

ATTACHMENT 3 to TXX-94048

AFFECTED TECHNICAL SPECIFICATION PAGE
(NUREG - 1468)

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 280°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods (consisting of Zircaloy-4 or stainless steel) may be made if justified by a cycle specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment not to exceed 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment not to exceed ~~4.3~~ weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing and may include clad surface treatment for wear mitigation.

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 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
- b. For a pressure of 2,485 psig, and
- c. For a temperature of 650°F, except for the pressurizer which is 680°F.

Enclosure 1 to TXX-94048

Criticality Safety Evaluation of Comanche Peak Fuel Storage
Facilities with Fuel of 5% Enrichment

Note: The enclosure has a typographical error in the last paragraph on page 4. The words "scattering matrices only at 20°F and 277°F C" should read "scattering matrices only at 20°C and 277°C".

CRITICALITY SAFETY EVALUATION OF
COMANCHE PEAK FUEL STORAGE FACILITIES
WITH FUEL OF 5% ENRICHMENT

Prepared for the
TU ELECTRIC COMPANY

by

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This document conforms to the requirements of the design specification and the applicable sections of the governing codes.

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1.0 INTRODUCTION

The present study was undertaken for the purpose of documenting the capability of the fuel storage facilities at Comanche Peak to safely store fuel of 5% initial enrichment. The fuel storage facilities include the New Fuel Storage Vault (NFV), the Spent Fuel Storage Pool (SFP), and the In-Containment Storage Rack. Criticality safety analyses and accident evaluations for each of the fuel storage facilities at Comanche Peak are presented in this report. These calculations confirm that all storage facilities can safely receive and store fuel up to 5% enrichment (including a manufacturing tolerance of 0.05%) within the limits of the USNRC guidelines.

2.0 SUMMARY OF COMANCHE PEAK FUEL RACK DESIGNS

2.1 New Fuel Storage Rack Design

The storage rack layout in the NFV is illustrated in Figure 1. The racks consist of stainless steel boxes (0.0747 inch thick) located on a 21-inch lattice spacing. The storage boxes are arranged in seven rows of two cells each as illustrated in Figure 1. Normally, fuel is stored in the dry condition. Examination of as-built dimensions indicate that the average lattice spacing is 21.01 ± 0.04 inches (two sided tolerance for 95% probability at the 95% confidence level), and the spacing between pairs of rows shown in Figure 1 is 36.0 ± 0.133 inches (95%/95%). These values were used in the evaluation of the small uncertainties in reactivity due to manufacturing tolerances.

2.2 Spent Fuel Storage Rack Design

Two separate storage pools exist, both of the same cell design. The in-containment rack is a single 5 x 5 module for temporary storage of fuel assemblies. In the storage pools, spent fuel is stored underwater in type 304 stainless steel racks. The spent fuel storage racks consist of square stainless steel boxes, nominally 9.000 inch inside dimension and 0.0747 inches thick located on a 16-inch lattice spacing. Figure 2 illustrates a cross-section of the SFP cells.

2.3 Fuel Assembly Specifications

Four 17 x 17 fuel assembly designs were considered in the criticality safety evaluation of the Comanche Peak storage facilities. These included both the Westinghouse Optimized Assembly (OFA) and standard design and the Siemens large and small fuel rod designs. Initial calculations established that the Westinghouse 17 x 17 (OFA) design exhibits the highest reactivity, and this fuel design was used for the remainder of the calculations.

3.0 Criticality Analyses

3.1 Introduction

The storage facilities were analyzed to assure that the most reactive fuel assembly with 5% enrichment could be safely stored within the limits established by USNRC guidelines. Applicable codes, standards, and regulations include the following:

- General Design Criteria 62, Prevention of Criticality in Fuel Storage and Handling.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.1, New Fuel Storage, Rev. 3 - July 1981.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3 - July 1981.
- USNRC 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 Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December 1981.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

3.2 Design Criteria

3.2.1 New Fuel Storage Vault

The new fuel storage vault is intended for the receipt and storage of fresh fuel under normally dry, low reactivity conditions. To assure criticality safety under accident conditions and to conform to the requirements of General Design Criterion 62, two criteria, as defined in NUREG-0800, Standard Review Plan 9.1.1, must be satisfied. These criteria are as follows:

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

3.2.2 Spent Fuel Storage Pool

The principal NRC guidance (and identification of requirements) for spent fuel storage racks is included in the April 14, 1978, NRC letter which provides the definitive interpretation of SRP 9.1-2 and Reg Guide 1.13. The limiting k_{eff} for water filled storage facilities is 0.95 including uncertainties evaluated for 95% probability at the 95% confidence level. The water in the spent fuel storage pool normally contains soluble boron which would result in large subcriticality margins under normal operating conditions. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 for normal storage be evaluated for the accident condition of the loss of soluble boron. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions since only a single independent accident need be considered at one time.

