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CAPSIZE: A Personal Computer Program and Cross-Section Library for Determining the Shielding Requirements, Size, and Capacity of Shipping Casks Subject to Various Proposed Objectives

J. A. Bucholz

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MARTIN MARIETTA ENERGY SYSTEMS, INC.
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CAPSIZE: A PERSONAL COMPUTER PROGRAM AND CROSS-SECTION LIBRARY FOR DETERMINING THE SHIELDING REQUIREMENTS, SIZE, AND CAPACITY OF SHIPPING CASKS SUBJECT TO VARIOUS PROPOSED OBJECTIVES

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May 1987

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MASTER

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ABSTRACT

A new interactive program called CAPSIZE has been written for the IBM-PC to rapidly determine the likely impact that proposed design objectives might have on the size and capacity of spent fuel shipping casks designed to meet those objectives. Given the burnup of the spent fuel, its cooling time, the thickness of the internal basket walls, the desired external dose rate, and the nominal weight limit of the loaded cask, the CAPSIZE program will determine the maximum number of PWR fuel assemblies that may be shipped in a lead-, steel-, or uranium-shielded cask meeting those objectives. The necessary neutron and gamma shield thicknesses are determined by the program in such a way as to meet the specified external dose rate while simultaneously minimizing the overall weight of the loaded cask.

The one-group cross-section library used in the CAPSIZE program has been distilled from the intermediate results of several hundred 1-D multigroup discrete ordinates calculations for different types of casks. Neutron and gamma source terms, as well as the decay heat terms, are based on ORIGEN-S analyses of PWR fuel assemblies having exposures of 10, 20, 30, 40, 50, and 60 gigawatt days per metric tonne of initial heavy metal (GWD/MTIHM). In each case, values have been tabulated at 17 different decay times between 120 days and 25 years. Other features of the CAPSIZE program include a steady-state heat transfer calculation which will minimize the size and weight of external cooling fins, if and when such fins are required.

Comparisons with previously reported results show that the CAPSIZE program can generally estimate the necessary neutron and gamma shield thicknesses to within 0.16 in. and 0.08 in., respectively. The corresponding cask weights have generally been found to be within 1000 lbs of previously reported results.

1. INTRODUCTION

1.1 ORGANIZATIONAL AND TECHNICAL REASONS FOR DEVELOPING THE CAPSIZE PROGRAM

Shipments of intact spent fuel assemblies from nuclear power plants to the proposed Monitored Retrievable Storage facility, repository, or reprocessing site, may be minimized by developing a new generation of truck and rail casks with greater payload capacities than those designed in the past for short-term cooled spent fuel. To that end, the Office of Civilian Radioactive Waste Management (OCRWM) spent much of FY 86 developing a request for proposals for such casks. This request for proposals[1] was to include certain design objectives, such as the type of fuel to be shipped, its burnup and cooling time, the desired external dose rate, the nominal weight limit of the loaded cask, etc. While the stated design objectives were to be consistent with long-term economic, logistical, and operational goals, they could not be overly restrictive. To aid OCRWM personnel in developing the proposal, a new interactive program called CAPSIZE was written for the IBM-PC. The CAPSIZE program, described in this document, may be used to rapidly determine the likely impact that proposed design objectives might have on the size and capacity of casks designed to meet those objectives--in short, to answer the numerous "what if" questions that could arise as the request for proposals was being developed.

1.2 OVERVIEW OF THE CAPSIZE PROGRAM

Given the burnup of the spent fuel, its cooling time, the thickness of the internal basket walls, the desired external dose rate, and the nominal weight limit of the loaded cask, the CAPSIZE program will determine the maximum number of PWR fuel assemblies that may be shipped in a lead-, steel-, or uranium-shielded cask meeting those criteria. The necessary neutron and gamma shield thicknesses are determined by the program in each case. The calculational response time required for determining the capacities and shield thicknesses for all three types of casks totals approximately 2 or 3 seconds for small truck casks, and approximately 8-12 seconds for large rail casks.* The user may then interactively change one or more of the specified criteria and readily determine the impact of those changes on the projected capacities.

Neutron and gamma source terms, as well as the decay heat terms, are based on SAS2/ORIGEN-S[2,3] analyses of PWR fuel assemblies having exposures of 10, 20, 30, 40, 50, and 60 gigawatt days per metric tonne of initial heavy metal (GWD/MTIHM). In each case, values have been tabulated at 17 different decay times between 120 days and 25 years. The CAPSIZE program then performs a 2-D interpolation of this tabulated data to obtain the source terms and decay heat loads for the conditions specified. Based on comparisons with numerous 1-D transport calculations for different types of fuel,[4] the neutron and gamma radiation levels impinging on the inner wall of a cask have been found to be proportional to $N^{0.75}$ and $N^{0.5}$, respectively, where N is the number of assemblies in the cask. These correlations are used in the CAPSIZE program to account for the spatial self-shielding by the fuel itself.

One-group dose attenuation factors (i.e., cross sections) are used to determine the necessary shield thicknesses. These cross sections for the neutron and gamma shields and each of the stainless-steel shells comprising the cask(s) are dependent on the age of the spent fuel, the type of shield material (Pb, Fe, or U-metal), and the nominal thickness of the neutron and gamma shields. The one-group data library used in the CAPSIZE program has been distilled from the intermediate results of several hundred 1-D multigroup transport calculations for Pb-, Fe-, and U-shielded casks containing different numbers of spent fuel assemblies with cooling times ranging from 1 to 10 years.[4]

*Based on an IBM-PC/XT equipped with an 8087 math coprocessor chip operating at 4.77 MHz.

Other features of the CAPSIZE program include (1) an automatic shielding optimization algorithm which determines the relative amounts of neutron and gamma shielding in such a way as to meet the specified external dose rate while simultaneously minimizing the overall weight of the loaded cask, and (2) a steady-state heat transfer calculation which will minimize the size and weight of external cooling fins, if and when such fins are required.

2. BASIC CASK DESIGN FEATURES ASSUMED BY THE CAPSIZE PROGRAM

The key physical features of those casks considered in this study are illustrated in Fig. 2.1. In each case there is an inner shell, a gamma shield, an outer shell, a neutron shield, and an outside liner. Depending on the amount of decay heat that must be dissipated, the cask(s) may or may not have fins as illustrated in Fig. 2.2. (Casks with forced circulation cooling systems are not considered.) The inner and outer shells and the outside liner were all assumed to be stainless steel. The thicknesses of these components vary, depending on the type of gamma shielding being used. Assumed thicknesses are shown in Table 2.1. For lead (Pb) casks, these dimensions are comparable to those used in the NLI-10/24 design.[5] For uranium-metal casks, the dimensions are comparable to (but slightly thicker than) those used in the IF-300 design.[6] For iron (Fe) casks, these dimensions are comparable to (but slightly thicker than) those used in an earlier conceptual design of an LMFBR spent fuel shipping cask described in ref. 7. In the latter case, the gamma shield is assumed to be ordinary carbon steel, with a relatively thin inner and outer shell being provided only as a corrosion barrier. In each case, the length of the cavity inside the cask (171.6 in.) was chosen so as to accommodate a 13.8-ft PWR fuel assembly while leaving an additional 6-in. space for an internal axial shock absorber.

Table 2.1. Thicknesses of various structural components for casks with different types of gamma shields

Type of γ shield	Inner shell	Outer shell	Outside liner
	Thickness in inches		
Lead	1.5	2.0	0.75
Iron	0.375	0.375	0.75
U-metal	0.75	2.0	0.75

For any set of design objectives specified by the user, the CAPSIZE program will survey three different types of casks--that is, those with Pb, Fe, and depleted uranium-metal gamma shields. In each case, the neutron shield is assumed to consist of 28.5 vol% water (1.0 g/cc), 66.0 vol% ethylene glycol (1.11 g/cc; $HOCH_2CH_2OH$) and 5.5 vol% potassium tetraborate (1.74 g/cc; $K_2B_4O_7 \cdot 8H_2O$) made with natural boron. By weight, this common mixture of water and antifreeze contains ~1% boron. Depending on the type of cask, the number of fuel assemblies, the fuel's burnup and cooling time, and the desired dose rate 10 ft from the centerline, the program will determine the relative amount of neutron and gamma shielding that should be used in each case so as to minimize the overall weight of the loaded cask.

Inside the cask, the fuel assemblies are assumed to be separated by means of a removable aluminum basket (insert), as illustrated in Fig. 2.3. Each of the 10 positions shown in Fig. 2.3 measures

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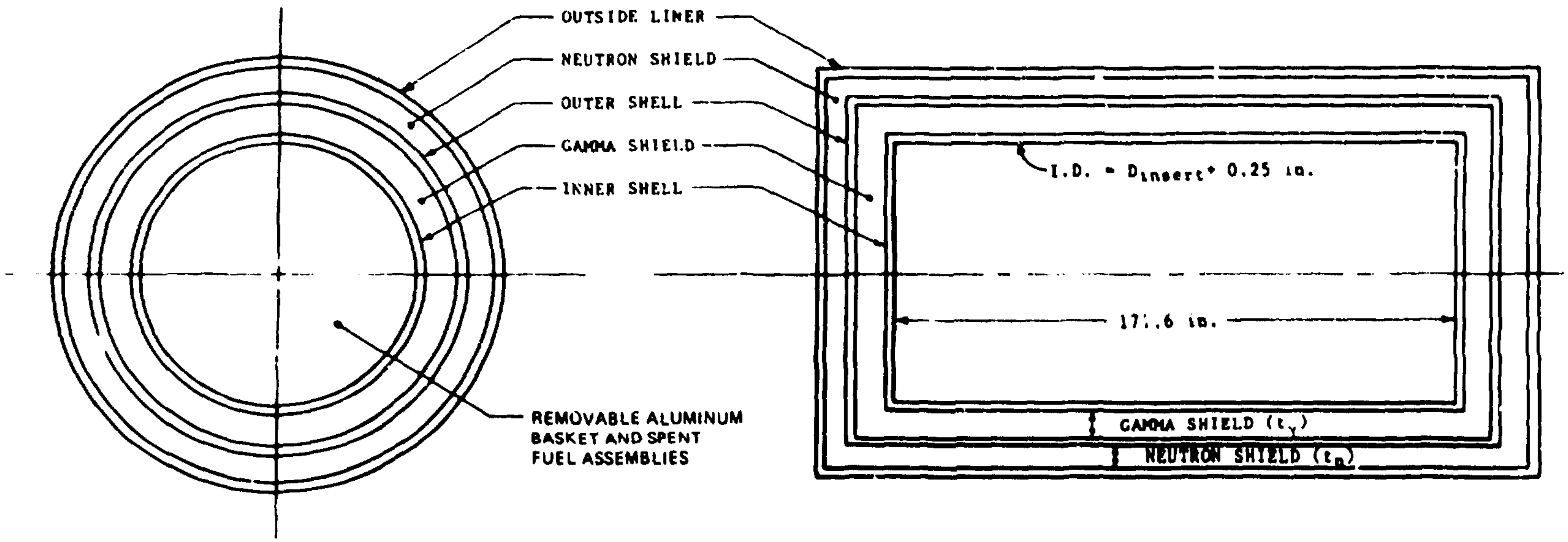

a) The removable aluminum insert containing the spent fuel is more fully illustrated in Fig. 2.3.
b) The thicknesses of the various components in the axial direction were assumed to be the same as in the radial direction.
c) Actual dimensions of the steel shells and outside liner depend on the type of cask, as shown in Table 2.1.

Fig. 2.1. Overview of a typical spent fuel shipping cask.

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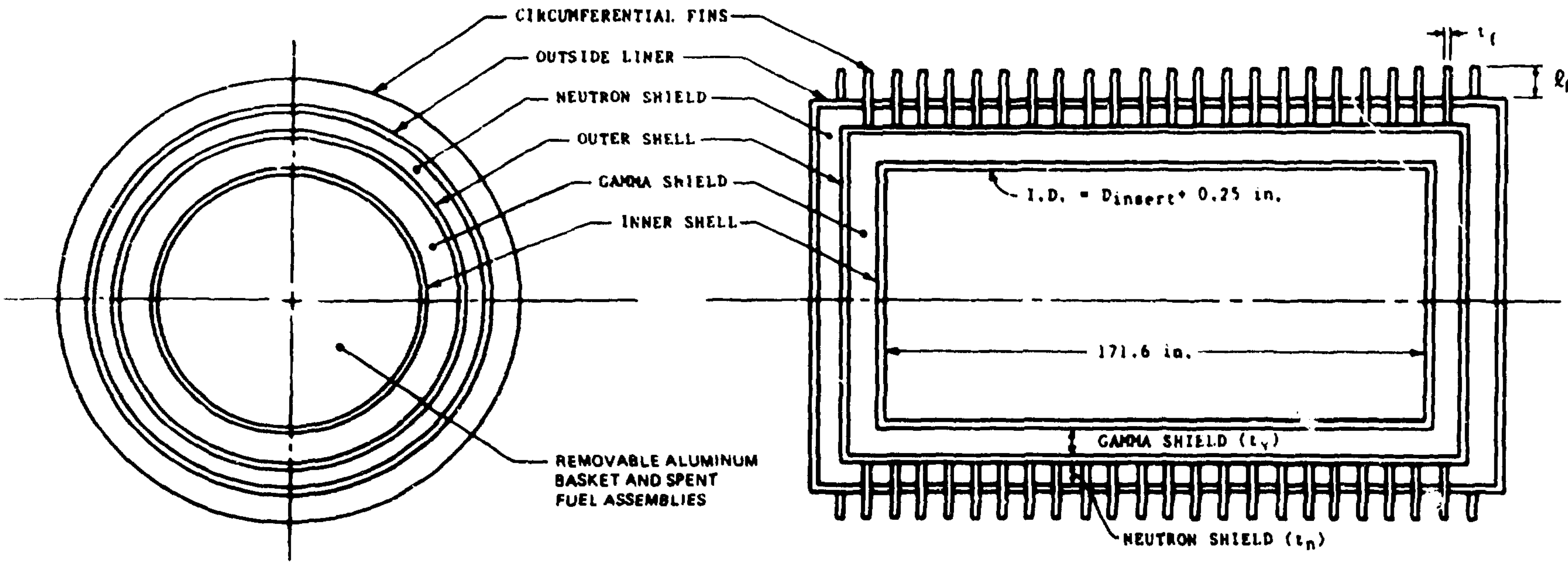

a) The removable aluminum insert containing the spent fuel is more fully illustrated in Fig. 2.3.
b) The thicknesses of the various components in the axial direction were assumed to be the same as in the radial direction.
c) Actual dimensions of the steel shells and outside liner depend on the type of cask, as shown in Table 2.1.
d) Sketch not drawn to scale; when necessary, casks typically have 45-50 fins; actual center-to-center spacing is 4.0 in.

Fig. 2.2. Overview of a spent fuel shipping cask with external cooling fins.

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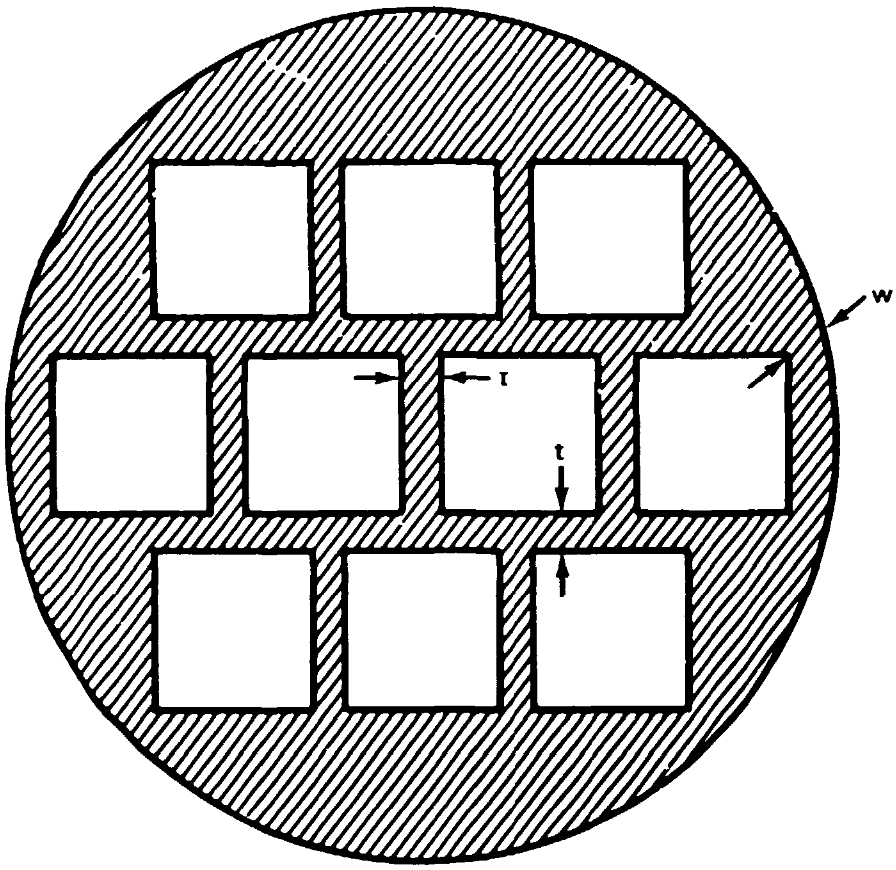

Fig. 2.3. Simplified diagram of a removable aluminum basket designed to hold 10 PWR fuel assemblies. Optimal packing arrangements for additional assemblies are illustrated in Fig. 2.4 a-e.

