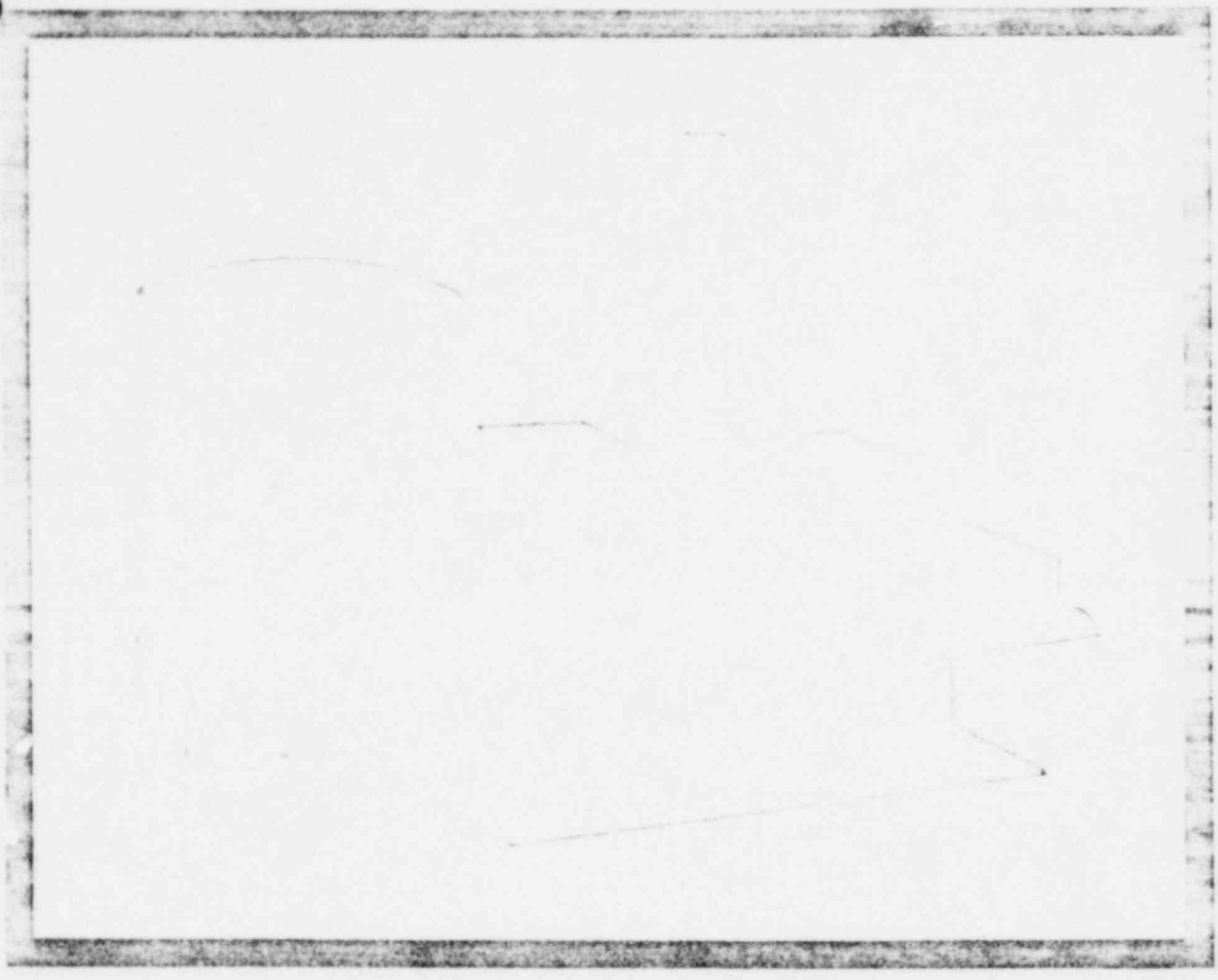


ANEFCO

ANEFCO INC. 222 Mamaroneck Avenue White Plains, New York 10605 914-948-4633

May 16, 1980

Charles McDonald, Chief
Nuclear Regulatory Commission
Division of Fuel Cycle and Material Safety
Transportation Branch
Washington, D.C. 20555

Subject: ANEFCO AP-101 Cask Supplement

Dear Mr. McDonald:

Enclosed are ten (10) copies each of Section 6 CRITICALITY EVALUATION, and Section 7 OPERATING PROCEDURES, to complete ANEFCO's submittal for a petition to amend the AP-101 cask Certificate of Compliance to include spent fuel assemblies as contents.

While the earlier submittal of December 20, 1979 indicated in Figure 1.6 (page 1-17) and on page 1-18 that 1/4" thick boral plates would be used for criticality control, we have added 1/8" thick plates of boral to the spent fuel basket design for PWR fuel. Revision A sheets for Figure 1.6 and page 1-18 are attached which incorporate that design modification.

The criticality analysis resulted in a maximum reactivity of 0.925 for the hypothetical accident condition of transport. We believe that the demonstrated safety from criticality along with the high degree of safety indicated in the earlier ANEFCO submittal support this petition to include spent LWR fuel assemblies as contents in the AP-101 cask.

We await any questions that you may have regarding this submittal and will respond in a timely fashion to expedite this petition.

Very truly yours,

ANEFCO INC.

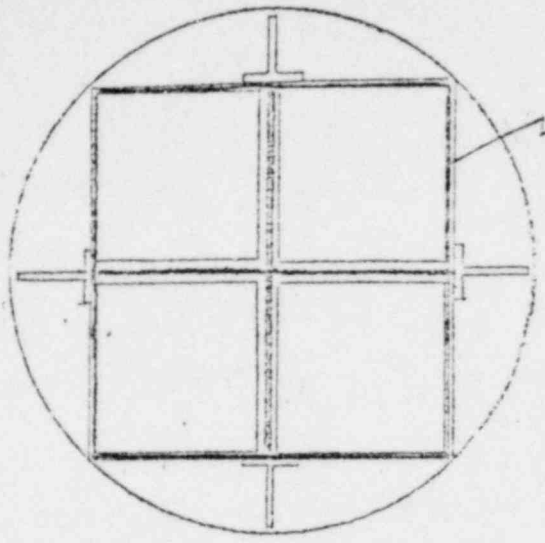
John D. Murphy
President

JDM/er

encl.

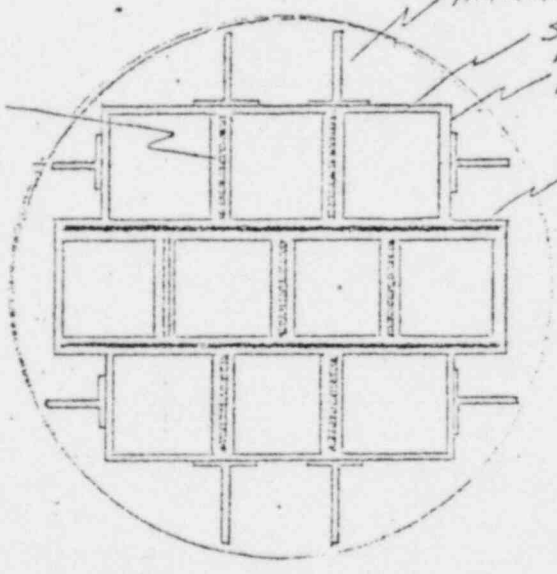
16113

FIG 1.6 TYPICAL FUEL BASKET CONFIGURATIONS



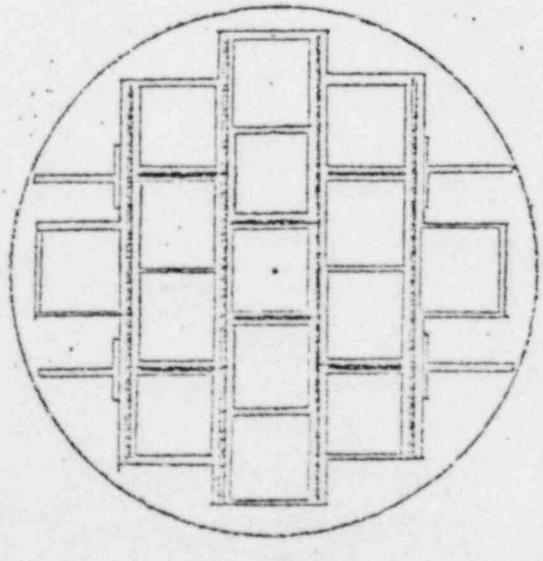
1/8" boral plates (4) (A)

4 PWR F/A's



10 BWR F/A's

NOTE: All boral plates 1/4" thick except as noted. (A)



15 6X6 BWR F/A's

REV. A. 4/28/80

Spacers will be attached to the cover of the fuel basket to fit into each fuel cannister and fill the space between the top of the fuel assembly and the cover of the fuel basket.

The fuel cannisters and fuel basket outer structure will be anodized aluminum to promote radiative transfer of heat from the contents to the AP-101 cask.

1.2.3.5.3 Criticality Control

Boral plates of nominal 1/4" and 1/8" thickness are used in the fuel baskets to provide criticality control. As noted in Figure 1.6, some of the plates will traverse the full width of the fuel basket in order to provide a conduction path for heat transfer from interior spent fuel assemblies. The boral plates will be affixed to the fuel cannisters with flat steel bands. The long boral plates will be banded to several fuel cannister tubes. Short boral plates will be banded to a single fuel cannister. (A)

The position of spent fuel assemblies and boral plates for criticality control are secured within the fuel basket outer structure, which is in turn secured within the cask cavity by the placement of aluminum tees where there are gaps between the fuel basket outer side walls and the cask cavity. This tight assemblage is based on close spacing between the individual fuel assemblies and the intervening structure (fuel cannisters and boral plate). Since no loss of integrity for the fuel assemblies is possible for the "normal" and "accident" conditions of transport, the array of fuel assemblies, boral criticality control, and fuel basket structure within the cask cavity represents the closest packing arrangement used as the basis for the criticality analysis presented in Section 6 of this Amendment to the AP-101 SAR.

AMENDMENT TO ANEFCO, INCORPORATED
SAFETY ANALYSIS REPORT
CASK AP-101
SECTION 6 CRITICALITY
SECTION 7 OPERATING PROCEDURES
MAY 15, 1980

16413

ANEFCO

TABLE OF CONTENTS

APPENDICES TO SECTION 6

APPENDIX 1. NULIF Code Description

APPENDIX 2. NUTAN Code Description

APPENDIX 3. Computer Output Images

3.1 Benchmark Calculation
KENO 4 Input
Cross-Sections
Array Description
Reactivity Summary

3.2 Criticality Analysis
3.2.1 Reactivity as a Function of Pitch
3.2.1.1 7x7 BWR 1.8745cm
7x7 BWR 1.9805cm
7x7 BWR 2.0864cm
3.2.1.2 6x6 BWR 1.3195cm
6x6 BWR 1.8957cm
6x6 BWR 2.0108cm
3.2.1.3 15x15 PWR 1.4256cm
15x15 PWR 1.4817cm
15x15 PWR 1.524cm
3.2.2 KENO 4 PWR Fuel in AP-101 Cask
Input
Cross-Sections
Array Descriptions
Reactivity Summary
3.2.3 KENO 4 Array of AP-101 Casks-PWR Fuel
Input
Array Description
Reactivity Summary

TABLES

FIGURES

6. CRITICALITY EVALUATION

6.1 Discussion and Results

The AP-101 cask is intended for the shipment of PWR and BWR spent fuel as Fissile Class I shipments. A separate package design and fuel loading is specified for each fuel type considered. Boral plates are incorporated in each package design to assure subcriticality in case of water moderation and reflection. The results of this criticality evaluation indicate that the requirements of 10 CFR 71.33 through 71.36 will be satisfied for single packages while 10 CFR 71.37 and 71.38 will be satisfied even for an infinite array of packages.

6.1.1 Criticality Design Features

The spent fuel basket designs for the typical PWR and BWR, and the smaller 6 x 6 BWR fuel assemblies considered in this analysis all share similar design features. As depicted in Figure 1-6, Section 1 of this Amendment, the spent fuel baskets each consists of an array of square aluminum tubes (fuel cannisters) each to house a single spent fuel assembly. The array of fuel cannisters are secured in a tight configuration by plates of aluminum which form the outer shell of the spent fuel basket. Plates of boral are provided between adjacent fuel cannisters or fuel basket outer shell walls as indicated to provide criticality control. The spent fuel basket is secured within the cask cavity by placement of aluminum structural tees between the fuel basket outer side walls and the cask cavity.

The spent fuel basket designs are tight assemblies to maintain the geometric arrangement under "normal" and "accident" conditions of transport. Therefore, the design array of fuel assemblies, boral criticality control, and fuel basket structure within the cask cavity represents the packing arrangement used as the basis for this criticality analysis.

6.1.2 Criticality Evaluation

The computational method for the criticality analysis employs up-to-date and fully verified techniques. A benchmark calculation was performed for a related criticality experiment to verify the computational method. The statistically close agreement for the benchmark calculation (discussed in Section 6.5) strongly justifies the adequacy of the criticality evaluation method.

The LWR fuel lattices considered are all somewhat undermoderated, so cold water at its maximum density was assumed in order to maximize reactivity for the water moderated and reflected accident conditions. Normal conditions of transport for the dry cask would be highly subcritical and consideration of a 5% reduction in fuel rod spacing would not alter the highly subcritical nature of the packages considered.

For the hypothetical accident conditions of transport the criticality analysis assumed fresh fuel with the cask flooded with water both internally and externally to represent the most reactive configuration for calculational purposes. The design arrangement of fuel, poison and basket materials within the cask apply for "accident" conditions of transport since the tight packing of fuel, poison and basket structures leave little room for movement. The reactivity effect of fuel rod pitch was considered separately in this analysis.

6.1.3 Results of Criticality Evaluation

The criticality evaluation considered separate loadings of four (4) PWR, ten (10) regular BWR, or fifteen (15) of the smaller size 6 x 6 BWR spent fuel assemblies. The maximum reactivity was calculated for the AP-101 cask containing four (4) PWR fuel assemblies. The reactivity for an infinite array of flooded AP-101 casks each containing four (4) PWR fuel assemblies was calculated to be 0.8926 with a standard deviation of ±0.0054. A statistically similar result of 0.8995 with a standard deviation of ±0.0059 was obtained for a single flooded AP-101 cask. The flooded single cask is essentially fully reflected and the presence of additional similar casks has little reactivity effect. Imposition of the calculational uncertainties presented in Section 6.4.3 of this Amendment to the SAR for the AP-101 cask results in a maximum adjusted reactivity of 0.9 25 for the hypothetical accident case. Thus, this criticality evaluation indicates that the AP-101 package satisfies the requirements for a Fissile Class I package.

6.2 PACKAGE FUEL LOADING

The criticality analysis was performed for the maximum fuel loadings of LWR fuels in the ANEFCO AP-101 cask. For PWR spent fuel the physical limitation is four (4) assemblies and ten (10) assemblies for typical BWR spent fuel. The smaller 6x6 BWR fuel can be loaded to a maximum of 15 assemblies. Figures 1.4 and 1.5 (Section 1, Amendment to AP-101 SAR) illustrate the maximum loading configurations which were used in this criticality analysis.

The specifications for individual spent fuel assemblies is presented in Table 1.2. Additionally, Table 6.1 in this section provides details of fuel assembly characteristics and dimensions used in this analysis. It is seen that a loading of four (4) design basis PWR fuel assemblies would result in a package loading of about 140 pounds of U-235 which is somewhat higher than that for BWR spent fuel as contents in the AP-101 cask.

TABLE 6.1

SPENT FUEL LOADING OF PACKAGE
AP-101 CASK

	PWR Fuel	Regular BWR Fuel	Small (6x6) BWR Fuel
F/A's in package, number	4	10	15
Max. U-235 enrichment, w/0	3.4	3.0	2.5
F/A UO_2 content, pound	1,170	490	355
UO_2 in package, pound	4,680	4,900	5,325
U-235 in package, pound	140	130	117

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

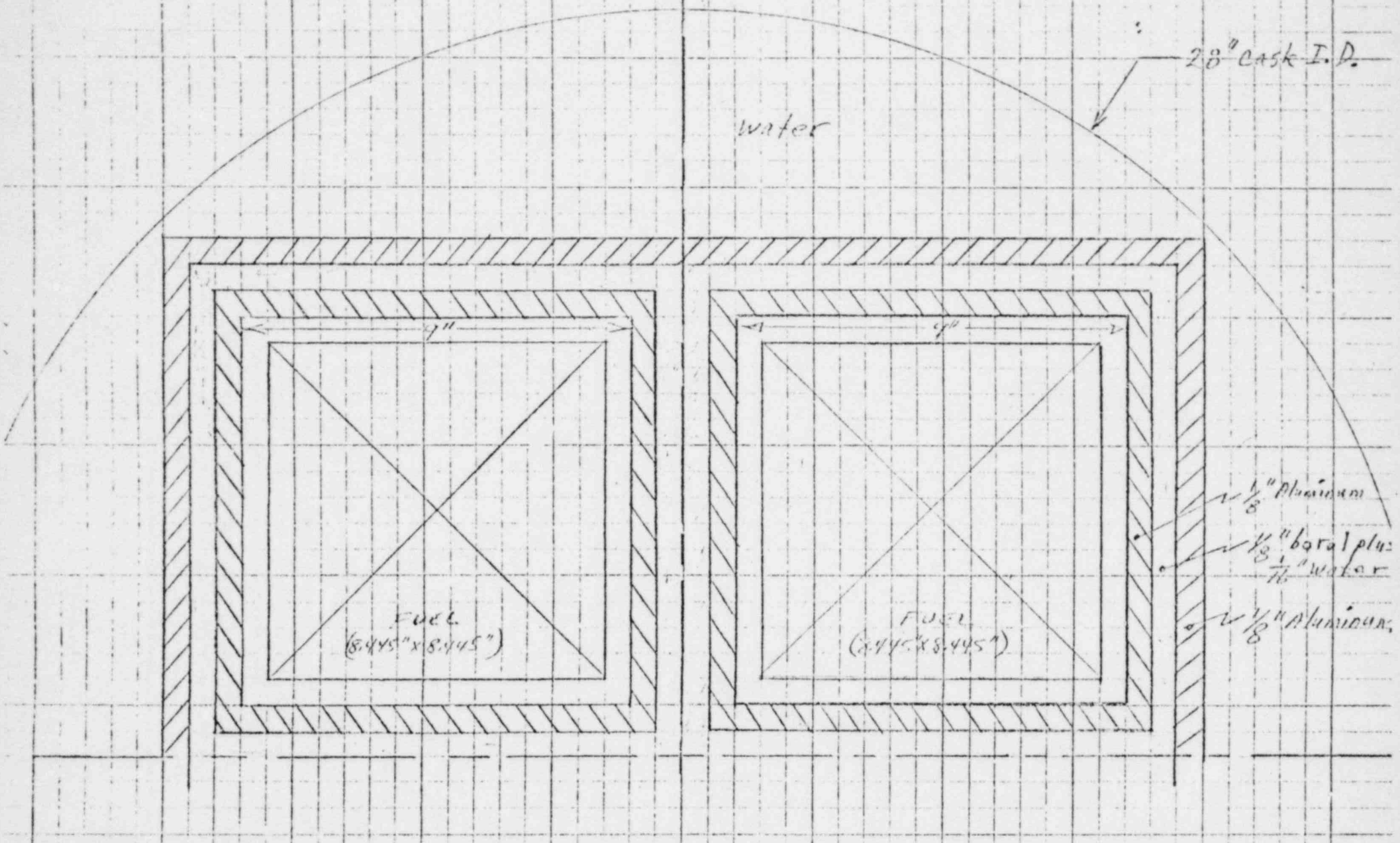
The KENO IV program can calculate the reactivity of complex three dimensional configurations. Full advantage was taken of this capability to closely model the three (3) different package fuel loadings considered in this criticality analysis. The essential features of the fuel, fuel basket, and AP-101 cask were exactly modeled for the KENO criticality calculations. The following is a discussion of the modeling procedures which apply for each of the fuel loadings considered.

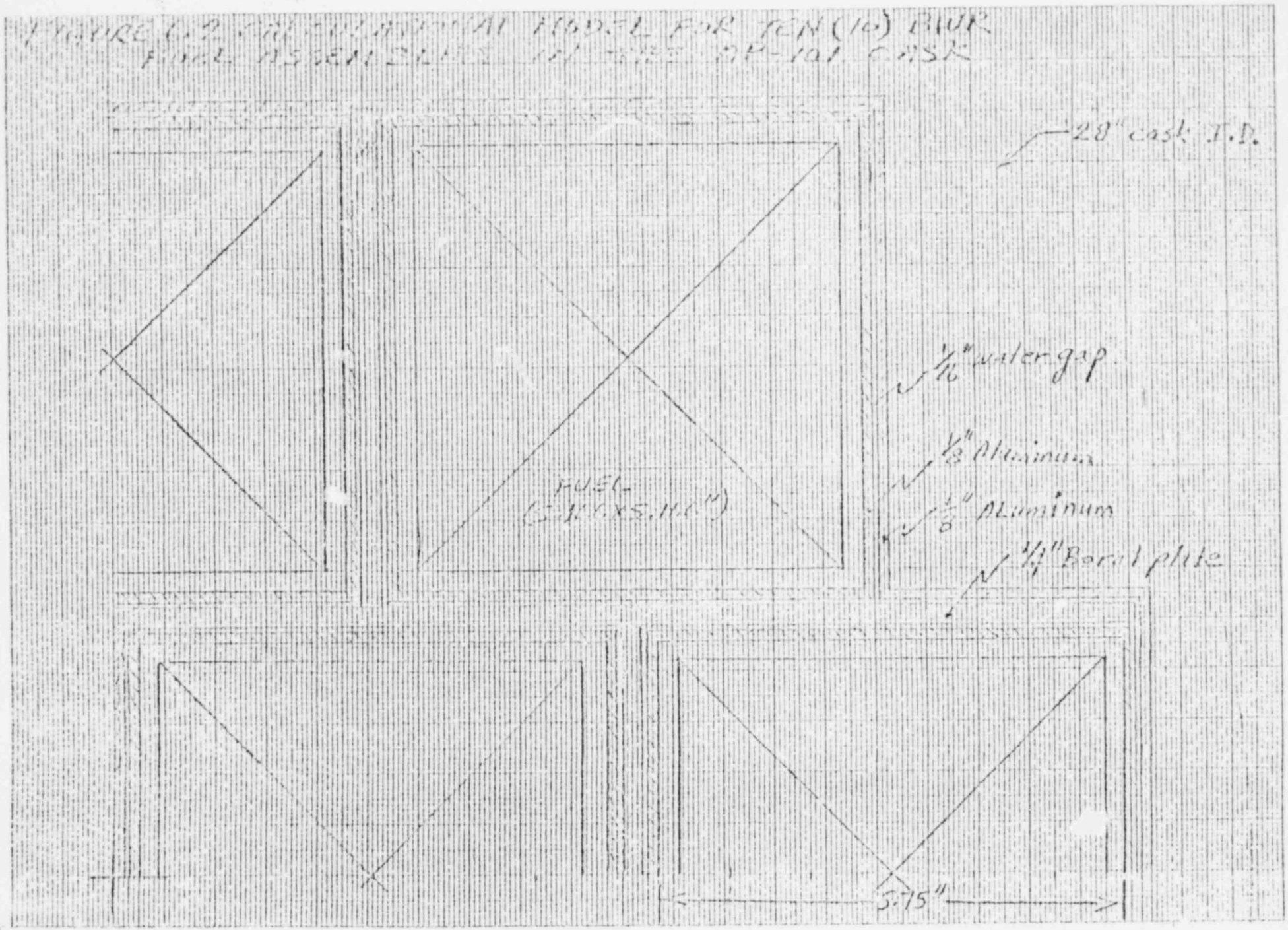
Figure 1.6 depicts the overall plan views of the loading configurations for the AP-101 containing regular BWR fuel, 6x6 BWR fuel, and PWR fuel. For calculational purposes, the fuel lattice was homogenized, but the overall dimensions of the fuelled regions were preserved exactly. Fuel assembly end fittings such as nozzles, etc. were not included, the replacement in the model for this omitted structure would result in a small positive reactivity effect. Each fuel assembly is contained within an aluminum fuel cannister. While the number, and square tube size of the aluminum fuel cannisters are different for each of the three types of fuel considered, the baskets all share similar characteristics.