3.3 Analytical Methods and Benchmark Experiments

In the fuel rack analyses, the primary criticality analyses were made with the three-dimensional Monte-Carlo code package NITAWL-KENO-5a^(1,2) using the 27-group SCALE* cross-section library⁽³⁾ and the Nordheim integral treatment for U-238 resonance shielding effects. In addition, tolerance effects were determined with CASMO-3^(4,8), a two dimensional multi-group transport theory code. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0101 ± 0.0020 (95%/95%)⁽⁹⁾ for NITAWL-KENO-5a. Since the SCALE cross-section library as used by NITAWL has scattering matrices only at 20°F and 277°F C, a special routine was developed to interpolate the scattering matrix for temperatures between those currently in NITAWL.

* "SCALE" is an acronym for Standardized Computer Analysis for Licensing Evaluation, a standard cross-section set developed by ORNL for the USNRC.

Monte Carlo (KENO-5a) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty, a minimum of 250,000 neutron histories in 500 generations of 500 neutrons each, were accumulated in each calculation.

CASMO-3 has been benchmarked against critical experiments with water gaps up to 2.576 inches (Appendix A). However, when large water gaps are present (as in the Comanche Peak racks), CASMO-3 will underpredict or overpredict k_{eff} , depending upon the size of the water-gap and the number of mesh intervals used. Incremental changes in k_{eff} , due to the small manufacturing tolerances can be calculated by CASMO-3 to obtain estimates of the associated uncertainties. A comparison of small incremental values calculated by CASMO-3 and KENO-5a for the flooded NFV is given in Table 1. CASMO-3 and KENO-5a show the same trend with temperature (Appendix A). These data confirm that the incremental reactivity effects calculated by CASMO-3 are reasonable estimates of the uncertainties. Consequently, CASMO-3 was used to evaluate small incremental reactivity effects from manufacturing tolerances.

3.4 New Fuel Vault Criticality Analyses

3.4.1 Calculational Model

The calculational model used for the analyses is indicated by the dashed lines in Figure 1, conservatively using an infinite array of 2x10 storage boxes. Each fuel rod (including cladding) and guide tube within each of the stainless steel boxes and the boxes themselves are explicitly described. The model describes the concrete reflector in one direction (as indicated in Figure 1) but uses reflecting boundary conditions in the other direction (long direction of the 2x10 storage cell array). This effectively creates an infinite array in the wide direction and is conservative since four of the rows have only 9 cells rather than the 10 described in the model.

3.4.2 New Fuel Vault Analysis Results For Normal Conditions

Fuel in the New Fuel Vault is normally stored in the dry condition. An infinite number of dry assemblies of this design would have a $k_{eff} < 0.5$ for enrichments up to 5%.

3.4.3 New Fuel Vault Under Accident Conditions

In the new fuel vault, the accident conditions considered are (1) the optimum moderation density and (2) flooded with clean unborated water. Figure 3 shows the variation of the calculated k_{eff} of the new fuel storage vault with water density (moderation). The maximum k_{eff} occurs at 5.3% of normal water density. Table 2 summarizes the calculated k_{eff} and uncertainties for both the low-density optimum moderation and the fully-flooded conditions, including calculational and manufacturing uncertainties. For these postulated accident conditions, the maximum calculated values of k_{eff} of the new fuel storage vault are 0.972 at the optimum low-density moderating condition and 0.929 for the fully flooded condition. These values of k_{eff} are within the USNRC guidelines (SRP 9.1.1) and indicate that fuel up to 5% enrichment can be safely stored.

3.5 Spent Fuel Rack Criticality Analysis

3.5.1 Calculational Model

The calculational model used for analysis of the spent fuel storage racks is shown in Figure 2. 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 at the centerline of the water-gap between cells which has the effect of creating an infinite array of storage cells. In the KENO-5a model, the active fuel length was used in the axial direction, assuming a thick (30 cm) water reflector, top and bottom.

3.5.2 Spent Fuel Rack Analysis Results for Normal Conditions

CASMO-3 calculations were made of the effect of temperature on the k_{eff} of the rack. These calculations showed that the highest k_{eff} occurs at 40°C and, therefore, this temperature was used for all subsequent calculations.

Under normal storage conditions, the presence of soluble boron in the pool water assures a very low value of k_{eff} (approximately 0.70 at 4000 ppm).

3.5.3 Spent Fuel Rack Under Accident Conditions

The results of the postulated loss of all soluble boron accident condition calculations, summarized in Table 3, indicate a maximum k_{eff} of 0.946, including calculational and manufacturing uncertainties (95% probability at the 95% confidence level). This is less than the regulatory limit of a k_{eff} of 0.95.

Under other accident conditions in the spent fuel pool, credit for the soluble poison in the water is acceptable (double contingency principle of the April 1978 USNRC letter), and will assure that the required subcritical reactivity margin is maintained for normal conditions and all credible accidents. Based on KENO-5a calculations with and without boron, the presence of 2000 ppm boron in the pool water reduced the k_{eff} by 0.251 δk .