8.7 in. x 8.7 in. and can accommodate most typical PWR fuel assemblies with 0.125-in. clearance on all four sides. Other "optimal" packing arrangements capable of holding many more fuel assemblies are also considered, as illustrated in Figs. 2.4a-c. In addition to physically separating the fuel assemblies, the basket must provide a low-resistance path by which the decay heat may be carried away from the innermost assemblies and, secondly, it must provide an inherently passive means of ensuring subcriticality under the most reactive conditions conceivable.[8,9]

To avoid the complexities of a forced circulation cooling system and the special licensing considerations associated with a natural convection water-filled cask, it was assumed that the spent fuel would be shipped dry. Most spent fuel shipping casks are, however, loaded under water. Inadvertent loading of the cask with fresh fuel (or fuel with very little accumulated burnup) is then usually taken as the most reactive condition considered in the licensing application.[8,9] Because of the larger payload associated with casks designed to carry older spent fuel assemblies, the problem of ensuring subcriticality becomes more complicated. This question has been examined in some detail and is discussed in Sect. III of ref. 4. In the CAPSIZE program, it is simply assumed that when a user specifies the thickness of the insert between assemblies, he has already made some provision for ensuring the criticality safety of the system under all conditions.

In general, fins will not be used on a cask if it can dissipate the decay heat to the environment (at 130°F) while maintaining an outer surface temperature less than or equal to 250°F. In those cases where fins are necessary, a numerical search is conducted to determine the particular fin dimensions that would satisfy the above criteria while minimizing the overall weight of the loaded cask. In all cases, the circumferential fins were assumed to be spaced on a 4-in. pitch.

3. BASIC DATA USED IN THE CAPSIZE PROGRAM

3.1 ORIGEN-S DATA FOR THE NEUTRON AND GAMMA SOURCE TERMS AND THE DECAY HEAT GENERATION RATE

The possible capacity of a cask designed for a particular type of spent fuel is strongly dependent on the weight limit to which the cask is designed, the amount of neutron and gamma shielding required, and the characteristics of any external cooling fins that might be required to dissipate the internal decay heat load. The optimal neutron and gamma shield thicknesses are calculated by the CAPSIZE program as noted in Sect. 4, and the optimal dimensions of the external cooling fins (when required) are calculated as shown in Sect. 5. The assumed neutron and gamma source terms, as well as the decay heat terms, are based on SAS2/ORIGEN-S[2,3] analyses of PWR assemblies irradiated to 10, 20, 30, 40, 50, and 60 gigawatt days per metric tonne of initial heavy metal (GWD/MTIHM). In each case, values have been tabulated at 17 different decay times between 120 days and 25 years. These decay heat generation rates, neutron source terms, gamma source terms, and gamma energy source terms are listed in Tables 3.1-3.4. The CAPSIZE program then performs a simple 2-D interpolation of these tabulated values to obtain the source terms and decay heat loads for the particular burnup and cooling time specified by the user. Based on graphical analysis of the data, the decay heat generation rate and the gamma source terms were both found to vary as $(ax+b)t^c$ between tabulated data points, where x represents the burnup, t represents the cooling time, and a, b, and c are empirical constants. The neutron source terms, however, were found to vary as ax^be^{ct} between tabulated data points. The remainder of this section briefly describes the SAS2/ORIGEN-S calculations used to generate the tabulated data shown in Tables 3.1-3.4, the assumptions that were used, and possible caveats that the user should be aware of.

The SAS2/ORIGEN-S[2,3] burnup and depletion calculations were based on an infinite lattice of fuel pins from a typical Westinghouse 15x15 fuel assembly. The pitch of the 0.422-in. OD zircalloy

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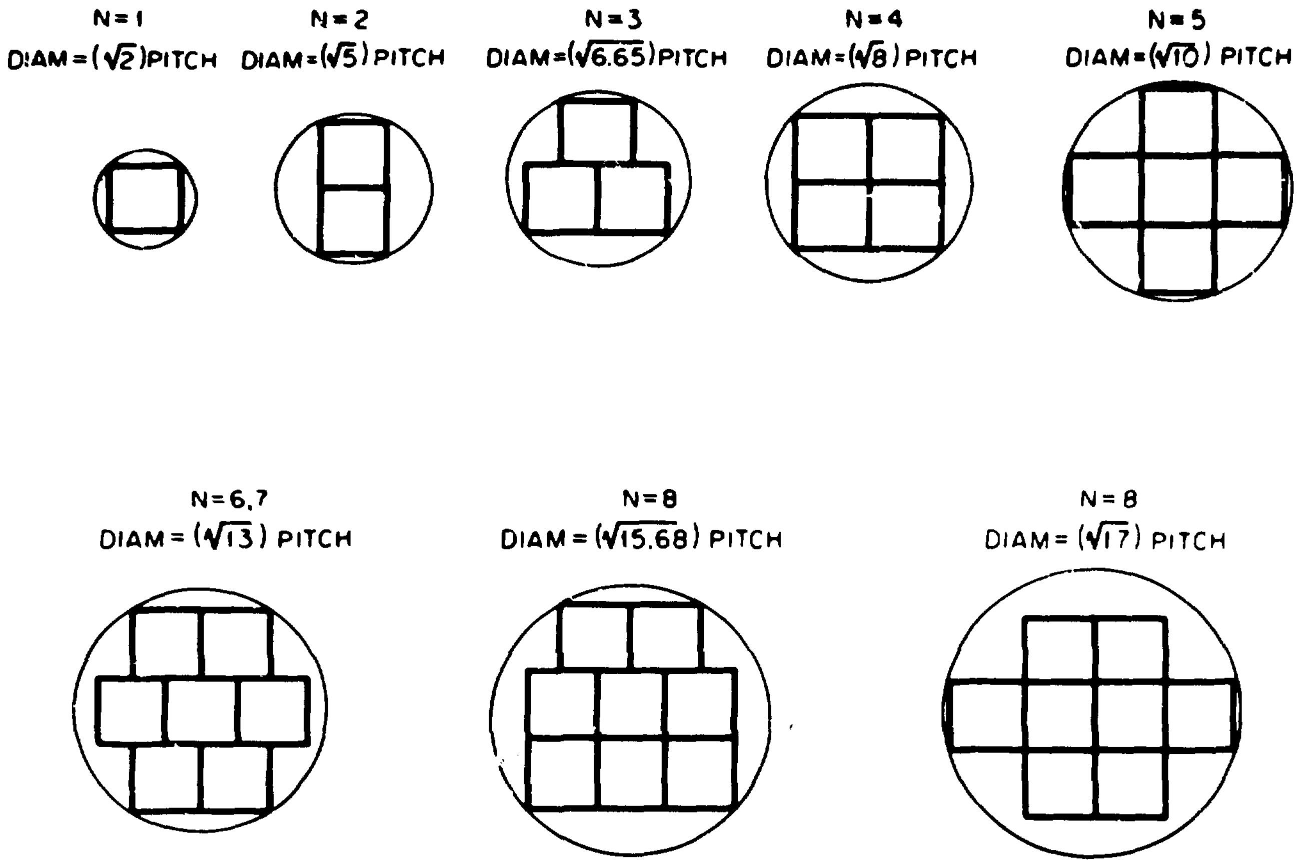

Fig. 2.4a. Optimal packing arrangements for 1-8 square assemblies in a cylindrical cask.

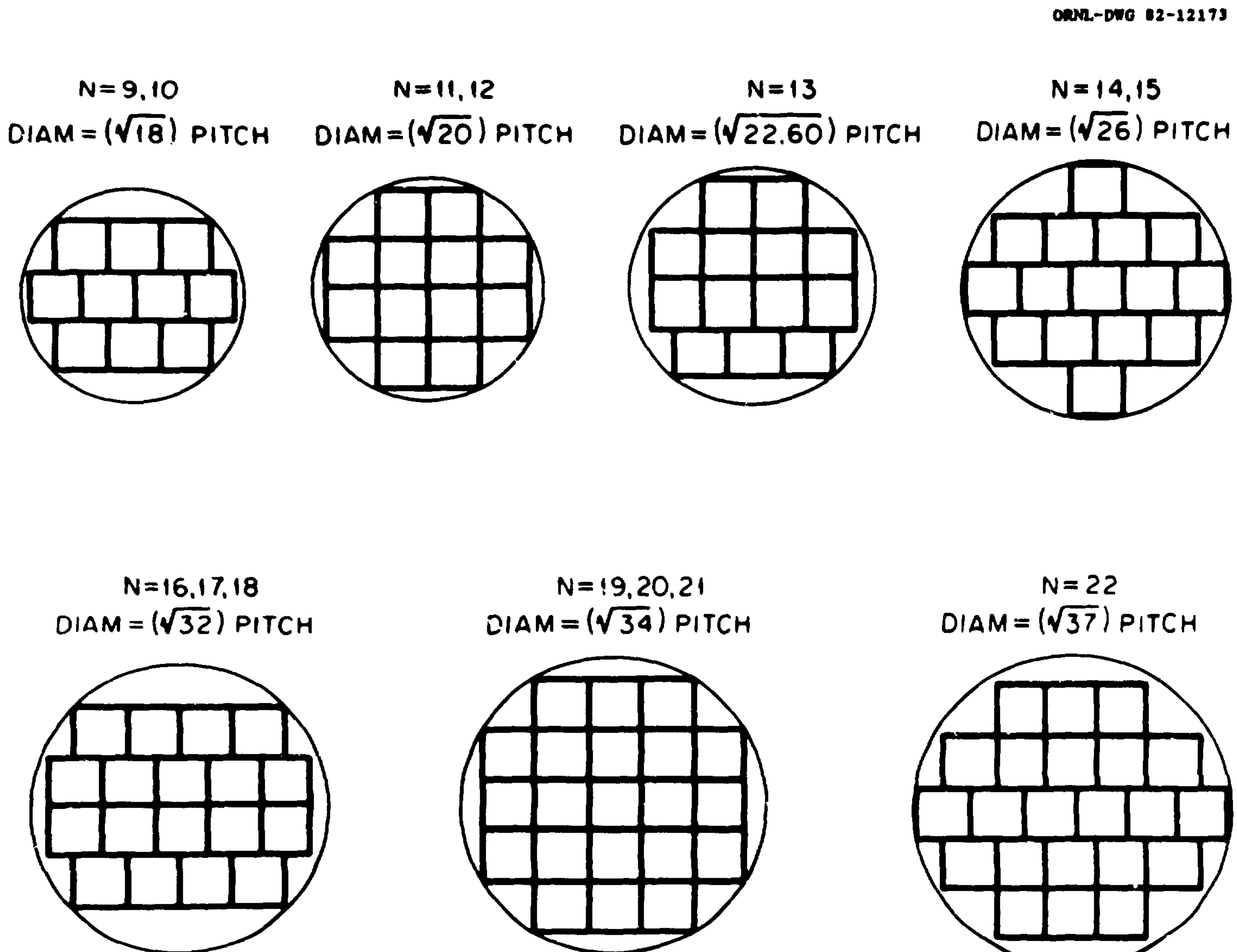

Fig. 2.4b. Optimal packing arrangements for 9-22 square assemblies in a cylindrical cask.

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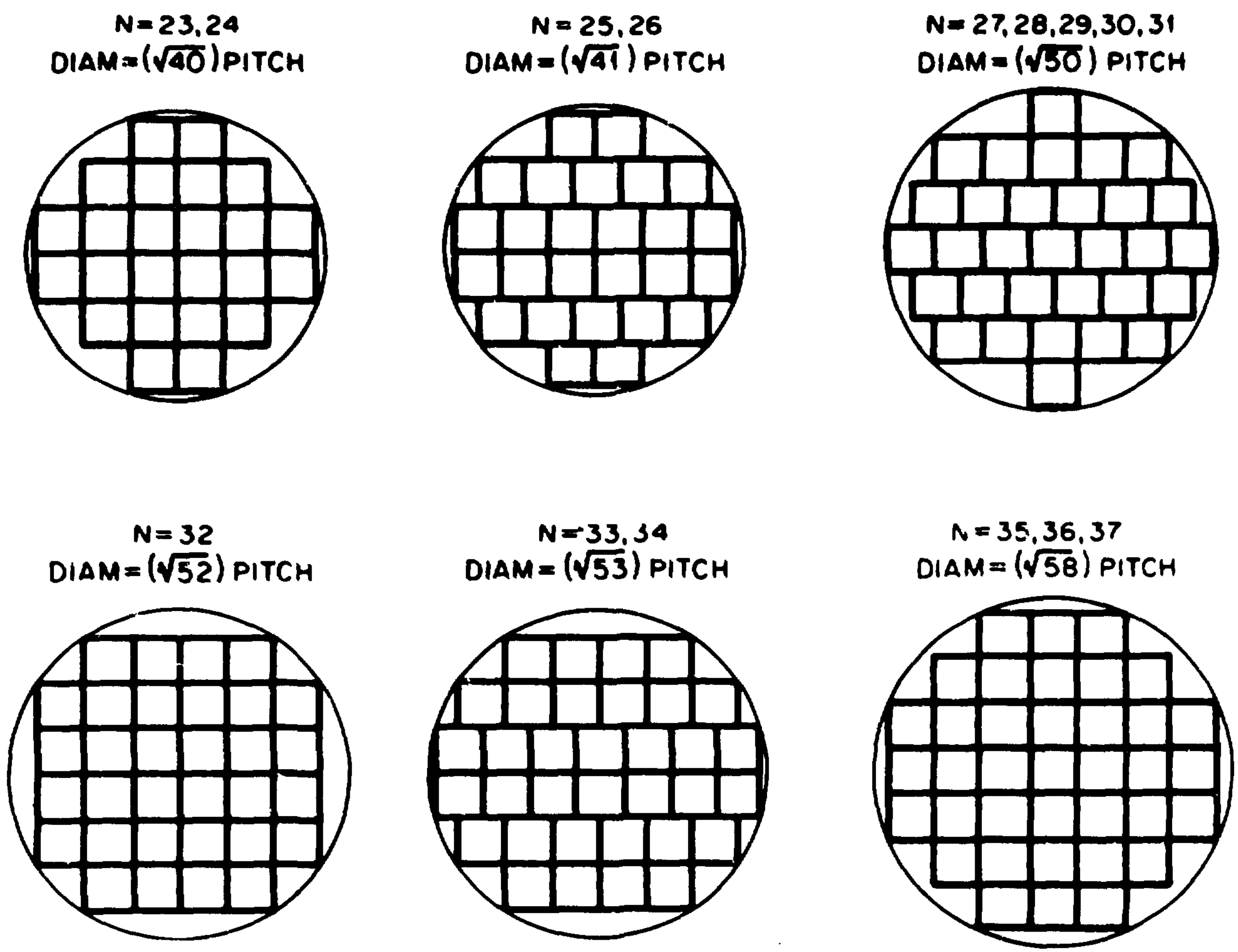

Fig. 2.4c. Optimal packing arrangements for 23-37 square assemblies in a cylindrical cask.