The fuel basket components are all constructed of the same materials, and thicknesses of similar plates comprising the three spent fuel baskets are identical except for the boral plates. All boral plates in the BWR baskets and the central cruciform in the PWR basket are nominally 1/4" thick. The outer boral plates in the PWR basket are 1/8" thick. All aluminum fuel cannisters are 1/8" thick and all basket structural (outside) plates are 1/8" thick. A 1/16" water gap is assumed to exist between adjacent structures of the spent fuel basket. That is, there is a 1/16" water gap between each aluminum fuel cannister side and the adjacent boral plate, and there are similar 1/16" water gaps between each aluminum fuel cannister side and adjacent basket structural (outside) plates. Adjacent aluminum fuel cannisters without intervening boral plates are also each separated by a 1/16" water gap. These features may be seen in Figures 6.1, 6.2, and 6.3.

Key fuel and aluminum fuel cannister dimensions are presented in Table 6.2. Dimensions of the AP-101 cask used in the criticality calculations are presented in Figure 6.4. Table 1.2 provides the overall specifications for the design basis fuel.

FIGURE 6-1 CALCULATIONAL MODEL FOR FOUR (4) PWR FUEL ASSEMBLIES IN THE AT-101 CASK




28" cask I.D.
1/16" water gap
1/8" Aluminum
1/8" Aluminum
1/4" Barrel plate
FUEL
(5.1" x 5.1" x)
5.75"

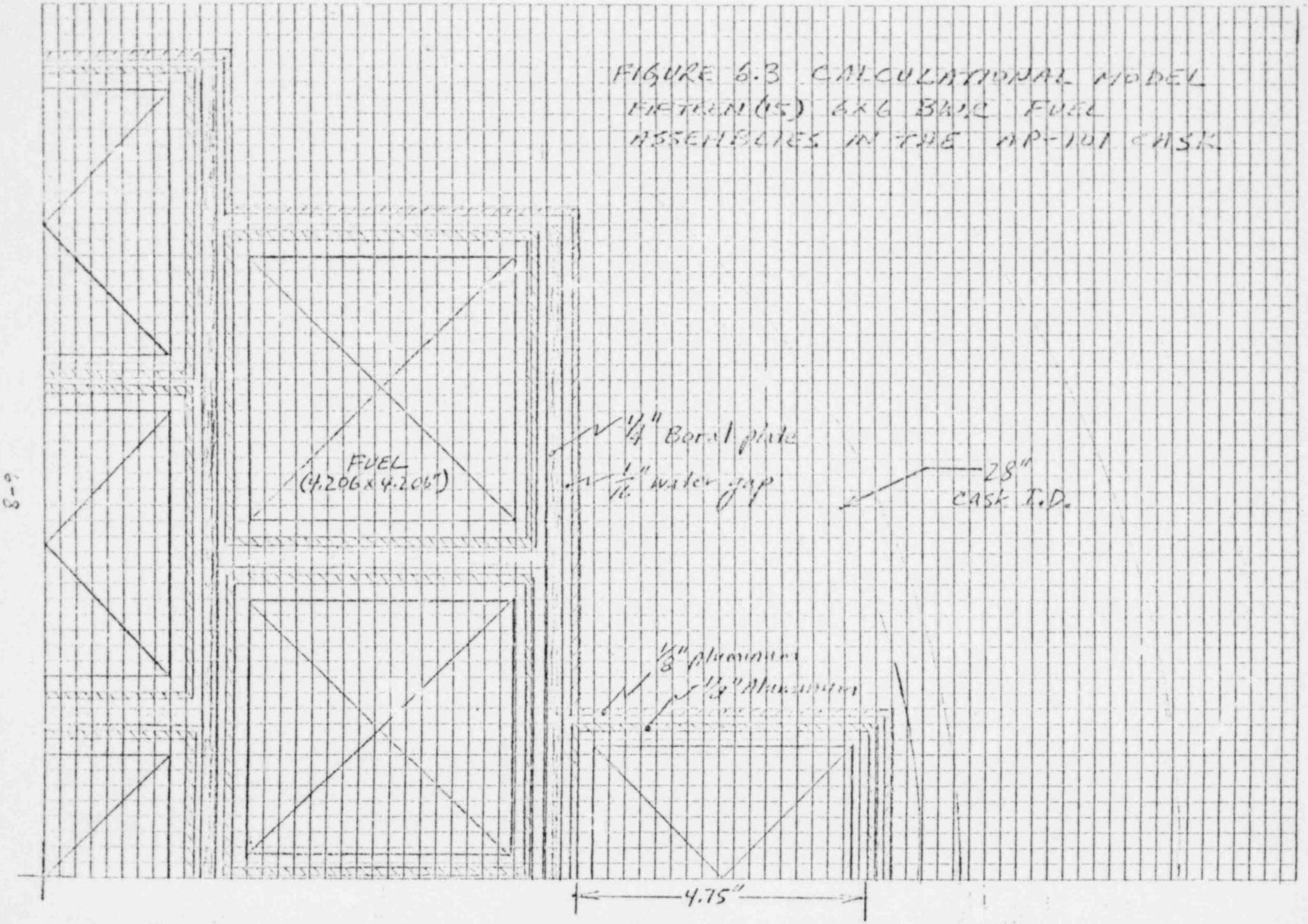


FIGURE 6.3 CALCULATIONAL MODEL
FACTORY (45) 6x6 BWR FUEL
ASSEMBLIES IN THE MP-101 CASK

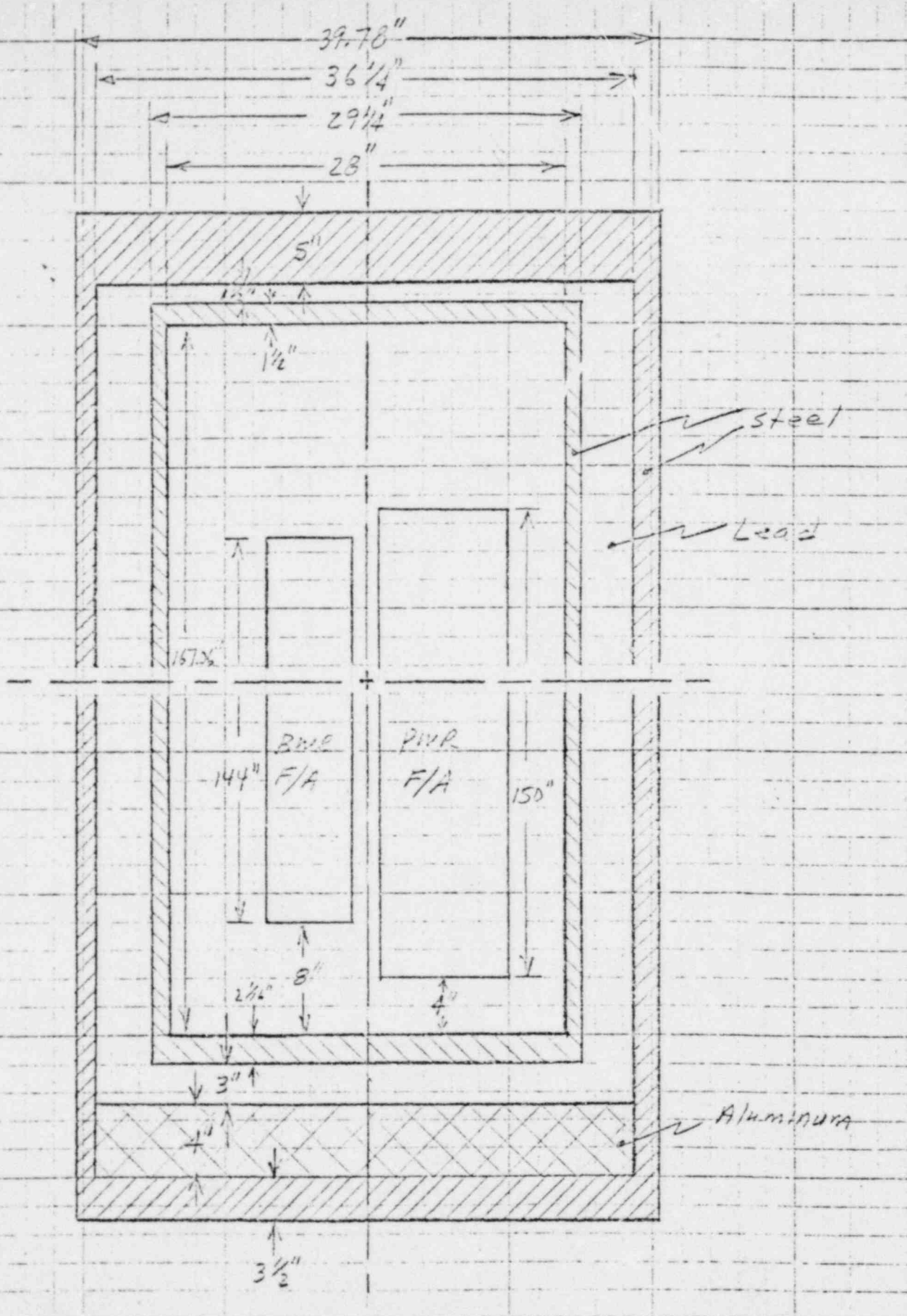


FIGURE 6.4 CALCULATIONAL MODEL

AP-101 CASK

TABLE 6.2

FUEL AND SQUARE ALUMINUM FUEL CANNISTER PHYSICAL DESCRIPTION

	15x15 PWR	7x7 BWR	6x6 BWR
1. Fuel Rods			
Fuel pellet O.D., inch	0.3655	0.488	0.480
Clad I.D., inch	0.374	0.498	0.493
Clad O.D., inch	0.422	0.562	0.563
Pin pitch, inch	0.563	0.738	0.701
2. Fuel Assemblies			
Number of Fuel Rods	204	49	36
Number of Instrument Cells	1	--	--
Number of Guide Tube Cells	20	--	--
Active Fuel Envelope, inch	8.445x8.445x150	5.166x5.166x144	4.206x4.206x144
3. Fuel Cannister			
Tube Square Dimension, inch	9.25x9.25	6.0x6.0	5.0x5.0
Tube thickness, inch	0.125	0.125	0.125
Length, inch	167	167	167

6.3.2 Package Regional Densities

The calculational model consists of regions of fuel, water gaps, boral plates, aluminum structure, and cask steel and lead structure. The material density and atomic number densities are presented in Table 6.3 for fuel regions and Table 6.4 for fuel basket and cask regions. Only PWR fuel have an instrument cell and guide tube cells in the fuel assembly.

TABLE 6.3

FUEL REGION MATERIAL AND ATOMIC NUMBER DENSITIES

	(Material Density, g/cm^3) Atom Density, atom/barn-cm		
	15x15 PWR	7x7 BWR	6x6 BWR
1. Fuel Pellet	(10.202 g/cm^3)	(10.400 g/cm^3)	(10.495 g/cm^3)
U-235	7.8351-4	6.9609-4	5.8536-4
U-238	2.1979-2	2.2507-2	2.2829-2
O	4.5525-2	4.6407-2	4.6828-2
2. Clad (6.57 g/cm^3)			
ZY-2	4.3380-2	4.3380-2	4.3380-2
3. Moderator (1.0 g/cm^3)			
H	6.6864-2	6.6743-2	6.6743-2
O	3.3432-2	3.3373-2	3.3373-2
4. Instrument Cell			
H	5.068-2	--	--
O	2.534-2	--	--
ZY-2	1.063-2	--	--
5. Guide Tube Cell			
H	6.145-2	--	--
O	3.073-2	--	--
ZY-2	3.630-2	--	--

TABLE 6.4

MATERIAL AND ATOMIC NUMBER DENSITIES FOR FUEL BASKET AND CASK REGIONS

		(Material Density) Atom Density, atom/barn-cm
1.	Water Gap (1.0 g/cm^3)	6.6864-2
	H	3.3432-2
	O	
2.	Boral Region* (2.50 g/cm^3-boral)	
	H	2.2285-2
	B-10	3.2973-3
	B-11	1.3662-2
	C	4.2422-3
	O	1.1143-2
	Al	2.8519-2
3.	Aluminum (2.69 g/cm^3)	6.0092-2
4.	Steel (8.03 g/cm^3)	8.7398-2
5.	Lead (11.2 g/cm^3)	3.2558-2

* Includes boral plate of nominal 1/4" thickness plus 1/16" of water on each side.

6.4 CRITICALITY CALCULATION

6.4.1 Calculational Method

6.4.1.1. Reactivity Programs

The computer program used to calculate reactivity for spent fuel in the AP-101 cask is KENO 4[2]. KENO 4 is a fully certified version of KENO-IV[3] by Babcock and Wilcox Company for the CDC-7600. The combinatorial geometry version[4] was also used for BWR fuel since this version is required to model the more complex arrangement of spent fuel assemblies within the cask. The powerful three dimension geometry capability of these codes permitted the virtually exact modeling of the key features of the fuel assemblies (homogenized fuel lattice), boral criticality control plates, basket structure and the AP-101 cask.

The criticality analysis is performed with thirteen (13) energy groups of neutrons, seven (7) epithermal and six (6) thermal (below 1.85 eV) groups. Although KENO can perform a discrete pin by pin calculation, a homogenized fuel lattice is considered in these calculations. According to Reference 1 (Page 34), the homogenized fuel assembly model can yield as accurate results as the discrete calculation which considers individual fuel pins. Neutron cross-sections are provided separately as input to the KENO calculation.

6.4.1.2 Cross Section Generation

Cross sections are supplied as input for the homogenized fuel lattice, boral poison plates, water, aluminum (basket structure) and cask structure of steel and lead. Cross-section are calculated using the Babcock and Wilcox computer program NULIF[5].

The NULIF code generates a microgroup neutron spectrum and calculates spectrum-weighted few-group parameters. A wide variety of fuel cells, non-fuel cells, and fuel lattices are treated. Variation with fuel burnup (not used in this analysis), fuel temperature, and moderator temperature are readily obtained. Heterogeneous effects, including resonance shielding and thermal flux depressions are treated. The affiliated program NUTAN[6] when run with NULIF computes and punches out in ANISN format. macroscopic cross-sections for a mixture for up to 50 energy groups. Both inelastic and elastic (both P0 and P1) transfer matrices were obtained for KENO input.

NULIF is a standard code used by B&W to generate cross-sections for reactor physics analysis of fuel assemblies, and reactor cores. The NUTAN code was newly developed to obtain punched card output of cross-section data for direct use in reactivity codes.

The use of these codes is widespread in B&W and both codes are fully certified by the B&W Quality Assurance Program.

Sections of the NULIF-NUTAN manuals are presented in the appendix to this section to detail some of the key features of these codes.

6.4.1.3 Nuclear Data Library

A standard B&W production library is used for the NULIF code. All thermal microgroup data (0 to 1.85 eV) is obtained from ENDF/B3. Epithermal data above 1.85 eV is obtained from ENDF/B3 data and from in-house B&W evaluations. Updating of this library may be performed occassionally.

6.4.2 Fuel Loading Optimization

Reactivity calculations were made with the AP-101 cask fully loaded with PWR or BWR fuel assemblies as specified in Tables 1.2 and 6.1 of this Amendment to the SAR for the AP-101 cask. The fuel loadings for calculational purposes were based on maximum values for the various parameters to maximize the calculated reactivity.

Normal conditions of transport are for dry fuel so that reactivity for the AP-101 cask with fissile contents will be extremely low. Small (5%) reductions in cask or contents volume will be insignificant. Criticality calculations were performed for the accident condition with the cask filled and surrounded with cold water. The essential features of the AP-101 packages considered were exactly modeled in the reactivity calculations except for homogenization of the individual spent fuel assemblies. No cross-section adjustments or any modification of the calculational technique were used in order to reduce the calculated reactivity. Normal leakage boundry conditions were used for the cases of a single AP-101 cask package. Specular reflection was specified for all cask surfaces when the reactivity was calculated for an infinate array of AP-101 packages.

The requirements of 10 CFR 71.33 through 71.36 for single packages were fully considered for normal and accident conditions. Reactivity was calculated for maximum fuel loadings for the design fuel configurations flooded and reflected with cold water. Cold water maximizes the reactivity since the LWR fuel assemblies are slightly undermoderated. An infinite array of AP-101 packages were considered with the most reactive fuel loading (four PWR fuel assemblies) to demonstrate compliance with 10 CFR 71.37 and 71.38 requirements for Fissile Class I shipments. Subcriticality

was also demonstrated for the case of an increase in the fuel rod pitch to completely fill the available space within the spent fuel cannister. This calculation was performed to demonstrate the inherent safety of the AP-101 package design even though this Amendment to the SAR indicated that fuel assembly integrity would be maintained even under accident conditions.

6.4.3 Criticality Results

The AP-101 cask is intended for dry shipments of spent fuel which precludes any possibility of criticality during "normal" conditions of transport of the low enrichment LWR fuels specified as contents. The criticality evaluation therefore only considered reactivity for the accident condition with the cask flooded internally and externally with cold water. Cold water at maximum density would tend to maximize reactivity for the undermoderated fuel. Reactivity calculations determined k-infinity for the LWR fuel lattices considered along with k-effective for AP-101 packages containing LWR fuel assemblies. Calculated reactivities are presented in Table 6.5 along with the calculated reactivity for an infinite array af AP-101 casks (each containing 4 PWR F/A's) in order to show requirements for Fissile Class I packaging is achieved for the intended shipment of spent LWR fuel in the AP-101 cask.

KENO output is reproduced and presented as an appendix to this section.

6.4.3.1 K-infinity for Lattices

Table 6.5 presents the calculated reactivity for 7x7 and 6x6 BWR, and 15x15 PWR fuel lattices as calculated with the NULIF computer code. These results are presented for normal fuel rod pitch. The effect of variation of fuel rod pitch is illustrated in Figure 6.5. Increase in fuel rod spacing results in an increase in reactivity for the range of fuel rod pitch. Fuel rod spacing ranged from a low corresponding to the normal fuel rod pitch to a maximum corresponding to fuel rods uniformly filling the available space within a spent fuel cannister.

6.4.3.2 Reactivity for LWR Fuel Loadings in the AP-101 Cask

Calculated reactivity for AP-101 packages containing fissile material are presented in Table 6.5. The KENO computer results indicate that BWR fuel packages would result in lower reactivity than for PWR fuel in the AP-101 cask.

