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TTC-0120

Nuclear Criticality Safety
Studies Applicable to Spent
Fuel Shipping Cask Designs
and Spent Fuel Storage

J. S. Tang

MASTER

*Prepared for Sandia National Laboratories'
Transportation Technology Center for the
United States Department of Energy under
Contract Number DE-AC04-76-DP00789.*

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
NTIS price codes—Printed Copy: A03; Microfiche A01

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ORNL/TM-7403
(TTC-0120)
Category Distribution UC-85

Contract No. W-7405-eng-26

**NUCLEAR CRITICALITY SAFETY STUDIES APPLICABLE
TO SPENT FUEL SHIPPING CASK DESIGNS
AND SPENT FUEL STORAGE**

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Manuscript Completed: October 1, 1980
Date Published: November 1980

*Prepared for Sandia National Laboratories'
Transportation Technology Center for the
United States Department of Energy under
Contract Number DE-AC04-76-DP00789.*

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Revisions to ORNL/TM-7403 (TTC-0120)

Attached are eight pages of ORNL/TM-7403 (TTC-0120), by J. S. Tang, which have been revised.

Please incorporate these corrected pages into your copy of this report.

G. E. Whitesides

G. E. Whitesides
Nuclear Engineering Applications
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NUCLEAR CRITICALITY SAFETY STUDIES APPLICABLE
TO SPENT FUEL SHIPPING CASK DESIGNS
AND SPENT FUEL STORAGE

J. S. Tang

ABSTRACT

Criticality analyses of water-moderated and reflected arrays of LWR fresh and spent fuel assemblies were carried out in this study. The calculated results indicate that using the assumption of fresh fuel loading in spent fuel shipping cask design leads to assembly spacings which are about twice the spacings of spent fuel loadings. Some shipping cask walls of composite lead and water are more effective neutron reflectors than water of 30.48 cm (12 in) thickness.

INTRODUCTION

In spent fuel shipping cask design, criticality control of loaded shipping casks is one of the primary safety considerations. This study deals with the effects of fuel assembly spacing and neutron reflectors on neutron multiplication factors of LWR fuel systems. This work is sponsored by the Transportation Technology Center (TTC) at Sandia National Laboratory.

In shipping cask design, the usual practice is to establish criticality safety criteria of a spent fuel cask based on fresh fuel loading.¹ This approach allows for extra criticality-safety margin since in the absence of burnable neutron absorbers, fresh fuel is in the most reactive state for criticality. Another advantage of analyzing fresh fuel is that its composition and physical configuration are more accurately known, and sometimes its nuclear behavior is available from

previous experiments for core physics calculations. However, there may be a severe economic penalty for being overly conservative in using this criteria for cask design.

Generally, criticality control of shipping casks can be achieved by: (1) limiting the mass of fissile material, (2) controlling the amount of moderator, (3) introducing neutron absorbers, and (4) regulating the effectiveness of the reflector, and (5) changing the spacing of fuel assemblies. This study focuses on the mass of fissile material loaded, the fuel assembly spacing, and the reflector effectiveness of spent fuel shipping casks. Since criticality control by neutron absorbers is dependent on individual cask design, the use of neutron absorbers is not considered. The effects of assembly spacing and the number of assem-

blies on reactivity were investigated for boiling-water-reactor (BWR) and pressurized-water-reactor (PWR) fresh fuel systems and for three PWR spent fuel systems. The performance of the shipping cask wall as a reflector was examined for the PWR fresh fuel system.

Neutron multiplication factors for arrays of fuel assemblies were calculated by KENO.² Isotopic compositions of the spent fuels was obtained from ORIGEN³ depletion calculations. Neutron cross-section

data were obtained from the Hansen-Roach library⁴ for the fresh fuels, and for the spent fuels cross-section data were generated from the 123-group XSDRN library.⁵ The SCALE⁶ computer code system was used to perform resonance treatment and group collapsing of cross-section data.

REACTIVITY EFFECTS OF NUMBER OF FUEL ASSEMBLIES AND ASSEMBLY SPACING

The reactivity effects of the number of assemblies and assembly spacing was examined by determining the spacing required for criticality

as a function of the number of assemblies. KENO calculations were carried out for water-moderated square arrays of fuel assemblies reflected by 30.48 cm (12 inches) of water. The critical spacing for each array was determined by an iterative process of the search option in KENO. Water was used as a reference reflector in the calculations since the effectiveness of water as a neutron reflector is well known and provides a comparative basis for the effectiveness of cask walls. In the following sections, details of calculations and discussions of results are described separately for fresh fuel arrays and spent fuel arrays.