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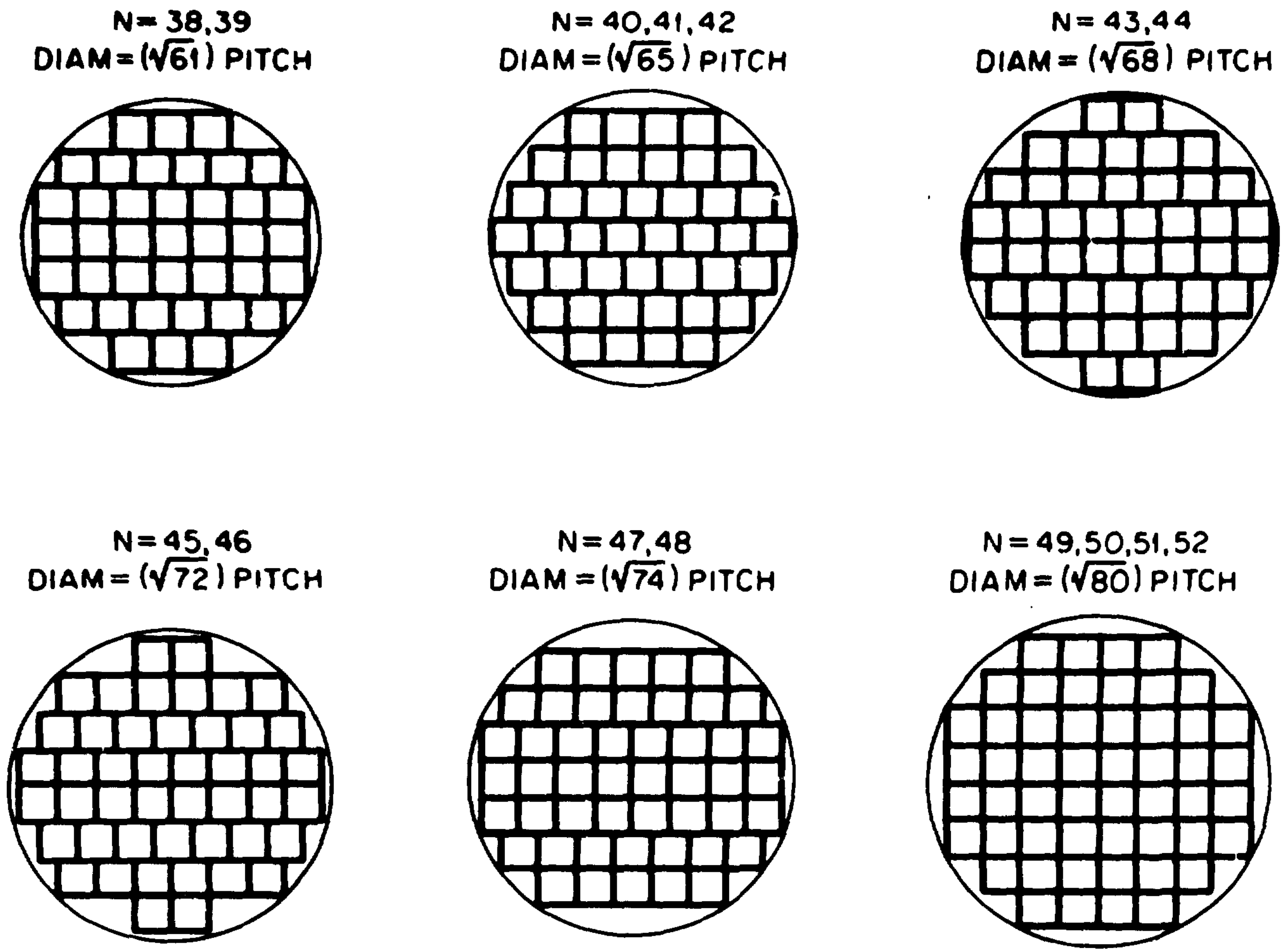

Fig. 2.4d. Optimal packing arrangements for 38-52 square assemblies in a cylindrical cask

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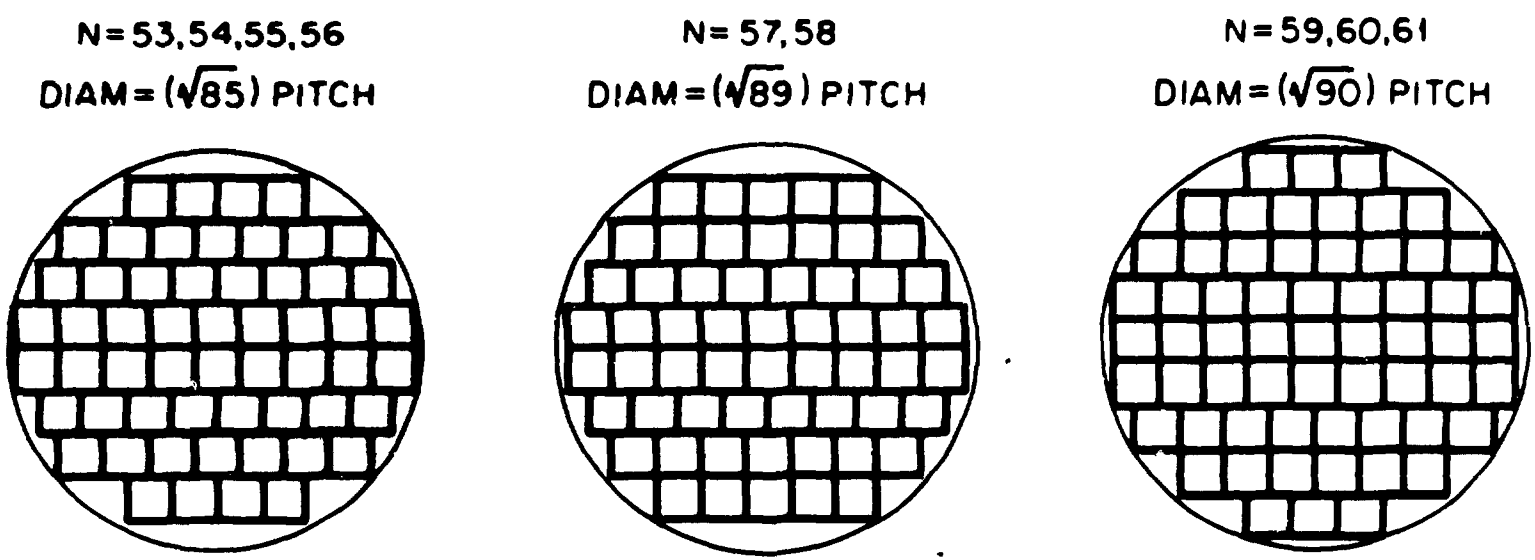

Fig. 2.4e. Optimal packing arrangements for 53-61 square assemblies in a cylindrical cask.

Table 3.1. Decay heat generation rates (watts/MTIHM) for PWR spent fuel as a function of burnup and cooling time[a]

Cooling time (yrs)	Burnup (MWD/MTIHM)					
	10000.	20000.	30000.	40000.	50000.	60000.
0.00	669043.	1286800.	1879250.	2457160.	3029130.	3503140.
0.33	6555.	13628.	21197.	29126.	37351.	45600.
0.41	5514.	11554.	18137.	25081.	32307.	39676.
1.00	2457.	5396.	8788.	12575.	16497.	20802.
2.00	1224.	2758.	4554.	6614.	8895.	11361.
3.00	763.	1722.	2861.	4197.	5712.	7386.
4.00	551.	1234.	2054.	3016.	4133.	5396.
5.00	447.	986.	1628.	2402.	3298.	4319.
7.00	360.	776.	1261.	1836.	2501.	3281.
8.00	340.	725.	1169.	1689.	2306.	3010.
10.00	315.	663.	1055.	1508.	2050.	2674.
13.00	289.	606.	958.	1359.	1827.	2370.
15.00	275.	578.	912.	1288.	1719.	2222.
18.00	258.	544.	855.	1203.	1599.	2053.
20.00	248.	524.	824.	1154.	1529.	1954.
23.00	234.	497.	779.	1091.	1438.	1836.
25.00	226.	481.	753.	1052.	1383.	1756.

[a]Each PWR fuel assembly initially contains 0.4614 metric tonnes of heavy metal.

Table 3.2. Neutron source (neut/sec/MTIHM) for PWR spent fuel as a function of burnup and cooling time[a,b]

Cooling time (yrs)	Burnup (MWD/MTIHM)					
	10000.	20000.	30000.	40000.	50000.	60000.
0.00	2.113E+07	1.638E+08	5.317E+08	1.241E+09	2.442E+09	4.370E+09
0.33	1.352E+07	1.118E+08	4.004E+08	1.018E+09	2.126E+09	3.953E+09
0.41	1.213E+07	1.022E+08	3.761E+08	9.764E+08	2.066E+09	3.871E+09
1.00	5.972E+06	5.974E+07	2.673E+08	7.865E+08	1.786E+09	3.474E+09
2.00	2.732E+06	3.670E+07	2.051E+08	6.696E+08	1.594E+09	3.158E+09
3.00	2.050E+06	3.109E+07	1.867E+08	6.262E+08	1.505E+09	2.979E+09
4.00	1.910E+06	2.921E+07	1.778E+08	5.992E+08	1.441E+09	2.839E+09
5.00	1.258E+06	2.813E+07	1.710E+08	5.763E+08	1.385E+09	2.715E+09
7.00	1.203E+06	2.645E+07	1.590E+08	5.347E+08	1.281E+09	2.494E+09
8.00	1.180E+06	2.567E+07	1.535E+08	5.152E+08	1.233E+09	2.394E+09
10.00	1.137E+06	2.421E+07	1.429E+08	4.785E+08	1.143E+09	2.211E+09
13.00	1.077E+06	2.221E+07	1.286E+08	4.284E+08	1.022E+09	1.969E+09
15.00	1.042E+06	2.099E+07	1.199E+08	3.981E+08	9.482E+08	1.826E+09
18.00	9.926E+05	1.932E+07	1.080E+08	3.568E+08	8.485E+08	1.632E+09
20.00	9.629E+05	1.831E+07	1.008E+08	3.317E+08	7.882E+08	1.515E+09
23.00	9.224E+05	1.692E+07	9.092E+07	2.976E+08	7.060E+08	1.357E+09
25.00	8.978E+05	1.607E+07	8.495E+07	2.769E+08	6.563E+08	1.261E+09

[a]Each PWR fuel assembly initially contains 0.4614 metric tonnes of heavy metal.

[b]Includes the (α,n) neutron source terms as well as the spontaneous fission source terms.

Table 3.3. Gamma source (photons/sec/MTIHM) for PWR spent fuel as a function of burnup and cooling time[a,b,c]

Cooling time (yrs)	Burnup (MWD/MTIHM)					
	10000.	20000.	30000.	40000.	50000.	60000.
0.00	4.5073E+18	8.8470E+18	1.3106E+19	1.7321E+19	2.1546E+19	2.5784E+19
0.33	4.3467E+16	8.9069E+16	1.3669E+17	1.8600E+17	2.3666E+17	2.8809E+17
0.41	3.5746E+16	7.3938E+16	1.1435E+17	1.5657E+17	2.0020E+17	2.4463E+17
1.00	1.4457E+16	3.1805E+16	5.1554E+16	7.3209E+16	9.6274E+16	1.2014E+17
2.00	7.0732E+15	1.6166E+16	2.6952E+16	3.9075E+16	5.2178E+16	6.5832E+16
3.00	4.3753E+15	1.0103E+16	1.6991E+16	2.4790E+16	3.3252E+16	4.2083E+16
4.00	3.1181E+15	7.1509E+15	1.1987E+16	1.7450E+16	2.3361E+16	2.9520E+16
5.00	2.4910E+15	5.6081E+15	9.2891E+15	1.3403E+16	1.7820E+16	2.2403E+16
7.00	1.9507E+15	4.2032E+15	6.7422E+15	9.4906E+15	1.2374E+16	1.5330E+16
8.00	1.8175E+15	3.8460E+15	6.0811E+15	8.4612E+15	1.0929E+16	1.3442E+16
10.00	1.6444E+15	3.3906E+15	5.2448E+15	7.1658E+15	9.1173E+15	1.1081E+16
13.00	1.4768E+15	2.9805E+15	4.5222E+15	6.0768E+15	7.6244E+15	9.1635E+15
15.00	1.3905E+15	2.7846E+15	4.1934E+15	5.5986E+15	6.9861E+15	8.3591E+15
18.00	1.2804E+15	2.5461E+15	3.8075E+15	5.0524E+15	6.2726E+15	7.4743E+15
20.00	1.2155E+15	2.4098E+15	3.5925E+15	4.7542E+15	5.8894E+15	7.0049E+15
23.00	1.1271E+15	2.2278E+15	3.3098E+15	4.3671E+15	5.3970E+15	6.4066E+15
25.00	1.0730E+15	2.1178E+15	3.1409E+15	4.1380E+15	5.1076E+15	6.0569E+15

[a]Each PWR fuel assembly initially contains 0.4614 metric tonnes of heavy metal.

[b]Includes contributions from all of the fission products, the actinides, and the light elements. Also includes contributions due to activation of the Fe, Ni, and Co in the structural materials comprising the fuel assemblies.

[c]These values are used directly by the CAPSIZE program as shown in Eq. 3.6 of Sect. 3.2.2.

Table 3.4. Gamma energy source (MeV/sec/MTIHM) for PWR spent fuel as a function of burnup and cooling time[a,b,c]

Cooling time (yrs)	Burnup (MWD/MTIHM)					
	10000.	20000.	30000.	40000.	50000.	60000.
0.00	1.9876E+18	3.8095E+18	5.5338E+18	7.1976E+18	8.8423E+18	1.0489E+19
0.33	1.7577E+16	3.6380E+16	5.6420E+16	7.7525E+16	9.9507E+16	1.2201E+17
0.41	1.3844E+16	2.9053E+16	4.5578E+16	6.3218E+16	8.1752E+16	1.0080E+17
1.00	4.1271E+15	9.8505E+15	1.6966E+16	2.5217E+16	3.4330E+16	4.3936E+16
2.00	2.0900E+15	5.3342E+15	9.5817E+15	1.4639E+16	2.0301E+16	2.6306E+16
3.00	1.5137E+15	3.8357E+15	6.8678E+15	1.0468E+16	1.4490E+16	1.8747E+16
4.00	1.2102E+15	2.9967E+15	5.2947E+15	7.9977E+15	1.0996E+16	1.4160E+16
5.00	1.0315E+15	2.4814E+15	4.3058E+15	6.4226E+15	8.7489E+15	1.1192E+16
7.00	8.3734E+14	1.9066E+15	3.1864E+15	4.6239E+15	6.1688E+15	7.7738E+15
8.00	7.7837E+14	1.7337E+15	2.8504E+15	4.0850E+15	5.3971E+15	6.7526E+15
10.00	6.9458E+14	1.4972E+15	2.3993E+15	3.3695E+15	4.3805E+15	5.4143E+15
13.00	6.1119E+14	1.2803E+15	2.0032E+15	2.7590E+15	3.5304E+15	4.3105E+15
15.00	5.6964E+14	1.1802E+15	1.8290E+15	2.4993E+15	3.1775E+15	3.8602E+15
18.00	5.1864E+14	1.0636E+15	1.6331E+15	2.2148E+15	2.7990E+15	3.3844E+15
20.00	4.8958E+14	9.9940E+14	1.5282E+15	2.0655E+15	2.6036E+15	3.1417E+15
23.00	4.5106E+14	9.1614E+14	1.3943E+15	1.8777E+15	2.3603E+15	2.8419E+15
25.00	4.2797E+14	8.6699E+14	1.3163E+15	1.7693E+15	2.2208E+15	2.6711E+15

[a]Each PWR fuel assembly initially contains 0.4614 metric tonnes of heavy metal.

[b]Includes contributions from all of the fission products, the actinides, and the light elements. Also includes contributions due to activation of the Fe, Ni, and Co in the structural materials comprising the fuel assemblies.