TABLE 6.5

CALCULATED REACTIVITY FOR LWR FUEL LATTICES
AND FUEL LOADINGS IN THE AP-101 CASK

I. K-infinity for Fuel Lattices (NULIF results)

Fuel Type	Enrichment %	K-infinity
7x7 BWR	3.0	1.3795
6x6 BWR	2.5	1.3164
15x15 PWR	3.4	1.4228

II. K-effective for AP-101 Packages (KENO results)

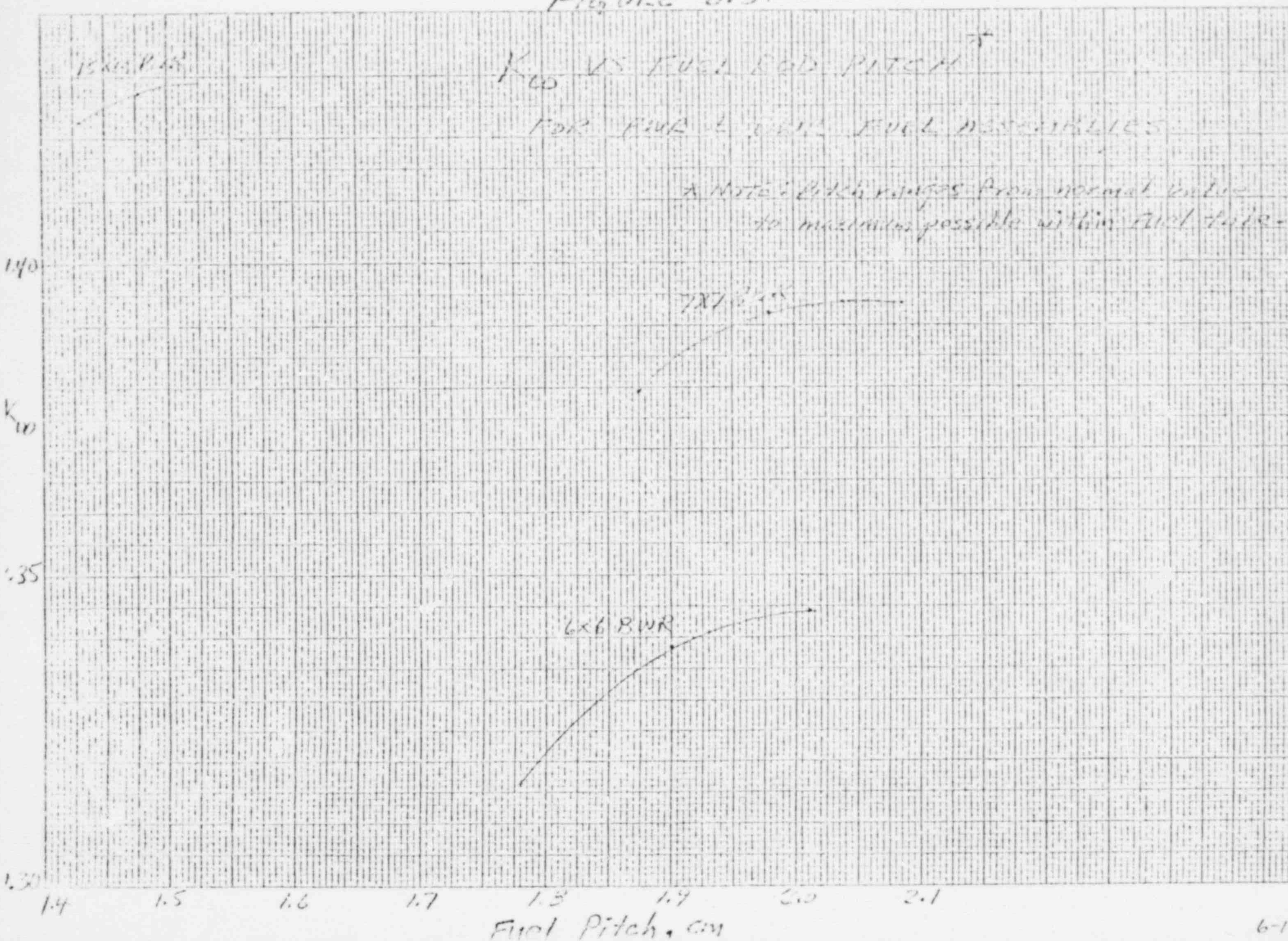
Fuel Type	Loading, F/A's	K-effective
7x7 BWR	10	0.71875 ± 0.00560*
6x6 BWR	15	0.77008 ± 0.01040*
15x15 PWR	4	0.89951 ± 0.00591*

III. K-infinity for Infinite Array of AP-101 Packages (KENO results)

Fuel Type	Loading, F/A's	K-infinity
15x15 PWR	4	0.89263 ± 0.00542*

* Standard Deviation

FIGURE 6.5


K∞ vs FUEL ROD PITCH*
FOR ... FUEL ASSEMBLIES
* Note: Pitch ranges from normal value to maximum possible within
7x7 BWR
6x6 BWR
1.40
K∞
1.35
1.30
1.4
1.5
1.6
1.7
1.8
1.9
2.0
2.1
Fuel Pitch, cm

Calculational results are presented for a single AP-101 package containing four (4) PWR assemblies, and an infinite array of similar AP-101 casks. The resulting reactivity for the two cases are statistically identical such that the reactivity of a water reflected single AP-101 package may be considered to be fully reflected. Therefore, multiple similar packages will not result in any increase in reactivity for the accident condition (flooded with water internally and externally) over that for a single AP-101 package containing fissile material.

6.4.3.3 Maximum Reactivity Calculated for AP-101 Package Containing Fissile Material

The maximum reactivity was calculated for four (4) PWR fuel assemblies in the AP-101 cask to be 0.89951 with a standard deviation of 0.00591. For a 95% confidence level this reactivity increases from the average calculated value of 0.89951 to a maximum value of 0.91133. Section 6.5.3 of this Amendment to the AP-101 SAR indicates that the calculational technique could introduce a bias of 1.0%, based on the benchmark calculation results. Including the calculational bias, a maximum reactivity of 0.921 (=0.911 + 0.010) is obtained for the accident condition of the AP-101 cask for shipment of LWR fuel assemblies.

Although this Amendment to the SAR indicates that integrity of fuel assemblies would be maintained under accident conditions, the inherent criticality safety of spent fuel shipments in the AP-101 cask is clearly demonstrated even if the effect of optimal fuel rod pitch within the fuel cannisters is imposed. From Figure 6.5, it is seen that k-infinity for the PWR fuel lattice increases from 1.423 for normal fuel rod spacing to a maximum value of 1.429 when the fuel rods uniformly fill the available space within the fuel cannister. Imposing a similar increase in reactivity, the previously stated maximum reactivity of 0.921 would be increased to 0.925 (=0.921 x 1.429/1.423). It is therefore the conclusion of this criticality evaluation that shipment of the specified LWR fuels in the LWR cask provide an adequate margin to preclude criticality under normal or accident conditions of transport.

6.5 CRITICAL BENCHMARK EXPERIMENT

A benchmark experiment was selected from the Report PNL-2438, Critical Separation Between Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons.(1)

The research program described in that report was funded by the U.S. Nuclear Regulatory Commission to provide "experimental criticality data on systems simulating conditions associated with fuel element shipping packages and fuel storage pools". The specific experiment selected as the benchmark for this criticality analysis is Experiment 017 (line 5 of Table V, page 19)-See attached. The benchmark experiment was selected since it is exactly defined in the reference report and is suitable for virtually exact three dimensional modeling with this criticality analysis computational technique.

The benchmark experiment shares many features in common with the loading configurations considered in this Safety Analysis Report. Boral plates are used for criticality control in the benchmark and the AP-101 loading configurations. The composition, thickness and proximity to the nearest fuel assembly in the benchmark experiment are all similar to the corresponding feature in the actual AP-101 packages considered herein. Additional features of the benchmark experiment which are similar to the AP-101 package conditions are similarities in fuel material, fuel radial dimensions, fuel rod pitch and water moderator at room temperature and atmospheric pressure.

Although the benchmark experiment is essentially a linear array of three super sized fuel clusters, the length and width of the critical assembly are not unlike that for the AP-101 loading configurations. However, the length of the benchmark fuel rods are only about 1/4 the length of comparable power reactor fuel rods. The three dimensional code used for the criticality analysis is rigorous in its treatment of fuel region length. Good agreement for the higher axial leakage condition of the benchmark experimental configuration should be indicative of even better agreement for the longer fuel region which characterize the actual AP-101 package configurations with spent fuel as contents.

The benchmark experiment is therefore seen to be quite similar in materials and geometry to the AP-101 package configurations of LWR fuel proposed as contents. The bias calculated by the computational technique of this criticality analysis should be indicative of the bias for the actual AP-101 loading configurations since the cross-section generating and criticality computer program are identical for the similar systems.

TABLE V

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 wt% ^{235}U ENRICHED UO_2 RODS IN WATER WITH BORAL PLATES BETWEEN FUEL CLUSTERS (1)

FUEL CLUSTERS		BORAL PLATES			
NUMBER IN ARRAY (2)	LENGTH x WIDTH 20.32mm SQ. PITCH (FUEL RODS)	THICKNESS (3) (tp, mm)	DISTANCE TO FUEL CLUSTER (4) (G, mm)	CRITICAL SEPARATION BETWEEN FUEL CLUSTERS (5) (Xc, mm)	EXPERIMENT NUMBER
3	20 x 17	7.13 ± 0.11	6.45 x 0.06	63.4 ± 0.2	020
3	20 x 17	7.13 ± 0.11	6.45 ± 0.06	63.2 ± 0.5	023 (6)
3	20 x 17	7.13 ± 0.11	44.42 ± 0.60	90.3 ± 0.5	016
3	22.21 ± 0.02 x 16 (7)	7.13 ± 0.11	6.45 ± 0.06	52.2 ± 0.3	017
3	22 x 16 (8)	7.13 ± 0.11	6.45 ± 0.06	50.5 ± 0.3	017
1	20 x 18.88 ± 0.10	7.13 ± 0.11	27.32 ± 0.50	∞	002
1	20 x 18.48 ± 0.05	7.13 ± 0.11	27.32 ± 0.50 (9)	∞	002

(1) ERROR LIMITS SHOWN ARE ONE STANDARD DEVIATION

(2) CLUSTERS OF FUEL RODS ALIGNED IN A SINGLE ROW

(3) INCLUDES 1.02 mm THICK CLADDING OF TYPE 1100 Al ON EITHER SIDE OF THE B_4C-Al CORE MATERIAL. PLATES 365mm WIDE BY 915 mm LONG.

(4) PERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARY OF THE CENTER FUEL CLUSTER AND THE NEAR SURFACE OF THE BORAL PLATE

(5) PERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARIES OF THE FUEL CLUSTERS

(6) RERUN OF EXPERIMENT 020

(7) CENTER FUEL CLUSTER AND SEPARATION BETWEEN CLUSTERS HELD CONSTANT AT 20 x 16 RODS AND 52.2 ± 0.3 mm RESPECTIVELY. TWO OUTER CLUSTERS LOADED TO CRITICAL AT 22.21 ± 0.02 x 16 RODS EACH

(8) CENTER FUEL CLUSTER AT 20 x 16 RODS. TWO OUTER FUEL CLUSTERS AT 22 x 16 RODS EACH

(9) BORAL PLATE ON ONE SIDE OF FUEL CLUSTER ONLY

6.5.2 Details of the Benchmark Calculations

6.5.2.1 Actual Benchmark Experiment

Table V of Reference 1 is attached to describe the benchmark experimental configuration. Also, attached from Reference 1 are Figure 5 (overall experimental arrangement), Table II (composition of boral plates), and Figure 4 (description of experimental fuel rods).

6.5.2.2 Input Data for Cross-Section Calculations

Cross-sections are obtained from the NULIF/NUTAN programs for the homogenized fuel cell, water, and boral plate. Fuel rod dimensions, and atom densities for the pellet, clad and moderator materials are presented in Table 6.6. Fuel radial dimensions and composition correspond exactly to the specifications for the experiment fuel rods. Water density presented in Table 6.6 corresponds to that of water at 68°F (density-0.99820 g/cm^3).

The boral region for calculational purposes is taken to be the boral plate plus an equal thickness of adjacent water. The volume percent of boral plate and of water in the boral region therefore are both 0.5.

6.5.2.3 KENO Model and Input

Figure 6.6 illustrates the plan view of the calculational model for the benchmark calculation. The array consists of nine regions in a linear array surrounded by a water moderator. The first and ninth region of the array are 22x16 rod fuel clusters. The second and eighth regions are identical water gaps, the third and seventh region are identical boral regions (consisting of boral plate with an equal thickness of water), the fourth and sixth region are identical thin water gaps, and the fifth region is the central 20x16 rod fuel cluster.

Acrylic plastic plates were used in the actual experiment for fuel rod spacing and support. The density and moderating property of acrylic plastic are virtually identical to water. The calculational model used water in the regions of the experiment occupied by acrylic plastic. Also, the aluminum structure in the experiment to support the grid structure was also taken to be water. These material changes in a region of low importance at the ends of the fuel rods would be insignificant.

TABLE II

COMPOSITION OF NEUTRON ABSORBER PLATES (1)

ELEMENT	BORAL (2.49 mg/mm³) wt%	COPPER-CADMIUM (8.910 mg/mm³) wt%	COPPER (8.913 mg/mm³) wt%	6061 ALUMINUM (2.692 mg/mm³) wt%	ZIRCALOY-4 (6.32 mg/mm³) wt%	304-L STEEL (7.930 mg/mm³) NO BORON wt%	304-L STEEL (7.900 mg/mm³) 1.1 wt% BORON wt%	304-L STEEL (7.770 mg/mm³) 1.6 wt% BORON wt%	CADMIUM (8.650 mg/mm³) wt%
Al	62.39 ± 2.8	-	-	97.15 ± 0.21	-	-	-	-	-
K B	28.70 ± 0.25	0.005	-	-	-	-	1.05 ± 0.08	1.62 ± 0.10	-
C	7.97 ± 0.41	0.002	0.340	-	-	-	-	-	-
Cd	-	0.989 ± 0.003	-	-	-	-	-	-	99.7 ± 0.3
Cr	0.05	-	-	0.21	0.13 ± 0.04	18.56 ± 0.10	19.03 ± 0.10	19.60 ± 0.10	-
Cu	0.09	98.685 ± 0.300	99.60 ± 0.14	0.12	-	0.27 ± 0.05	0.28 ± 0.05	0.26 ± 0.05	-
Fe	0.33 ± 0.04	0.020	0.004	0.82	0.21 ± 0.03	68.24 ± 0.34	68.04 ± 0.34	66.40 ± 0.33	-
Mg	0.05	-	0.002	-	-	-	-	-	-
Mn	0.05	0.009	-	0.21	-	1.58 ± 0.05	1.58 ± 0.05	1.69 ± 0.05	-
Mo						0.26 ± 0.05	0.49 ± 0.05	0.31 ± 0.05	
Na	0.02	-	0.002	-	-	-	-	-	-
Ni	0.02	0.010	-	-	-	11.09 ± 0.06	9.53 ± 0.05	10.12 ± 0.05	-
O	-	0.019	0.030	-	-	-	-	-	-
Si	0.20	0.004	0.020	0.82	-	-	-	-	-
Sn	-	0.250	-	-	1.50 ± 0.27	-	-	-	-
S	0.03	-	0.002	0.06	-	-	-	-	-
Ti	-	-	-	0.61	-	-	-	-	-
Zn	0.10	0.007	-	-	-	-	-	-	0.3
Zr	-	-	-	-	98.16 ± 0.35	-	-	-	-

(1) ERROR LIMITS WHERE SHOWN ARE ONE STANDARD DEVIATION BASED ON MULTIPLE CHEMICAL ANALYSES. ERROR LIMITS NOT SHOWN FOR MINOR IMPURITIES. IMPURITIES DISTRIBUTION BASED ON SPARK SOURCE MASS SPECTROGRAPHIC ANALYSES AND REPRESENT BEST ESTIMATE OF MAXIMUM CONCENTRATION FOR EACH ELEMENT PRESENT IN SIGNIFICANT QUANTITY

FIGURE 5

GRAPHICAL ARRANGEMENT OF SIMULATED SHIPPING PACKAGE CRITICAL EXPERIMENTS

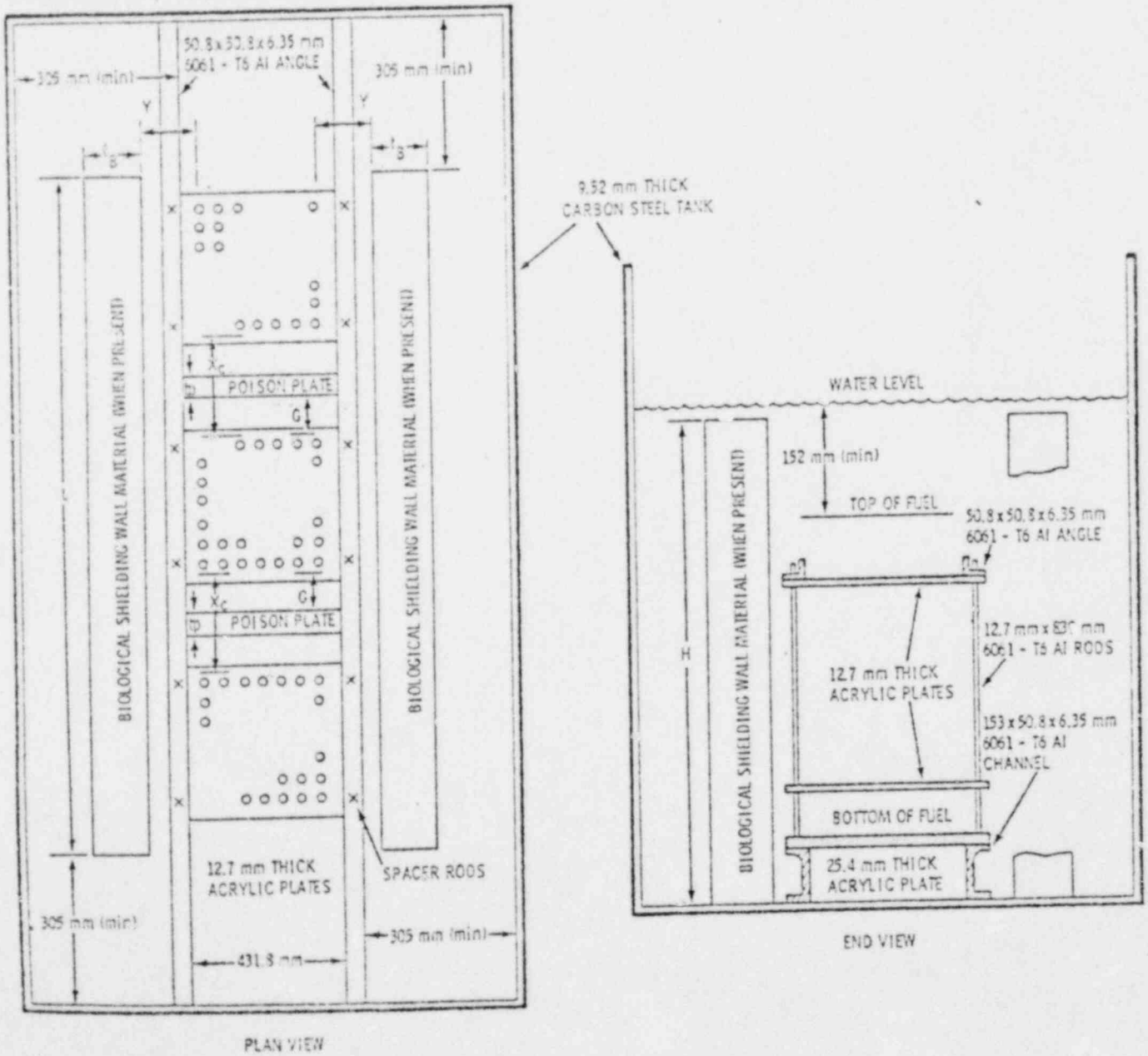
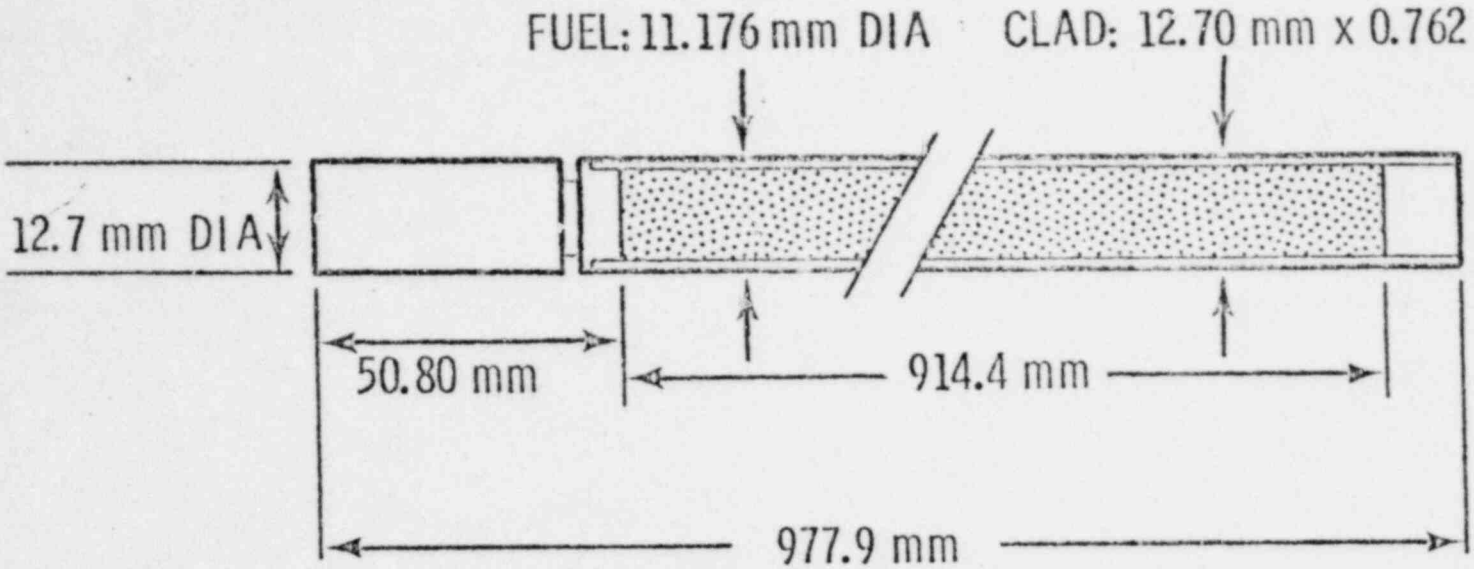


FIGURE 4

DESCRIPTION OF 2.35 wt% ^{235}U ENRICHED UO_2 RODS



CLADDING: 6061 ALUMINUM TUBING SEAL WELDED WITH A LOWER END PLUG OF 5052-H32 ALUMINUM AND A TOP PLUG OF 1100 ALUMINUM

TOTAL WEIGHT OF LOADED FUEL RODS: 917 gm (AVERAGE)

LOADING:

825 gm OF UO_2 POWDER/ROD, 726 gm OF U/ROD, 17.08 gm OF U-235/ROD
ENRICHMENT - 2.35 ± 0.05 w/o U-235
FUEL DENSITY - 9.20 mg/mm^3 (84% THEORETICAL DENSITY)

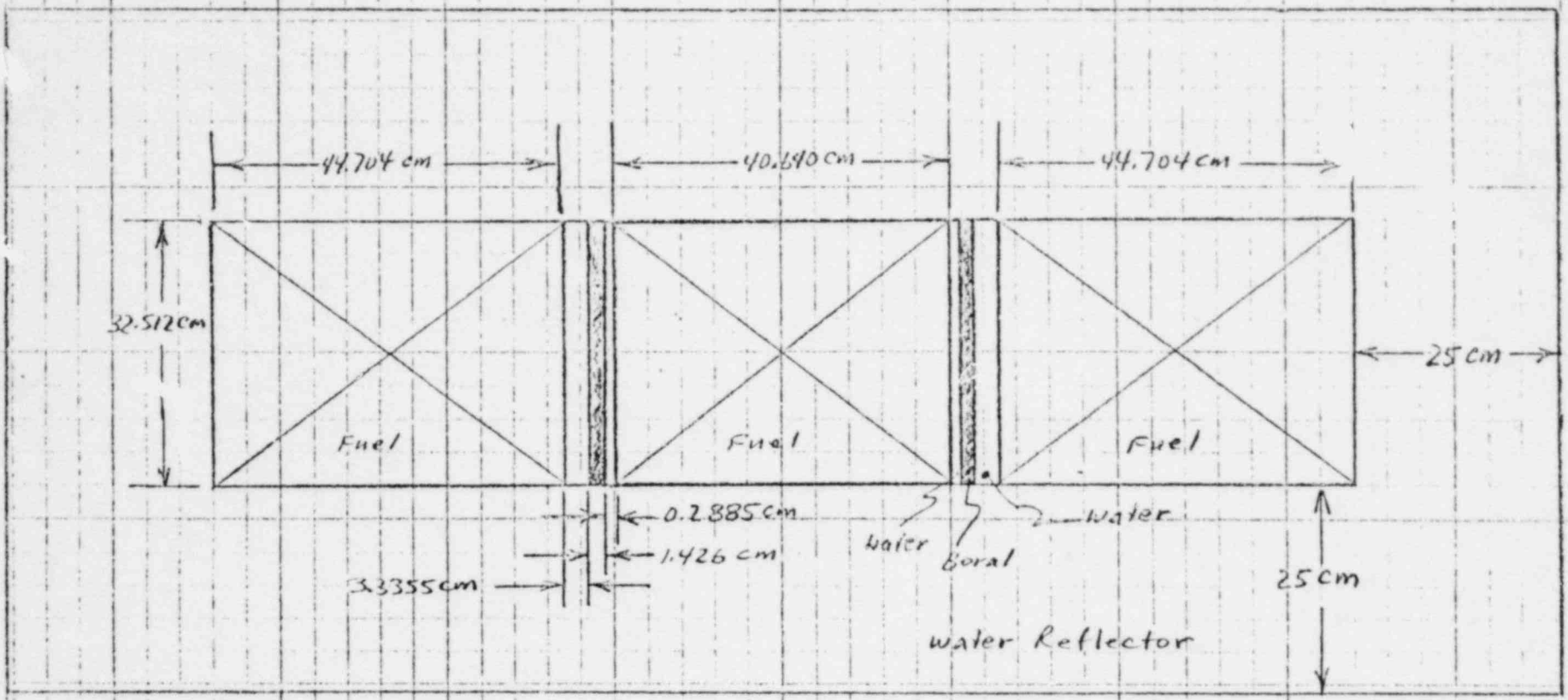


FIGURE 6.6 PLAN VIEW OF BENCHMARK CALCULATIONAL MODEL

Material cross-sections for the benchmark experiment fuel mix, water and boral region are presented as an Appendix to this section. The cross-sections are presented in ANISN format. KENO input and output for the benchmark calculation are also attached as an Appendix to this section. The input cross-section data has been deleted for compactness since that data appears separately in the Appendix.