KENO-5a calculations of the accidental mis-loading of a fuel assembly of 5% enrichment outside and adjacent to the storage rack were made in the presence of 2000 ppm soluble boron. A mis-loaded fuel assembly could theoretically be situated in an inside corner adjacent to two rack modules. Assuming all of the fuel assemblies were of the highest permissible reactivity, the calculated k_{eff} is 0.708, which is well below the NRC limit.

The soluble boron in the pool water would more than adequately compensate for a fuel assembly of 5% enrichment dropped and assumed to come to rest on top of a filled rack module. The consequences of this accident have been assessed assuming 2000 ppm soluble boron with pool water. In its final assumed position lying on top of the rack, the calculated k_{eff} is 0.697, which is well within the NRC limits.

4.0 Summary and Conclusions

Criticality safety analyses of the Comanche Peak fuel storage facilities resulted in k_{eff} values less than the regulatory limits for all conditions considered with fuel types expected to be loaded in CPSES of up to 5.0% enrichment. These facilities include both the new fuel vault, the in-containment storage rack, and the spent fuel pool. Tables 2 and 3 summarize the calculated k_{eff} of the facilities, confirming that the Comanche Peak fuel storage facilities can safely accommodate fuel of 5% enrichment or any fuel of lower enrichment or reactivity.

1. R.M. Westfall, et. al., "NITAWL-S: Scale System Module for Performing Resonance Shielding and Working Library Production" in SCALE: A Modular Code System for performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200, 1979
2. L.M. Petric and N.F. Landers, "KENO Va. An Improved Monte Carlo Criticality Program with Supergrouping" in Scale: A Modular Code System for performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200, 1979.
3. R.M. Westfall et al., "SCALE: A Modular Code System for performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, 1979.
4. A. Ahlin, M. Edenius, H. Haggblom, "CASMO - A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik report (proprietary).
5. A. Ahlin and M. Edenius, "CASMO - A Fast Transport Theory Depletion Code for LWR Analysis," ANS Transactions, Vol. 26, p. 604, 1977.
6. M. Edenius et al., "CASMO Benchmark Report," Studsvik/ RF-78-6293, Aktiebolaget Atomenergi, March 1978.
7. "CASMO-3 A Fuel Assembly Burnup Program, Users Manual", Studsvik/NFA-87/7, Studsvik Energitechnik AB, November 1986
8. M. Edenius and A. Ahlin, "CASMO-3: New Features, Benchmarking, and Advanced Applications", Nuclear Science and Engineering, 100, 342-351, (1988)
9. M.G. Natrella, Experimental Statistics National Bureau of Standards, Handbook 91, August 1963.

Table 1

UNCERTAINTIES FROM MANUFACTURING
TOLERANCES USING CASMO-3 AND KENO-5a

	<u>CASMO3</u>	<u>KENO5a</u>
UO ₂ density	0.0031	.0023
Lattice Spacing	0.0000	.0004
Enrichment	0.0015	.0019
Box I.D.	0.0003	.0017

Table 2

SUMMARY OF CRITICALITY SAFETY ANALYSES
NEW FUEL VAULT - 5% ENRICHED FUEL
(Under Accident Conditions)

	Optimum ⁽³⁾ Moderation	Flooded Condition
Temperature for analysis (68°F)	20°C (68°F)	20°C
Reference k_{∞} (KENO-5a)	0.9513	0.9143
Calculational bias, δk	0.0101	0.0101
Uncertainties		
In the Bias ⁽¹⁾	± 0.0020	± 0.0020
KENO Statistics ⁽¹⁾	± 0.0047	± 0.0022
Lattice spacing	± 0.0006	Negligible
Box I.D.	± 0.0005	± 0.0003
Spacing between rows	± 0.0010	NA
SS thickness	± 0.0094	± 0.0016
Fuel enrichment	± 0.0011	± 0.0015
Fuel density	± 0.0017	± 0.0031
Eccentric fuel	Negligible	Negligible
Statistical combination of uncertainties ⁽²⁾	± 0.0110	± 0.0048
Total	0.9614 ± 0.0110	0.9244 ± 0.0048
Maximum Reactivity (k_{eff})	0.972	0.929

(1) With two-sided factor for 95%/95% tolerance.

(2) Square root of sum of squares.

(3) 5.3% of normal water density.