[c]While these values are not used directly by the CAPSIZE program, they can be used in conjunction with the data in Table 3.3 to determine the shift in the average photon energy as a function of cooling time.

clad rods was 0.563 in., the diameter of the 10.43-g/cc UO_2 fuel pellet was 0.366 in., the cladding was assumed to be 0.0243 in. thick, and the uranium initially contained 3.3 wt% ^{235}U. While in the reactor, the coolant water was assumed to have an average density of 0.71 g/cc, and an average (unchanged) boron concentration of 550 wppm in solution. The SAS2 module[2] (employing the NITAWL resonance self-shielding code, the XSDRNPM discrete ordinates cell-averaging code, and the ORIGEN-S depletion code) was then used to generate burnup-dependent cross-section libraries at 0, 15, 25, 35, 45, and 55 GWD/MTIHM, and nuclide concentration files at 10, 20, 30, 40, 50, and 60 GWD/MTIHM. The burnup-dependent cross-section files generated by NITAWL and XSDRNPM at the end of each burnup interval were used by ORIGEN-S in the depletion analysis for the following burnup interval. Resonance self-shielded, cell-averaged data for several of the more absorptive fission products (^{134}Cs, ^{147}Pm, ^{153}Eu, and ^{154}Eu) were also updated periodically. The infinite-lattice neutron multiplication factor (k_∞) and the corresponding fissile atom concentrations at various burnup intervals are shown in Table 3.5. The extensive nuclide concentration files saved at 10, 20, 30, 40, 50, and 60 GWD/MTIHM were used by later stand-alone ORIGEN-S calculations to generate the neutron and gamma source terms, and the decay heat source terms, at various cooling times from 120 days to 25 years.

Table 3.5. Infinite-lattice neutron multiplication factor (k_∞) and fissile atom concentrations for typical PWR fuel pins as a function of burnup

Burnup (GWD/MTIHM)	Pin cell k_∞	Number density (atoms/barn-cm) in UO_2*		
		N^{U-235}	N^{Pu-239}	N^{Pu-241}
0	1.24725	7.77E-4	---	---
5	1.16827	6.49E-4	5.21E-5	1.29E-6
10	---	5.43E-4	9.01E-5	6.87E-6
15	1.09960	4.54E-4	1.09E-4	1.52E-5
20	---	3.77E-4	1.30E-4	2.12E-5
25	1.04728	3.12E-4	1.38E-4	2.95E-5
30	---	2.56E-4	1.47E-4	3.40E-5
35	1.00111	2.09E-4	1.51E-4	4.04E-5
40	---	1.69E-4	1.54E-4	4.32E-5
45	0.962170	1.36E-4	1.56E-4	4.76E-5
50	---	1.09E-4	1.56E-4	4.92E-5
55	0.928163	8.70E-5	1.56E-4	5.20E-5
60	---	6.87E-5	1.56E-4	5.28E-5

*N^U = 2.327E-2 atoms/(barn-cm) initially.

The data shown in Table 3.1-3.4 are good, extremely useful, and probably adequate for the type of simplistic analysis performed by the CAPSIZE program. There are, however, certain deficiencies that should be noted:

1. Most pressurized water reactors operate with a nominal specific power of about 32-38 MW(t)/MTIHM. This nominal limit is dictated by the thermal-hydraulic design of the reactor. Fuel burned to 10 or 20 GWD/MTIHM (as opposed to 33 GWD/MTIHM) may have been in the reactor for less than three full years, or the reactor may have been operating at less than full power over most of that time. Any number of different operating scenarios are possible. Fuel burned to 55 GWD/MTIHM, however, will generally have been operating at or near the maximum specific power (38 MW(t)/MTIHM) for an extended period of time (36-60 months). [Typically, extended burnup fuels would also have a somewhat higher initial enrichment--i.e.,

3.6-4.2 wt% ^{235}U, as opposed to 3.3 wt% ^{235}U.] Ideally, a number of SAS2/ORIGEN-S burnup and depletion calculations would have been performed to investigate the effect of the different realistic scenarios that one could postulate. Because the CAPSIZE program was developed without explicit programmatic funding, maximum use had to be made of already existing data. In this case, the SAS2/ORIGEN-S calculations previously described had been performed assuming that each of the six irradiation cycles (ΔBU = 10 GWD/MTIHM) occurred over a 153-day period followed by a 30-day downtime. This unfortunately resulted in a somewhat unrealistic specific power of about 65.4 MW(t)/MTIHM. While the total number of fission products produced as a function of burnup will still be correct, the shorter than average cycle time affords less decay time for the fission products produced near the beginning of the cycle. As a result, the calculated short-lived gamma source terms during the first few months after shutdown may be slightly higher than in a more realistic scenario. At longer cooling times, the results should agree quite well.

2. Extended burnup fuels would typically have a somewhat higher initial enrichment--i.e., 3.6-4.2 wt% ^{235}U, as opposed to the 3.3 wt% ^{235}U used in the present SAS2/ORIGEN-S calculations. Because of these higher enrichments, lower flux levels would be required to achieve a given specific power density. With lower overall flux levels in these more highly enriched systems, fewer of the higher-order actinides (like ^{244}Cm) will be produced per unit burnup.* This will reduce the (α,n) neutron source following the α-decay of these actinides, as well as the neutron source resulting from the spontaneous fission of these actinides. Since the present SAS2/ORIGEN-S calculations were all based on the use of low enriched fuel (3.3 wt % ^{235}U), the resulting neutron source terms shown in Table 3.2 may be significantly (20-60%) higher than what one might expect from fuels with higher initial enrichments.

3. In a typical PWR fuel assembly, a number of fuel pins will have been removed to permit room for a cluster of small-diameter control rods which, when not inserted in the core, leave empty holes that are normally filled with water. A typical 15x15 fuel assembly, for example, consists of 204 actual fuel pins and 21 water holes. The SAS2/ORIGEN-S calculations described above were based on the nominal pitch between fuel pins in the square lattice, and did not account for the extra water associated with the 21 water holes in each assembly. The resulting neutron spectrum was, therefore, slightly harder than what one might realistically expect. The harder spectrum, in turn, enhances the parasitic capture by the ^{238}U and the other higher-order actinides. Ultimately this approximation may yield ^{244}Cm concentrations that are 30-40% too large.' Thus, compared to reality, the spent fuel neutron source terms shown in Table 3.2 may be too high by a similar amount, even if the initial enrichment was the same as the 3.3 wt % used in these early SAS2/ORIGEN-S analyses.

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*More recent data in Appendix A of Ref. 10 show that after 33 GWD/MTIHM and a cooling time of two years, the ^{244}Cm concentration in PWR fuel initially enriched to 4.0 wt% ^{235}U will be a factor of 1.77 times lower than the ^{244}Cm concentration in PWR fuel initially enriched to 3.0 wt% ^{235}U.

'This estimate is based on calculations using the more recent SAS2H/ORIGEN-S analytic sequence, which accounts for the extra moderation afforded by the water holes in a typical PWR fuel assembly.

3.2 BASIC CROSS-SECTION DATA AND OTHER CORRELATIONS RELATED TO THE NEUTRON AND GAMMA SOURCE TERMS

Effective one-group cross-section data have been developed and are used by the CAPSIZE program to determine the neutron and gamma dose attenuation rates in each of the structural components comprising a typical cask. These cross sections are dependent on the age (i.e., cooling time) of the spent fuel, the type of shield material (Pb, Fe, or U metal), and the nominal thicknesses of the neutron and gamma shields. The one-group data library used by the CAPSIZE program has been derived from the intermediate results of several hundred 1-D multigroup transport calculations for Pb-, Fe-, and U-shielded casks containing different numbers of PWR spent fuel assemblies, with cooling times ranging from 1 to 10 years (cf. Sect. IV and Appendix D of ref. 4). These S_8P_3 discrete ordinates calculations used the DLC-23/CASK cross-section library having 22 neutron groups and 18 gamma groups, as shown in Table 3.6. Section 3.2.1 lists the one-group cross sections derived from these extensive calculations, while Sect. 3.2.3 provides a brief description of the calculational procedure used in the 1-D shielding analyses (including some of the more useful intermediate results). Lastly, Sect. 3.2.4 describes how the current one-group cross sections were derived from the intermediate results of the detailed shielding calculations.

Variations of the neutron and gamma volumetric source terms in spent PWR fuel assemblies are shown in Tables 3.2 and 3.3 as a function of burnup and cooling time. The simplistic shielding calculation performed by the CAPSIZE program, however, is primarily dependent on the neutron and gamma fluxes impingent on the inner wall of the cask. The correlations in Sect. 3.2.2 show how the neutron and gamma dose rates on the inner wall of the cask depend on the actual volumetric source terms, the number of assemblies in the cask, and the spectral hardening that occurs in the gamma source over the first few years following irradiation. The derivations of these correlations are described briefly in Sect. 3.2.4.

3.2.1 Listing of the Basic Cross-Section Data

Table 3.7 shows the macroscopic cross sections (inch^{-1}) used by CAPSIZE for calculating the spatial attenuation of the gamma dose rate in the various components (inner steel shell, gamma shield, outer steel shell, neutron shield, and outside liner) of Pb, Fe, and U-metal casks containing PWR spent fuel that has been out of the reactor for 1, 3, 5, and 10 years. Values at intermediate times are obtained by interpolation, while the values at 1 and 10 years are assumed to apply to fuel that has been out of the reactor for less than 1 year or more than 10 years. The gamma dose rate on the outer surface of the inner steel shell may, for example, be calculated as

$$D^{\gamma}_{outer} = D^{\gamma}_{inner} \exp\{- \Sigma^{i.shell}_{\gamma} t^{i.shell}\} \tag{3.1}$$

where D^{γ}_{inner} is the gamma dose rate on the inner surface, and $t^{i.shell}$ is the thickness of the inner steel shell in inches. The same basic formula applies for attenuation through the outer steel shell, the neutron shield, and the outside liner. Attenuation through the gamma shield, however, is given by a slightly more complicated expression:

$$D^{\gamma}_{outer} = D^{\gamma}_{inner} \exp\{-[\Sigma^{g.shld}_{\gamma,avg} t^{g.shld}_{ref} + \Sigma^{g.shld}_{\gamma,dif} (t_{\gamma} - t^{g.shld}_{ref})]\} \tag{3.2}$$

where D^{γ}_{inner} and D^{γ}_{outer} are the gamma dose rates on the inner and outer surfaces, t_{γ} is the actual thickness of the gamma shield, and $t^{g.shld}_{ref}$ depends on the type of cask as shown in Table 3.7. Since $\Sigma^{g.shld}_{\gamma,dif}$ is less than $\Sigma^{g.shld}_{\gamma,avg}$, the "effective" average cross section decreases as the shield thickness increases. The

Table 3.6. Energy group structure and ANSI standard flux-to-dose conversion factors corresponding to the coupled (22n-18γ) DLC-23/CASK cross-section library

Neutron group	Energy range		ANSI standard dose factor[a]	Gamma group	Energy range		ANSI standard dose factor[b]
1	14.92 MeV --	12.20 MeV	194.49	1	10.00 MeV --	8.00 MeV	8.7716
2	12.20 MeV --	10.00 MeV	159.71	2	8.00 MeV --	6.50 MeV	7.4785
3	10.00 MeV --	8.18 MeV	147.06	3	6.50 MeV --	5.00 MeV	6.3748
4	8.18 MeV --	6.36 MeV	147.73	4	5.00 MeV --	4.00 MeV	5.4136
5	6.36 MeV --	4.96 MeV	153.39	5	4.00 MeV --	3.00 MeV	4.6221
6	4.96 MeV --	4.06 MeV	150.62	6	3.00 MeV --	2.50 MeV	3.9596
7	4.06 MeV --	3.01 MeV	138.92	7	2.50 MeV --	2.00 MeV	3.4686
8	3.01 MeV --	2.46 MeV	128.43	8	2.00 MeV --	1.66 MeV	3.0192
9	2.46 MeV --	2.35 MeV	125.27	9	1.66 MeV --	1.33 MeV	2.6276
10	2.35 MeV --	1.83 MeV	126.32	10	1.33 MeV --	1.00 MeV	2.2051
11	1.83 MeV --	1.11 MeV	128.94	11	1.00 MeV --	0.80 MeV	1.8326
12	1.11 MeV --	0.55 MeV	116.85	12	0.80 MeV --	0.60 MeV	1.5228
13	550 KeV --	111 KeV	65.2090	13	0.60 MeV --	0.40 MeV	1.1725
14	111 KeV --	3.35 KeV	9.1878	14	0.40 MeV --	0.30 MeV	0.87594
15	3350 eV --	583 eV	3.7134	15	0.30 MeV --	0.20 MeV	0.63061
16	583 eV --	101 eV	4.0086	16	0.20 MeV --	0.10 MeV	0.38338
17	101 eV --	29 eV	4.2946	17	0.10 MeV --	0.05 MeV	0.26693
18	29 eV --	10.100 eV	4.4761	18	0.05 MeV --	0.01 MeV	0.93477
19	10.100 eV --	3.060 eV	4.5673				
20	3.060 eV --	1.120 eV	4.5355				
21	1.120 eV --	0.414 eV	4.3701				
22	0.414 eV --	0.010 eV	3.7142				

[a]Flux-to-dose conversion factor given in (μRem/hr)/(neut/sec/cm^2)

[b]Flux-to-dose conversion factor given in (μRem/hr)/(photon/sec/cm^2)

Table 3.7. Macroscopic cross sections (inch^{-1}) for spatial attenuation of the gamma dose rate in the various components of different casks as a function of the spent fuel's cooling time*

Type of cask	$t_{ref}^{g.shld}$	Time (yrs)	$\Sigma_\gamma^{i.shell}$	$\Sigma_{\gamma,avg}^{g.shld}$	$\Sigma_{\gamma,dif}^{g.shld}$	$\Sigma_\gamma^{o.shell}$	$\Sigma_\gamma^{n.shld}$	$\Sigma_\gamma^{o.liner}$
Pb	4.50 in.	1	1.0993	1.5924	1.3114	1.0993	0.1119	0.7558
		3	1.1146	1.6499	1.4057	1.1146	0.1195	0.8015
		5	1.1255	1.7059	1.4876	1.1255	-	-
		10	1.1502	1.7382	1.5437	1.1502	0.1305	0.8634
Fe	10.00 in.	1	1.0855	0.9951	0.8566	1.0855	0.1237	0.8381
		3	1.1003	1.0207	0.9182	1.1003	0.1327	0.9035
		5	1.1103	1.0434	0.9692	1.1103	-	-
		10	1.1351	1.0623	0.9973	1.1351	0.1452	0.9906
U	2.75 in.	1	1.1029	2.9372	2.3494	1.1029	0.1071	0.7404
		3	1.1181	3.0532	2.5170	1.1181	0.1129	0.7771
		5	1.1285	3.1645	2.6952	1.1285	-	-
		10	1.1540	3.2274	2.7873	1.1540	0.1248	0.8465

*$\Sigma_{\gamma,avg}^{g.shld}$ and $\Sigma_{\gamma,dif}^{g.shld}$ are average and differential cross sections for the gamma shield, and $t_{ref}^{g.shld}$ is the reference thickness of the gamma shield to be used with these cross sections in calculating the attenuation through a gamma shield of thickness t_γ. See text for additional details.

effect, of course, is equivalent to using a constant mass attenuation coefficient in conjunction with a dose buildup factor which increases with the thickness of the shield. The object of both approaches is to account for radiation that escapes through the gamma shield after being scattered several times. Because of that, the effect is only noticeable in highly attenuating media of significant thickness.

Table 3.8 shows the macroscopic cross sections (inch^{-1}) used by CAPSIZE for calculating the spatial attenuation of the neutron dose rate in the various components of Pb, Fe, and U-metal casks containing PWR spent fuel that has been out of the reactor for 1, 3, 5, and 10 years. Values at intermediate times are obtained by interpolation, while the values at 1 and 10 years are assumed to apply to fuel that has been out of the reactor for less than 1 year or more than 10 years. The neutron dose rate on the outer surface of the inner shell may, for example, be calculated as

$$D_{outer}^{n} = D_{inner}^{n} \exp\{- \Sigma_n^{i.shell} t^{i.shell}\} \tag{3.3}$$

where D_{inner}^{n} is the neutron dose rate on the inner surface, and $t^{i.shell}$ is the thickness of the inner steel shell in inches. The same basic formula applies for attenuation through the gamma shield, the outer steel shell, and the outside liner. Attenuation through the neutron shield, however, is given by

$$D_{outer}^{n} = D_{inner}^{n} \exp\{-[\Sigma_{n,avg}^{n.shld} t_{ref}^{n.shld} + \Sigma_{n,dif}^{n.shld}(t_n - t_{ref}^{n.shld})]\} \tag{3.4}$$

where D_{inner}^{n} and D_{outer}^{n} are the neutron dose rates on the inner and outer surfaces, t_n is the actual thickness of the neutron shield, and $t_{ref}^{n.shld}$ depends on the type of cask as shown in Table 3.8.