6.5.3 Results of the Benchmark Calculations

As seen in the output reproduction of the reactivity summary for the KENO benchmark calculation, for 30,000 histories, an average k-effective of 1.00312 was calculated. For a 95 percent confindence interval the reactivity range is from 0.99557 to 1.01048.

The KENO calculation model for the benchmark experiment is a close representation of the actual experiment. The result of the KENO calculation is in close agreement with the experiment such that the total uncertainty in the calculation is about 1.0% in reactivity. This 1% uncertainty is comprised of about 0.31% above unity for the calculated average k-effective (1.00312) and about 0.74% for two standard deviations (2x.00368).

TABLE 6.6
INPUT DATA FOR CROSS-SECTION COLLAPSING
BENCHMARK CALCULATIONS

1. Fuel Cell Dimensions:

Pellet O.D., cm	1.1176
Cladding I.D., cm	1.1176
Cladding O.D., cm	1.2700
Rod Pitch, cm	2.0320

2. Concentrations, atoms/

	Pellet	Clad	Moderator
U-235	4.8807×10^{-4}		
U-238	2.0002×10^{-2}		
Oxygen	4.1560×10^{-2}		6.6743×10^{-2}
Hydrogen			3.3373×10^{-2}
Aluminum		6.0092×10^{-2}	

REFERENCES FOR SECTION 6

1. S.R. Bierman, E.D. Clayton, B.M. Durst, Critical Separation Between Subcritical Clusters of 2.35 Wt% 235U Enriched UO_2 Rods in Water With Fixed Neutron Poisons, PNL-2438, October 1977.

2. KENO 4, An Improved Monte Carlo Criticality Program (B&W Version of ORNL Code - KENO IV), NPGD-TM-503, April 1979, Babcock & Wilcox, Nuclear Power Generation Division.

3. L.M. Petrie, N.F. Cross, KENO IV, An Improved Monte Carlo Criticality Program, ORNL-4938, November 1975.

4. J.T. West, L.M. Petrie, S.K. Fraley, KENO-IV/CG, The Combinational Version of the KENO Monte Carlo Criticality Safety Program, NUREG/CR-0709, September 1979.

5. W.A. Wittkopf, NULIF - Neutron Spectrum Generator, Few-Group Constant Calculator, and Fuel Depletion Code, BAW-426, August 1976, Bobcock and Wilcox, Nuclear Power Generation Division.

6. W.A. Wittkopf, ANISN-DOT-KENO Library Generation from NULIP (NUTAN Program), LR-79221-02, December 14, 1979, Babcock and Wilcox, Power Generation Group.

Clark

BAW-426

August 1976

Fuel Utilization and Performance Analysis Code (FUPAC) System

NULIF — Neutron Spectrum Generator, Few-Group Constant Calculator, and Fuel Depletion Code

Babcock & Wilcox
Power Generation Group
Nuclear Power Generation Division
Lynchburg, Virginia

Report BAW-426

August 1976

Fuel Utilization and Performance Analysis Code (FUPAC) System

NULIF — Neutron Spectrum Generator, Few-Group Constant Calculator, and Fuel Depletion Code

W. A. Wittkopf, J. M. Tilford, J. B. Andrews, II, G. Kirschner, N. M. Hassan, P. N. Colpo

Key Words: Neutron Spectrum, Few-Group Constants, Fuel Depletion

ABSTRACT

The NULIF code generates a microgroup neutron spectrum and calculates spectrum-weighted few-group parameters for use in a spatial diffusion code. A wide variety of fuel cells, non-fuel cells, and fuel lattices, typical of PWR (or BWR) lattices, are treated. A fuel depletion routine and change card capability allow a broad range of problems to be studied. Coefficient variation with fuel burnup, fuel temperature change, moderator temperature change, soluble boron concentration change, burnable poison variation, and control rod insertion are readily obtained. Heterogeneous effects, including resonance shielding and thermal flux depressions, are treated. Coefficients are obtained for one thermal group and up to three epithermal groups. A special output routine writes the few-group coefficient data in specified format on an output tape for automated fitting in the PDQ07-HARMONY system of spatial diffusion-depletion codes.

1. INTRODUCTION

1.1. General Considerations

Reactor design engineers attempt to predict core power densities to within about 5% and core reactivity to within 0.5% or less throughout the core operating life. The most direct approach to achieving these accuracies suggests the use of high-order, three-dimensional transport theory using fine spatial and energy mesh structures, explicit representation of the heterogeneous fuel assemblies, and short-time depletion steps. We recognize that this approach must be rejected for practical reasons. Instead, we use the more practical three-dimensional, few-group diffusion theory with relatively coarse spatial mesh to describe the expected behavior of the reactor core. In addition to the diffusion approximation itself, the few-group constants that govern the diffusion equation must include, in some approximate manner, the effects of spatial homogenization and neutron cross-section weighting over relatively large (broad-group) ranges of the neutron energy spectrum.

A typical pressurized water reactor (PWR) core may consist of 200 or more fuel assemblies about 12 feet long arranged on a square lattice pitch of approximately 8.5 inches. Each fuel assembly itself may, for example, consist of a square lattice of 225 positions containing 208 fuel pins, one instrument tube, and 16 water holes, control rods, or lumped burnable poison (LBP) pins. Thus, even if diffusion theory provided an adequate model for neutron transport, spatial "homogenization" would still be required. This is because the spatial variation of neutron flux within each fuel cell (moderator, cladding, fuel) cannot be represented explicitly in a diffusion theory spatial problem. Consequently, a cell (or cluster of cells) must be represented as a "homogenized" region. The few-group constants corresponding to these regions must include the effects of flux depressions in the pins as well as transport and neutron spectrum effects.

The determination of few-group constants requires some model for determining a multigroup (fine energy group) neutron flux spectrum used to average the energy-dependent neutron cross sections. One such model (which is widely used) assumes that the spectrum for a given region depends only on the composition of that region. Actually, the spectrum for a given region also depends on the fixed and distributed neutron sources and the neutron leakage into (or out of) the region. Normally, these latter effects are smaller than the effects of composition. The source spectrum is often taken to be a uranium-235 fission spectrum, and leakage effects may be approximated by using a fundamental mode buckling for all energy groups. The spectrum is calculated for the "homogenized" region, and fine structure flux effects are usually applied by modification of the multigroup cross sections prior to the spectrum calculation.

When the core operates at power, the fuel is depleted and fission products build up. The composition at each point in the reactor is changing continuously, as are the few-group constants. The most obvious method to account for these changes is to recompute the neutron spectrum and few-group constants at each time step for each depletion block (or point). With a large number of depletion blocks (or points), current thinking is that this method would result in unacceptable computer running times (however, with the advent of fast, large-capacity computers the method may become practical for certain applications).

An alternate approach (which is used here) is to fit the few-group cross sections as a function of burnup and use these fits in the spatial diffusion calculation.

In summary, the process for obtaining few-group coefficients is as follows:

1. Define a region (cell or supercell cluster).
2. Determine fine structure effects (resonance self-shielding, flux depression, etc.) and obtain modified multigroup cross sections.
3. Compute the neutron flux energy spectrum and average the neutron cross sections to obtain few-group coefficients.
4. Deplete the region and repeat steps 1 through 3 to obtain the burnup dependence of the few-group cross sections.
5. Make any other desired changes (i.e., temperature, control rods, soluble poison, etc.) and repeat steps 1 through 3 (or 4) to obtain few-group constants as functions of these variables.

1.2. NULIF General Description

The primary objectives of the NULIF code package is to provide burnup-dependent, spectrum-weighted, few-group neutron cross sections for fuel cells, fuel clusters, and non-fuel cells. Using HAFIT[1], these cross sections are fit according to the HARMONY[2] specifications for use in the PDQ7[3] (B&W version of PDQ07[4]) multi-dimensional spatial diffusion code for core design.

1.2.1. Code Input

The basic input to NULIF includes the following:

1. A production library of multigroup neutron cross sections for all materials of interest. A master library of multigroup cross sections (consisting of 127 epithermal groups ranging from 15 MeV to 1.85 eV and 80 thermal groups ranging from 1.85 to zero eV) is generated by ETOGM[5] and THOR[6] from ENDF/B neutron data files maintained by the National Neutron Data Center at Brookhaven National Laboratories. Using the PROLIB[7] code, the master library data is collapsed to a fewer number of groups and materials and designated as a production library, which is used as direct input to NULIF. One of our present production libraries consists of data for 31 epithermal groups and 80 thermal groups for all materials of interest.

2. The other basic input to NULIF, provided by the user at run time, consists of the physical and geometric description of the fuel cluster, fuel cells, or other region for which the few-group constants are desired.

1.2.2. Code Output

The output from NULIF consists of the lattice volume fractions; the reactivity, conversion ratio, and few-group diffusion parameters for the region; the concentrations and few-group parameters for each isotope or material in the region; and the epithermal and thermal neutron flux spectra for the region. All these are given as functions of fuel burnup (depletion time) if requested.

1.2.3. Method of Solution

The neutron spectrum is computed by solving the multigroup P1 approximation to the space-independent neutron transport equation. Leakage effects are approximated by input of a fundamental mode buckling for the region. Self-shielding effects caused by the heterogeneous lattice configuration and the

different compositions of unit cells are taken into account in the spectrum calculation and are described later.

The neutron source for the spectrum calculation is a normalized fission source distribution. Theoretically, the slowing-down treatment for hydrogen moderator is exact, and the Greuling-Goertzel[8] approximation is used for other materials. For materials with neutron resonances in the epithermal range, the effective resonance integral for each resonance peak in each microgroup is computed by Dresner's[9] method using Sauer's[10] model to compute the Dancoff factor for closely packed fuel pin lattices. Both self-shielding and Doppler broadening are taken into account. The absorption rate in each isotope is given as the sum of a smooth absorption term and a resonance absorption term. The formulation for the resonance absorption term is given in terms of a resonance escape probability, following the method used in the MUFT[11] code. In computing the epithermal spectrum, both inelastic scattering and (n,2n) reactions are taken into account by the use of down-scatter matrices.

Below 1.85 eV both upscatter and downscatter may occur; this is designated as the "thermal" energy range. Because of the upscatter effects, the neutron spectrum in this range is solved by an iterative procedure. The initial flux guess for the iterative procedure is a Maxwellian distribution at the temperature of the moderator. Since upscattering above 1.85 eV is neglected, the sources for the thermal spectrum calculation are those neutrons scattered from above 1.85 eV into each microgroup below 1.85 eV. In the thermal region, the group structure is fine enough (80 groups from 0 to 1.85 eV) to mock up all the resonances explicitly by group cross section values. Of particular importance is the 1.05 eV resonance in plutonium-240, where the group structure is fine enough to give a good representation of this rapidly varying cross section.

Heterogeneous fuel cell local flux depression effects are obtained by the method of Amouyal, Benoist, and Horowitz[12] (ABH method), which, on a one-group basis, uses escape probabilities for the fuel pellet and diffusion theory for the water region of a fuel cell. In this manner, flux depression factors for each of the 80 thermal groups are applied to the cross sections of each cell region before computation of the thermal flux spectrum. The thermal scattering matrices (scattering kernels) are generated from the scattering law data of ENDF/B by integration over the scattering angle. For water moderated

lattices, only the scattering law (Haywood model) for H_2O is considered in determining the thermal spectrum. Energy exchange from scattering on heavy elements is neglected.

In some lattice studies, it is desirable to include the effects of water holes (vacant control rod positions), control rod cells, LBP cells, and instrument cells on the "homogenized" coefficients for the lattice. In this case, provision is made to use the "supercell" model, which has also been called the "subregion x" model. In this model, one cell type (usually the fuel cell) is designated as the primary cell. Other cells in the lattice are called X_1, X_2, X_3 depending on the number of different cell types (up to three are allowed). By previous detailed spatial calculations on the lattice, flux depression factors for each of the x-cells relative to the primary cell are obtained. By inputting these flux depression factors along with the descriptions of the subregion x cells, the flux depression factors are combined with those calculated for the primary cell to give overall flux depression factors for each material in the lattice. These depression factors are applied to each of the 80 groups before computation of the flux spectrum. In this way the final lattice spectrum is influenced by the number and type of each different cell in the lattice. For example, water holes interspersed within a lattice of fuel cells will soften the overall spectrum of the lattice when compared to an all-fuel cell lattice.

After the neutron spectra are obtained, few-group neutron cross sections are obtained by spectrum-weighting the microgroup cross sections over the appropriate energy range of the broad groups.

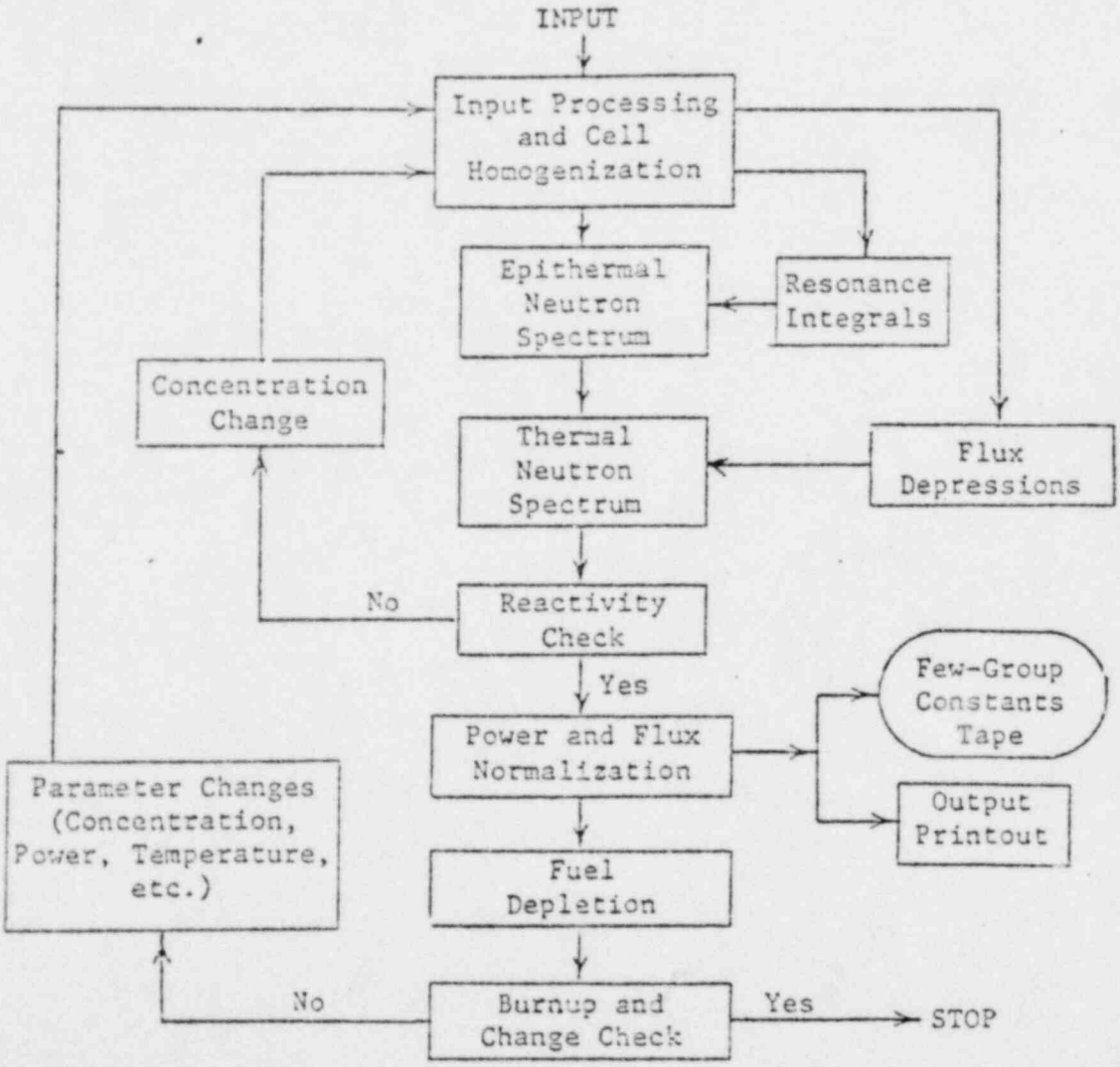
After obtaining the spectrum-averaged cross sections for each material in the lattice, the fuel may be depleted (burned) according to the power level specified in the input. The depletion equations are solved numerically by assuming constant flux over a small finite time interval. The flux is normalized to power after each time interval. The buildup and burnout of higher isotopes are based on a second-order approximation to the integral of the time-dependent differential depletion equations. After each depletion step, the neutron spectrum may be recalculated to take into account the buildup of fission products, the conversion of fertile material, and the burnout of fissionable isotopes. At each time step, the concentrations of each material and their few-group constants are stored on a special output tape (and printed if requested)

for later use in the HARMONY system of cross section fits used by the PDQ07 multidimensional, spatial diffusion code.

Special provisions in NULIF allow changes to be made at any time step during depletion. Application of this feature allows the user to change power level, soluble boron concentration, fuel temperature, moderator temperature, etc. The code will also iterate on a special material concentration to give a specified eigenvalue.

Figure 1-1 is a simple block diagram of the NULIF calculational flow.

Figure 1-1. Block Diagram of NULIF Calculations



HE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

AMERGO

DISTRIBUTION

From: W. A. WITTKOPF, TECHNICAL STAFF
M. M. HASSAN, COMPUTER SERVICES

BDS 663.5

Dist. ALL

File No. or Ref. LR-79221-02

Subj. ANISN-DOT-KENO LIBRARY GENERATION FROM NULIF

Date DECEMBER 14, 1979

This letter to cover one customer and one subject only.

DISTRIBUTION

Alcorn, F. M. (8)
Andrews, J. B.
Coleman, T. A.
Colpo, P. N.
Gudorf, M. R.
Hanson, G. E. (7)
Hassan, H. A.
Hassan, N. M.
Morgan, C. D.
Holman, P. L.
Hove, C. M.
Jones, H. M.
Kelly, S. M.
Kirschner, G.
Library
Livolsi, A. Z.
Mays, C. W.
Records Center (2A12/0221)
Rombough, C. T. (4)
Schwenk, G. A.
Shelburne, D. D. (10)
Sun, J. S.
Toops, E. C.
Uotinen, V. O.
Wilks, B. L.
Woods, J. J. (7)
Worsham, J. R.

SUMMARY

A new cross section preparation package called NUTAN is available for generating input libraries for the transport codes ANISN-DOT-KENO using output data from NULIF. Based on the NULIF multigroup structure, NUTAN generates a 111-group ANISN format library which includes resonance shielding and flux depression effects for heterogenous cells. In addition to generating a 111-group library for each material, NUTAN will collapse the cross sections to a preselected number of groups between 2 and 59 using the NULIF flux and current as weighting functions. Finally, NUTAN will compute and punch out in ANISN format, macroscopic cross sections for a mixture, based on concentrations from the NULIF run. This report describes the library generation and library collapse methods.