Fresh Fuel Arrays

Both PWR and BWR fresh fuel assemblies were examined. The PWR assemblies are the Westinghouse type containing 17×17 rods,⁷ and the BWR assemblies are the General Electric type containing 8×8 rods.⁸ The fissile material for these assemblies are uranium oxides with enrichments of 3.20% for PWR and 2.75% for BWR. The physical characteristics of these assemblies are summarized in Table 1. The PWR assembly is larger in size and contains more than twice the amount of fissile material as the BWR, but the BWR assembly is slightly longer. Table 2 contains fuel rod physical characteristics. The PWR rods are smaller and more closely spaced than the BWR rods.

In calculating the neutron multiplication factor of each array, details of each fuel rod and each assembly were modeled using the three dimensional capability of the KENO code. The fuel channels of BWR assemblies were also included. Each PWR assembly has 24 control cluster elements and a central instrumentation sheath. These 25 non-fuel elements were replaced by water in the calculations. Figure 1 shows the top view of the BWR and PWR assemblies. In the axial direction, only

Table 1. Physical Characteristics of LWR Fuel Assemblies*

	BWR	PWR
Fuel rod array	8 x 8	17 x 17
Fuel rods per assembly	63	264
Cross section dimensions, cm	14.0 x 14.0	21.4 x 21.4
Overall assembly length, cm	447.0	405.9
Fuel rod length, cm	406.4	385.1
Active fuel height, cm	375.9	365.8
Fuel rod OD, cm	1.252	0.950
Assembly total weight, kg	319.9	657.9
UO ₂ /assembly, kg	208.0	523.4
Total other metal/assembly, kg	111.9	134.5

*Data obtained from Ref. 9, Table 4.5

Table 2. Fuel Rod Data of LWR*

	BWR	PWR
Rod OD, cm	1.25222	0.94996
Cladding thickness, cm	0.08636	0.05715
Pellet OD, cm	1.05664	0.81916
Fuel rod pitch, cm	1.62560	1.23984
Density (% of theoretical)	95	95
UO ₂ enrichment (wt % ²³⁵ U)	2.75	3.20
Cladding material	Zircaloy	Zircaloy

*Refs. 7 and 8.

BWR FUEL ASSEMBLY, 8 x 8 (FUEL CHANNEL INCLUDED)

PWR FUEL ASSEMBLY, 17 x 17 (ONLY OUTER FUEL RODS SHOWN)

⊗ WATER ROD

Fig. 1. Top view of BWR and PWR fuel assemblies.

the active fuel height was modeled in the calculations. The oxides were assumed to be 95% theoretical density.

Neutron cross-section data were obtained from the 16-group Hansen-Roach library of the SCALE code system, and the Criticality Safety Analytic Sequence 2 (CSAS2) module was used to treat resonance self-shielding of resonance nuclei and to prepare a mixing table and microscopic cross sections for KENO calculations. The calculations were performed for water moderated fuel assemblies arranged with centers on a square pattern and reflected by 30.48 cm water. The water reflector was simulated by the differential albedo option in order to save computational time that otherwise would be required for neutron tracking in water. Calculations of a 2×2 array of fresh PWR assemblies at surface spacing of 0.1016 cm gave neutron multiplication factors of 1.1568 ± 0.0041 with differential albedo and 1.1517 ± 0.0047 with neutron tracking in the water reflector. Therefore, the differential albedo option is a very good approximation for this study. The critical spacing for each array was determined by the KENO search option. The determined surface spacings for criticality as a function of the number of assemblies for BWR and PWR systems are presented in Tables 3 and 4, respectively, and displayed in Fig. 2.

The spacings of PWR arrays are about twice those of BWR arrays for the same number of assemblies. Also, criticality could not be achieved for a 2×2 array of BWR assemblies at any spacing. It is evident that a PWR assembly is neutronicallly more reactive than BWR assembly; this is to be expected since the PWR assembly has a smaller rod diameter and a higher fissile material content per unit volume than the BWR assembly.