Table 3.8. Macroscopic cross sections (inch^{-1}) for spatial attenuation of the neutron dose rate in the various components of different casks as a function of the spent fuel's cooling time.*

Type of cask	$t_{ref}^{n,shld}$	Time (yrs)	$\Sigma_n^{i,shell}$	$\Sigma_n^{g,shld}$	$\Sigma_n^{o,shell}$	$\Sigma_{n,avg}^{n,shld}$	$\Sigma_{n,dif}^{n,shld}$	$\Sigma_n^{o,liner}$
Pb	4.00 in.	1	0.1530	0.0617	0.1530	0.9266	0.7894	0.1530
		3	0.1546	0.0643	0.1546	0.9140	0.7837	0.1546
		5	---	---	---	0.9019	0.7760	---
		10	0.1565	0.0723	0.1565	0.8871	0.7610	0.1565
Fe	3.75 in.	1	0.1530	0.1530	0.1530	1.1125	1.0113	0.1530
		3	0.1546	0.1546	0.1546	1.0828	0.9791	0.1546
		5	---	---	---	1.0636	0.9624	---
		10	0.1565	0.1565	0.1565	1.0359	0.9236	0.1565
U	3.25 in.	1	0.1530	0.2735	0.1530	0.9083	0.7765	0.1530
		3	0.1546	0.2781	0.1546	0.9001	0.7774	0.1546
		5	---	---	---	0.8918	0.7729	---
		10	0.1565	0.2860	0.1565	0.8802	0.7643	0.1565

*$\Sigma_{n,avg}^{n,shld}$ and $\Sigma_{n,dif}^{n,shld}$ are average and differential cross sections for the neutron shield, and $t_{ref}^{n,shld}$ is the reference thickness of the neutron shield to be used with these cross sections in calculating the attenuation through a neutron shield of thickness t_n. See text for additional details.

3.2.2 Listing of the Correlations Related to the Neutron and Gamma Source Terms

In the CAPSIZE program, the thicknesses of the neutron and gamma shields will be calculated so that, when used in conjunction with the inner and outer steel shells and the outside liner described in Table 2.1, the combined neutron and gamma dose rates 10 ft from the centerline of the cask will be reduced to some prescribed level. The unshielded dose rates with none of these five components present are assumed to vary as

$$D_n^o = (100.0 \text{ mrem/hr}) \frac{S_n(B,T)}{S_n(B_o,T_o)} \frac{F_n(N)}{F_n(N_n)} \tag{3.5a}$$

and

$$D_\gamma^o = (1.376E+6 \text{ mrem/hr}) \frac{S_\gamma(B,T)}{S_\gamma(B_o,T_o)} \frac{F_\gamma(N)}{F_\gamma(N_\gamma)} \frac{\chi(T)}{\chi(T_\gamma)} \tag{3.5b}$$

where B is the burnup of the spent fuel (MWD/MTIHM), T is the cooling time in years, N is the number of PWR assemblies in the cask, $S_n(B,T)$ is the number of neutrons/sec/MTIHM in the spent fuel as given by the ORIGEN data, $S_\gamma(B,T)$ is the number of photons/sec/MTIHM in the spent fuel as given by the ORIGEN data, $S_n(B_o,T_o)$ and $S_\gamma(B_o,T_o)$ are reference values at B_o = 33,000 MWD/MTIHM and T_o = 10 years,

$$F_n(N) = N^{0.73643} \tag{3.6a}$$

is a geometric/self-attenuation factor for the neutrons with $F_n(N_n) = 9.413$ being a reference value based on $N_n = 21$ assemblies,

$$F_\gamma(N) = \sqrt{N} + (0.0919/\sqrt{N})$$

is a geometric/self-attenuation factor for the gammas with $F_\gamma(N_\gamma) = 3.491$ being a reference value based on $N_\gamma = 12$ assemblies, and $\chi(T)$ is an empirical factor that accounts for spectral hardening of the gamma source in fuel that has cooled for 5-7 years. This spectral correction factor is given by

$$\chi(T) = 1.0 \qquad T < 1 \text{ year} \tag{3.7a}$$
$$= 1.101429 - (0.17533)T + (0.073905)T^2 \qquad 1 \text{ year} < T < 2.5 \text{ years} \tag{3.7b}$$
$$= 0.465568 + (0.3567)T - (0.037173)T^2 \qquad 2.5 \text{ years} < T < 5 \text{ years} \tag{3.7c}$$
$$= 1.405376 - (0.003452)T - (0.002734)T^2 \qquad 5 \text{ years} < T < 10 \text{ years} \tag{3.7d}$$
$$= 1.09744 \qquad T > 10 \text{ years}, \tag{3.7e}$$

with a reference value of $\chi(T_\gamma) = 1.09744$ at $T_\gamma = 10$ years. The bases for Eqs. 3.5-3.7 are described in Sects. 3.2.4.4, 3.2.4.7, and 3.2.4.9.

3.2.3 Brief Description of the Numerous 1-D Shielding Calculations From Which the Present Data Was Distilled

The CAPSIZE program was originally developed as an independent effort, without explicit programmatic funding. All of the data and correlations described in Sects. 3.2.1 and 3.2.2 were therefore distilled from the intermediate results of a large number of shielding calculations previously performed in 1980 for another project (cf. ref. 4). At that time, over 2600 1-D shielding calculations were performed for Pb, Fe, and U-metal casks containing 1 to 26 PWR fuel assemblies that had been out of the reactor for 1, 2, 3, 5, 7, and 10 years. These were all S_8P_3 discrete ordinates calculations using 22 neutron groups and 18 gamma groups. In each case, the inner cavity of the (dry) cask was modeled as an homogenized region consisting of the prescribed number of spent PWR fuel assemblies (burnup = 33,000 MWD/MTIHM) with a 1-in.-thick aluminum insert between the assemblies. Each of several hundred series of calculations were conducted in the following fashion:

1. The neutron source terms in the homogenized fuel region were set to zero, leaving only the gamma source terms. Also, the thickness of the neutron shield and the outside liner were both set to zero.

2. Given configuration 1, the thickness of the gamma shield was varied until some prescribed dose rate, called the initial design point (D_γ), was achieved at a point 10 ft from the centerline. In a typical zone width search, the prescribed dose rate ($\pm$ 1%) could usually be obtained after 4 or 5 iterations. [Each of these "iterations" required a completely converged 1-D shielding calculation for the 18 gamma groups.] In the first iteration, the gamma shield was assumed to be 0.001 in.

thick, while in the second it was assumed to be 2.0 in. thick. In subsequent iterations, the thickness of the gamma shield was adjusted as necessary until the prescribed gamma dose rate (D'_γ) was achieved [cf. Table 3.9, iteration numbers 1-5].

Table 3.9. Intermediate results from a series of 1-D shielding calculations for a Pb cask containing 21 10-year-old PWR fuel assemblies (with D'_γ set at 30 mrem/hr)

Iteration number	Gamma shield thickness (inches)	Neutron shield thickness (inches)	Total dose at 10 ft. (mrem/hr)	Total γ dose at 10 ft. (mrem/hr)	Total n dose at 10 ft. (mrem/hr)
1	0.0010	0.0	33323.4	33323.4	0.0
2	2.0000	0.0	659.251	659.251	0.0
3	3.5745	0.0	54.9784	54.9784	0.0
4	3.9584	0.0	30.5099	30.5099	0.0
5	3.9694	0.0	30.0392	30.0392	0.0
6	3.9694	0.0010	54.8724	15.6979	39.1746
7	3.9694	2.0000	17.4944	12.1776	5.31679
8	3.9694	2.9780	12.9972	10.6589	2.33834
9	3.9694	3.8408	10.6809	9.48553	1.19536
10	3.9694	4.1304	10.0366	9.07304	0.96354

3. Having established the thickness of the gamma shield, the second phase of the calculational procedure could begin. In this phase, both the neutron and gamma source terms were assumed to be present in the homogenized fuel region. The neutron shield and outside liner were also included in the calculational model.

4. Given configuration 3, the thickness of the neutron shield was varied until the total combined dose rate 10 ft from the centerline was reduced to 10 mrem/hr ($\pm 1\%$). This value includes the dose due to neutrons and secondary gammas as well as that due to primary gammas. The prescribed dose rate of 10 mrem/hr could usually be obtained in 4 or 5 iterations. [Each of these "iterations" required a completely converged 1-D shielding calculation for all 22 neutron energy groups and all 18 gamma groups.] In the first iteration, the neutron shield was assumed to be 0.001 in. thick, while in the second it was assumed to be 2.0 in. thick. In subsequent iterations, the thickness of the neutron shield was adjusted as necessary until the prescribed total dose rate (10 mrem/hr) was achieved [cf. Table 3.9, iteration numbers 6-10].

Table 3.9 shows the intermediate results from one series of 1-D shielding calculations for a Pb cask carrying 21 10-year-old PWR fuel assemblies. Here the initial design point (D'_γ) was set at 30 mrem/hr. Altogether, this calculational procedure was repeated for over 268 cases of interest, each requiring 8-12 separate 1-D shielding calculations. In many cases, alternate initial design points were used. These ranged from 5 mrem/hr to 300 mrem/hr. Others involved different types of casks (Pb, Fe, or U-metal), a different number of fuel assemblies per cask ($1 \leq N \leq 26$), or spent fuel at different cooling times. Tables 3.10-3.18, taken from Appendix D of ref. 4, show the intermediate and final results for 101 cases of interest involving Pb, Fe, and U-metal casks containing 13, 15, or 21 PWR fuel assemblies that have been out of the reactor for 1, 3, or 10 years. With no neutron source, no neutron shield, and no outside liner, the indicated gamma shield thickness caused the gamma dose rate 10 ft from the centerline to be the same as the initial design point. Including the neutron source and the outside liner, but setting the thickness of the neutron shield to 0.001 in., the neutron and gamma dose rates 10 ft from the centerline were found to be the same as the values shown for the "accidental" neutron and gamma dose rates.* With the neutron and gamma shield thicknesses shown, the "nominal" neutron and gamma dose rates were obtained 10 ft from the centerline.

The data in Tables 3.10-3.18, coupled with the other 267 sets of unpublished intermediate results similar to those shown in Table 3.9, formed a large data base from which the one-group cross sections in Sect. 3.2.1 and the correlations in Sect. 3.2.2 were derived. The derivations of those cross sections and correlations are reviewed briefly in Sect. 3.2.4.

Figures 3.1, 3.2, and 3.3 are based on the data in Tables 3.10-3.18 and show the overall loaded weight of different casks as a function of the initial design point (D'_γ). Based on these results, Table 3.19 shows recommended values of the initial design point (D'_γ) that will minimize the weight of Pb, Fe, and U-metal casks carrying 1-, 2-, 3-, 5-, 7-, or 10-year-old spent fuel. As noted in Sect. 4, these values are used by the CAPSIZE program to determine the relative amount of neutron and gamma shielding to be used in meeting the prescribed external dose rate while minimizing the overall weight of a loaded cask.

Multigroup S_8P_3 discrete ordinates shielding calculations like those described above were also performed to determine the optimal neutron and gamma shield thicknesses for Pb, Fe, and U-metal casks as a function of the number of spent fuel assemblies in the cask. Using the optimal initial design points shown in Table 3.19, these assessments were made for 1-, 2-, 3-, 5-, 7-, and 10-year-old spent fuel. The final results are summarized in Tables 3.20-3.25. Many of these results were subsequently used to develop correlations for the parametric variation of the unshielded neutron and gamma dose rates as a function of the number of assemblies in a cask.

3.2.4 Origin of the Basic Cross-Section Data and the Correlations Related to the Neutron and Gamma Source Terms

3.2.4.1 Determination of $\Sigma_\gamma^{o.liner}$

Table 3.16 shows the intermediate and final results for a Pb cask carrying 21 10-year-old PWR assemblies. The optimal initial design point (D'_γ) corresponds to 30 mrem/hr. [The optimal initial design points for various cask/fuel combinations are listed in Table 3.19. Alternately, from Table 3.16, one can see that the optimal initial design point corresponds to that which minimizes the total weight of the loaded cask.] This implies that the inner and outer steel shells, together with the corresponding gamma shield, were able to reduce the gamma dose rate down to 30 mrem/hr at a point 10 ft from the centerline. If one included the 0.75-in.-thick outside liner, but no neutron shield, the corresponding (accidental) gamma dose rate at this point would be 15.7 mrem/hr [see Table 3.16, or iterations 5 and 6 in Table 3.9]. The corresponding cross section is then given by

$$\Sigma_\gamma^{o.liner} = [\ln(30.0/15.7)]/(0.75 \text{ inch}) = 0.8634/\text{inch}. \quad (3.8)$$

*This term is used throughout this report to denote the dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.10. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for a Pb cask containing 13 1-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
5	7.21	10	1.80	76.33	2.69	7.31	2.99	49.8
7	6.95	10	1.96	74.45	3.54	6.43	4.12	50.5
12	6.54	10	2.39	71.65	5.49	4.54	6.93	51.8
20	6.14	10	3.34	69.61	7.95	2.10	11.4	52.9
25	5.97	10	4.11	69.13	8.93	1.17	14.2	53.5
30	5.83	10	5.15	69.21	9.40	0.565	17.0	53.9
37	5.67	10	6.52	69.53	9.82	0.242	20.9	54.4
45	5.52	10	7.98	70.13	9.90	0.109	25.4	54.9
75	5.13	10	11.98	72.31	10.01	0.0173	42.3	56.1
110	4.84	10	15.02	74.44	10.01	0.0052	61.8	57.1
150	4.61	10	17.45	76.39	10.00	0.0022	84.1	57.9

[a]Radius of cask inner cavity = 58.6 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.11. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for an Fe cask containing 13 1-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
5	14.46	10	0.38	89.10	2.71	7.24	2.78	12.4
9	13.75	10	0.72	84.73	4.62	5.36	4.93	13.9
13	13.31	10	1.10	82.28	6.29	3.71	7.10	14.9
20	12.80	10	2.00	80.04	8.49	1.49	10.8	16.2
25	12.54	10	2.85	79.34	9.43	0.631	13.4	16.8
30	12.32	10	3.94	79.26	9.81	0.219	16.1	17.4
37	12.08	10	5.41	79.51	9.98	0.0608	19.8	18.1
50	11.72	10	7.69	80.24	10.00	0.0124	26.6	19.2
85	11.10	10	11.70	81.98	10.00	0.0019	44.9	21.1
150	10.45	10	15.91	84.46	10.00	0.0004	78.6	23.4
300	9.66	10	20.91	88.26	9.98	0.0001	155.3	26.5

[a]Radius of cask inner cavity = 58.6 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.12. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for a U-metal cask containing 13 1-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
3	4.45	10	1.03	67.26	1.72	8.36	1.79	25.1
5	4.23	10	1.24	65.05	2.71	7.36	2.94	26.8
10	3.93	10	1.75	62.31	4.94	5.04	5.82	29.2
16	3.72	10	2.50	60.83	7.17	2.82	9.27	31.0
20	3.63	10	3.15	60.40	8.26	1.74	11.6	31.9
25	3.53	10	4.18	60.36	9.13	0.857	14.4	32.8
30	3.45	10	5.30	60.63	9.57	0.436	17.2	33.5
37	3.36	10	6.84	61.27	9.81	0.196	21.2	34.4
45	3.28	10	8.31	61.96	9.96	0.102	25.8	35.3
70	3.09	10	11.93	64.20	10.04	0.0247	40.0	37.2
90	2.98	10	14.05	65.76	9.98	0.0116	52.0	38.5
150	2.76	10	18.09	69.22	10.01	0.0031	85.8	40.9

[a]Radius of cask inner cavity = 58.6 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.13. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for a Pb cask containing 15 3-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
7	6.00	10	1.58	72.26	3.46	6.57	3.96	35.3
13	5.54	10	2.11	69.26	5.76	4.21	7.23	36.2
19	5.27	10	2.78	67.81	7.62	2.40	10.5	36.8
23	5.13	10	3.36	67.35	8.53	1.52	12.7	37.1
28	4.99	10	4.21	67.18	9.25	0.795	15.4	37.5
34	4.85	10	5.34	67.36	9.68	0.369	18.7	37.8
41	4.72	10	6.61	67.78	9.82	0.172	22.4	38.0
48	4.60	10	7.76	68.29	9.90	0.0928	26.2	38.3
59	4.46	10	9.22	68.97	10.02	0.0459	32.1	38.7
70	4.33	10	10.52	69.71	10.02	0.0259	38.1	38.9
130	3.90	10	15.11	72.83	9.99	0.0044	70.3	40.0