W. A. Wittkopf

N. M. Hassan

WAW,NMH:ae

1. INTRODUCTION

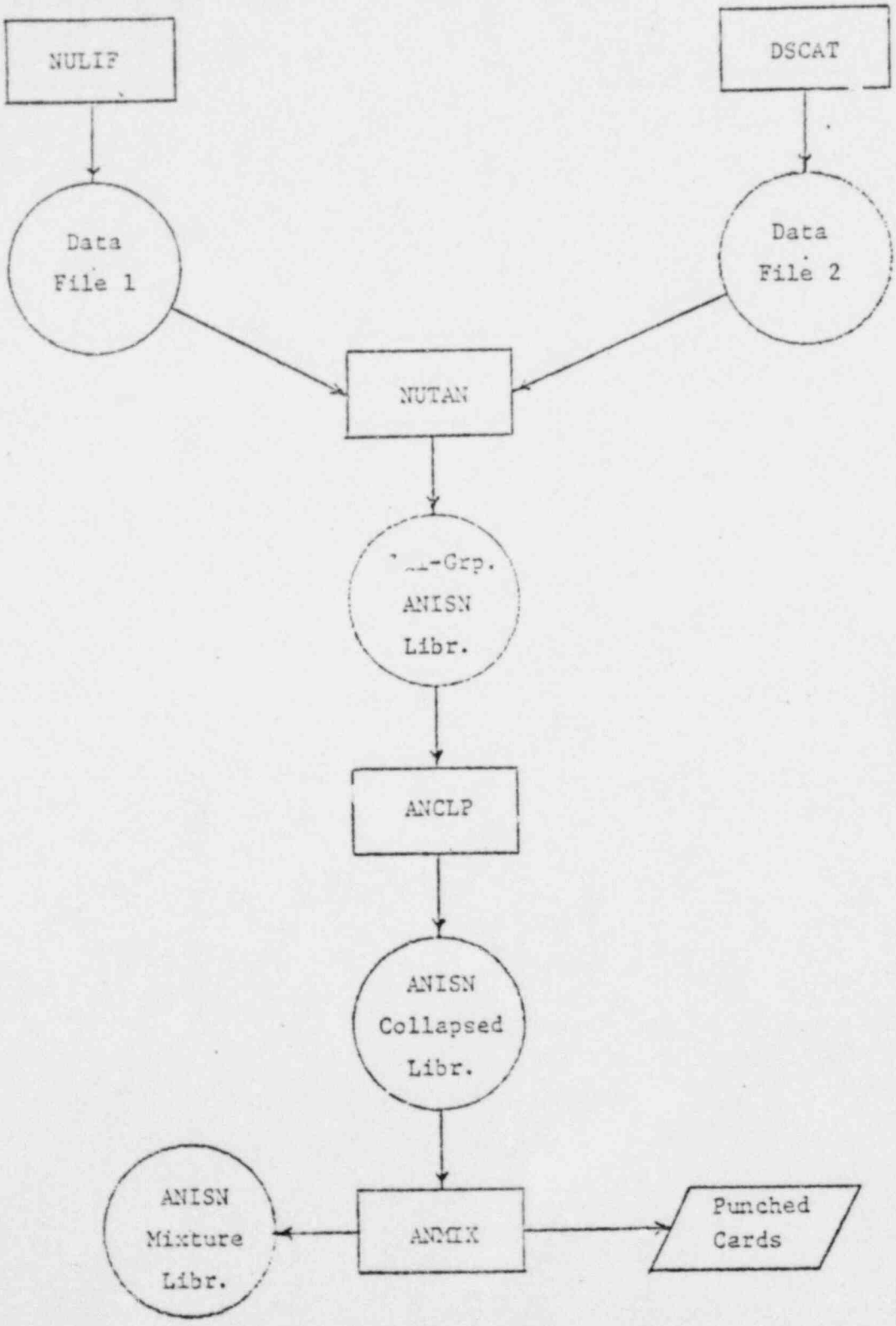
The completion of the NUTAN package is another milestone in the ongoing effort to update and improve nuclear design capability at the NPGD. When compared with previous methods of ANISN library generation at NPGD, completion of the NUTAN package is expected to:

1. Save approximately 3000 manhours per year based on estimated use of ANISN-DOT-KENO,
2. Provide job turnaround time savings up to a factor of ten for specific jobs and provide fast response to urgent jobs,
3. Provide increased quality assurance by automation of data transfer and elimination of large amounts of user input,
4. Provide for more consistent results due to the use of standard NULIF library basic data.

The first step in the library generation is the transfer of material concentrations, material multigroup cross sections and scattering matrixes, and multigroup neutron fluxes and currents from NULIF[1] calculations to a special output tape file. An independent subroutine called DSCAT generates a second file containing full elastic scattering downscatter matrixes (both P0 and P1) for each material. The NUTAN program processes these two files and generates a 111-group ANISN format library for each material. This library consists of 31 epithermal groups and 80 thermal groups with a 1.85 eV thermal cutoff. If no other input is present, the 111-group library may be output and saved on a data tape. When the group collapse option is selected, a subroutine called ANCLP collapses the 111-group library to the specified MG-group library where $2 \leq MG \leq 50$. The number of epithermal collapsed groups (NE) and the number of thermal collapsed groups (NT) must also be specified, and NE + NT = MG. Finally, if the mixture option is selected, a subroutine called ANMIX will compute macroscopic cross sections for the mixture using the concentrations from the NULIF data tape. These data may be punched on cards in the ANISN

format. Both the collapsed material cross sections and the collapsed mixture cross sections may be saved on output data tapes. A flow chart of the NUTAN code package is shown in Figure 1-1.

Figure 1-1. Flow Chart of NUTAN Processing



FOLLOWING IS A CARD IMAGE LISTING OF THE INPUT DATA

```
                                          C O L U M N   N U M B E R
CARD              11111111112222222222333333333344444444445555555555666666666677777777778
NUMBER   12345678901234567890123456789012345678901234567890123456789012345678901234567890

  1      CASE 1 BENCHMARK PNL-2438 PAGE 19 EXPERIMENT 17
  2       8 103 300 3 13 13 3 3  3  7 5 9 1 1 0 0 0 0000 00 4*0 5 0 10 9 15
  3      1.87+9 4.18+8 1.54+8 5.65+7 2.08+7 7.66+6  2.82+6 1.73+6 1.45+6 1.13+6 7.60+5
  4             5.42+5 2.20+5
  5      1 -1 1.0
  6      2 2 1.0
  7      3 3 1.0
  8        13   27    0  100 BENCHMARK FUEL MIX                                   1
  9       0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +24197- 7 0 +     0+ 0 0 +51200-18      1
 10       0 +24197- 7 0 +44495- 7 0 +33433- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      2
 11       0 +     0+ 0 0 +     0+ 0 0 +25385- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      3
 12       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      4
 13       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0      5
 14       0 +14059- 7 0 +     0+ 0 0 +64031-19 0 +14059- 7 0 +48556- 8 0 +71019- 5      6
 15       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +47488- 5      7
 16       0 +68799- 6 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      8
 17       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      9
 18       0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +32239- 7 0 +     0+ 0 0 +11701-18     10
 19       0 +32239- 7 0 +88239- 8 0 +10383- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     11
 20       0 +     0+ 0 0 +     0+ 0 0 +66450- 5 0 +20303- 5 0 +80227- 7 0 +     0+ 0     12
 21       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     13
 22       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0     14
 23       0 +50855- 7 0 +     0+ 0 0 +24213-18 0 +50855- 7 0 +18244- 7 0 +11299- 4     15
 24       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +71687- 5     16
 25       0 +32143- 5 0 +26699- 6 0 +10675- 7 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     17
 26       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     18
 27       0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +11710- 6 0 +     0+ 0 0 +72558-18     19
 28       0 +11710- 6 0 +54671- 7 0 +11325- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     20
 29       0 +     0+ 0 0 +     0+ 0 0 +70963- 5 0 +35382- 5 0 +42511- 6 0 +36133- 7     21
 30       0 +14447- 8 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     22
 31       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0     23
 32       0 +24758- 6 0 +     0+ 0 0 +16718-17 0 +24758- 6 0 +12597- 6 0 +11574- 4     24
 33       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +72044- 5     25
 34       0 +35680- 5 0 +46805- 6 0 +57533- 7 0 +48901- 8 0 +19552- 9 0 +     0+ 0     26
 35       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     27
 36       0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +24478- 6 0 +     0+ 0 0 +12362-17     28
 37       0 +24478- 6 0 +93143- 7 0 +11653- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     29
 38       0 +     0+ 0 0 +     0+ 0 0 +72680- 5 0 +35895- 5 0 +47226- 6 0 +63343- 7     30
 39       0 +77862- 8 0 +66180- 9 0 +26461-10 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     31
 40       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +10000- 4     32
 41       0 +     0+ 0 0 +66587- 7 0 +59963-18 0 +66587- 7 0 +45181- 7 0 +11323- 4     33
 42       0 +12140-26 0 +22902-25 0 +16584-22 0 +34317-14 0 +17220- 7 0 +33264- 5     34
 43       0 +13133- 5 0 +16160- 6 0 +21728- 7 0 +29143- 8 0 +35823- 9 0 +30448-10     35
 44       0 +12174-11 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     36
 45       0 +10000- 4 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +14444- 6 0 +24985-17     37
 46       0 +14444- 6 0 +18826- 6 0 +11625- 4 0 +     0+ 0 0 +26295-20 0 +49625-19     38
 47       0 +35667-16 0 +13973- 7 0 +35204- 5 0 +26395- 5 0 +83568- 6 0 +10963- 6     39
 48       0 +14745- 7 0 +19771- 8 0 +24383- 9 0 +20656-10 0 +82592-12 0 +     0+ 0     40
 49       0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +10000- 4     41
 50       0 +     0+ 0 0 +15405- 6 0 +26381-17 0 +15405- 6 0 +19877- 6 0 +12214- 4     42
 51       0 +     0+ 0 0 +     0+ 0 0 +38220-11 0 +79752-13 0 +90193- 7 0 +62849- 5     43
```

```
F O L L O W I N G   I S   A   C A R D   I M A G E   L I S T I N G   O F   T H E   I N P U T   D A T A

                                                  C O L U M N   N U M B E R
CARD                      11111111112222222222333333333344444444445555555555666666666677777777778
NUMBER           12345678901234567890123456789012345678901234567890123456789012345678901234567890

 358             0 +20680-15 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0       50
 359             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0       51
 360             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +69974- 6 0 +10373- 4        52
 361             1 +20563- 5 0 +37904- 6 0 +19648- 6 0 +11559- 6 0 +19533- 7 0 +11565- 8        53
 362             0 +66006-10 0 +35862-11 0 +16877-12 0 +50484-14 0 +53377-16 0 +     0+ 0       54
 363             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0       55
 364             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0       56
 365             0 +     0+ 0 0 +     0+ 0 0 +99636- 5 0 +64892- 6 0 +48876- 8 0 -13614- 7        57
 366             0 +19970- 7 0 +10946- 7 0 +63228- 8 0 +37435- 9 0 +21366-10 0 +11608-11        58
 367             0 +54630-13 0 +16342-14 0 +17278-16                                             59
 368             BOX TYPE  1
 369             CUBOID  1 44.704 0.0 32.512 0.0 45.72 -45.72 13*0.5
 370             BOX TYPE  2
 371             CUBOID  2 3.3355 0.0 32.512 0.0 45.72 -45.72 13*0.5
 372             BOX TYPE  3
 373             CUBOID  3 1.426  0.0 32.512 0.0 45.72 -45.72 13*0.5
 374             BOX TYPE  4
 375             CUBOID  2 0.2885 0.0 32.512 0.0 45.72 -45.72 13*0.5
 376             BOX TYPE  5
 377             CUBOID  1 40.64  0.0 32.512 0.0 45.72 -45.72 13*0.5
 378             CORE BDY  0 70.074 -70.074 16.256 -16.256 45.72 -45.72 13*0.5
 379             CUBOID  2 95.074 -95.074 41.256 -41.256 60.92 -68.64 13*0
 380             1 1 9 8 1 1 1 1 1 1 0  2 2 8 6 1 1 1 1 1 1 0
 381             3 3 7 4 1 1 1 1 1 1 0  4 4 6 2 1 1 1 1 1 1 0
 382             5 5 5 1 1 1 1 1 1 1 1
 383             END CASE
 384             END KENO

********************************** E N D   O F   I N P U T   L I S T I N G **********************************
```

CASE 1 BENCHMARK PHL-243R PAGE 19 EXPERIMENT 17

NUMBER OF GENERATIONS	103	START TYPE	0
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	13	LIST 1-D MIXTURE X SECTIONS	YES
MAX. NUMBER OF ENERGY TRANSFERS	13	LIST 2-D MIXTURE X-SECTIONS	YES
NUMBER OF INPUT NUCLIDES	3	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	3	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	3	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	7	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	5	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	9	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	0	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	YES
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	YES

ANISN TYPE CROSS SECTIONS WERE READ. POSITION OF SIGMA TOTAL WAS 9 , POSITION OF SIGMA GG WAS 15

VELOCITIES WILL BE READ FROM CARDS.

MAXIMUM TIME = 8.0000 MINUTES

STORAGE LOCATIONS REQUIRED FOR THIS JOB = 5512
REMAINING AVAILABLE LOCATIONS= 35963

REQUIRED LCM STORAGE LOCATIONS = 508 (000007748)
REMAINING LCM AVAILABLE 126468 (001ED7004B)

NUCLIDE 1 17 27 0 100 BENCHMARK FUEL MIX

GP.	ABSORPTION	(n,f) FISSION	TOTAL
1	2.4197E-01	4.4495E-03	3.3433E-01
2	1.4859E-01	4.8556E-04	7.1019E-01
3	3.2239E-03	9.8239E-04	1.0[illegible]E+00
4	5.3855E-01	1.3244E-03	1.1279E+00
5	1.1710E-02	5.4671E-03	1.1325E+00
6	2.4758E-02	1.2537E-02	1.1574E+00
7	2.4478E-02	9.3143E-03	1.1653E+00
8	5.6587E-03	4.5181E-03	1.1323E+00
9	1.4444E-02	1.0826E-02	1.1525E+00
10	1.5485E-02	1.9877E-02	1.2214E+00
11	3.1932E-02	4.5535E-02	1.4295E+00
12	3.9747E-02	5.5628E-02	1.6945E+00
13	7.9481E-02	1.1316E-01	2.5893E+00

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	2.5385E-01	6.8799E-02	8.8227E-03	1.8075E-03	1.4447E-04	1.9552E-05	2.6461E-06	1.2174E-07	8.2592E-08	1.2033E-07
2	4.7488E-01	2.9313E-01	2.6592E-02	1.6143E-03	4.8901E-04	6.6180E-05	3.0448E-06	2.0656E-06	3.0095E-06	1.0856E-06
3	6.6458E-01	3.2147E-01	4.2511E-02	5.7533E-03	7.7862E-04	3.5823E-05	2.4303E-05	3.5407E-05	1.2772E-05	5.8107E-06
4	7.1697E-01	4.5182E-01	4.6805E-02	6.3343E-03	2.9143E-04	1.9771E-04	2.8805E-04	1.0391E-04	4.7272E-05	3.1811E-05
5	7.0963E-01	3.5688E-01	4.7226E-02	2.1729E-03	1.4740E-03	2.1475E-03	7.7447E-04	3.5244E-04	2.3717E-04	0.
6	7.2344E-01	3.5845E-01	1.6158E-02	1.0963E-02	1.5972E-02	5.7614E-03	2.6212E-03	1.7639E-03	0.	0.
7	7.2680E-01	1.3143E-01	4.3569E-02	1.2175E-01	4.3919E-02	1.9981E-02	1.3446E-02	0.	0.	0.
8	3.3264E-01	2.6195E-01	3.2364E-01	1.1674E-01	5.3265E-02	3.5420E-02	0.	0.	0.	0.
9	3.5204E-01	5.8176E-01	1.6594E-01	7.6897E-02	5.0486E-02	1.7220E-03	0.	0.	0.	0.
10	6.2849E-01	3.4364E-01	1.4151E-01	9.0958E-02	1.3973E-03	3.4317E-10	0.	0.	0.	0.
11	7.4771E-01	4.1021E-01	2.3961E-01	9.0193E-03	3.5667E-12	1.6584E-18	0.	0.	0.	0.
12	9.8083E-01	6.4295E-01	3.0957E-02	7.9752E-06	4.9625E-15	2.2902E-21	0.	0.	0.	0.
13	2.4272E+00	8.1995E-02	1.2148E-03	3.8220E-07	2.6295E-16	1.2140E-22	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	4.3406E-08	1.9748E-08	1.3284E-18
2	4.9389E-07	3.3236E-07	0.
3	3.9192E-06	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE = 11 27 0 .94 PURE WATER

GP.	ABSORPTION	NU FISSION	TOTAL
1	6.9719E-04	0.	3.5405E-01
2	9.3727E-06	0.	6.7667E-01
3	3.9456E-05	0.	1.3342E+00
4	9.1829E-05	0.	1.4667E+00
5	2.1948E-04	0.	1.4837E+00
6	5.9488E-04	0.	1.5017E+00
7	1.5908E-03	0.	1.5283E+00
8	2.8328E-03	0.	1.5676E+00
9	3.4582E-03	0.	1.5332E+00
10	4.5634E-03	0.	1.6170E+00
11	6.8496E-03	0.	1.8893E+00
12	9.9115E-03	0.	2.2553E+00
13	2.0469E-02	0.	3.4001E+00

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	2.5641E-01	8.4838E-02	1.8643E-02	1.4403E-03	1.9493E-04	2.6380E-05	3.5702E-06	1.6426E-07	1.1143E-07	1.6235E-07
2	5.4156E-01	2.9842E-01	4.8635E-02	5.2287E-03	7.0763E-04	9.5768E-05	4.4061E-06	2.9891E-06	4.3549E-06	1.5709E-06
3	8.8144E-01	4.6155E-01	5.1525E-02	8.3265E-03	1.1269E-03	5.1845E-05	3.5172E-05	5.1243E-05	1.8485E-05	8.4096E-06
4	8.7986E-01	5.8846E-01	6.7777E-02	9.1726E-03	4.2201E-04	2.8630E-04	4.1711E-04	1.5046E-04	6.8453E-05	4.6065E-05
5	8.8971E-01	5.1474E-01	6.8633E-02	3.1577E-03	2.1422E-03	3.1210E-03	1.1258E-03	5.1220E-04	3.4468E-04	0.
6	9.2138E-01	5.2177E-01	2.3653E-02	1.6047E-02	2.3379E-02	8.4332E-03	3.8[illegible]7E-03	2.5819E-03	0.	0.
7	9.2382E-01	1.8630E-01	1.2988E-01	1.7611E-01	6.3528E-02	2.8902E-02	1.9449E-02	0.	0.	0.
8	3.1957E-01	3.6289E-01	4.6622E-01	1.6826E-01	7.6768E-02	5.1049E-02	0.	0.	0.	0.
9	4.8752E-01	7.0984E-01	2.3783E-01	1.0905E-01	7.2358E-02	2.2204E-03	0.	0.	0.	0.
10	7.4096E-01	4.8627E-01	2.0307E-01	1.3050E-01	1.8173E-03	4.8049E-10	0.	0.	0.	0.
11	9.2652E-01	6.3522E-01	3.4072E-01	9.9452E-03	4.1942E-12	7.9503E-18	0.	0.	0.	0.
12	1.2641E+00	9.4930E-01	3.2314E-02	8.8104E-06	5.5246E-15	2.5486E-21	0.	0.	0.	0.
13	3.2759E+00	1.0214E-01	1.5595E-03	4.9107E-07	3.3889E-16	1.5653E-22	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	5.8564E-08	2.6644E-08	1.7930E-08
2	7.1470E-07	4.8075E-07	0.
3	5.6591E-06	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE 3 13 27 0 299 NORAL-WATER MIX

GP.	ABSORPTION	NU FISSION	TOTAL
1	1.5758E-01	0.	2.9521E-01
2	7.5378E-03	0.	6.1312E-01
3	2.0505E-02	0.	8.3566E-01
4	5.5366E-02	0.	8.9623E-01
5	1.4621E-01	0.	9.7761E-01
6	3.9472E-01	0.	1.2623E+00
7	9.7319E-01	0.	1.8592E+00
8	1.9108E+00	0.	2.7782E+00
9	2.2716E+00	0.	3.1517E+00
10	2.9881E+00	0.	3.9937E+00
11	4.4239E+00	0.	5.4500E+00
12	6.2443E+00	0.	7.4394E+00
13	1.1116E+01	0.	1.2708E+01

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	2.3614E-01	5.0857E-02	5.7367E-03	7.7628E-04	1.0506E-04	1.4218E-05	1.9242E-06	8.8529E-08	6.0059E-08	8.7501E-08
2	4.3091E-01	1.5281E-01	1.9338E-02	2.6172E-03	3.5420E-04	4.7935E-05	2.2054E-06	1.4962E-06	2.1798E-06	7.8631E-07
3	5.4341E-01	2.3655E-01	3.8442E-02	4.1199E-03	5.5756E-04	2.5653E-05	1.7403E-05	2.5355E-05	9.1460E-06	4.1610E-06
4	5.5857E-01	2.5242E-01	1.2111E-02	4.4495E-03	2.0430E-04	1.3860E-04	2.0193E-04	7.2841E-05	3.3139E-05	2.2301E-05
5	5.5167E-01	2.4347E-01	4.1569E-02	1.4525E-03	9.8536E-04	1.4356E-03	5.1785E-04	2.3560E-04	1.5854E-04	0.
6	6.1782E-01	2.2666E-01	1.0939E-02	6.8103E-03	9.9220E-03	3.5791E-03	1.6283E-03	1.0958E-03	0.	0.
7	6.5522E-01	7.4398E-02	4.6235E-02	6.7336E-02	2.4290E-02	1.1651E-02	7.4364E-03	0.	0.	0.
8	1.1728E-01	1.7555E-01	2.2467E-01	8.2663E-02	3.7708E-02	2.5078E-02	0.	0.	0.	0.
9	3.2747E-01	3.4404E-01	1.1755E-01	5.3883E-02	3.5757E-02	1.3235E-03	0.	0.	0.	0.
10	5.4181E-01	2.1661E-01	9.3245E-02	6.0474E-02	1.4873E-03	3.9217E-10	0.	0.	0.	0.
11	6.2137E-01	2.5613E-01	1.4982E-01	9.0906E-03	3.7807E-12	1.7580E-18	0.	0.	0.	0.
12	7.8448E-01	3.7354E-01	3.7894E-02	9.4671E-06	5.8185E-15	2.6869E-21	0.	0.	0.	0.
13	1.4958E+00	9.4641E-02	1.2838E-03	4.0295E-07	2.7240E-16	1.2550E-22	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	3.1554E-08	1.4369E-08	9.6534E-09
2	3.5774E-07	2.4073E-07	0.
3	2.8901E-06	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