Table 3. Calculated Surface Spacing of Arrays of Fresh BWR Assemblies at Critical

Array	Spacing (cm)	No. of Iterations	K-effective
3 x 3	3.2942	8	1.00060 $\pm$ 0.00348
4 x 4	4.2714	3	1.00204 $\pm$ 0.00374
5 x 5	4.6910	4	1.00470 $\pm$ 0.00346
6 x 6	4.9158	5	1.00200 $\pm$ 0.00393
10 x 10	5.2188	5	1.00271 $\pm$ 0.00442
Infinite	5.5080	3	1.00146 $\pm$ 0.00334

Table 4. Calculated Surface Spacing of Arrays of Fresh PWR Assemblies at Critical

Array	Spacing (cm)	No. of Iterations	K-effective
2 x 2	7.2517	4	1.00013 $\pm$ 0.00432
3 x 3	8.5673	4	1.00234 $\pm$ 0.00472
4 x 4	9.4046	4	0.99686 $\pm$ 0.00357
5 x 5	9.6107	3	1.00170 $\pm$ 0.00403
6 x 6	9.6500	1	1.00284 $\pm$ 0.00397
7 x 7	9.7444	4	1.00151 $\pm$ 0.00368
Infinite	10.0500	6	0.99946 $\pm$ 0.00389

Spent PWR Fuel Arrays

Three types of PWR spent fuels were examined. They were 3.20% enriched fuel at 15,000 and 33,000 MWD/MT burnups and 4.20% enriched fuel at 50,000 MWD/MT burnup. One year of decay time was chosen for comparison of the effects of the three exposure times. The isotopic concentrations of the spent fuel materials were obtained from results of ORIGEN depletion calculations, and these are listed in Table 5. Since according to ORIGEN results the isotopic concentrations of the spent fuel materials do not change significantly between the first year and the first few decades after discharge, results of this section are expected to be applicable for spent fuels with longer decay time.

Fig. 2. Surface spacing versus number of assemblies for critical arrays of LWR fresh fuel reflected by 30.48 cm H_2O .

Table 5. Isotopic Concentrations of PWR Spent Fuels
One Year After Discharge (atom/barn-cm)

	Burnup Level (MWD/MT)		
	15,000 [U(3.2)]	33,000 [U(3.2)]	50,000 [U(4.2)]
<u>Actinides</u>			
²³⁴ U	5.6107-6*	4.2675-6	4.3012-6
²³⁵ U	4.2531-4	1.8723-4	1.5903-4
²³⁶ U	5.7710-5	9.2756-5	1.3140-4
²³⁸ U	2.2249-2	2.1929-2	2.1448-2
²³⁷ Np	3.3416-6	1.0326-4	1.7667-5
²³⁸ Pu	4.3056-7	3.1835-6	7.5068-6
²³⁹ Pu	9.4912-5	1.1647-4	1.3560-4
²⁴⁰ Pu	2.3001-5	5.3161-5	6.2741-5
²⁴¹ Pu	1.0000-5	2.6738-5	3.7987-5
²⁴² Pu	1.3637-6	1.0514-5	1.9673-5
<u>Fission Products</u>			
¹⁰³ Rh	1.2852-5	1.7800-5	3.2249-5
¹³³ Cs	2.3322-5	4.2907-5	6.5394-5
¹⁴³ Nd	1.7518-5	3.0033-5	4.0171-5
¹⁴⁹ Sm	1.2830-7	1.3361-7	1.5588-7
¹³¹ Xe	9.8782-6	1.9839-5	2.3510-5
¹⁴⁷ Pm	3.6826-6	4.0408-6	4.2608-6
¹⁵² Sm	2.3615-6	4.1917-6	6.2022-6
¹⁵¹ Sm	3.6091-7	4.5195-7	6.8932-7
¹⁵³ Eu	1.3090-6	3.3504-6	6.4399-6
¹⁴⁵ Nd	1.2968-5	2.4488-5	3.6605-5
¹⁵⁵ Eu	1.1702-7	3.4571-7	8.3857-7
¹⁵⁰ Sm	4.8877-6	1.0409-5	1.7097-5
¹⁵⁴ Eu	2.5671-7	1.0608-6	2.4029-6
¹⁰⁹ Ag	1.2559-6	2.0713-6	5.7157-6
¹³⁴ Cs	8.4188-7	3.2481-6	6.7992-6
⁸³ Kr	1.5975-6	4.8346-6	3.6760-6
<u>Light Elements</u>			
Zr	1.3925-2	1.3925-2	1.4632-2
Ni	1.1702-3	1.1702-3	1.1702-3
Fe	3.1293-3	3.1298-3	3.1233-3
Cr	1.1691-3	1.1686-3	1.1702-3
O	4.6516-2	4.6511-2	4.6516-2

*Read as 5.6107×10^{-6} .