[a]Radius of cask inner cavity = 62.8 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.14. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for an Fe cask containing 15 3-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
8	12.45	10	0.47	81.99	4.03	5.94	4.22	11.2
12	12.00	10	0.81	79.41	5.70	4.28	6.27	12.1
19	11.49	10	1.55	76.99	8.00	2.05	9.78	13.1
23	11.28	10	2.14	76.32	8.93	1.13	11.8	13.5
27	11.10	10	2.84	76.01	9.48	0.565	13.8	13.9
32	10.92	10	3.79	75.95	9.80	0.232	16.3	14.3
39	10.70	10	5.09	76.15	9.94	0.0766	19.8	14.8
45	10.54	10	6.08	76.41	9.97	0.0368	22.8	15.2
56	10.30	10	7.61	76.91	10.00	0.0142	28.3	15.8
75	9.99	10	9.66	77.73	9.98	0.0052	37.8	16.6
140	9.32	10	13.96	79.90	9.99	0.0011	69.7	18.5
300	8.51	10	19.08	83.39	9.99	0.0003	147.6	21.0

[a]Radius of cask inner cavity = 62.8 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.15. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for a U-metal cask containing 15 3-year-old PWR fuel asemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
3	3.88	10	0.79	66.34	1.68	8.32	1.74	19.5
6	3.59	10	1.07	63.42	3.13	6.88	3.42	21.2
11	3.34	10	1.57	61.14	5.30	4.70	6.21	22.8
16	3.19	10	2.20	60.05	7.07	2.84	8.97	23.8
20	3.10	10	2.76	59.58	8.22	1.85	11.2	24.4
24	3.02	10	3.57	59.49	9.03	1.03	13.6	25.0
30	2.93	10	4.82	59.74	9.62	0.461	17.0	25.6
37	2.85	10	6.34	60.39	9.78	0.202	20.8	26.3
46	2.76	10	7.88	61.13	9.96	0.0993	25.8	26.9
80	2.55	10	12.04	63.74	9.98	0.0196	44.4	28.7
130	2.36	10	15.68	66.59	9.99	0.0057	71.6	30.3

[a]Radius of cask inner cavity = 62.8 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.16. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for a Pb cask containing 21 10-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask[a] in metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
5	5.15	10	1.48	77.21	2.43	7.66	2.71	36.4
8	4.84	10	1.70	74.78	3.59	6.40	4.26	37.1
13	4.52	10	2.10	72.50	5.38	4.62	6.88	37.8
20	4.24	10	2.80	70.89	7.36	2.62	10.5	38.5
25	4.09	10	3.40	70.34	8.41	1.63	13.1	38.9
30	3.97	10	4.13	70.18	9.07	0.964	15.7	39.2
37	3.83	10	5.25	70.34	9.55	0.460	19.3	39.5
45	3.71	10	6.43	70.73	9.78	0.229	23.4	39.8
70	3.42	10	9.38	72.13	9.99	0.0537	36.3	40.6
150	2.92	10	14.62	75.64	9.98	0.0069	77.7	41.9

[a]Radius of cask inner cavity = 71.8 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.17. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for an Fe cask containing 21 10-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask in[a] metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
5	11.35	10	0.40	87.28	2.40	7.58	2.47	13.1
8	10.87	10	0.63	84.19	3.67	6.29	3.92	14.1
13	10.38	10	1.01	81.28	5.53	4.44	6.33	15.3
20	9.94	10	1.67	79.11	7.67	2.38	9.64	16.4
25	9.72	10	2.25	78.34	8.66	1.35	12.0	17.0
30	9.54	10	2.96	77.96	9.28	0.701	14.4	17.5
40	9.25	10	4.46	77.91	9.84	0.194	19.1	18.3
50	9.02	10	5.84	78.21	9.93	0.0715	23.8	18.9
68	8.71	10	7.79	78.82	9.98	0.0231	32.2	19.9
90	8.43	10	9.59	79.51	10.01	0.0102	42.6	20.8
150	7.93	10	12.83	81.09	9.99	0.0031	70.4	22.6
300	7.24	10	17.10	83.69	10.04	0.0009	139.9	25.2

[a]Radius of cask inner cavity = 71.8 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

Table 3.18. Alternate sets of shielding thicknesses and the subsequent neutron and gamma dose rates for a U-metal cask containing 21 10-year-old PWR fuel assemblies

Initial design point (mrem/hr)	Resulting gamma shield thickness (inches)	Final design point (mrem/hr)	Resulting neutron shield thickness (inches)	Total mass of loaded cask in[a] metric tonnes	Nominal gamma dose 10 ft from centerline (mrem/hr)	Nominal neut. dose 10 ft from centerline (mrem/hr)	Accidental gamma dose[b] 10 ft from centerline (mrem/hr)	Accidental neut. dose[b] 10 ft from centerline (mrem/hr)
5	3.04	10	1.14	68.23	2.48	7.59	2.74	24.5
8	2.86	10	1.40	66.40	3.75	6.29	4.34	25.8
12	2.72	10	1.78	64.97	5.25	4.72	6.45	26.9
15	2.64	10	2.09	64.29	6.27	3.73	8.05	27.6
20	2.53	10	2.67	63.62	7.66	2.40	10.7	28.4
25	2.45	10	3.42	63.41	8.62	1.40	13.3	29.1
30	2.39	10	4.27	63.52	9.14	0.801	15.9	29.6
37	2.31	10	5.42	63.85	9.67	0.413	19.7	30.3
50	2.20	10	7.38	64.77	9.91	0.158	26.5	31.3
80	2.03	10	10.76	66.90	9.97	0.0399	42.8	32.8
150	1.81	10	15.02	70.24	9.98	0.0089	78.8	35.0

[a]Radius of cask inner cavity = 71.8 cm; length of cavity = length of active fuel = 12 ft.

[b]Dose rates that would exist if the borated water and ethylene glycol mixture comprising the neutron shield was no longer present due to some postulated accident.

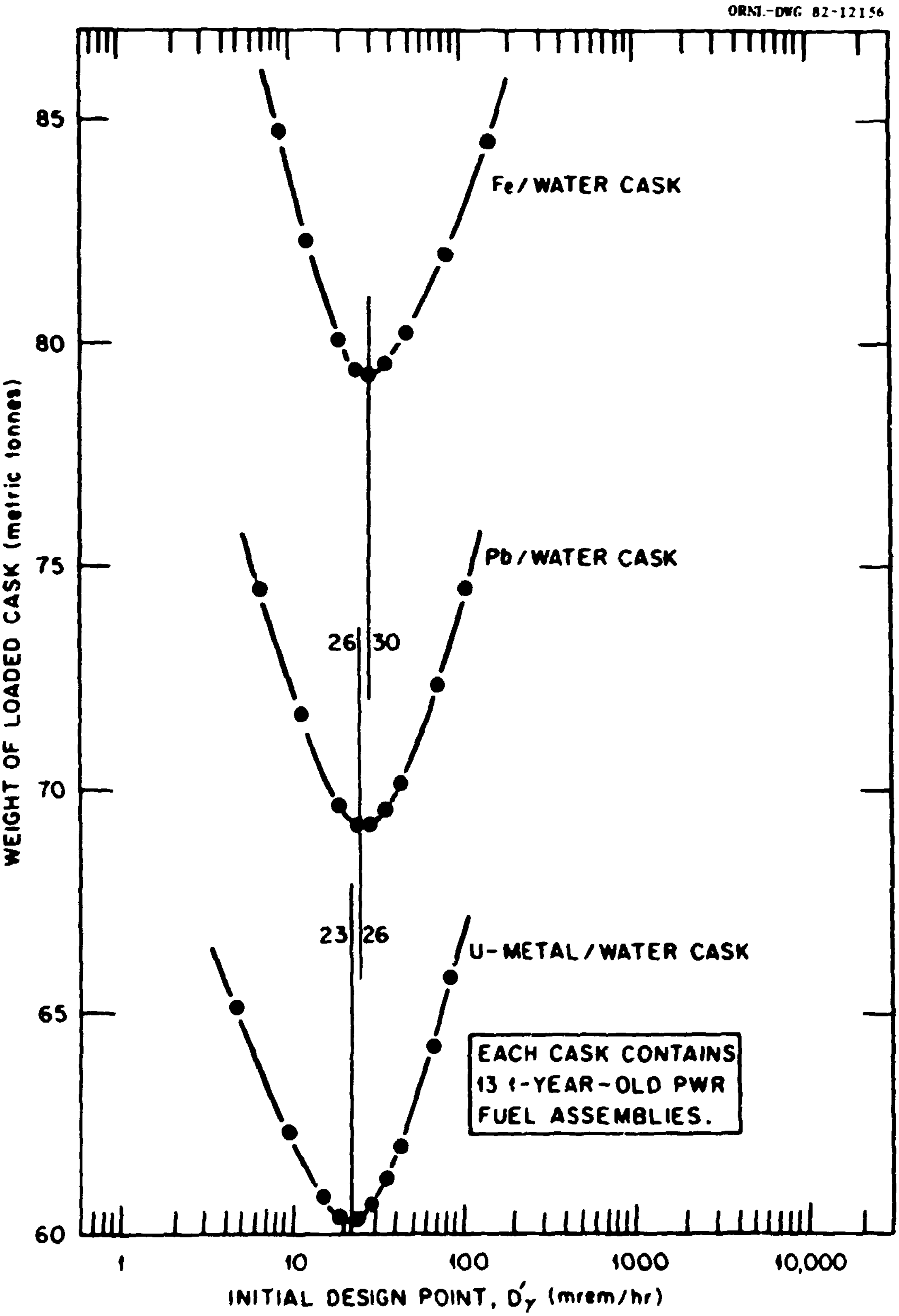

Fig. 3.1. Weight of Pb, Fe, and U-metal casks for 1-year-old PWR spent fuel, as a function of the initial design point used to determine the relative amount of neutron and gamma shielding.

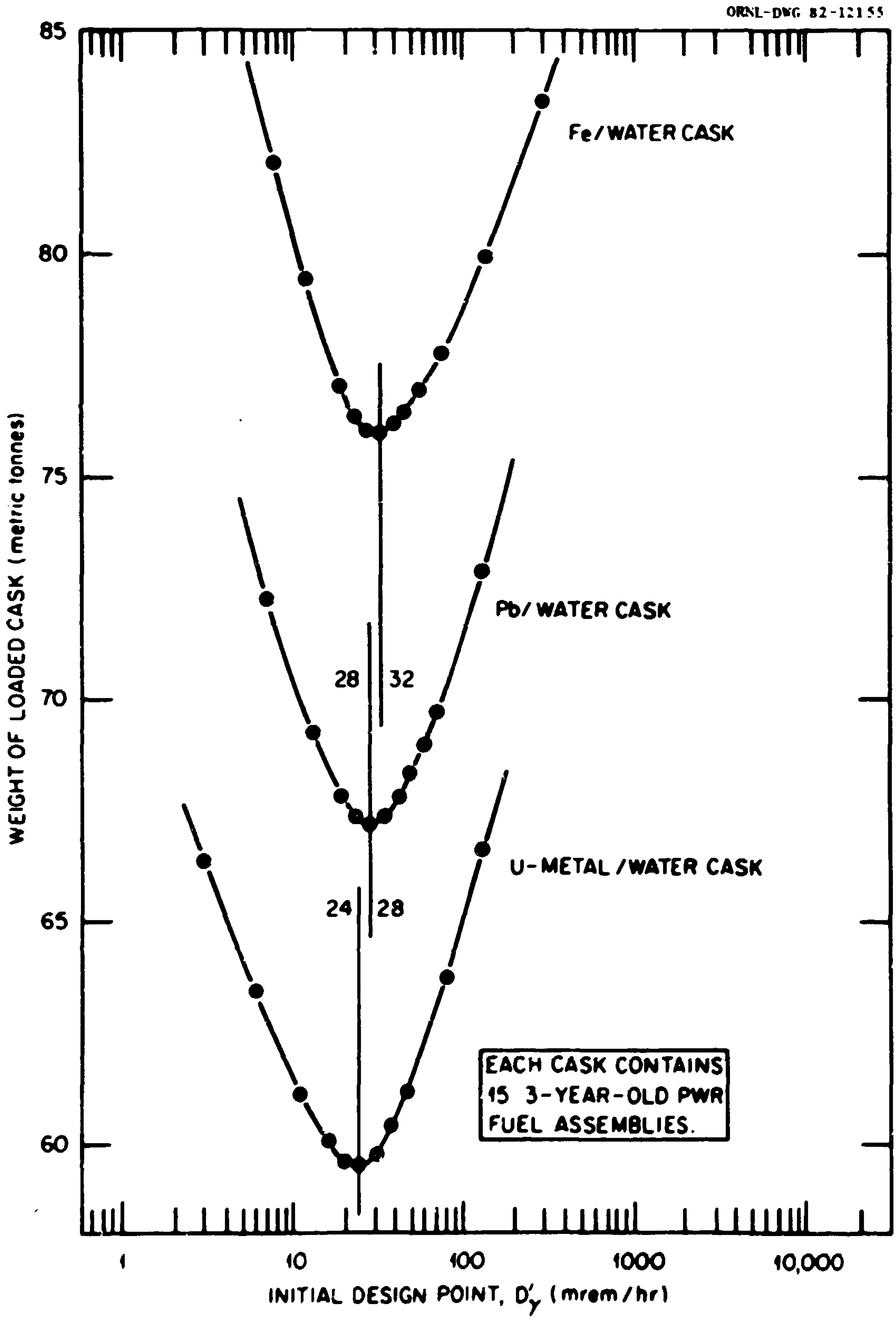

Fig. 3.2. Weight of Pb, Fe, and U-metal casks for 3-year-old PWR spent fuel, as a function of the initial design point used to determine the relative amount of neutron and gamma shielding.

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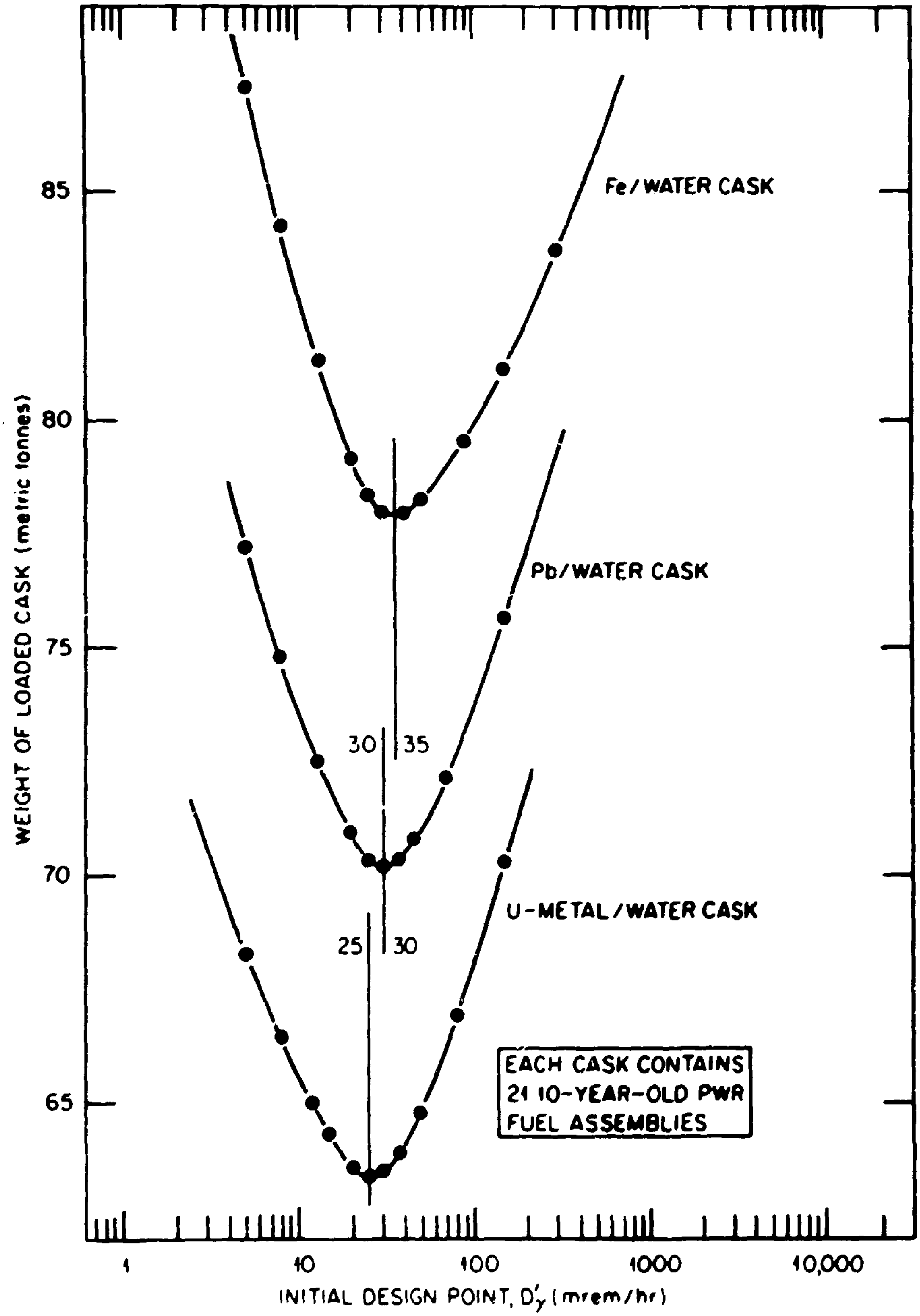

Fig. 3.3. Weight of Pb, Fe, and U-metal casks for 10-year-old PWR spent fuel, as a function of the initial design point used to determine the relative amount of neutron and gamma shielding.