Table 3

SUMMARY OF CRITICALITY SAFETY ANALYSES
 SPENT FUEL POOL - 5% W OFA ENRICHED FUEL
 (Without Soluble Boron)

		SFP
Temperture for analysis		40°C (104°F)
Reference k_{∞} (KENO)		0.9310
Calculational bias, σk		0.0101
Uncertainties ⁽¹⁾		
In the Bias ⁽¹⁾	± 0.0020	
KENO Statistics	± 0.0022	
Lattice spacing	± 0.0003	
Box T. D.	± 0.0002	
SS thickness	± 0.0024	
Fuel enrichment	± 0.0019	
Fuel density	± 0.0021	
Eccentric Fuel	<u>Negligible</u>	
Statistical combination of uncertainties		± 0.0048
Total		0.9411 ± 0.0048
Maximum Reactivity (k_{eff})		0.946

- (1) With two-sided factor for 95%/95% tolerance.
 (2) Square root of sum of squares.

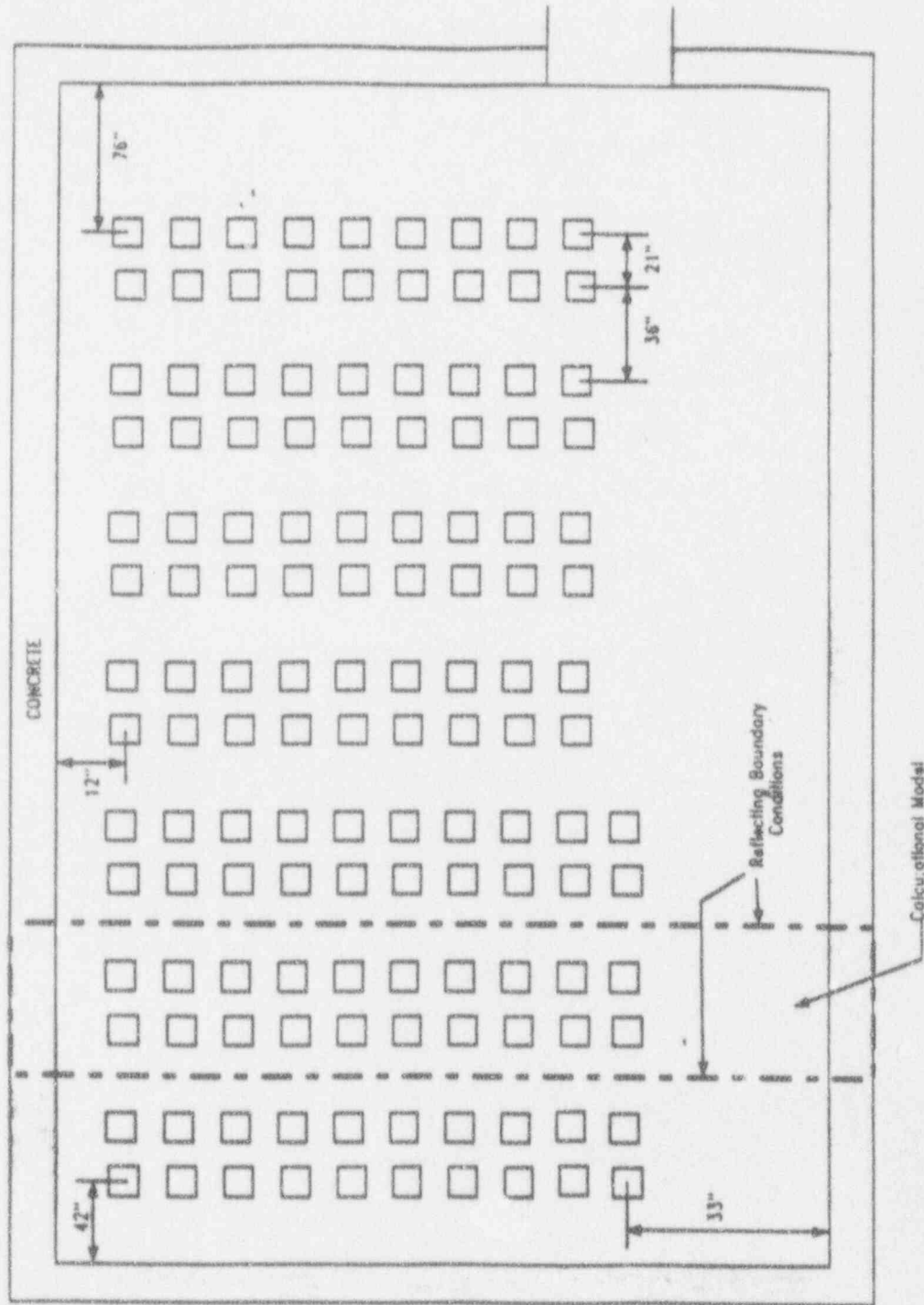


Fig. 1 NEW FUEL ASSEMBLY LAYOUT AND CALCULATIONAL MODEL

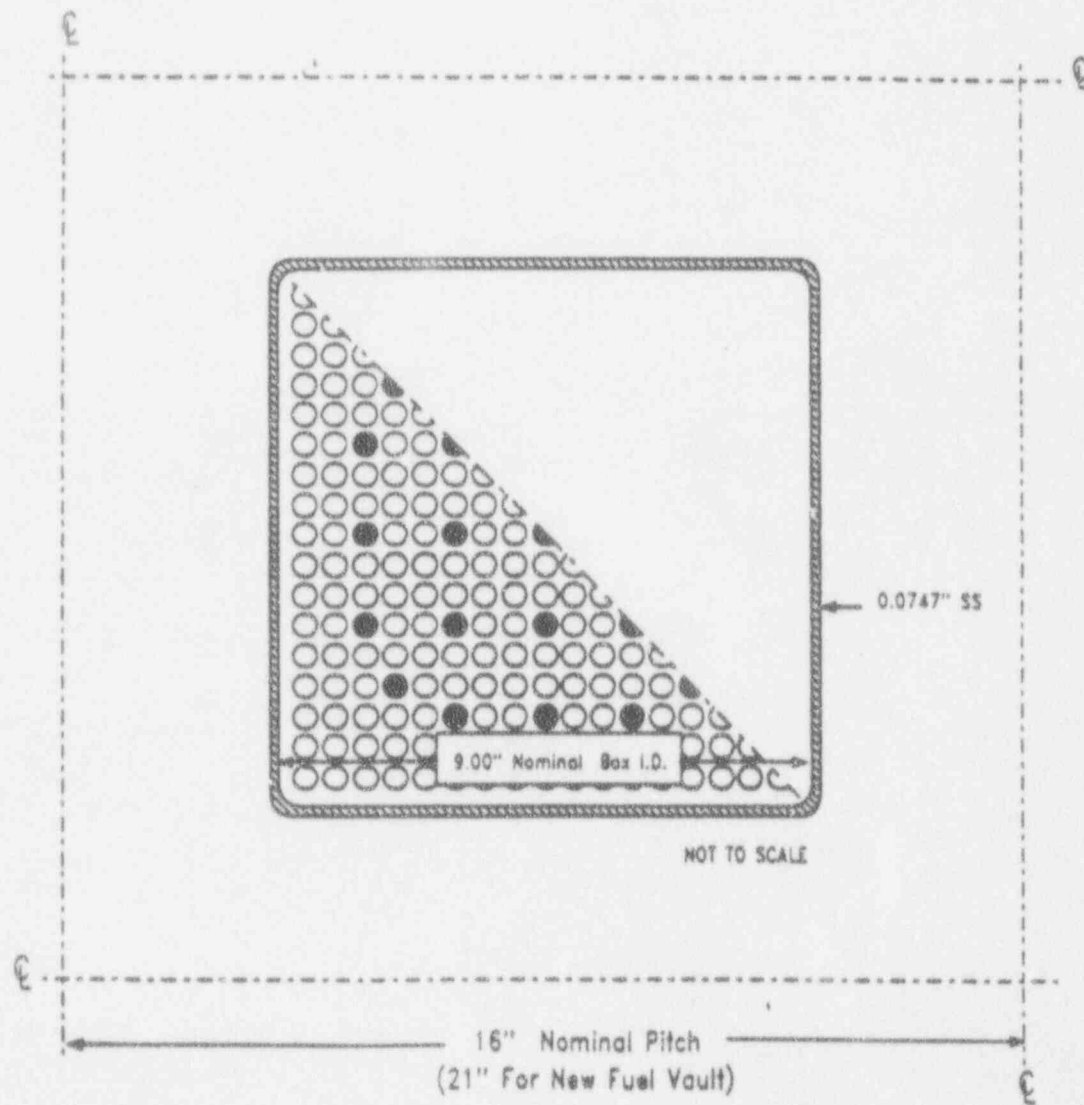


FIG. 2 CROSS SECTION OF SPENT FUEL STORAGE CELL

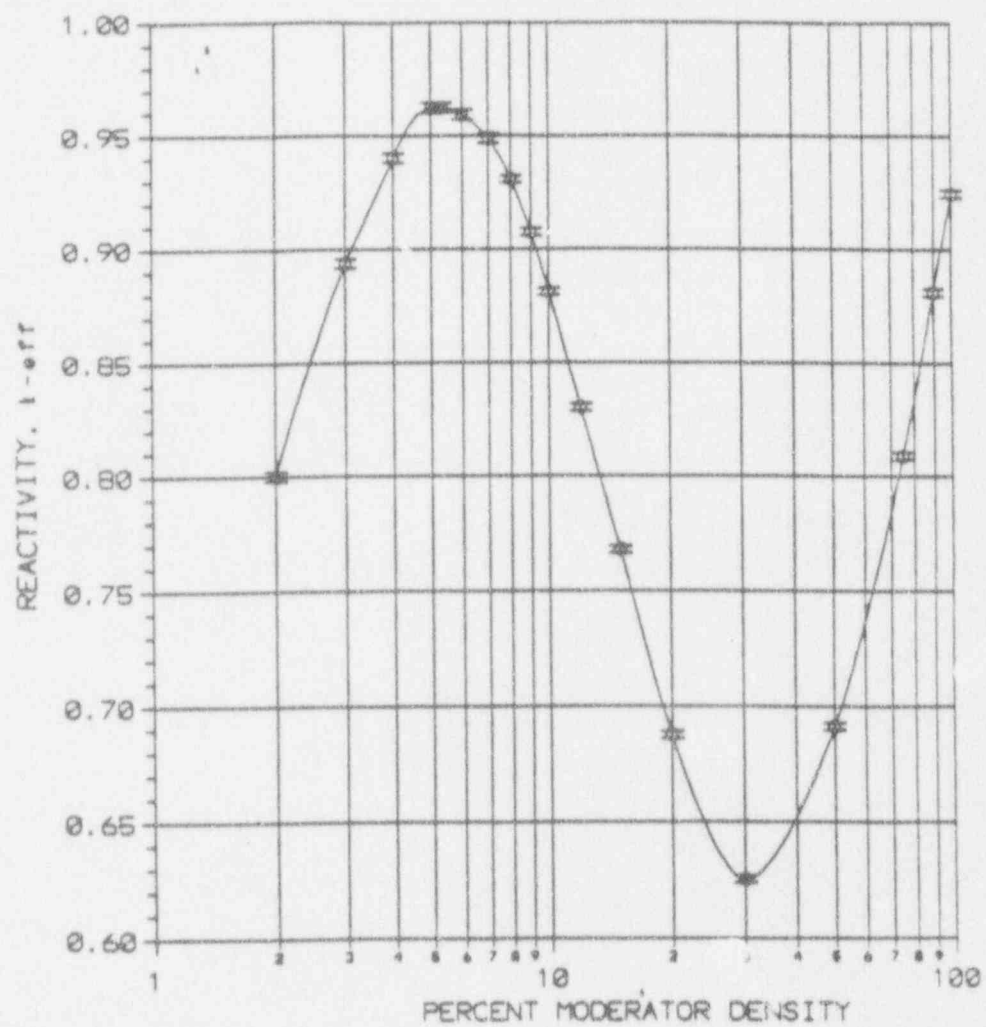


FIG. 3 REACTIVITY VARIATION WITH MODERATOR DENSITY

APPENDIX A

BENCHMARK CALCULATIONS

by

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August, 1992

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 CASMO-3 code (3) for use in criticality safety calculations of spent fuel racks. Both calculational methods are based on 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.