CASE 1 BENCHMARK PNL-2438 PAGE 19 EXPERIMENT 17

ARRAY DESCRIPTION

Z = 1

1 2 3 4 5 4 3 2 1

CASE 1 BENCHMARK PWR-2X40 PASS 14 EXPERIMENT 17

LIFETIME = 1.0E+99 + OR - 8.07533E-07 GENERATION TIME = 5.94014E-05 + OR - 3.83742E-07

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	1.00312	+ OR - .00369	.99944 TO 1.00680	.99577 TO 1.01048	.99209 TO 1.01416	30000
4	1.00274	+ OR - .00370	.99904 TO 1.00643	.99535 TO 1.01013	.99165 TO 1.01383	29700
5	1.00221	+ OR - .00370	.99852 TO 1.00591	.99482 TO 1.00960	.99112 TO 1.01330	29400
6	1.00223	+ OR - .00373	.99850 TO 1.00597	.99477 TO 1.00970	.99103 TO 1.01343	29100
7	1.00270	+ OR - .00374	.99896 TO 1.00644	.99521 TO 1.01018	.99147 TO 1.01393	28800
8	1.00294	+ OR - .00379	.99916 TO 1.00662	.99528 TO 1.01040	.99150 TO 1.01418	28500
9	1.00289	+ OR - .00382	.99907 TO 1.00671	.99525 TO 1.01053	.99143 TO 1.01435	28200
10	1.00317	+ OR - .00385	.99931 TO 1.00702	.99546 TO 1.01087	.99161 TO 1.01472	27900
11	1.00345	+ OR - .00388	.99957 TO 1.00733	.99569 TO 1.01122	.99180 TO 1.01510	27600
12	1.00415	+ OR - .00386	1.00031 TO 1.00802	.99645 TO 1.01188	.99259 TO 1.01574	27300
17	1.00440	+ OR - .00382	1.00058 TO 1.00822	.99676 TO 1.01204	.99294 TO 1.01586	25800
22	1.00268	+ OR - .00388	.99881 TO 1.00656	.99493 TO 1.01043	.99106 TO 1.01431	24300
27	1.00413	+ OR - .00389	1.00024 TO 1.00802	.99634 TO 1.01191	.99245 TO 1.01580	22800
32	1.00507	+ OR - .00408	1.00099 TO 1.00915	.99691 TO 1.01323	.99283 TO 1.01731	21300
37	1.00588	+ OR - .00431	1.00157 TO 1.01019	.99726 TO 1.01450	.99296 TO 1.01881	19800
42	1.00557	+ OR - .00464	1.00093 TO 1.01021	.99630 TO 1.01485	.99166 TO 1.01949	18300
47	1.00356	+ OR - .00470	.99886 TO 1.00826	.99416 TO 1.01296	.98946 TO 1.01766	16800
52	1.00389	+ OR - .00489	1.00900 TO 1.00878	.99511 TO 1.01467	.99021 TO 1.01956	15300
57	1.00111	+ OR - .00493	.99622 TO 1.00601	.99132 TO 1.01091	.98642 TO 1.01581	13800
62	.99998	+ OR - .00534	.99464 TO 1.00532	.98930 TO 1.01066	.98396 TO 1.01600	12300
67	.99703	+ OR - .00575	.99953 TO .99953	.98454 TO 1.00452	.97954 TO 1.00952	10800
72	.99569	+ OR - .00609	.99940 TO 1.00979	.98370 TO 1.00648	.97801 TO 1.01217	9300
77	.99571	+ OR - .00671	.99901 TO 1.00242	.98230 TO 1.00913	.97560 TO 1.01583	7800
82	.99634	+ OR - .00641	.99123 TO 1.00485	.98442 TO 1.01166	.97761 TO 1.01847	6300
87	.99434	+ OR - .00693	.99743 TO .99939	.98160 TO 1.00492	.97577 TO 1.01075	4800

=CASE 4 BWR 7X7 F/A

---COMPOSITION AND CELL DATA---

COMPOSITION IS A - HETEROGENEOUS SINGLE CELL

SOURCE SPECTRUM IS - U235 PROMPT + DELAYED

CELL TYPE IS A - FUEL CELL

PELLET O.D. = .12395E+01 (CM)
CLAD I.D. = .12649E+01 (CM)
CLAD O.D. = .14275E+01 (CM)
LATTICE PITCH = .18745E+01 (CM)
LATTICE GEOMETRY = SQUARE

PELLET TEMPERATURE = .68000E+02 (DEGREES F)
CLAD TEMPERATURE = .68000E+02 (DEGREES F)
COOLANT TEMPERATURE = .68000E+02 (DEGREES F)
DANCOFF FACTOR OPTION - USE SAUER S METHOD

PELLET	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION	RESONANCE REDUCTION	RESONANCE CALCULATION
1	O	.46407E-01	.10000E+01	.10000E+01	RESON
2	U235	.69609E-03	.10000E+01	.10000E+01	RESON
3	U238	.22907E-01	.10000E+01	.10000E+01	RESON

CLAD

	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION
1	ZY2	.43380E-01	.10000E+01

COOLANT

	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION
1	H1	.66743E-01	.10000E+01
2	O	.33373E-01	.10000E+01

VOLUME FRACTIONS

PELLET	-	.34341E+00
GAP	-	.14219E-01
CLAD	-	.97854E-01
COOLANT	-	.54452E+00
TOTAL	-	.10000E+01

TIME = 0.000 CASE NO. 1

G E N E R A L D A T A

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.13795E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.13795E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.11576E+01
AVERAGE ENERGY PER FISSION (MEV)	.20147E+03	NEUTRON AGE (SQ-CM)	.34000E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.37518E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.11123E+00
THERMAL NEUTRON LIFETIME (SEC)	.23049E-04	SLOWING-DOWN LIFETIME (SEC)	.55840E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

F E W G R O U P M A C R O S C O P I C C O N S T A N T S

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.11888E+01	.10326E-01	.30112E-02	.24638E-01	.77574E-02	.97635E-13	.15753E+12
2		.33088E+00	.94054E-01	.63564E-01		.15451E+00	.20506E-11	.41266E+11

TIME = 6.000 CASE NO. 1

GENERAL DATA

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.13920E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.13920E+01
INTEGRATED CONVERSION RATIO		THERMAL K-INFINITE	.12007E+01
AVERAGE ENERGY PER FISSION (MEV)	.20144E+03	NEUTRON AGE (SQ-CM)	.32611E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.36056E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.99417E-01
THERMAL NEUTRON LIFETIME (SEC)	.25976E-04	SLOWING-DOWN LIFETIME (SEC)	.54435E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

FEW GROUP MACROSCOPIC CONSTANTS

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.11998E+01	.93908E-02	.27281E-02	.27399E-01	.70378E-02	.88463E-13	.14821E+12
2		.30414E+00	.88273E-01	.58565E-01		.14231E+00	.18887E-11	.46003E+11

TIME = 6.000 CASE NO. 1

GENERAL DATA

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.13936E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.13936E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.12265E+01
AVERAGE ENERGY PER FISSION (MEV)	.20143E+03	NEUTRON AGE (SQ-CM)	.31556E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.34982E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.90562E-01
THERMAL NEUTRON LIFETIME (SEC)	.29086E-04	SLOWING-DOWN LIFETIME (SEC)	.53469E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

FEW GROUP MACROSCOPIC CONSTANTS

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.12091E+01	.85661E-02	.24793E-02	.29750E-01	.64033E-02	.80401E-13	.14203E+12
2		.28304E+00	.82600E-01	.53694E-01		.13048E+00	.17316E-11	.51154E+11

---COMPOSITION AND CELL DATA---

COMPOSITION IS A - HETEROGENEOUS SINGLE CELL

SOURCE SPECTRUM IS - U235 PROMPT + DELAYED

CELL TYPE IS A - FUEL CELL

PELLET O.D. = .12192E+01 (CM)

CLAD I.D. = .12522E+01 (CM)

CLAD O.D. = .14300E+01 (CM)

LATTICE PITCH = .17800E+01 (CM)

LATTICE GEOMETRY = SQUARE

PELLET TEMPERATURE = .68000E+02 (DEGREES F)

CLAD TEMPERATURE = .68000E+02 (DEGREES F)

COOLANT TEMPERATURE = .68000E+02 (DEGREES F)

DANCOFF FACTOR OPTION - USE SAUER S METHOD

PELLET	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION	RESONANCE REDUCTION	RESONANCE CALCULATION
1	O	.46828E-01	.10000E+01	.10000E+01	RESON
2	U235	.58536E-03	.10000E+01	.10000E+01	RESON
3	U238	.22829E-01	.10000E+01	.10000E+01	RESON

CLAD

	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION
1	ZY2	.43380E-01	.10000E+01

COOLANT

	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION
1	H1	.66743E-01	.10000E+01
2	O	.33373E-01	.10000E+01

VOLUME FRACTIONS

PELLET	-	.36847E+00
GAP	-	.20217E-01
CLAD	-	.11822E+00
COOLANT	-	.49310E+00
TOTAL	-	.10000E+01

TIME = 0.000 CASE NO. 1

GENERAL DATA

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.13164E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.13164E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.10887E+01
AVERAGE ENERGY PER FISSION (MEV)	.20149E+03	NEUTRON AGE (SQ-CM)	.36078E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.40072E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.11548E+00
THERMAL NEUTRON LIFETIME (SEC)	.23943E-04	SLOWING-DOWN LIFETIME (SEC)	.66380E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

FEW GROUP MACROSCOPIC CONSTANTS

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.11797E+01	.10572E-01	.28782E-02	.22127E-01	.74459E-02	.93370E-13	.17673E+12
2		.35496E+00	.83861E-01	.58832E-01		.14296E+00	.18974E-11	.44007E+11

TIME = 0.000 CASE NO. 1

GENERAL DATA

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.13385E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.13385E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.11478E+01
AVERAGE ENERGY PER FISSION (MEV)	.20146E+01	NEUTRON AGE (SQ-CM)	.34019E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.37840E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.10064E+00
THERMAL NEUTRON LIFETIME (SEC)	.27303E-04	SLOWING-DOWN LIFETIME (SEC)	.57828E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

FEW GROUP MACROSCOPIC CONSTANTS

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+31	.11934E+01	.95104E-02	.25805E-02	.25571E-01	.66892E-02	.83723E-13	.16178E+12
2		.31891E+00	.83472E-01	.54092E-01		.13144E+00	.17445E-11	.49559E+11

TIME = 0.000 CASE NO. 1

G E N E R A L D A T A

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.13442E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.13442E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.11814E+01
AVERAGE ENERGY PER FISSION (MEV)	.20144E+03	NEUTRON AGE (SQ-CM)	.32552E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.36.99E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.90224E-01
THERMAL NEUTRON LIFETIME (SEC)	.30885E-04	SLOWING-DOWN LIFETIME (SEC)	.56147E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

F E W G R O U P M A C R O S C O P I C C O N S T A N T S

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.12047E+01	.85827E-02	.23212E-02	.28426E-01	.60269E-02	.75322E-13	.15254E+12
2		.29202E+00	.77933E-01	.49328E-01		.11987E+00	.15909E-11	.55637E+11

---COMPOSITION AND CELL DATA---

COMPOSITION IS A - SUBREGION X (SUPERCELL)
SOURCE SPECTRUM IS - U235 PROMPT + DELAYED
CENTRAL CELL TYPE IS A - FUEL CELL

PELLET O.D. =	.9285 00 (CM)	PELLET TEMPERATURE =	.68000E+02 (DEGREES F)
CLAD I.D. =	.94996E+00 (CM)	CLAD TEMPERATURE =	.68000E+02 (DEGREES F)
CLAD O.D. =	.10719E+01 (CM)	COOLANT TEMPERATURE =	.68000E+02 (DEGREES F)
LATTICE PITCH =	.14300E+01 (CM)	DANCOFF FACTOR OPTION - USE SAUER S METHOD	
LATTICE GEOMETRY =	SQUARE	DISADVANTAGE FACTOR =	.10000E+01

PELLET	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION	RESONANCE REDUCTION	RESONANCE CALCULATION
1	)	.45525E-01	.10000E+01	.10000E+01	RESON
2	U235	.77392E-03	.10000E+01	.10000E+01	RESON
3	U238	.21989E-01	.10000E+01	.10000E+01	RESON

CLAD

	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION
1	ZYZ	.43380E-01	.10000E+01

COOLANT

	MATERIAL I.D.	CONCENTRATION	FLUX DEPRESSION
1	H1	.66743E-01	.10000E+01
2	O	.33373E-01	.10000E+01

SUBREGION X DATA

THERE ARE 2 DIFFERENT SUBREGION X CELLS

CELL NUMBER 1

DISADVANTAGE FACTOR = .10000E+01

NUMBER PER CENTRAL CELL = .49020E-02

CELL TYPE IS A - H2O CELL

CELL AREA = .20450E+01 (CM**2)

CELL TEMPERATURE = .68000E+02 (DEGREES F)

RESONANCE TEMPERATURE = .68000E+02 (DEGREES F)

	MATERIAL I.D.	CONCENTRATION	A0	A1	A2	RESONANCE REDUCTION	RESONANCE CALCULATION
1	H1	.50680E-01	.10000E+01	0.	0.	.10000E+01	RESON
2	O	.25340E-01	.10000E+01	0.	0.	.10000E+01	RESON
3	ZY2	.10630E-01	.10000E+01	0.	0.	.10000E+01	RESON

CELL NUMBER 2

DISADVANTAGE FACTOR = .10000E+01

NUMBER PER CENTRAL CELL = .98039E-01

CELL TYPE IS A - H2O CELL

CELL AREA = .20450E+01 (CM**2)

CELL TEMPERATURE = .68000E+02 (DEGREES F)

RESONANCE TEMPERATURE = .68000E+02 (DEGREES F)

	MATERIAL I.D.	CONCENTRATION	A0	A1	A2	RESONANCE REDUCTION	RESONANCE CALCULATION
1	H1	.61450E-01	.10000E+01	0.	0.	.10000E+01	RESON
2	O	.30703E-01	.10000E+01	0.	0.	.10000E+01	RESON
3	ZY2	.36300E-02	.10000E+01	0.	0.	.10000E+01	RESON

VOLUME FRACTIONS

PELLET	-	.30012E+00
GAP	-	.14121E-01
CLAD	-	.85837E-01
COOLANT	-	.50659E+00
TOTAL CENTRAL CELL	-	.90667E+00
SUBREGION 1	-	.44444E-02
SUBREGION 2	-	.88889E-01
TOTAL SUBREGION X	-	.93333E-01
TOTAL SUPERCELL	-	.10000E+01

TIME = 0.000 CASE NO. 1

G E N E R A L D A T A

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.14228E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.14228E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.12241E+01
AVERAGE ENERGY PER FISSION (MEV)	.20144E+03	NEUTRON AGE (SQ-CM)	.33074E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.36377E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.10651E+00
THERMAL NEUTRON LIFETIME (SEC)	.23133E-04	SLOWING-DOWN LIFETIME (SEC)	.52623E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

F E W G R O U P M A C R O S C O P I C C O N S T A N T S

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.12131E+01	.96584E-02	.28366E-02	.27020E-01	.72888E-02	.91943E-13	.14539E+12
2		.31630E+00	.95768E-01	.65486E-01		.15913E+00	.21119E-11	.41021E+11

TIME = 0.000 CASE NO. 1

G E N E R A L D A T A

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.14273E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.14273E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.12447E+01
AVERAGE ENERGY PER FISSION (MEV)	.20142E+03	NEUTRON AGE (SQ-CM)	.32356E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.35632E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.10026E+00
THERMAL NEUTRON LIFETIME (SEC)	.24745E-04	SLOWING-DOWN LIFETIME (SEC)	.52042E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

F E W G R O U P M A C R O S C O P I C C O N S T A N T S

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.10000E+01	.12183E+01	.91529E-02	.26740E-02	.28501E-01	.68754E-02	.86675E-13	.14110E+12
2		.30240E+00	.92277E-01	.62446E-01		.15174E+00	.20139E-11	.43502E+11

TIME = 0.000 CASE NO. 1

GENERAL DATA

POWER DENSITY (WATTS/CC)	.10000E+00	EFFECTIVE MULTIPLICATION	.14287E+01
FUEL BURNUP (MWD/T)	0.	K-INFINITE	.14287E+01
INTEGRATED CONVERSION RATIO	0.	THERMAL K-INFINITE	.12578E+01
AVERAGE ENERGY PER FISSION (MEV)	.20142E+03	NEUTRON AGE (SQ-CM)	.31845E+02
AVERAGE FUEL TEMPERATURE (DEG F)	.68000E+02	MIGRATION AREA (SQ CM)	.35111E+02
AVERAGE COOLANT TEMPERATURE (DEG F)	.68000E+02	AVERAGE THERMAL SPECTRUM ENERGY (EV)	.95830E-01
THERMAL NEUTRON LIFETIME (SEC)	.26110E-04	SLOWING-DOWN LIFETIME (SEC)	.51656E-05

NO RECOVERY CHECKPOINT REQUESTED
NO FIT FILE REQUESTED FOR THIS CASE

FEW GROUP MACROSCOPIC CONSTANTS

GROUP NO	SOURCE	DIFFUSION (CM)	SIG-A	SIG-F	SIG-R	NU SIG-F	KAPPA SIG-F	FLUX (N/CM2-SEC)
1	.13040E+01	.12222E+01	.87625E-02	.25498E-02	.29618E-01	.65593E-02	.82653E-13	.13825E+12
2		.29213E+00	.89449E-01	.59997E-01		.14579E+00	.19349E-11	.45776E+11

FOLLOWING IS A CARD IMAGE LISTING OF THE INPUT DATA

```
                                                      COLUMN   NUMBER
 CARD                 111111111122222222223333333333444444444455555555556666666666777777777778
NUMBER       12345678901234567890123456789012345678901234567890123456789012345678901234567890

   1         CASE 30 4 PWR F/A,S  FLOODED AP-101 CASK
   2         5 53 300 3 13 13 6 6 6 17 2 2 1 1 0 1 0 2000 00 4*0 5 0 10 9 15
   3         0.0 0.0 0.0 -1.0 0.0 0.0
   4         1.87+9 4.18+8 1.54+8 5.65+7 2.08+7 7.66+6 2.82+6 1.73+6 1.45+6 1.13+6
   5         7.60+5 5.42+5 2.20+5
   6         1 -1 1.0 2 2 1.0 3 3 1.0 4 4 1.0 5 5 1.0 5 6 1.0
   7            13   27     0 1140 PWR 15X15 FUEL 3.4W                          1
   8          0 +10000- 4 0 +10009- 4 0 +     0+ 0 0 +31943- 7 0 +     0+ 0 0 +71693-18      1
   9          0 +31943- 7 0 +62060- 7 0 +34282- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      2
  10          0 +     0+ 0 0 +     0+ 0 0 +26696- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      3
  11          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      4
  12          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0      5
  13          0 +21433- 7 0 +     0+ 0 0 +12979-18 0 +21433- 7 0 +98424- 8 0 +67414- 5      6
  14          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +47124- 5      7
  15          0 +64340- 6 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      8
  16          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0      9
  17          0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +49188- 7 0 +     0+ 0 0 +23714-18     10
  18          0 +49188- 7 0 +17883- 7 0 +96787- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     11
  19          0 +     0+ 0 0 +     0+ 0 0 +64531- 5 0 +17441- 5 0 +72185- 7 0 +     0+ 0     12
  20          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     13
  21          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0     14
  22          0 +80427- 7 0 +     0+ 0 0 +49091-18 0 +80427- 7 0 +36991- 7 0 +10488- 4     15
  23          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +69118- 5     16
  24          0 +27569- 5 0 +22781- 6 0 +95030- 8 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     17
  25          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     18
  26          0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +19242- 6 0 +     0+ 0 0 +14745-17     19
  27          0 +19242- 6 0 +11110- 6 0 +10476- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     20
  28          0 +     0+ 0 0 +     0+ 0 0 +67694- 5 0 +30343- 5 0 +36277- 6 0 +30831- 7     21
  29          0 +12861- 8 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     22
  30          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0     23
  31          0 +38974- 6 0 +     0+ 0 0 +33569-17 0 +38974- 6 0 +25293- 6 0 +10758- 4     24
  32          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +68672- 5     25
  33          0 +30508- 5 0 +39957- 6 0 +49096- 7 0 +41725- 8 0 +17405- 9 0 +     0+ 0     26
  34          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     27
  35          0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +37628- 6 0 +     0+ 0 0 +24979-17     28
  36          0 +37628- 6 0 +18822- 6 0 +10794- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     29
  37          0 +     0+ 0 0 +     0+ 0 0 +68649- 5 0 +30508- 5 0 +40205- 6 0 +54076- 7     30
  38          0 +66444- 8 0 +55469- 9 0 +23556-10 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     31
  39          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +10000- 4     32
  40          0 +     0+ 0 0 +96295- 7 0 +12169-17 0 +96295- 7 0 +91692- 7 0 +10340- 4     33
  41          0 +11121-26 0 +22033-25 0 +14891-22 0 +29958-14 0 +15769- 7 0 +33962- 5     34
  42          0 +11371- 5 0 +13679- 6 0 +18498- 7 0 +24879- 8 0 +30570- 9 0 +25980-10     35
  43          0 +10838-11 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     36
  44          0 +10000- 4 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +24866- 6 0 +50624-17     37
  45          0 +24866- 6 0 +38144- 6 0 +10701- 4 0 +     0+ 0 0 +24095-20 0 +47735-19     38
  46          0 +32028-16 0 +12720- 7 0 +35602- 5 0 +23215- 5 0 +71427- 6 0 +92801- 7     39
  47          0 +12549- 7 0 +16878- 8 0 +20739- 9 0 +17625-10 0 +73523-12 0 +     0+ 0     40
  48          0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +10000- 4     41
  49          0 +     0+ 0 0 +25442- 6 0 +53499-17 0 +25442- 6 0 +40310- 6 0 +11203- 4     42
  50          0 +     0+ 0 0 +     0+ 0 0 +35124-11 0 +76953-10 0 +84054- 7 0 +59808- 5     43
  51          0 +43686- 5 0 +27690- 5 0 +19406- 5 0 +13520- 6 0 +18283- 7 0 +24590- 8     44
```

FOLLOWING IS A CARD IMAGE LISTING OF THE INPUT DATA

```
                                                    COLUMN NUMBER
CARD          1111111111222222222233333333334444444444555555555566666666667777777778
NUMBER  1234567890123456789012345678901234567890123456789012345678901234567890123456789012345678901234567890

 715     0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0       51
 716     0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +10348- 8 0 +55764- 7       52
 717     0 +45442- 8 0 +82111- 9 0 +39875- 9 0 +22187- 9 0 +61236-10 0 +30866-11       53
 718     0 +15329-12 0 +75676-14 0 +32896-15 0 +98853-17 0 +40892-18 0 +    0+ 0       54
 719     0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0       55
 720     0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0 0 +    0+ 0       56
 721     0 +    0+ 0 0 +    0+ 0 0 +54889- 7 0 +15509- 8 0 +41413-10 0 -53812-10       57
 722     0 +40730-10 0 +21257-10 0 +19822-10 0 +99912-12 0 +49618-13 0 +24496-14       58
 723     0 +10648-15 0 +31998-17 0 +13236-18                                           59
 724    BOX TYPE  1
 725    CUBOID  1    22.9489    1.4986   22.9489    1.4986  190.5     -190.5     13*0.5
 726    CUBOID  2    23.65375    .79375  23.65375    .79375 223.6724  -200.66    13*0.0
 727    CUBOID  3    23.97125    .47625  23.97125    .47625 223.6724  -200.66    13*0.0
 728    CUBOID  4    24.4475     0.0     24.4475     .0     223.6724  -200.66    13*0.0
 729    CUBOID  3    24.765      0.0     24.765      .0     223.6724  -200.66    13*0.0
 730    BOX TYPE  2
 731    CUBOID  1    23.2664    1.8161   22.9489    1.4986  190.5     -190.5     13*0.5
 732    CUBOID  2    23.97125   1.11125  23.65375    .79375 223.6724  -200.66    13*0.0
 733    CUBOID  3    24.28875    .79375  23.97125    .47625 223.6724  -200.66    13*0.0
 734    CUBOID  4    24.765      .3175   24.4475     .0     223.6724  -200.66    13*0.0
 735    CUBOID  3    24.765     0.0      24.765      .0     223.6724  -200.66    13*0.0
 736    CORE BDY  0  24.765   -24.765    24.765      .0     223.6724  -200.66    13*0.0
 737    ZHEMICYL+Y  2 35.56    223.6724 -200.66      13*0.0
 738    ZHEMICYL+Y  5 37.1475  227.482  -205.899     13*0.0
 739    ZHEMICYL+Y  6 46.0375  231.292  -213.519     13*0.0
 740    ZHEMICYL+Y  3 46.0375  231.292  -223.679     13*0.0
 741    ZHEMICYL+Y  5 50.5206  243.992  -232.569     13*0.0
 742    CUBOID  2    70.5206  -70.5206   70.5206    0.0     264.0     -252.6     13*0.0
 743    2 9*1 0 1 2 2 1 6*1 1
 744    END CASE
 745    END KENO
```

************************************ E N D O F I N P U T L I S T I N G ************************************

CASE 30 4 PWR F/A.S FLOODED AP-101 CASK

NUMBER OF GENERATIONS	53	START TYPE	0
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	13	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	13	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	6	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	6	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	6	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	17	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	2	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	0	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	YES
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	YES

ANISN TYPE CROSS SECTIONS WERE READ, POSITION OF SIGMA TOTAL WAS 9 , POSITION OF SIGMA GG WAS 15

VELOCITIES WILL BE READ FROM CARDS.