Table 3.19. Recommended dose rates to be used as the initial design point (D'_γ) for optimally designed casks containing 1-, 2-, 3-, 5-, 7-, or 10-year-old spent fuel* (D'_γ in mrem/hr)

Cooling time	Type of cask: Pb	Fe	U
1 yr	26	30	23
2 yr	27	31	24
3 yr	28	32	24
5 yr	29	33	24
7 yr	29	34	25
10 yr	30	35	25

*Values for the 1-, 3-, and 10-year-old spent fuel were independently determined; the others were simply interpolated.

Table 3.20. Optimized neutron and gamma shielding requirements* for bare or copper-finned Pb, Fe, and U-metal spent fuel shipping casks as a function of the number of 1-year-old PWR assemblies in the cask

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield	G-shield	N-shield	G-shield	N-shield
1	5.00	3.77	10.91	3.83	3.03	2.98
4	5.48	4.02	11.64	3.88	3.30	3.33
8	5.73	4.14	12.01	3.91	3.44	3.52
10	5.83	4.29	12.17	3.94	3.50	3.63
12	5.92	4.38	12.30	3.97	3.56	3.72
13	5.94	4.40	12.32	3.94	3.57	3.72
15	5.99	4.41	12.40	3.96	3.60	3.73

*Neutron and gamma shield thicknesses are given in inches.

Table 3.21. Optimized neutron and gamma shielding requirements* for Pb, Fe, and U-metal spent fuel shipping casks, as a function of the number of 2-year-old PWR assemblies in the cask

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield	G-shield	N-shield	G-shield	N-shield
1	4.47	3.71	10.11	3.81	2.72	3.10
4	4.93	3.98	10.81	3.86	2.98	3.32
8	5.18	4.06	11.18	3.86	3.13	3.49
10	5.28	4.13	11.33	3.89	3.18	3.58
12	5.37	4.25	11.46	3.92	3.23	3.65
15	5.44	4.24	11.56	3.90	3.27	3.67
18	5.49	4.24	11.64	3.93	3.31	3.67
21	5.58	4.38	11.77	3.93	3.36	3.76

*Neutron and gamma shield thicknesses are given in inches.

Table 3.22. Optimized neutron and gamma shielding requirements* for Pb, Fe, and U-metal spent fuel shipping casks, as a function of the number of 3-year-old PWR assemblies in the cask

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield	G-shield	N-shield	G-shield	N-shield
1	4.06	3.72	9.52	3.68	2.49	2.86
4	4.50	3.97	10.19	3.76	2.74	3.18
8	4.74	4.05	10.55	3.74	2.88	3.30
12	4.92	4.19	10.82	3.79	2.99	3.48
15	4.99	4.21	10.92	3.79	3.02	3.57
18	5.05	4.24	10.99	3.82	3.06	3.58
21	5.12	4.34	11.11	3.87	3.11	3.61

*Neutron and gamma shield thicknesses are given in inches.

Table 3.23. Optimized neutron and gamma shielding requirements* for Pb, Fe, and U-metal spent fuel shipping casks, as a function of the number of 5-year-old PWR assemblies in the cask

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield	G-shield	N-shield	G-shield	N-shield
1	3.55	3.60	8.79	3.44	2.21	2.54
4	3.97	3.85	9.42	3.48	2.45	2.85
8	4.19	3.95	9.76	3.51	2.58	3.09
12	4.37	4.07	10.02	3.57	2.67	3.30
15	4.43	4.08	10.11	3.56	2.71	3.33
18	4.49	4.07	10.19	3.57	2.74	3.35
21	4.56	4.21	10.30	3.62	2.79	3.45
24	4.60	4.25	10.35	3.65	2.81	3.44

*Neutron and gamma shield thicknesses are given in inches.

Table 3.24. Optimized neutron and gamma shielding requirements* for Pb, Fe, and U-metal spent fuel shipping casks, as a function of the number of 7-year-old PWR assemblies in the cask

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield	G-shield	N-shield	G-shield	N-shield
1	3.28	3.54	8.35	3.49	2.05	2.67
4	3.69	3.70	8.98	3.50	2.28	2.93
8	3.91	3.82	9.31	3.55	2.40	3.14
12	4.08	3.94	9.56	3.59	2.50	3.34
15	4.15	3.95	9.65	3.59	2.53	3.37
18	4.20	4.01	9.73	3.58	2.56	3.40
21	4.27	4.03	9.84	3.62	2.61	3.49
26	4.35	4.16	9.95	3.68	2.65	3.54

*Neutron and gamma shield thicknesses are given in inches.

Table 3.25. Optimized neutron and gamma shielding requirements* for Pb, Fe, and U-metal spent fuel shipping casks, as a function of the number of 10-year-old PWR assemblies in the cask

Number of assemblies	Pb cask		Fe cask		U-metal cask	
	G-shield	N-shield	G-shield	N-shield	G-shield	N-shield
1	2.98	3.69	7.90	3.61	1.90	2.61
4	3.39	3.83	8.52	3.66	2.13	2.86
8	3.61	3.93	8.85	3.66	2.25	3.06
12	3.78	4.04	9.10	3.68	2.34	3.26
18	3.90	4.02	9.27	3.68	2.41	3.33
21	3.97	4.13	9.39	3.71	2.45	3.42
26	4.04	4.25	9.49	3.75	2.49	3.47

*Neutron and gamma shield thicknesses are given in inches.

Because the first value (30 mrem/hr) only accounts for the decay gamma source in the fuel while the second value (15.7 mrem/hr) also accounts for secondary gammas due to the neutron source, this "effective" attenuation cross section will be less than that which one would have if there were no neutron source, and less than that given for the other steel components in the cask (cf. Table 3.7). Because it accounts for secondary gamma production (and attenuation) in the cask, it is more sensitive to the age of the spent fuel and the type of gamma shield used in the cask.

Data in Tables 3.10-3.18 may be used in a similar fashion to obtain values of $\Sigma_\gamma^{o.liner}$ for the other cask/fuel combinations listed in Table 3.7. In two or three cases however, Tables 3.10-3.18 do not contain data corresponding exactly to the optimal initial design points given in Table 3.19. In these cases, the necessary data at those design points was extracted from the more complete set of original (unpublished) data sheets, of which Table 3.9 is a single example. Using the appropriate data in Tables 3.10-3.18 will, however, yield very similar results.

3.2.4.2 Determination of $\Sigma_\gamma^{n.shld}$

Table 3.16 shows the intermediate and final results for a Pb cask carrying 21 10-year-old PWR assemblies. As previously noted, the optimal initial design point in this case is 30 mrem/hr. With a 4.13-in.-thick neutron shield in place, the nominal gamma dose rate 10 ft from the centerline is 9.07 mrem/hr while, without the neutron shield, the accidental gamma dose rate at this point is 15.7 mrem/hr. (Both values include the effect of secondary gammas.) The corresponding dose attenuation cross section is therefore

$$\Sigma_\gamma^{n.shld} = [\ln(15.7/9.07)]/(4.13 \text{ inch}) = 0.1329/\text{inch}. \tag{3.9}$$

Data in Tables 3.10-3.18 may be used in a similar fashion to obtain values of $\Sigma_\gamma^{n.shld}$ for the other cask/fuel combinations listed in Table 3.7. The values so obtained, however, will generally be about 2% higher than the average values shown in Table 3.7. Indeed, the average values shown in Table 3.7 were based on a more complete set of results for casks containing 1, 4, 8, 12, 15, 18, 21, and 26 assemblies. These more complete results were also based on the use of the optimal initial design point in each case. As noted above, however, use of the appropriate data in Tables 3.10-3.18 will yield very similar results.

3.2.4.3 Initial determination of $\Sigma_\gamma^{i.shell}$ and $\Sigma_\gamma^{o.shell}$

The combined thicknesses of the inner and outer steel shells range from 0.75 in. for Fe casks, to 2.75 in. for U-metal casks, to 3.5 in. for Pb casks. Because the 8- to 12-in.-thick gamma shields typical of Fe casks are considerably thicker, cross sections based on the gamma attenuation through these thick shields will be less than those characteristic of the thinner steel shells. Realistic values may be obtained, however, by examining the intermediate results for the gamma dose rate as the thickness of the Fe gamma shield is increased from 0.001 in. to 2.0 in. These intermediate results are summarized in Table 3.26. In casks with 10-year-old fuel, for example, the inferred dose attenuation cross section is given by

$$\Sigma_\gamma^{steel} = [\ln(5.874\text{x}10^5/6.073\text{x}10^4)]/(2.00-0.001) = 1.1351/\text{inch}. \tag{3.10}$$

Table 3.26. Calculated gamma dose rates using 0.001 and 2.00 in. of gamma shielding in an Fe cask with 12 spent fuel assemblies whose cooling times vary from 1 to 10 years

Cooling time	Gamma dose rates[a] (mrem/hr)		Σ_γ
	t_γ = 0.001 in.	t_γ = 2.00 in.	(inch^{-1})[b]
1 yr	4.52262+6	5.16397+5	1.08[illegible]
2 yr	2.53325+6	2.85247+5	1.09[illegible]
3 yr	1.83912+6	2.03885+5	1.1003
5 yr	1.14608+6	1.24537+5	1.1103
7 yr	8.22775+5	8.78636+4	1.1190
10 yr	5.87351+5	6.07322+4	1.1351

[a]Calculated gamma dose rate, 10 ft from the centerline.

[b]Σ_γ is an inferred cross section given by
$\Sigma_\gamma = \ln[D(0.001)/D(2.00)]/(2.00\text{-}0.001)$.

Slightly smaller values were obtained for shorter cooling times where the gamma spectra were somewhat harder. As a first approximation, the values shown in Table 3.26 may also be used for the inner and outer shells which are normally made of stainless steel. As noted in Sect. 3.2.4.5, some very slight adjustments were later made to account for the different thicknesses of the steel shells used in Pb and U-metal casks.

3.2.4.4 Determination of the gamma source strength and an auxiliary correlation to account for spectral hardening of the source over time

Column 2 of Table 3.26 (t_γ = 0.001 in.) shows the resulting gamma dose rate 10 ft from the centerline of an Fe cask with 12 assemblies, no neutron source, no neutron shield, no outside liner, and essentially no gamma shield. The "cask" in this case consists simply of the inner steel shell and the outer steel shell, with a combined thickness of 0.75 in. Multiplying these values by $\exp\{+0.75\Sigma_\gamma\}$ then yields the unshielded gamma dose rate (D_γ^o) that one would expect 10 ft from the centerline if these two steel shells were not present. These unshielded values are shown in column 2 of Table 3.27. Note that the unshielded value for 12 10-year-old PWR fuel assemblies is 1.37606 x 10^6 mrem/hr--the

Table 3.27. The unshielded gamma dose rate [D^o_γ], ORIGEN source terms (photons/sec/assembly), and spectral quality factor [$\chi(T)$] for 12 PWR fuel assemblies as a function of cooling time*

Cooling time	D^o_γ (mrem/hr)	S_γ (p/s/assy)	D^o_γ/S_γ	$\chi(T)$
1 yr	1.02087+7	3.98961+16	2.55881-10	1.00000
2 yr	5.74814+6	2.14683+16	2.67750-10	1.04638
3 yr	4.19758+6	1.36576+16	3.07344-10	1.20112
5 yr	2.63551+6	7.80427+15	3.37701-10	1.31976
7 yr	1.90443+6	5.96726+15	3.19147-10	1.24724
10 yr	1.37606+6	4.90010+15	2.80823-10	1.09744

*Burnup = 33,000 MWD/MTIHM.

same value used as a base reference point in Eq. 3.5b (cf. Sect. 3.2.2). Column 3 of Table 3.27 also shows the volumetric source terms (photons/sec/assembly) as given by the ORIGEN results for PWR fuel irradiated to 33,000 MWD/MTIHM and allowed to cool for 1, 2, 3, 5, 7, and 10 years. [These values were taken from Table C.1 of ref. 4.] Dividing column 2 by column 3 yields the effective dose rate per source photon (cf. column 4). Normalizing those values to 1.0 for 1-year-old fuel then yields an empirical quality factor that accounts for the spectral hardening of the gamma source and the diminished self-shielding of the fuel over time. This spectral quality factor, $\chi(T)$, is given in column 5. The correlation given in Sect. 3.2.2 is simply a good numerical fit of that data. While the total amount of gamma radiation generated by the spent fuel always decays monotonically with time, the fraction of that which escapes from the fuel and impinges on the inner wall of the cask tends to peak when the fuel is 5-7 years old.

The amount of spatial self-shielding afforded by the spent fuel is also dependent on the number of spent fuel assemblies in the cask. This phenomenon is discussed separately in Sect. 3.2.4.7.

3.2.4.5 Final determination of $\Sigma^{i.shell}_\gamma$ and $\Sigma^{o.shell}_\gamma$

Table 3.28 shows the calculated gamma dose rate 10 ft from the centerline of various Pb, Fe, and U-metal casks containing 12 spent fuel assemblies whose cooling times varied from 1 to 10 years. These results are based on calculations with no neutron source, no neutron shield, no outside liner, and essentially no Pb, Fe, or U-metal gamma shield. [In this respect, they are similar to the calculation described as iteration number 1 in Table 3.9.] They include only the 18-group gamma source in the homogenized fuel region and the shielding provided by the inner and outer steel shells whose combined thicknesses range from 0.75 in. for Fe casks, to 2.75 in. for U-metal casks, to 3.5 in. for Pb casks. Using this data and the unshielded gamma dose rates shown in column 2 of Table 3.27, one may calculate Σ_γ for the inner and outer shells of the Pb, Fe, and U-metal casks as a function of decay time. These values for a Pb cask with 10-year-old spent fuel would be calculated as

$$\Sigma^{i.shell}_\gamma = \Sigma^{o.shell}_\gamma$$

$$= [\ln(1.37606 \times 10^6/2.45642 \times 10^4)]/(3.50\text{inch}) = 1.1502/\text{inch}. \quad (3.11)$$

Table 3.28. Calculated gamma dose rates using 0.001 in. of gamma shielding in Pb, Fe, and U-metal casks containing 12 spent fuel assemblies whose cooling times varied from 1 to 10 years

Cooling time	Gamma dose rate (mrem/hr)* at 10 ft Pb cask (t^{stl}=3.50 in.)	Fe cask (t^{stl}=0.75 in.)	U cask (t^{stl}=2.75 in.)
1 yr	2.17808+5	4.52262+6	4.91730+5
2 yr	1.19908+5	2.53325+6	2.71486+5
3 yr	8.48683+4	1.83912+6	1.93910+5
5 yr	5.12899+4	1.14608+6	1.18339+5
7 yr	3.59572+4	8.22775+5	8.34306+4
10 yr	2.45642+4	5.87351+5	5.75912+4

*Assumes no neutron source, no neutron shield, no outside liner, and essentially no Pb, Fe, or U-metal gamma shield; includes only the gamma source and the inner and outer steel shells.