Benchmark calculations were performed for critical experiments that are representative of realistic fuel storage racks and poison worths. Results of these calculations show that the 27 group (SCALE) NITAWL-KENO-5a calculations consistently underpredict the critical eigenvalue by 0.0101 ± 0.0020 (with a two-sided tolerance factor for 95% probability at a 95% confidence level).

Extensive benchmarking calculations of critical experiments with CASMO-3 have been reported (5), giving a bias in k_{eff} of 0.0004 ± 0.0011 for 37 cases. The 95%/95% bias for the CASMO-3 data is 0.0000 ± 0.0027 (neglecting the small overprediction in k). CASMO-3 and NITAWL-KENO-5a intercomparison calculations of infinite arrays of poisoned cell configurations (representative of typical fuel storage rack designs) confirm that the reported bias is reasonable for use in CASMO-3 calculations. Additionally, Reference 5 documents good agreement between CASMO-3 calculations and measurements of heavy nuclide concentrations for Yankee core isotopics.

The benchmark calculations reported here confirm that either the 27 group (SCALE) NITAWL-KENO-5a or CASMO-3 calculations are acceptable for criticality analysis of spent fuel storage racks.

For configurations with large water gaps (> 2 or 3 inches), results of CASMO-3 calculations differ from corresponding KENO-5a results. The difference observed depends on the size of the water gap and the number of mesh intervals used in the CASMO-3 calculation. However, as discussed in Section 3, CASMO-3 calculations will provide reasonable estimates of uncertainties for small perturbations of input parameters (e.g., manufacturing tolerances).

2.0 NITAWL-KENO-5a BENCHMARK CALCULATIONS

Table 1 summarizes results of analysis of a series of Babcock & Wilcox (B&W) critical experiments (4), including some with absorber panels typical of a poisoned spent fuel rack. These critical experiments were calculated with KENO-5a, the 27 group SCALE cross-section library, and the Nordheim resonance integral treatment in NITAWL. Dancoff factors used in NITAWL were calculated with Oak Ridge SUPERDAN routine (from the SCALE system of codes). The mean k_{eff} for these calculations is 0.9899.

Similar calculational bias has been reported by ORNL (7) for 54 critical experiments. The bias for these mostly clean criticals without strong absorbers is 0.0100 ± 0.0015 (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-KENO-5a calculational model for use in spent fuel rack criticality analysis. Further, results reported in the present evaluation show no abnormal deviations in k_{eff} with intra-assembly water gap, absorber panel worth, enrichment, or poison concentration and do not show the trends previously presented using the 123-group GAM-THERMOS cross-section library (8).