A 16-group neutron cross-section library containing all isotopes listed in Table 5 was generated from the XSDRN 123-group library. Generation of the 16-group library was accomplished in two steps: (1) resonance self-shielding of resonance nuclei was treated by the Nordheim integral method, (2) group collapse from 123 to 16 groups was performed using the fine group flux as the weighting function. These computational steps were carried out via the CSAS2 module of SCALE code system. The energy group boundaries of the new library were chosen as close as possible to those of the Hansen-Roach library so that the albedo data, which are represented in the Hansen-Roach group structure, could be used in calculations of the spent fuel arrays. For comparison, the energy group structures of the two libraries are shown in Table 6 and the calculated results of fresh PWR assemblies using the two cross-section libraries and albedo reflection are presented in Table 7. With the exception of the 4×4 array, the results in Table 7 agree very well. These results indicate that the bias of using the new cross section library, instead of using the Hansen-Roach library, to calculate the k-effectives of the systems is negligible.

Calculated results of assembly spacings as a function of the number of assemblies and burnup are plotted in Fig. 3 and the corresponding data are given in Tables 8 through 10. The curve for fresh PWR arrays from Fig. 2 is reproduced for comparison. Each curve in the figure is given a number in order to facilitate discussion of results. It is generally true that the assembly spacings of the fresh PWR arrays are about twice those of the spent fuel arrays. This indicates that using the fresh fuel assumption to assess the criticality of spent fuels can result in assembly spacings about double those required.

Table 6. Energy Group Structure of Two 16-Group Libraries

Energy Group	Upper Energy Boundaries (eV)	
	Hansen-Roach	New Sixteen Group
1	1.50 + 7*	1.4918 + 7
2	3.00 + 6	3.0119 + 6
3	1.40 + 6	1.3534 + 6
4	9.00 + 5	9.0718 + 5
5	4.00 + 5	4.0762 + 5
6	1.00 + 5	1.1109 + 5
7	1.70 + 4	1.5034 + 4
8	3.00 + 3	2.6126 + 3
9	5.50 + 2	5.8295 + 2
10	1.00 + 2	1.0130 + 2
11	3.00 + 1	2.9023 + 1
12	1.00 + 1	1.0677 + 1
13	3.00 + 0	3.0590 + 0
14	1.00 + 0	1.0000 + 0
15	4.00 - 1	3.5000 - 1
16	1.00 - 1	1.0000 - 1

*Read as 1.50×10^7 .

Table 7. Comparison of K-effectives for Arrays of Fresh PWR Assemblies Calculated with Two Different Cross-Section Data Sets

Array	Spacing (cm)	K-effective	
		Hansen-Roach	New 16-Group
2 × 2	7.2517	1.00013 ± 0.00432	1.00101 ± 0.00403
3 × 3	8.5673	1.00234 ± 0.00472	1.00183 ± 0.00387
4 × 4	9.4046	0.99686 ± 0.00357	0.98391 ± 0.00372
5 × 5	9.6107	1.00170 ± 0.00403	----
6 × 6	9.6500	1.00284 ± 0.00397	1.00317 ± 0.00389
7 × 7	9.7444	1.00151 ± 0.00368	----
Infinite	10.0500	0.99946 ± 0.00389	----

Curves (2) through (8) are results of critical and subcritical (k-effective = 0.9) arrays of spent fuel assemblies with and without fission products. For arrays greater than 5 × 5 assemblies, the spacing changes very slowly and approaches the infinite-array spacing. Of the three spent fuels with fission products, the 15,000 MWD/MT fuel has the

Fig. 3. Surface spacing versus number of assemblies for arrays of PWR spent fuel reflected by 30.48 cm H_2O .

highest reactivity, followed respectively by the 50,000 MWD/MT fuel and the 33,000 MWD/MT fuel. The primary reason for the 50,000 MWD/MT fuel being more reactive than the 33,000 MWD/MT fuel is that the former was initially more enriched (4.20%) than the latter (3.20%). Finite arrays of the last two spent fuel assemblies did not achieve criticality when fission products were included in the fuels. It can be inferred by comparing curve (2) with curve (5) and curve (6) with curve (7) that the negative reactivities associated with the fission products in the 15,000 MWD/MT and 33,000 MWD/MT fuels are worth less than 0.10. On the other

Table 8. Calculated Spacing for Arrays of PWR Spent Fuel Assemblies of 3.20% Enrichment at 15,000 MWD/MT Burnup

