO AND KENO5a CALCULATIONS
AT VARIOUS ENRICHMENTS IN REPRESENTATIVE FUEL STORAGE RACK

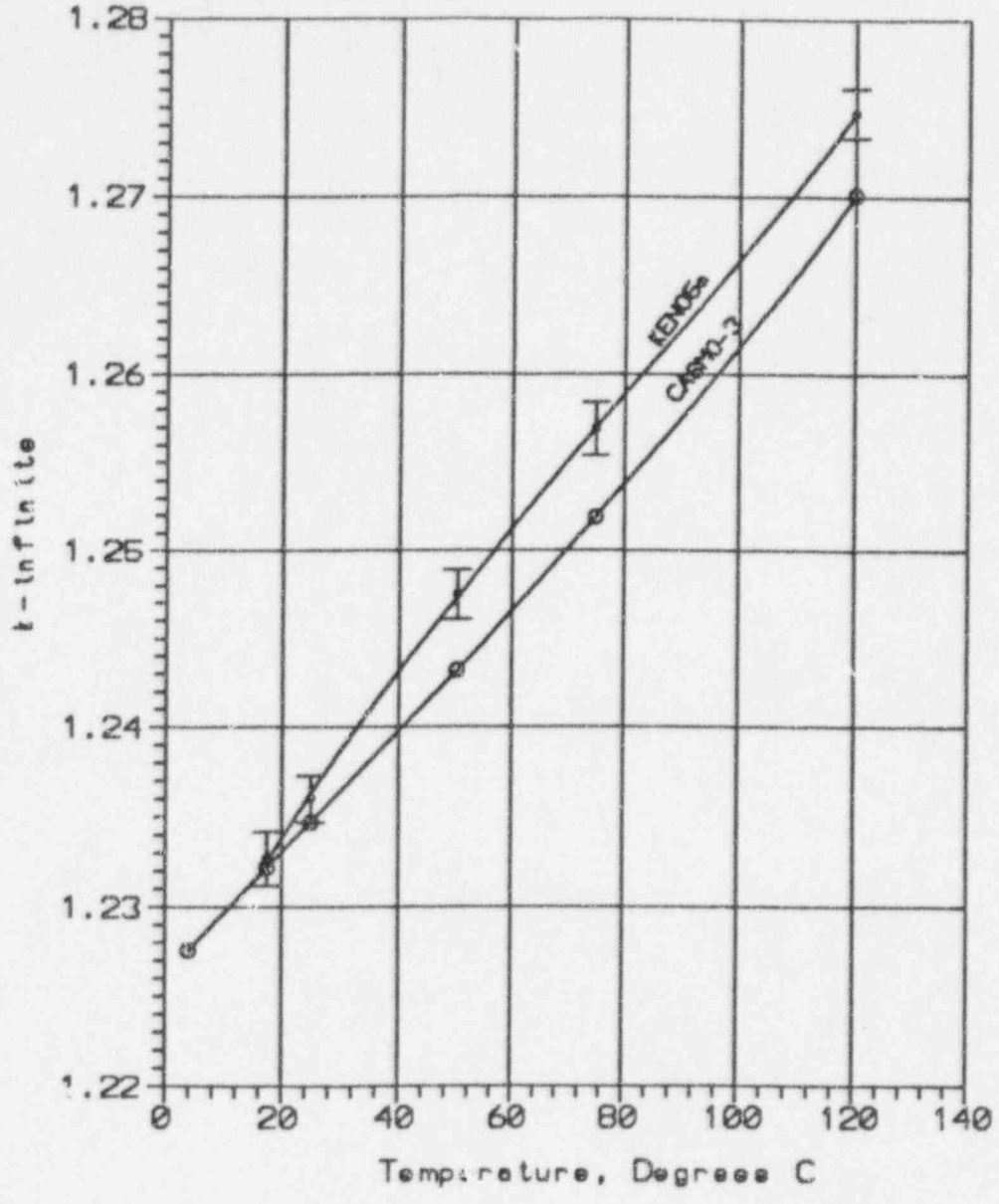

Fig. 2 COMPARISON OF CASMO-3 and KENO5a TEMPERATURE DEPENDENCE

ATTACHMENT 3
PROPRIETARY INFORMATION