Since the B&W critical experiments were made with fuel of 2.459 % U-235 enrichment, additional benchmarking calculations were performed to confirm that enrichment is not a significant factor in the KENO-5a bias. The additional calculations were for a series of French critical experiments (9) at 4.75% enrichment and for several BNWL criticals (11) with 4.26 % enrichment.

The results of the French criticals, presented in Table 2, show an overprediction of k_{eff} . Further, the calculated k_{eff} show a trend toward higher values with decreasing core size. ORNL has reported similar results (10). These critical experiments are for very small cores, and the overprediction suggests that NITAWL-KENO-5a has an inadequate treatment of the very large leakage from very small cores. Since the analysis of fuel storage racks does not entail large neutron leakage, the observed inadequacy will not affect fuel storage rack analysis.

The results shown in Table 2 for the French criticals and BNWL experiments (also small cores, but significantly larger than the French criticals) suggest that enrichment has no effect on the KENO-5a bias. Or, in the case of the French criticals, any small enrichment effect would yield a more conservative value of k_{eff} .

Subsequent CASMO-3 calculations (discussed in Section 5) indicate that enrichment has no significant effect on modeling critical experiments with CASMO-3.

3.0 INTERPOLATION ROUTINE

A special routine was developed to interpolate the hydrogen scattering matrix for temperatures between 20 °C and 277 °C in NITAWL. This special routine corrects a deficiency noted in NRC Information Notice 91-66 (October 18, 1991.) Benchmark calculations were made using CASMO-3 for comparison based on the assumption that two independent methods of analysis would not exhibit the same error.

Results of the benchmark calculations shown in Table 3 confirm that the trend with temperature obtained by both codes is comparable over the range investigated. This agreement establishes the validity of

the interpolation routine used in conjunction with NITAWL-KENO-5a to calculate k_{eff} for temperatures between 20 °C and 277 °C.

The deficiency in the hydrogen scattering matrix does not appear except in the presence of large water gaps where the scattering matrix is important. However, the value of k_{inf} from CASMO-3, in the presence of a large water gap, differs from the KENO-5a value. Table 3 and Figure 1 show results for both codes for a water gap of 2.6 inches. The absolute values of k differ, but the trend with temperature is very similar for both codes. The agreement in the trend with temperature lends further credibility to the interpolation routine, and also supports the use of CASMO-3 to provide reasonable estimates of changes in k due to minor perturbations in input values (e.g., manufacturing tolerances).

4.0 CLOSE PACKED ARRAYS

The B&W close-packed series of critical experiments (12) simulate consolidated fuel. These experiments were analyzed using NITAWL-KENO-5a. Results of these analyses, shown in Table 4, suggest a slightly higher bias than that for fuel with normal lattice spacing. ORNL (13) has obtained similar results. Because very few cases are available for analysis, a maximum bias for close packed lattices of 0.0155 including uncertainties should be used. This value of bias would conservatively encompass all but one of the cases measured.

5.0 THE CASMO-3 BENCHMARK CALCULATIONS

The CASMO-3 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, CASMO-3 is well suited for criticality analysis of fuel storage racks since general practice treats these racks as an infinite medium of storage cells, neglecting leakage effects.

CASMO-3 is a modification of CASMO-2E code and has been extensively benchmarked against both mixed oxide and hot and cold critical experiments by Studsvik Energiteknik. Reported analyses (5) of 37 critical experiments indicate a mean k_{eff} of 1.0004 +/- 0.0027 (95%/95%). To independently confirm the validity of CASMO-3 and to investigate any effects of enrichment, a series of calculations was made with CASMO-3 and with NITAWL-KENO-5a for identical poisoned storage cells representative of a typical spent fuel storage rack.

Results of these intercomparison calculations* shown in Table 5 are within the normal statistical variation of KENO calculations. Since two independent methods of analysis are not expected to have the same error, the agreement between CASMO-3 and KENO-5a indicate that fuel enrichment does not have a significant effect over the range of enrichment of power reactor fuel (2.5% to 5%).

Results of these intercomparison calculations* shown in Table 5 are within the normal statistical variation of KENO calculations. Since two independent methods of analysis are not expected to have the same error, the agreement between CASMO-3 and KENO-5a indicate that fuel enrichment does not have a significant effect over the range of enrichment of power reactor fuel (2.5% to 5%).

A second series of CASMO-3-KENO-5a intercomparison calculations consist of analysis of the central cell only of 5 B&W experiments. The calculated results, shown in Table 5, indicate a mean delta k difference that lies within the 95% confidence limit of the KENO-5a calculations.