Array	Spacing (cm)	No. of Iterations	K-effective
With Fission Products, at Critical			
3 × 3	Not critical at any spacings		
4 × 4	0.9900	3	1.00209 ± .00283
5 × 5	2.7995	3	1.00117 ± .00274
6 × 6	3.2336	5	1.00254 ± .00248
Infinite	3.9022	5	0.99948 ± .00268
With Fission Products, at K-effective = 0.9			
3 × 3	4.3564	5	0.90262 ± .00287
4 × 4	5.1610	4	0.89777 ± .00323
5 × 5	5.4501	3	0.90112 ± .00282
6 × 6	5.6572	7	0.89508 ± .00297
Infinite	6.0150	3	0.89816 ± .00263
No Fission Products, at Critical			
3 × 3	2.5164	4	1.00174 ± .00347
4 × 4	3.6661	4	1.00128 ± .00296
5 × 5	4.1654	2	0.99992 ± .00274
6 × 6	4.4272	3	0.99886 ± .00306
Infinite	4.8438	2	0.99998 ± .00305

Table 9. Calculated Spacing for Arrays of PWR Spent Fuel Assemblies of 3.20% Enrichment at 33,000 MWD/MT Burnup

Array	Spacing (cm)	No. of Iterations	K-effective
With Fission Products, at Critical			
Infinite	0.3245	5	1.00131 ± .00264
Finite arrays would not go critical			
With Fission Products, at K-effective = 0.9			
3 × 3	0.1977	11	0.90091 ± .00305
4 × 4	3.4100	7	0.89801 ± .00278
5 × 5	3.8212	8	0.90106 ± .00277
6 × 6	4.0638	7	0.90673 ± .00271
Infinite	4.5742	4	0.90143 ± .00255
No Fission Products, at Critical			
3 × 3	Did not achieve criticality after 12 iterations		
4 × 4	3.0024	9	0.99839 ± .00313
5 × 5	3.4782	9	0.99985 ± .00288
6 × 6	3.6630	5	1.00130 ± .00290
Infinite	4.2724	6	1.00213 ± .00291

Table 10. Calculated Spacing for Arrays of PWR Spent Fuel Assemblies of 4.20% Enrichment at 50,000 MWD/MT Burnup

Array	Spacing (cm)	No. of Iterations	K-effective
<u>With Fission Products, at Critical</u>			
Infinite	2.4932	5	1.00212 ± .00246
Finite arrays would not go critical			
<u>With Fission Products, at K-effective = 0.9</u>			
3 × 3	2.6120	2	0.90240 ± .00286
4 × 4	3.7533	3	0.90063 ± .00257
5 × 5	4.3007	3	0.89955 ± .00261
6 × 6	4.5000	1	0.90014 ± .00268
Infinite	4.9506	6	0.90000 ± .00260
<u>No Fission Products, at Critical</u>			
3 × 3	3.6430	3	0.99925 ± .00327
4 × 4	4.3215	3	1.00229 ± .00334
5 × 5	4.6353	4	0.99749 ± .00293
6 × 6	4.7576	7	1.00153 ± .00312
Infinite	5.4000	7	0.99798 ± .00312

hand, from curves (3) and (4) it can be seen that the fission products in the 50,000 MWD/MT fuel are worth more than 0.10 in negative reactivity.

The oxygen content in the spent fuels was inadvertently left out in the above analyses. Results of subsequent calculations of 33,000 MWD/MT fuel with oxygen included are presented in Table 11. It is evident that oxygen in the spent fuel reduces the k_{∞} 's of infinite lattices of unit cells by 1.2%, and it has an insignificant negative reactivity effect on finite arrays. Therefore, the results of spent fuel systems are still valid, but have a small (less than 1.2%) conservative bias in reactivity. The reactivity effect of oxygen in spent fuel would be much larger in a dry system because, in the absence of water, this oxygen plays a very significant role in slowing down neutrons.

Table 11. Comparison of K-effectives of 33,000 MWD/MT Spent Fuels With and Without Oxygen

Array	Spacing (cm)	k	Δk
With Fission Products			
3 $\times$ 3 arrays of assemblies:			
no oxygen in fuel	0.1977	$0.90091 \pm .00305$	0.00629
with oxygen in fuel	0.1977	$0.89462 \pm .00279$	
Infinite lattice of unit cell*			
no oxygen in fuel	1.25984**	0.98348	0.01163
with oxygen in fuel	1.25984	0.97185	
No Fission Products			
4 $\times$ 4 arrays of assemblies:			
no oxygen in fuel	3.0024	$0.99839 \pm .00313$	0.00230
with oxygen in fuel	3.0024	$0.99609 \pm .00271$	
Infinite lattice of unit cell*			
no oxygen in fuel	1.25984**	1.08851	0.01295
with oxygen in fuel	1.25984	1.07556	

*XSDRNPM one-dimensional S_n calculations.