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 0. -X = 0. +Y = 0. -Y = 1.00000E+00 +Z = 0. -Z = 0.

MAXIMUM TIME = 5.0000 MINUTES

STORAGE LOCATIONS REQUIRED FOR THIS JOB = 6238
REMAINING AVAILABLE LOCATIONS= 35234

REQUIRED LCM STORAGE LOCATIONS 508 (000007748)
REMAINING LCM AVAILABLE 126468 (003670048)

CASE 30 4 PWR F/A,S FLOODED AP-101 CASK

MIXTURE	NUCLIDE	DENSITY
1	-1	1.00000E+00
2	2	1.00000E+00
3	3	1.00000E+00
4	4	1.00000E+00
5	5	1.00000E+00
6	6	1.09000E+00

NUCLIDE 1 13 27 0 1140 PWR 15X15 FUEL 3.4W

GP.	ABSORPTION	NU FISSION	TOTAL
1	3.1943E-03	6.2060E-03	3.4282E-01
2	2.1433E-03	9.8424E-04	6.7414E-01
3	4.9188E-03	1.7883E-03	9.6787E-01
4	8.0427E-03	3.6991E-03	1.0488E+00
5	1.9242E-02	1.1110E-02	1.0476E+00
6	3.8974E-02	2.5293E-02	1.0759E+00
7	3.7628E-02	1.8822E-02	1.0794E+00
8	9.6295E-03	9.1692E-03	1.0340E+00
9	2.4866E-02	3.8144E-02	1.0701E+00
10	2.5442E-02	4.0310E-02	1.1203E+00
11	5.6409E-02	9.2429E-02	1.3177E+00
12	6.8312E-02	1.1227E-01	1.5457E+00
13	1.3735E-01	2.3053E-01	2.3317E+00

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	2.6696E-01	6.4340E-02	7.2185E-03	9.5030E-04	1.2861E-04	1.7405E-05	2.3556E-06	1.0838E-07	7.3523E-08	1.0712E-07
2	4.7124E-01	1.7441E-01	2.2781E-02	3.0831E-03	4.1725E-04	5.6469E-05	2.5980E-06	1.7625E-06	2.5679E-06	9.2629E-07
3	6.4531E-01	2.7569E-01	3.6277E-02	4.9096E-03	6.6444E-04	3.0570E-05	2.0739E-05	3.0215E-05	1.0899E-05	4.9586E-06
4	6.9118E-01	3.0343E-01	3.9957E-02	5.4076E-03	2.4879E-04	1.6878E-04	2.4590E-04	8.8703E-05	4.0356E-05	2.7157E-05
5	6.7694E-01	3.0508E-01	4.0205E-02	1.8498E-03	1.2549E-03	1.8283E-03	6.5951E-04	3.0005E-04	2.0191E-04	0.
6	6.8672E-01	3.0508E-01	1.3679E-02	9.2801E-03	1.3520E-02	4.8771E-03	2.2189E-03	1.4932E-03	0.	0.
7	6.8649E-01	1.1371E-01	7.1427E-02	1.0406E-01	3.7538E-02	1.7078E-02	1.1492E-02	0.	0.	0.
8	3.3962E-01	2.3215E-01	2.7690E-01	9.9853E-02	4.5558E-02	3.0295E-02	0.	0.	0.	0.
9	3.5602E-01	4.3686E-01	1.4227E-01	6.5234E-02	4.3285E-02	1.5769E-03	0.	0.	0.	0.
10	5.9808E-01	2.9655E-01	1.2111E-01	7.7865E-02	1.2729E-03	2.9958E-10	0.	0.	0.	0.
11	7.0382E-01	3.5226E-01	1.9679E-01	8.4054E-03	3.2028E-12	1.4891E-18	0.	0.	0.	0.
12	9.0748E-01	5.3906E-01	3.0805E-02	7.6953E-06	4.7735E-15	2.2033E-21	0.	0.	0.	0.
13	2.1159E+00	7.7322E-02	1.1172E-03	3.5124E-07	2.4095E-16	1.1121E-22	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	3.8640E-08	1.7579E-08	1.1830E-08
2	4.2142E-07	2.8359E-07	0.
3	3.3368E-06	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE 2 13 27 0 1040 PURE WATER

GP.	ABSORPTION	NU FISSION	TOTAL
1	4.9709E-04	0.	3.5485E-01
2	9.3727E-06	0.	8.7667E-01
3	3.0456E-05	0.	1.3342E+00
4	8.1829E-05	0.	1.4667E+00
5	2.1948E-04	0.	1.4837E+00
6	5.9888E-04	0.	1.5017E+00
7	1.5900E-03	0.	1.5203E+00
8	2.8328E-03	0.	1.5076E+00
9	3.3582E-03	0.	1.5332E+00
10	4.5634E-03	0.	1.6170E+00
11	6.8490E-03	0.	1.8893E+00
12	9.9115E-03	0.	2.2553E+00
13	2.0469E-02	0.	3.4001E+00

TRANSFER CROSS SECTIONS

FROM	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1										
1	2.5641E-01	8.4938E-02	1.0643E-02	1.4403E-03	1.9493E-04	2.6380E-05	3.5702E-06	1.6426E-07	1.1143E-07	1.6235E-07
2	5.4156E-01	2.9042E-01	3.8635E-02	5.2287E-03	7.0763E-04	9.5768E-05	4.4061E-06	2.9891E-06	4.3549E-06	1.5709E-06
3	8.0149E-01	4.6155E-01	6.1525E-02	8.3265E-03	1.1269E-03	5.1845E-05	3.5172E-05	5.1243E-05	1.8485E-05	8.4096E-06
4	8.7980E-01	5.0846E-01	6.7777E-02	9.1726E-03	4.2201E-04	2.8630E-04	4.1711E-04	1.5046E-04	6.8453E-05	4.6065E-05
5	8.8971E-01	5.1478E-01	6.8633E-02	3.1577E-03	2.1422E-03	3.1210E-03	1.1258E-03	5.1220E-04	3.4468E-04	0.
6	9.0138E-01	5.2177E-01	2.3653E-02	1.6047E-02	2.3379E-02	8.4332E-03	3.8367E-03	2.5819E-03	0.	0.
7	9.2382E-01	1.8600E-01	1.2088E-01	1.7611E-01	6.3528E-02	2.8902E-02	1.9449E-02	0.	0.	0.
8	3.7957E-01	3.6289E-01	4.6622E-01	1.6826E-01	7.6768E-02	5.1049E-02	0.	0.	0.	0.
9	4.9752E-01	7.0088E-01	2.3783E-01	1.0905E-01	7.2358E-02	2.2204E-03	0.	0.	0.	0.
10	7.9096E-01	4.8507E-01	2.0307E-01	1.3050E-01	1.8173E-03	4.8049E-10	0.	0.	0.	0.
11	9.2652E-01	6.0522E-01	3.4072E-01	9.9452E-03	4.1942E-12	1.9500E-18	0.	0.	0.	0.
12	1.2641E+00	9.4900E-01	3.2314E-02	8.8104E-06	5.5246E-15	2.5486E-21	0.	0.	0.	0.
13	3.2759E+00	1.0214E-01	1.5595E-03	4.9107E-07	3.3889E-16	1.5653E-22	0.	0.	0.	0.

FROM	TO I= 10	I= 11	I= 12
1			
1	5.8564E-08	2.6644E-08	1.7930E-08
2	7.1470E-07	4.8095E-07	0.
3	5.6591E-06	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE 3 13 27 0 1119 ALUM. REGION

GP.	ABSORPTION	NU FISSION	TOTAL
1	1.1545E-04	0.	2.0373E-01
2	1.9380E-04	0.	2.2232E-01
3	3.9693E-04	0.	8.6390E-02
4	6.6093E-04	0.	8.6416E-02
5	6.6154E-04	0.	8.6274E-02
6	6.6173E-04	0.	8.6291E-02
7	9.8742E-04	0.	8.6634E-02
8	1.7001E-03	0.	8.7929E-02
9	2.1100E-03	0.	8.8369E-02
10	2.7840E-03	0.	8.9112E-02
11	4.1185E-03	0.	9.0692E-02
12	5.8030E-03	0.	9.2693E-02
13	9.8831E-03	0.	9.7504E-02

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	1.9435E-01	9.2317E-03	2.2364E-05	2.6572E-06	3.5961E-07	4.8668E-08	6.5865E-09	3.0303E-10	2.0558E-10	2.9952E-10
2	2.1403E-01	8.0392E-03	5.2815E-05	7.1477E-06	9.6734E-07	1.3092E-07	6.8232E-09	4.0862E-09	5.9532E-09	2.1475E-09
3	8.2629E-02	3.2887E-03	6.4483E-05	8.7160E-06	1.1796E-06	5.4271E-08	3.6818E-08	5.3640E-08	1.9349E-08	8.8030E-09
4	8.2319E-02	3.3693E-03	6.5112E-05	8.9473E-06	4.1165E-07	2.7927E-07	4.0687E-07	1.4677E-07	6.6772E-08	4.4934E-08
5	8.2162E-02	3.3732E-03	6.7009E-05	3.0830E-06	2.0915E-06	3.0472E-06	1.0992E-06	5.0007E-07	3.3652E-07	0.
6	8.2171E-02	3.3824E-03	2.3084E-05	1.5560E-05	2.2816E-05	8.2302E-06	3.7443E-06	2.5197E-06	0.	0.
7	8.2361E-02	2.8966E-03	1.1505E-04	1.6762E-04	6.0465E-05	2.7509E-05	1.8512E-05	0.	0.	0.
8	8.5240E-02	3.0156E-04	4.2019E-04	1.5149E-04	6.9062E-05	4.5942E-05	0.	0.	0.	0.
9	8.5152E-02	6.8957E-04	2.3540E-04	1.0791E-04	7.1609E-05	2.6152E-06	0.	0.	0.	0.
10	8.5573E-02	4.4096E-04	1.8896E-04	1.2239E-04	2.7601E-06	7.2813E-13	0.	0.	0.	0.
11	8.5737E-02	5.1695E-04	3.0188E-04	1.7534E-05	7.3011E-15	3.3949E-21	0.	0.	0.	0.
12	8.6067E-02	7.4967E-04	7.3375E-05	1.8742E-08	1.1522E-17	5.3207E-24	0.	0.	0.	0.
13	8.7421E-02	1.9735E-04	2.6633E-06	8.3582E-10	5.6429E-19	2.5995E-25	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	1.0804E-10	4.9154E-11	3.3078E-11
2	9.7700E-10	6.5746E-10	0.
3	5.9239E-09	.	0.
4	0.	3.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE 4 13 27 0 1070 BORAL REGION

GP.	ABSORPTION	NU FISSION	TOTAL
1	1.3600E-03	0.	2.6522E-01
2	5.9565E-03	0.	4.9797E-01
3	1.7302E-02	0.	6.2621E-01
4	4.6410E-02	0.	6.4638E-01
5	1.2192E-01	0.	7.1169E-01
6	3.2025E-01	0.	9.4810E-01
7	8.1423E-01	0.	1.4477E+00
8	1.5999E+00	0.	2.2189E+00
9	1.9041E+00	0.	2.5316E+00
10	2.4950E+00	0.	3.1447E+00
11	3.7065E+00	0.	4.4379E+00
12	5.2131E+00	0.	6.0705E+00
13	9.1518E+00	0.	1.0455E+01

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	2.2102E-01	3.8149E-02	4.0569E-03	5.4888E-04	7.4283E-05	1.0053E-05	1.3605E-06	6.2596E-08	4.2466E-08	6.1869E-08
2	3.7220E-01	1.0492E-01	1.2369E-02	1.7551E-03	2.3753E-04	3.2147E-05	1.4790E-06	1.0034E-06	1.4618E-06	5.2732E-07
3	4.2556E-01	1.5989E-01	2.0294E-02	2.7465E-03	3.7170E-04	1.7101E-05	1.1602E-05	1.6903E-05	6.0973E-06	2.7740E-06
4	4.0613E-01	1.6574E-01	2.1711E-02	2.9382E-03	1.3518E-04	9.1709E-05	1.3361E-04	4.8197E-05	2.1928E-05	1.4756E-05
5	4.0479E-01	1.6114E-01	2.0699E-02	9.5233E-04	6.4607E-04	9.4127E-04	3.3954E-04	1.5447E-04	1.0395E-04	0.
6	4.5768E-01	1.4864E-01	6.5312E-03	4.4308E-03	6.4553E-03	2.3286E-03	1.0594E-03	7.1291E-04	0.	0.
7	4.8247E-01	4.9252E-02	3.0085E-02	4.3811E-02	1.5804E-02	7.1899E-03	4.8384E-03	0.	0.	0.
8	2.5361E-01	1.1644E-01	1.5230E-01	5.4951E-02	2.5066E-02	1.6679E-02	0.	0.	0.	0.
9	2.5946E-01	2.2913E-01	7.8311E-02	3.5897E-02	2.3821E-02	8.8991E-04	0.	0.	0.	0.
10	4.8274E-01	1.4382E-01	6.1973E-02	4.0203E-02	1.0081E-03	2.6577E-10	0.	0.	0.	0.
11	4.5522E-01	1.7037E-01	9.9689E-02	6.1102E-03	2.5405E-12	1.1813E-18	0.	0.	0.	0.
12	5.6391E-01	2.4867E-01	2.4820E-02	6.3321E-06	3.8914E-15	1.7970E-21	0.	0.	0.	0.
13	1.0397E+00	6.2945E-02	8.5400E-04	2.6804E-07	1.8121E-16	8.3491E-23	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	2.2318E-08	1.0153E-08	6.8326E-09
2	2.3991E-07	1.6144E-07	0.
3	1.8667E-06	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE S 13 21 0 1090 STEEL REGION

GP.	ABSORPTION	NU FISSION	TOTAL
1	6.4378E-04	0.	2.7086E-01
2	9.8742E-04	0.	3.6510E-01
3	2.3247E-03	0.	8.0992E-01
4	7.0703E-03	0.	8.7356E-01
5	6.8192E-03	0.	1.0102E+00
6	7.1971E-03	0.	9.1483E-01
7	1.8398E-02	0.	9.2536E-01
8	3.1759E-02	0.	9.3695E-01
9	4.0226E-02	0.	9.4545E-01
10	5.2486E-02	0.	9.5777E-01
11	7.8173E-02	0.	9.8371E-01
12	1.1039E-01	0.	1.0162E+00
13	1.9388E-01	0.	1.1005E+00

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	2.5927E-01	1.0910E-02	2.7236E-05	3.6860E-06	4.9884E-07	6.7511E-08	9.1366E-09	4.2036E-10	2.8517E-10	4.1548E-10
2	3.5778E-01	6.2917E-03	4.0525E-05	5.4844E-06	7.4224E-07	1.0045E-07	4.6216E-09	3.1353E-09	4.5679E-09	1.6477E-09
3	7.9378E-01	1.3742E-02	5.6964E-05	7.7092E-06	1.0433E-06	4.8002E-08	3.2565E-08	4.7444E-08	1.7114E-08	7.7862E-09
4	8.5478E-01	1.1688E-02	6.5453E-05	8.8581E-06	4.0755E-07	2.7648E-07	4.0281E-07	1.4530E-07	6.6107E-08	4.4486E-08
5	9.8722E-01	1.6123E-02	6.6369E-05	3.0535E-06	2.0715E-06	3.0181E-06	1.0887E-06	4.9530E-07	3.3331E-07	0.
6	8.9388E-01	1.3769E-02	2.2085E-05	1.498_E-05	2.1828E-05	7.8739E-06	3.5823E-06	2.4106E-06	0.	0.
7	8.9676E-01	9.8631E-03	1.0111E-04	1.4731E-04	5.3138E-05	2.4175E-05	1.6268E-05	0.	0.	0.
8	9.0426E-01	2.8125E-04	4.0109E-04	1.4452E-04	6.5884E-05	4.3805E-05	0.	0.	0.	0.
9	9.0412E-01	6.8433E-04	2.3427E-04	1.0738E-04	7.1261E-05	2.8073E-06	0.	0.	0.	0.
10	9.0456E-01	4.2474E-04	1.8377E-04	1.1935E-04	3.2299E-06	8.5118E-13	0.	0.	0.	0.
11	9.0471E-01	5.0616E-04	2.9671E-04	1.5_11E-05	7.8952E-15	3.6712E-21	0.	0.	0.	0.
12	9.0504E-01	7.3925E-04	7.6327E-05	1.9438E-08	1.1940E-17	5.5138E-24	0.	0.	0.	0.
13	9.0642E-01	1.9556E-04	2.6373E-06	8.2760E-10	5.5896E-19	2.5751E-25	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	1.4987E-10	6.8185E-11	4.5884E-11
2	7.4965E-10	5.0447E-10	0.
3	5.2396E-09	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOVE

NUCLIDE 6 13 27 0 1100 LEAD REGION

GP.	ABSORPTION	NU FISSION	TOTAL
1	1.2309E-04	0.	1.8367E-01
2	1.3380E-04	0.	3.2249E-01
3	1.0879E-04	0.	3.5148E-01
4	1.0524E-04	0.	3.6911E-01
5	5.7966E-05	0.	3.6608E-01
6	1.5692E-04	0.	3.6612E-01
7	4.2056E-04	0.	3.6639E-01
8	7.2505E-04	0.	3.6668E-01
9	8.7731E-04	0.	3.6686E-01
10	1.1676E-03	0.	3.6722E-01
11	1.7218E-03	0.	3.6812E-01
12	2.4283E-03	0.	3.6912E-01
13	4.2168E-03	0.	3.7143E-01

TRANSFER CROSS SECTIONS

FROM I	TO I= 0	I= 1	I= 2	I= 3	I= 4	I= 5	I= 6	I= 7	I= 8	I= 9
1	1.8878E-01	2.7038E-03	5.6452E-05	3.7555E-06	5.0825E-07	6.8785E-08	9.3090E-09	4.2829E-10	2.9056E-10	4.2332E-10
2	3.2044E-01	1.8695E-03	4.0933E-05	5.5397E-06	7.4972E-07	1.0146E-07	4.6682E-09	3.1669E-09	4.6139E-09	1.6644E-09
3	3.4919E-01	2.1049E-03	6.1088E-05	8.2674E-06	1.1189E-06	5.1477E-08	3.4923E-08	5.0879E-08	1.8353E-08	8.3499E-09
4	3.6666E-01	2.2706E-03	6.7624E-05	9.1519E-06	4.2106E-07	2.8565E-07	4.1617E-07	1.5012E-07	6.8299E-08	4.5961E-08
5	3.6368E-01	2.2592E-03	6.8432E-05	3.1484E-06	2.1359E-06	3.1119E-06	1.1225E-06	5.1070E-07	3.4367E-07	0.
6	3.6366E-01	2.2293E-03	2.3403E-05	1.5877E-05	2.3131E-05	8.3438E-06	3.7960E-06	2.5545E-06	0.	0.
7	3.6377E-01	1.8128E-03	1.1674E-04	1.7008E-04	6.1351E-05	2.7912E-05	1.8703E-05	0.	0.	0.
8	3.6489E-01	3.3614E-04	4.4504E-04	1.6053E-04	7.3236E-05	4.8686E-05	0.	0.	0.	0.
9	3.6487E-01	6.9414E-04	2.3649E-04	1.0838E-04	7.1915E-05	2.4371E-06	0.	0.	0.	0.
10	3.6528E-01	4.5397E-04	1.9307E-04	1.2478E-04	2.4409E-06	6.4440E-13	0.	0.	0.	0.
11	3.6555E-01	5.2796E-04	3.0707E-04	1.6168E-05	6.7490E-15	3.1381E-21	0.	0.	0.	0.
12	3.6585E-01	7.6552E-04	6.9262E-05	1.7766E-08	1.0935E-17	5.0493E-24	0.	0.	0.	0.
13	3.6702E-01	1.8528E-04	2.5295E-06	7.9412E-10	5.3704E-19	2.4744E-25	0.	0.	0.	0.