The combined data in Table 5 show a mean difference in k_{eff} values between CASMO-3 and KENO-5a of 0.0003 ± 0.0012 (two-sided 95%/95% confidence, CASMO yielding the higher value). Combined with the uncertainty in the KENO-5a bias, the CASMO-3 bias is 0.0000 ± 0.0023 .

* Intercomparison between analytical methods is a technique endorsed by Reg. Guide 5.14, "Validation of Computational Methods for Nuclear Criticality Safety."

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Table 1

RESULTS OF 27-GROUP (SCALE) NITAWL-KENO-5a 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	0.9890	± 0.0006
Mean	$0.9899 \pm 0.0007^{(1)}$	$\pm 0.0006^{(2)}$
Bias	$0.0101 \pm 0.0020^{(3)}$	

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

(2) $(\sum (\sigma)^2 / 16)^{1/2}$

(3) With two-sided factor ($K=2.903$) for 95%/95% tolerance.

Table 2

RESULTS OF 27-GROUP (SCALE) NITAWL-KENO-5a 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

Intercomparison of NITAWL-KENO-5a (with Interpolation Routine)
and CASMO-3 Calculations at Various Temperatures

<u>Temperature</u>	<u>CASMO 3</u>	<u>W-N-KENO-5a^(*)</u>
4°C	1.2276	-
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 4

Reactivity Calculations for Close-Packed
Critical Experiments

Calc. No.	B&W 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

Table 5

RESULTS OF CASMO-3 AND NITAWL-KENO-5a
BENCHMARK (INTERCOMPARISON) CALCULATIONS

Enrichment ⁽¹⁾ Wt. % U-235	NITAWL-KENO-5a ⁽²⁾ k_{∞}	CASMO-3	δk CASMO-KENO
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
B&W 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	0.0003 ± 0.0012 (2-sided 95%/95%)

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

(2) k_{∞} from NITAWL-KENO-5a corrected for bias.

(3) Central Cell from B&W Critical Experiments

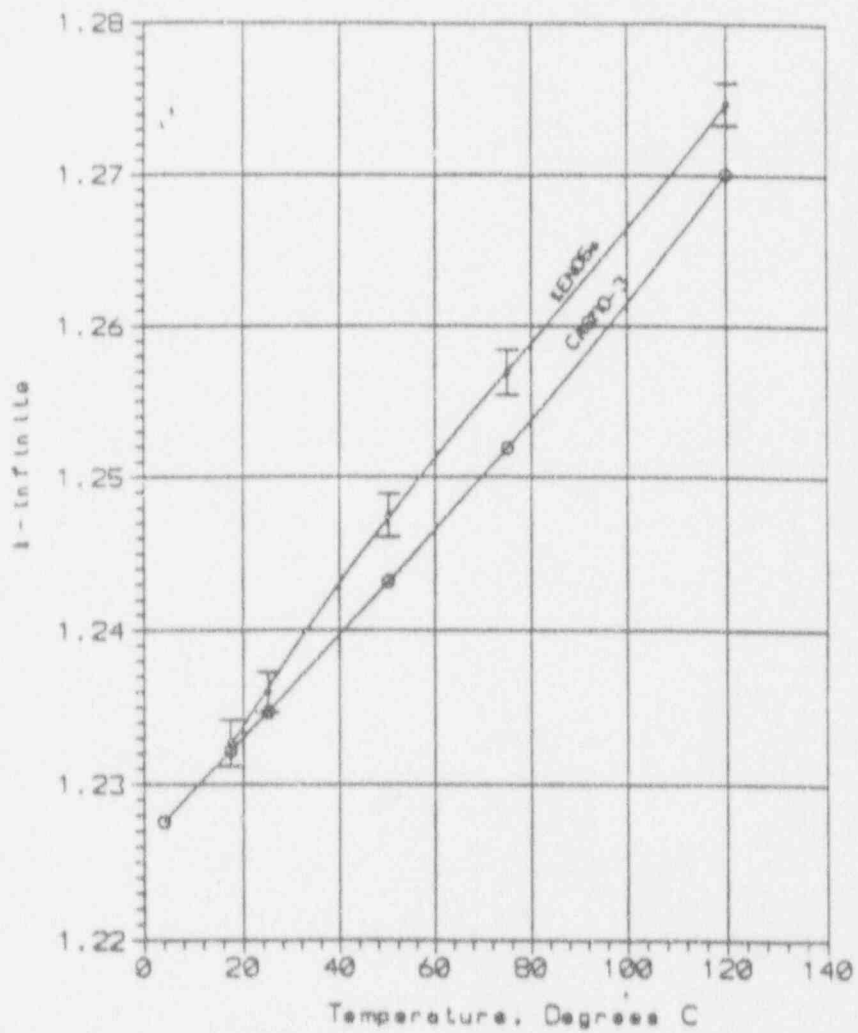


Fig. 1 Comparison of CASMO-3 and KENO5a
Temperature Dependence