**Dimension of fuel rod pitch.

EFFECT OF SHIPPING CASK WALL AS REFLECTOR MATERIAL

The reactivity effect of the wall of a lead-water shipping cask as a reflector material of arrays of fresh PWR assemblies was examined. The materials and dimensions of the cask wall are given in Table 12. Neutron multiplication factors for the arrays in Table 4 under two reflector conditions were calculated. The two reflector conditions are: (1) the cask wall touching the outer assemblies, and (2) a water gap of $S/2$ thickness between the cask wall and the outer assemblies where S is the surface separation of assemblies. Results of the calculations are presented in Table 13. It is evident that water backed by a lead-steel shipping cask wall is a more effective neutron reflector configuration

than 30.48 cm of water; and, in addition, the cask wall is also most effective when touching fuel assemblies. Since the relative effectiveness of different reflectors does not depend on fissile compositions, similar performance of the shipping cask wall as reflector for PWR spent fuels and BWR fuels can be expected. Furthermore, some variation in the thicknesses of wall materials would not significantly change the reflector characteristics of the cask wall configuration.

Table 12. Materials and Dimensions of Lead-Water Shipping Cask Wall

Component	Material	Thickness (cm)
Inner shell	stainless steel	3.810
Gamma shield	lead	17.907
Outer shell	stainless steel	5.080
Neutron shield	water*	10.922
Outer liner	stainless steel	1.905

*By volume, the neutron shield consists of 28.5% pure water (.9978 g/cc), 66.0% ethylene glycol (1.1088 g/cc) HOCH₂CH₂OH, and 5.5% potassium tetraborate (1.74 g/cc) K₂B₄O₇ · 8H₂O made with natural boron (17.3 wt% B¹⁰, 82.7 wt% B¹¹).

Table 13. Neutron Multiplication Factors for Arrays of Fresh PWR Assemblies with Three Different Reflector Conditions

Array	S, Spacing (cm)	Reflector Condition		
		30.48 cm Water	Cask Wall	S/2 of Water + Cask Wall
2 × 2	7.2517	1.00013 ± .00432	1.03208 ± .00475	1.02331 ± .00483
3 × 3	8.5673	1.00234 ± .00472	1.02535 ± .00449	1.01207 ± .00381
4 × 4	9.4046	0.99686 ± .00357	1.01141 ± .00410	1.00232 ± .00386
5 × 5	9.6107	1.00170 ± .00403	1.01210 ± .00413	0.99920 ± .00439
6 × 6	9.6500	1.00284 ± .00397	1.00436 ± .00514	0.99977 ± .00406
7 × 7	9.7444	1.00151 ± .00368	1.00678 ± .00502	1.00380 ± .00425
Infinite	10.0500	-----0.99946 ± .00389-----		

CONCLUSIONS

Reactivity effects of assembly spacing and the number of assemblies for LWR fuels moderated and reflected by water have been investigated. The results indicate that the critical assembly spacings of fresh PWR arrays are approximately two times the assembly spacings of the corresponding PWR spent fuel arrays. Therefore, design of spent fuel shipping cask based on fresh fuel loading can be overly conservative and uneconomical. Given the same number of assemblies, the critical assembly spacings of fresh BWR arrays are comparable to those of arrays of PWR spent fuel assemblies. Fission products in PWR spent fuels are worth more than 10% in negative reactivity for the 50,000 MWD/MT U(4.2) fuel and less than 10% for the 15,000 and 33,000 MWD/MT U(3.2) fuels. Finally, shipping cask walls composed of lead and water as shielding materials are more effective neutron reflectors than water of 30.48 cm thickness.

This study shows that, given a transport container and the number of assemblies of a specific spent fuel, there is a correspondence between the megawatt-day burnup and negative reactivity. The use of this negative reactivity as an element in the definition of an acceptable margin of subcriticality for specific transport containers appears reasonable.

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ACKNOWLEDGMENT

I am very grateful to J. T. Thomas for his many helpful discussions during the course of this work.

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