FROM I	TO I= 10	I= 11	I= 12
1	1.5279E-10	6.9471E-11	4.6750E-11
2	7.5721E-10	5.0955E-10	0.
3	5.6190E-09	0.	0.
4	0.	0.	0.

I = 5 THRU I= 13 SAME AS ABOV

CASE 30 4 PWR F/A.S FLOODED AP-101 CASK

ARRAY DESCRIPTION

Z = 1

2 1

CASE 39 4 PWR F/A.S FLOODED AP-101 CASK

LIFETIME = .71337E-05 + OR - 7.40861E-07 GENERATION TIME = 2.65289E-05 + OR - 2.55916E-07

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	.89951	+ OR - .00591	.89359 TO .90542	.88768 TO .91133	.88177 TO .91724	15000
4	.90012	+ OR - .00600	.89412 TO .90612	.88812 TO .91212	.88212 TO .91812	14700
5	.89961	+ OR - .00610	.89351 TO .90572	.88740 TO .91182	.88130 TO .91793	14400
6	.89903	+ OR - .00621	.89283 TO .90524	.88662 TO .91145	.88041 TO .91766	14100
7	.89769	+ OR - .00619	.89149 TO .90388	.88530 TO .91007	.87911 TO .91627	13800
8	.89718	+ OR - .00631	.89087 TO .90350	.88456 TO .90981	.87825 TO .91612	13500
9	.89661	+ OR - .00643	.89018 TO .90304	.88375 TO .90947	.87732 TO .91590	13200
10	.89529	+ OR - .00644	.88885 TO .90173	.88241 TO .90817	.87597 TO .91461	12900
11	.89418	+ OR - .00650	.88769 TO .90068	.88119 TO .90718	.87469 TO .91368	12600
12	.89499	+ OR - .00661	.88838 TO .90160	.88178 TO .90821	.87517 TO .91481	12300
17	.89870	+ OR - .00677	.89193 TO .90547	.88515 TO .91225	.87838 TO .91902	10800
22	.89374	+ OR - .00743	.88632 TO .90117	.87889 TO .90859	.87147 TO .91602	9300
27	.89650	+ OR - .00761	.88889 TO .90411	.88127 TO .91173	.87366 TO .91934	7800
32	.88739	+ OR - .00790	.87950 TO .89529	.87160 TO .90318	.86371 TO .91108	6300
37	.88624	+ OR - .00904	.87720 TO .89527	.86816 TO .90431	.85913 TO .91335	4800
42	.89190	+ OR - .01238	.87952 TO .90428	.86713 TO .91667	.85475 TO .92905	3300
47	.88543	+ OR - .01723	.86820 TO .90267	.85096 TO .91990	.83373 TO .93713	1800

```
FOLLOWING IS A CARD IMAGE LISTING OF THE INPUT DATA

                                   C O L U M N   N U M B E R
 CARD                    11111111112222222222333333333344444444445555555555666666666677777777778
 NUMBER         12345678901234567890123456789012345678901234567890123456789012345678901234567890

   1            CASE 31 AP-101 CASK FOUR PWR F/A FLOODED K-INFINITY
   2            5 53 300 3 13 13 6 6 6 17 2 2 1 1 0 1 0 2000 00 4*0 5 0 10 9 15
   3            -1.0 -1.0 -1.0 -1.0 -1.0 -1.0
   4            1.87+9 4.16+8 1.54+8 5.65+7 2.08+7 7.66+6 2.82+6 1.73+6 1.45+6 1.13+6
   5            7.60+5 5.42+5 2.20+5
   6            1 -1 1.0 2 2 1.0 3 3 1.0 4 4 1.0 5 5 1.0 6 6 1.0
   7              13   27    0 1140 PWR 15X15 FUEL 3.4W                            1
   8             0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +31943- 7 0 +     0+ 0 0 +71693-18     1
   9             0 +31943- 7 0 +62060- 7 0 +34282- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     2
  10             0 +     0+ 0 0 +     0+ 0 0 +26696- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     3
  11             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     4
  12             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0     5
  13             0 +21433- 7 0 +     0+ 0 0 +12979-18 0 +21433- 7 0 +98424- 8 0 +67414- 5     6
  14             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +47124- 5     7
  15             0 +64 40- 6 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     8
  16             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0     9
  17             0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +49188- 7 0 +     0+ 0 0 +23714-18    10
  18             0 +49188- 7 0 +17883- 7 0 +96787- 5 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    11
  19             0 +     0+ 0 0 +     0+ 0 0 +64531- 5 0 +17441- 5 0 +72185- 7 0 +     0+ 0    12
  20             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    13
  21             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0    14
  22             0 +80427- 7 0 +     0+ 0 0 +49091-18 0 +80427- 7 0 +36991- 7 0 +10488- 4    15
  23             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +69118- 5    16
  24             0 +27569- 5 0 +22781- 6 0 +95030- 8 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    17
  25             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    18
  26             0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +19242- 6 0 +     0+ 0 0 +14745-17    19
  27             0 +19242- 6 0 +11110- 6 0 +10476- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    20
  28             0 +     0+ 0 0 +     0+ 0 0 +67694- 5 0 +30343- 5 0 +36277- 6 0 +30831- 7    21
  29             0 +12861- 8 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    22
  30             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +10000- 4 0 +     0+ 0    23
  31             0 +38974- 6 0 +     0+ 0 0 +33569-17 0 +38974- 6 0 +25293- 6 0 +10758- 4    24
  32             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +68672- 5    25
  33             0 +30508- 5 0 +39957- 6 0 +49096- 7 0 +41725- 8 0 +17405- 9 0 +     0+ 0    26
  34             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    27
  35             0 +10000- 4 0 +10000- 4 0 +     0+ 0 0 +37628- 6 0 +     0+ 0 0 +24979-17    28
  36             0 +37628- 6 0 +18822- 6 0 +10794- 4 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    29
  37             0 +     0+ 0 0 +     0+ 0 0 +68649- 5 0 +30508- 5 0 +40205- 6 0 +54076- 7    30
  38             0 +66444- 8 0 +56469- 9 0 +23556-10 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    31
  39             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +10000- 4    32
  40             0 +     0+ 0 0 +96295- 7 0 +12169-17 0 +96295- 7 0 +91692- 7 0 +10340- 4    33
  41             0 +11121-26 0 +22033-25 0 +14891-22 0 +29958-14 0 +15769- 7 0 +33962- 5    34
  42             0 +11371- 5 0 +13679- 6 0 +18498- 7 0 +24879- 8 0 +30570- 9 0 +25980-10    35
  43             0 +10838-11 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +     0+ 0    36
  44             0 +10000- 4 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +24866- 6 0 +50624-17    37
  45             0 +24866- 6 0 +38144- 6 0 +10701- 4 0 +     0+ 0 0 +24095-20 0 +47735-19    38
  46             0 +32028-16 0 +12720- 7 0 +35602- 5 0 +23215- 5 0 +71427- 6 0 +92801- 7    39
  47             0 +12549- 7 6 +16878- 8 0 +20739- 9 0 +17625-10 0 +73523-12 0 +     0+ 0    40
  48             0 +     0+ 0 0 +     0+ 0 0 +     0+ 0 0 +10000- 4 0 +     0+ 0 0 +10000- 4    41
  49             0 +     0+ 0 0 +25442- 6 0 +53499-17 0 +25442- 6 0 +40310- 6 0 +11203- 4    42
  50             0 +     0+ 0 0 +     0+ 0 0 +35124-11 0 +76953-10 0 +84054- 7 0 +59808- 5    43
  51             0 +43686- 5 0 +27690- 5 0 +10496- 5 0 +13520- 6 0 +18283- 7 0 +24590- 8    44
```

```
F O L L O W I N G   I S   A   C A R D   I M A G E   L I S T I N G   O F   T H E   I N P U T   D A T A

                                      C O L U M N   N U M B E R
 CARD                    111111111122222222223333333333444444444455555555556666666666777777777 8
NUMBER          1234567890123456789012345678901234567890123456789012345678901234567890123456789 0

  715           0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0        51
  716           0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *10348- 8 0 *55764- 7        52
  717           0 *45442- 8 0 *82111- 9 0 *39875- 9 0 *22187- 9 0 *61236-10 0 *30866-11        53
  718           0 *15329-12 0 *75676-14 0 *32896-15 0 *98853-17 0 *40892-18 0 *     0* 0        54
  719           0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0        55
  720           0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0 0 *     0* 0        56
  721           0 *     0* 0 0 *     0* 0 0 *54889- 7 0 *15509- 8 0 *41413-10 0 -53812-10        57
  722           0 *40730-10 0 *21257-10 0 *19822-10 0 *99912-12 0 *49618-13 0 *24496-14        58
  723           0 *19648-15 0 *31998-17 0 *13236-18                                              59
  724           BOX TYPE  1
  725           CUBOID  1    22.9489     1.4986   22.9489     1.4986  190.5      -190.5       13*0.5
  726           CUBOID  2    23.65375    .79375   23.65375    .79375  223.6724   -200.66      13*0.0
  727           CUBOID  3    23.97125    .47625   23.97125    .47625  223.6724   -200.66      13*0.0
  728           CUBOID  4    24.4475     0.0      24.4475     .0      223.6724   -200.66      13*0.0
  729           CUBOID  3    24.765      0.0      24.765      .0      223.6724   -200.66      13*0.0
  730           BOX TYPE  2
  731           CUBOID  1    23.2664     1.8161   22.9489     1.4986  190.5      -190.5       13*0.5
  732           CUBOID  2    23.97125    1.11125  23.65375    .79375  223.6724   -200.66      13*0.0
  733           CUBOID  3    24.28875    .79375   23.97125    .47625  223.6724   -200.66      13*0.0
  734           CUBOID  4    24.765      .3175    24.4475     .0      223.6724   -200.66      13*0.0
  735           CUBOID  3    24.765      0.0      24.765      .0      223.6724   -200.66      13*0.0
  736           CORE BDY  0  24.765     -24.765   24.765      .0      223.6724   -200.66      13*0.0
  737           ZHEMICYL+Y  2 35.56     223.6724 -200.66     13*0.0
  738           ZHEMICYL+Y  5 37.1475   227.482  -205.899    13*0.0
  739           ZHEMICYL+Y  6 46.0375   231.292  -213.519    13*0.0
  740           ZHEMICYL+Y  3 46.0375   231.292  -223.679    13*0.0
  741           ZHEMICYL+Y  5 50.5206   243.992  -232.569    13*0.0
  742           CUBOID  2  51.5206  -51.5206  51.5206  0.0  244.992  -233.569  13*0.0
  743           2 9*1 0 1 2 2 1 6*1 1
  744           END CASE
  745           END KENO

*********************************** E N D   O F   I N P U T   L I S T I N G ***********************************
```

CASE 31 AP-101 CASK FOUR PWR F/A FLOODED K-INFINITY

NUMBER OF GENERATIONS	53	START TYPE	0
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	13	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	13	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	6	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	6	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	6	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	17	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	2	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	0	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	YES
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	YES

ANISN TYPE CROSS SECTIONS WERE READ. POSITION OF SIGMA TOTAL WAS 9 , POSITION OF SIGMA GG WAS 15

VELOCITIES WILL BE READ FROM CARDS.

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00

MAXIMUM TIME = 5.0000 MINUTES

STORAGE LOCATIONS REQUIRED FOR THIS JOB = 6238
REMAINING AVAILABLE LOCATIONS= 35234

REQUIRED LCM STORAGE LOCATIONS 508 (000009748)
REMAINING LCM AVAILABLE 126468 (003670048)

CASE 31 AP-101 CASK FOUR PWR F/A FLOODED K-INFINITY

ARRAY DESCRIPTION

Z = 1

2 1

CASE 31 AP-101 CASK FOUR PER F/A FLOODED K-INFINITY

LIFETIME = 4.64804E-05 +OR- 6.25553E-07 GENERATION TIME = 2.64483E-05 +OR- 2.23526E-07

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	.89263	+OR- .00542	.88721 TO .89805	.88179 TO .90347	.87637 TO .90888	15000
4	.89096	+OR- .00526	.88570 TO .89622	.88043 TO .90148	.87517 TO .90675	14700
5	.89139	+OR- .00536	.88593 TO .89666	.88057 TO .90202	.87521 TO .90738	14400
6	.89080	+OR- .00545	.88535 TO .89626	.87989 TO .90171	.87444 TO .90717	14100
7	.89047	+OR- .00556	.88491 TO .89604	.87934 TO .90160	.87378 TO .90717	13800
8	.89157	+OR- .00558	.88600 TO .89715	.88042 TO .90273	.87485 TO .90830	13500
9	.89162	+OR- .00570	.88591 TO .89732	.88021 TO .90303	.87450 TO .90873	13200
10	.89055	+OR- .00574	.88481 TO .89628	.87908 TO .90202	.87334 TO .90776	12900
11	.89018	+OR- .00586	.88432 TO .89604	.87845 TO .90190	.87259 TO .90776	12600
12	.89070	+OR- .00597	.88481 TO .89676	.87883 TO .90273	.87286 TO .90870	12300
17	.89057	+OR- .00634	.88423 TO .89691	.87789 TO .90325	.87155 TO .90959	10800
22	.89544	+OR- .00680	.88864 TO .90224	.88185 TO .90903	.87505 TO .91583	9300
27	.89801	+OR- .00773	.89028 TO .90575	.88254 TO .91348	.87481 TO .92121	7800
32	.90683	+OR- .00776	.89908 TO .91459	.89132 TO .92235	.88356 TO .93010	6300
37	.90373	+OR- .00746	.89627 TO .91119	.88881 TO .91865	.88135 TO .92611	4800
42	.90880	+OR- .00925	.89954 TO .91805	.89029 TO .92731	.88103 TO .93656	3300
47	.90809	+OR- .00849	.89959 TO .91657	.89111 TO .92505	.88262 TO .93354	1800

7. OPERATING PROCEDURES

Operating procedures for spent fuel contents are presented in this section. Operating procedures for non-fissile contents are covered in the Safety Analysis Report, Cask AP-101 These operating procedures are in compliance with the requirements of Subpart D of 10 CFR 71. A copy of the ANEFCO Quality Assurance Program for shipping packages is on file with the Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission.

Prior to the first use, the spent fuel baskets will be checked for cracks, pinholes, uncontrolled voids or other defects which could significantly reduce the effectiveness of the package. Also, prior to the first use of the AP-101 cask for the shipment of spent fuel, the containment vessel will be checked at a pressure of 41 psia (50% higher than the maximum normal operating pressure) in order to ascertain that it will not leak.

The following sections detail operating procedures for the AP-101 cask for shipping of spent fuel. It should be noted that the procedures are consistent with maintaining occupational radiation exposure to levels which are as low as is reasonably achievable. Toward this objective all operations are performed with full benefit of the shielding provided by the cask, or shielding of water in the loading pool. Also, fission gases are vented through closed lines in order to avoid exposure to personnel.

7.1 PROCEDURES FOR LOADING THE PACKAGE

7.1.1 Routine Determinations

The simple design of the cask results in simplified operating procedures and routine determinations prior to shipment. The cask has no liquid coolant and is sealed for shipment without vents or valves for shipment; therefore, the routine determinations that are to be made are rather straight-forward.

A primary concern for any fissile material shipment is criticality safety. The presence of the specified criticality control materials in the cask are essential for the fuel loading configurations considered. These criticality control materials are incorporated as an integral part of the spent fuel baskets described in Section 1 of this amendment to the Safety Analysis Report for the shipment of spent fuel in the AP-101 cask. That is, the design of the spent fuel baskets for use with the AP-101 cask are constructed as permanent structures which cannot be disassembled during normal operations. Once the presence of the required criticality control materials is assured in the spent fuel basket structure, then the continued presence of the criticality control materials is assured for subsequent shipments.

7.1.1 Routine Determinations - (Cont'd)

Routine determinations that are to be made prior to each shipment of fissile material shall include:

1) inspection of the package to determine that it has not been significantly damaged.

2) inspection of the closure of the package and sealing gaskets to ascertain their presence and good condition.

3) a check that fuel loading restrictions included in the operating procedures have been performed, and are properly recorded and maintained for future inspection.

7.1.2 Cask Loading Procedures

7.1.2.1 Spent Fuel Loading Restrictions

Individual fuel loadings are to be analyzed prior to loading the AP-101 cask to ascertain that the requirements of the license and Safety Analysis Report for shipment of spent fuel in the AP-101 cask are satisfied.

1) The spent fuel basket shall be as prescribed for the type of spent fuel to be shipped and shall be in place within the cavity of the cask.

2) The spent fuel batch to be shipped shall be in conformance with the spent fuel specification in the license and the SAR for spent fuel shipping in the AP-101 cask.

3) The total heat loading of the spent fuel batch and the individual fuel assemblies shall be determined based on the detailed exposure history. Fuel assembly maximum decay heat loads are to be no greater than those specified in Figure 1.1 of the SAR for spent fuel shipping in the AP-101 cask.

   Total heat loads for a spent fuel batch in excess of 2.2 KW will require the use of an expanded metal shield for personnel protection.

4) Loading configuration of the spent fuel batch will be in accordance with Figure 1.4 and 1.5 of this amendment to the SAR for the AP-101 cask.

7.1.2.1 Spent Fuel Loading Restrictions - (Cont'd)

5) Prior determination of the compatability of the spent fuel batch with the shielding provided by the AP-101 cask is to be established by ascertaining that total dose rates as predicted by Figure 5.1 of this amendment to the SAR for the AP-101 cask are within the allowable range.

7.1.2.2 Spent Fuel Loading Procedures

1) Receive cask and trailer at handling area and perform radiation survey.

2) Release cask tie-down.

3) Detach impact limiter from cask.

4) Using an overhead crane, attach the lifting yoke to upper trunnions on cask.

5) Rotate cask to the vertical position on the trailer.

6) Remove cask from trailer.

7) Inspect cask for damage.

8) Attach lid lifting eyes to lid.

9) Attach lid lifting spider to lifting eyes.

10) Open cask drains and cask cover plug.

11) Remove cask lid bolts. Remove cask lid and inspect for damage.

12) Lower cask to cask pad on bottom of pool.

13) Load spent fuel basket with spent fuel in accordance with license restrictions and requirements of Section 7.1.2.1 of Amendment to the SAR for the AP-101 cask.

14) Place lid on cask.

15) Raise cask to surface of pool and monitor radiation.

16) Install four (4) lid bolts to secure cover.

17) Raise cover over pool and allow water to drain out.

18) Move cask to decon area.

7.1.2.2 Spent Fuel Loading Procedures - (Cont'd)

19) Close cask cover plug.

20) Remove lid lifting spider.

21) Remove lid lifting eyes.

22) Install and torque all remaining lid bolts to 100 ft-pounds.

23) Connect vacuum hose to one drain connection and close other drain connection.

24) Maintain vapor pressure in cask consistent with water temperature for at least 90 minutes.

25) Decon all exposed cask surfaces.

26) Pressurize cask internal cavity to 41 psia with air and bubble test all closures for tightness.

27) Remove hose connection and close drain plug to seal cask when bubble test is satisfactorily completed.

28) Check cask surface for contamination. (Decontaminate again if necessary).

29) Load cask onto trailer.

30) Remove yoke from cask.

31) Latch trunnion tie-downs.

32) Attach impact limiters to cask.

33) Perform neutron and gamma radiation survey of cask.

34) Attach expanded metal personnel barrier if decay heat load is greater than 2.2 KW.

35) Authorize shipment to receiving facility.

7.2 CASK UNLOADING PROCEDURES

1) Receive cask and trailer at receiving site.

2) Remove personnel protection barrier, if used.

3) Perform radiation survey.

4) Release cask tie-down.

5) Detach impact limiter from cask.

6) Using overhead crane, attach the lifting yoke to upper trunnions on the cask.

7) Rotate the cask to the vertical position on the trailer.

8) Remove cask from trailer.

9) Inspect cask for damage.

10) Attach vent line with pressure gauge to cover plug and vent cask cavity to controlled exhaust.

11) Attach lid lifting eyes to lid.

12) Attach lid lifting spider to lifting eyes.

13) Open cask drains.

14) Loosen all lid bolts. Remove lid bolts except for four (4) lid bolts that are left in place.

15) Lower cask in pool until water level is just below the bottom of the fuelled region within the cask.

16) Lower cask slowly so that water level within cask makes contact only with portions of the fuel assemblies that have been cooled by proximity to pool water within the cask. Monitor internal cask pressure (vent line) to control rate of lowering so that pressure does not build-up within cask.

17) When the cask is resting on the cask pad on the bottom of the pool and the cask is full of water, remove the remaining four (4) lid bolts.

18) Remove cask lid from cask and remove it from the pool.

19) Remove vent line and close cask cover plug.

7.2 CASK UNLOADING PROCEDURES - (Cont'd)

20) Unload individual spent fuel assemblies from cask.

21) With all spent fuel assemblies removed, raise cask so that drains are above water level. Rinse sides of cask with clean water.

22) Allow cask to completely drain.

23) Move cask to set-down area.

24) Close cask drains.

25) Inspect cask seal for satisfactory condition and replace cask lid. Install bolts torquing to 100 ft-pounds.

26) Remove lid lifting spider.

27) Remove lid lifting eyes.

7.3 PREPARATION OF EMPTY PACKAGES FOR TRANSPORT

1) Decon all exposed cask surfaces.

2) Load cask onto trailer.

3) Remove yoke from cask.

4) Latch trunnion tie-down.

5) Visually inspect that cask closures are secured and not leaking.

6) Attach impact limiters to cask.

7) Replace expanded metal personnel barrier, if used.

8) Label cask empty.