The other values shown in Table 3.7 may be calculated in a similar fashion. As one would expect, the values obtained for use with different types of casks are all within 1-2% of the values initially obtained in Sect. 3.2.4.3 for the Fe casks. Slightly smaller values were again obtained for shorter cooling times where the gamma flux spectra impingent on the inner wall of the cask was somewhat harder.

3.2.4.6 Determination of $\Sigma^{g\text{-}shld}_{\gamma,dif}$ and $\Sigma^{g\text{-}shld}_{\gamma,avg}$

The effective dose rate attenuation cross section to be used in simple exponential calculations of attenuation must depend on the total thickness of the shield, with the "effective" cross section being slightly less for thicker shields. [This reduced effective cross section accounts for radiation that escapes through the gamma shield after being scattered several times.] The net attenuation through the shield is therefore represented by the following expression:

$$D^{\gamma}_{outer} = D^{\gamma}_{inner} \exp\{- [\Sigma^{g\text{-}shld}_{\gamma,avg}\ t^{g\text{-}shld}_{ref} + \Sigma^{g\text{-}shld}_{\gamma,dif}\ (t_{\gamma} - t^{g\text{-}shld}_{ref})]\} \tag{3.12}$$

where $\Sigma^{g\text{-}shld}_{\gamma,avg}$ is the average cross section over some typical reference thickness, and $\Sigma^{g\text{-}shld}_{\gamma,dif}$ is the differential cross section applied to each additional increment of gamma shielding. Alternately, one could use a constant mass attenuation coefficient for the entire shield and then multiply the resulting dose rate by a buildup factor that increases with the thickness of the shield. Since the gamma shield thicknesses of most shipping casks vary over a relatively narrow and well-known range, Eq. 3.12 was deemed a more desirable approach. Because the effect of scattered radiation only becomes noticeable in highly absorbing shields of significant thickness, this approach was restricted to (1) attenuation of the gamma dose rate in the gamma shield and (2) attenuation of the neutron dose rate in the neutron shield.

3.2.4.6.1 Determination of $\Sigma^{g\text{-}shld}_{\gamma,dif}$

Table 3.16 shows the intermediate and final results for a Pb cask carrying 21 10-year-old PWR assemblies. As previously noted, the optimal initial design point in this case is 30 mrem/hr. This corresponds to a 3.97-in.-thick Pb gamma shield. To estimate $\Sigma^{g\text{-}shld}_{\gamma,dif}$, consider the results corresponding to initial design points of 20 mrem/hr and 45 mrem/hr. The first case required a 4.24-in.-thick gamma shield, while the second required a 3.71-in.-thick gamma shield. The differential cross section may then be estimated as

$$\Sigma_{\gamma,dif}^{g,shld} = [\ln(45/20)]/(4.24-3.71) = 1.5301/\text{inch}. \tag{3.13}$$

This value, however, only accounts for attenuation of the gammas emanating directly from the spent fuel; it does not account for the attenuation of secondary gammas generated in the gamma shield by the neutrons since the neutron source in the fuel had been set to zero in the first phase of the respective calculations. The neutron and gamma source terms were both used, however, in the second phase of the calculations. With both source terms, the 0.75-in.-thick outside liner, and no neutron shield, Table 3.16 shows the corresponding "accidental" gamma dose rates as being 10.5 and 23.4 mrem/hr. Including the effect of secondary gammas, the differential cross section for the gamma shield may then be estimated as

$$\Sigma_{\gamma,dif}^{g,shld} = [\ln(23.4/10.5)]/(4.24-3.71) = 1.5120/\text{inch}. \tag{3.14}$$

Other data in Tables 3.10-3.18 may be used in a similar fashion to estimate values of $\Sigma_{\gamma,dif}^{g,shld}$ for Pb, Fe, and U-metal casks containing spent fuel that has been out of the reactor for 1, 3, or 10 years. Tables 3.29 and 3.30 show the results obtained if one does and does not account for secondary gammas. Interestingly, the results are essentially the same, with those that do account for secondary gammas being only 1-2% lower than those that do not.

While the results in Tables 3.29 and 3.30 are interesting, the somewhat better values of $\Sigma_{\gamma,dif}^{g,shld}$ shown in Table 3.7 were obtained after considering the intermediate (unpublished) results from a more extensive range of calculations. These calculations were for optimized Pb, Fe, and U-metal casks containing 1-, 3-, 5-, and 10-year-old fuel. In each case, the value of $\Sigma_{\gamma,dif}^{g,shld}$ was calculated for casks containing 1 assembly, 12 assemblies, and 15, 18, 21, 24, or 26 assemblies (depending on the maximum practical capacity of the cask, as dictated by the amount of shielding required given the type of cask and the age of the spent fuel). The type of "intermediate" data used in each case was similar in nature to that shown in iterations 2-5 of Table 3.9. The calculated values of $\Sigma_{\gamma,dif}^{g,shld}$ were found to be surprisingly insensitive to the number of assemblies and the resulting gamma shield thickness. A Pb cask with 21 3-year-old assemblies, for example, required a 5.12-in.-thick gamma shield while a similar cask with just one assembly required just 4.06 in. of Pb; yet, the calculated values of $\Sigma_{\gamma,dif}^{g,shld}$ only ranged from 1.4006/inch for the large cask to 1.4185/inch for the smaller cask. The final values of $\Sigma_{\gamma,dif}^{g,shld}$ reported in Table 3.7, are average values for the cases studied. The data upon which these values are based did not include the effect of secondary gammas which, as noted above, might have caused the resulting cross sections to be 1-2% lower.

3.2.4.6.2 Determination of $\Sigma_{\gamma,avg}^{g,shld}$

The average dose attenuation cross section for a thick gamma shield will be somewhat greater than the differential cross section applicable to the last increment of the shield. To illustrate, consider the intermediate results shown in Table 3.9 for a series of 1-D shielding calculations for a Pb cask carrying 21 10-year-old PWR fuel assemblies. With 0.001 in. of Pb, no neutron source, no neutron shield, and no outside liner, the gamma dose rate 10 ft from the centerline was 33323.4 mrem/hr. With 3.9694 in. of Pb, the resulting gamma dose rate at that point was 30.0392 mrem/hr. The average dose attenuation cross section might therefore be estimated as

$$\Sigma_{\gamma,avg}^{g,shld} = [\ln(33323.4/30.0392)]/(3.9694-0.001) = 1.7668/\text{inch}. \tag{3.15}$$

Table 3.29. Estimates of $\Sigma_{\gamma,dif}^{g\text{-}shld}$ (inch^{-1}) for different types of casks, based on the data in Tables 3.10-3.18 if one does not account for the effect of secondary gammas

Cooling time	Type of cask		
	Pb	Fe	U
1 yr	1.2958	0.8484	2.3287
3 yr	1.3896	0.9076	2.4657
10 yr	1.5301	0.9960	2.7445

Table 3.30. Estimates of $\Sigma_{\gamma,dif}^{g\text{-}shld}$ (inch^{-1}) for different types of casks based on the data in Tables 3.10-3.18 if one does account for the effect of secondary gammas

Cooling time	Type of cask		
	Pb	Fe	U
1 yr	1.2734	0.8346	2.2978
3 yr	1.3697	0.8910	2.4737
10 yr	1.5120	0.9824	2.7174

As expected, this value is significantly greater than the corresponding differential cross section (1.5437/inch). Using similar data for Pb, Fe, and U-metal casks carrying different numbers of spent fuel assemblies cooled for 1, 3, 5, and 10 years, one could likewise estimate the corresponding value of $\Sigma_{\gamma,avg}^{g\text{-}shld}$ for each case. Unfortunately, the reference thickness ($t_{ref}^{g\text{-}shld}$) of the gamma shield would be different in each case. To simplify the problem, a slightly different approach was taken. After surveying a large number of casks (cf. Table IV.19 in ref. 4), it was found that Pb gamma shields typically range from 3.0 to 6.0 in. in thickness, Fe gamma shields typically range from 7.9 to 12.4 in. in thickness, and depleted U-metal gamma shields typically range from 1.9 to 3.6 in. in thickness. Subsequent to that survey, constant reference thicknesses of 4.5, 10.0, and 2.75 in. were selected for Pb, Fe, and U-metal casks. Average dose attenuation cross sections were then defined in terms of those reference thicknesses. Combining Eq. 3.12 with the data in Eq. 3.15, the average cross section in this example could be written as

$$\Sigma_{\gamma,avg}^{g\text{-}shld} = \{\ln(33323.4/30.0392) - \Sigma_{\gamma,dif}^{g\text{-}shld}(t_\gamma - t_{ref}^{g\text{-}shld})\}/t_{ref}^{g\text{-}shld} \quad (3.16a)$$

$$= \{\ln(33323.4/30.0392) - (1.5437)(3.9694-4.5)\}/(4.5) \quad (3.16b)$$

$$= 1.7401/\text{inch}. \quad (3.16c)$$

The actual values shown in Table 3.7 were derived so as to be consistent with results obtained from the detailed S_3P_3 multigroup shielding calculations for Pb, Fe, and U-metal casks carrying 12 1-, 3-, 5-, and 10-year-old spent fuel assemblies. Table 3.31, for example, shows the gamma shield thicknesses (t_γ) required in each case to reduce the gamma dose rate at 10 ft from the centerline down to the optimal initial design points ($D_\gamma^{'}$) shown in Table 3.19. Given the corresponding unshielded gamma dose rates (D_γ^o) shown in column 2 of Table 3.27, the thickness of the inner and outer steel shells (cf. Table 2.1), the cross sections for the steel shells ($\Sigma_\gamma^{i.shell}$ and $\Sigma_\gamma^{o.shell}$), the differential dose attenuation cross section for the gamma shield ($\Sigma_{\gamma,dif}^{g.shld}$), and the reference thickness assigned to each type of gamma shield ($t_{ref}^{g.shld}$), the average dose attenuation cross section for the gamma shield may be calculated as

$$\Sigma_{\gamma,avg}^{g.shld} = \{\ln(D_\gamma^o/D_\gamma^{'}) - [\Sigma_\gamma^{i.shell} t_\gamma^{i.shell} + \Sigma_\gamma^{o.shell} t_\gamma^{o.shell}$$

$$+ \Sigma_{\gamma,dif}^{g.shld}(t_\gamma - t_{ref}^{g.shld})]\}/t_{ref}^{g.shld} \ . \tag{3.17}$$

Indeed, this equation was used to calculate all of the values for $\Sigma_{\gamma,avg}^{g.shld}$ shown in Table 3.7 of Sect. 3.2.1. In the case of a Pb cask with 12 10-year-old fuel assemblies, this would yield:

$$\Sigma_{\gamma,avg}^{g.shld} = \{\ln(1.37606\mathrm{x}10^6/29.987) - [(1.1502)(1.5) + (1.1502)(2.0)$$

$$+ (1.5437)(3.7787-4.5)]\}/(4.5 \text{ inch}) \tag{3.18a}$$

$$= 1.7382/\text{inch}. \tag{3.18b}$$

Because of the standardization introduced by the use of $t_{ref}^{g.shld}$ and $\Sigma_{\gamma,dif}^{g.shld}$, this value differs by only 0.1% from the earlier value based on a large cask with 21 assemblies and a thicker gamma shield.

Table 3.31. Gamma shield thicknesses (inches) required for various types of casks* in order to reduce the dose rate at 10 ft down to the optimal initial design points shown in Table 3.19

Cooling time	Type of cask		
	Pb	Fe	U
1 yr	5.9226	12.3012	3.5557
3 yr	4.9195	10.8181	2.9889
5 yr	4.3662	10.0202	2.6734
10 yr	3.7787	9.1027	2.3429

*Each cask contains 12 PWR spent fuel assemblies.

3.2.4.7 Determination of $F_\gamma(N)$—a correlation for the parametric variation of the unshielded gamma dose rate as a function of the number of assemblies in the cask

Casks containing more spent fuel assemblies will require more gamma shielding than casks with fewer assemblies, although the amount of additional shielding needed will increase more and more slowly with the number of assemblies because of the spatial self-shielding afforded the gamma radiation by the heavy metal in the fuel itself. (Indeed, previous calculations have shown that 85% of all photons emitted by the spent fuel in a single assembly will be reabsorbed by the same assembly.) A simple correlation was therefore needed to estimate the variation in the effective gamma radiation load on the inner wall of a cask as a function of the number of assemblies. Such a correlation was developed by first postulating a crude conceptual model and then evaluating existing data to determine the necessary constants. Assume that the radius R_2 corresponds to the inner wall of the cask and that all gamma radiation impinging on the inner wall of the cask is emitted by fuel only in the outermost region of the homogenized fuel zone between R_1 and R_2 ($0 < R_1 < R_2$). The effective volume-integrated source would then be given by

$$S_\gamma^{eff} = a\pi(R_2^2 - R_1^2), \tag{3.19}$$

where "a" is a simple constant. Since the surface area of the inner wall is proportional to $2\pi R_2$, the impingent flux is given by

$$\phi_\gamma = S_\gamma^{eff}/A = b(R_2^2 - R_1^2)/R_2 \ , \tag{3.20}$$

where "b" is a simple constant. Noting that the number of assemblies in the cask (N) is proportional to πR_2^2, the square root of N times the gamma flux impingent on the inner wall of the cask should vary as

$$\sqrt{N}\ \phi_\gamma(N) = cN + d \ , \tag{3.21}$$

where the constants c and d may be found by plotting $\sqrt{N}\phi_\gamma(N)$ as a function of N. Assuming that the conceptual model is reasonably valid, a single set of constants should allow Eq. 3.21 to fit the data over a broad range of cases.

Table 3.25 shows the optimal amount of gamma shielding required in Pb, Fe, and U-metal casks as a function of the number of 10-year-old spent fuel assemblies in the cask. If $t_{\gamma,21}^{Pb}$, $t_{\gamma,21}^{Fe}$, and $t_{\gamma,21}^{U}$ represent the amount of Pb, Fe, or U-metal gamma shielding required for 21 fuel assemblies, and $t_{\gamma,N}$ is the amount of gamma shielding required for N assemblies, then the gamma flux on the inner wall of the cask, $\phi_\gamma(N)$, must be proportional to $\phi(N)$, where

$$\phi(N) = \exp[+\Sigma_{\gamma,dif}^{g,shld}(t_{\gamma,N} - t_{\gamma,21})] \ . \tag{3.22}$$

Table 3.32 shows the corresponding values of $\sqrt{N}\phi(N)$ for the Pb, Fe, and U-metal casks in Table 3.25. Since $\phi(N)$ should be proportional to the gamma flux on the inner wall of the cask, it was not surprising to see that the corresponding values of $\sqrt{N}\phi(N)$ were essentially independent of the type of cask. Column 5 of Table 3.32 shows the average values for all three types of casks. Following this same procedure, values of $\sqrt{N}\phi(N)$ were also calculated for the 3-year-old spent fuel casks described in Table 3.22. In this case, however, the amount of shielding required for 15 assemblies

Table 3.32. Calculated values of $\sqrt{N}\phi(N)$ for Pb, Fe, and U-metal casks containing 1 to 26 10-year-old PWR spent fuel assemblies[a]

N	Pb cask	Fe cask	U cask	Average values
1	0.2238	0.2314	0.2243	0.2265
4	0.8321	0.8508	0.8383	0.8404
8	1.6412	1.6640	1.6425	1.6492
12	2.5991	2.6054	2.5691	2.5912
18	3.8165	3.7709	3.8057	3.7977
21	4.5826	4.5826	4.5826	4.5826
26	5.6683	5.6252	5.6845	5.6593

[a]Where $\sqrt{N}\phi(N) = \sqrt{N}\ \exp[+\Sigma_{\gamma,dif}^{e,shld}(t_{\gamma,N} - t_{\gamma,21})]$.

($t_{\gamma,15}$) was used as the reference value. The values obtained for $\sqrt{N}\phi(N)$ were again nearly independent of the type of cask, although the average values in each case were a factor of 1.2194 higher than the average values for the 10-year-old fuel. These normalized average values are shown in Table 3.33. Lastly, values of $\sqrt{N}\phi(N)$ were calculated for the 1-year-old spent fuel casks in Table 3.20, with $t_{\gamma,13}$ being used as the reference value. Again, the values obtained for $\sqrt{N}\phi(N)$ were nearly independent of the type of cask, although the average values were a factor of 1.3132 higher than those for the 10-year-old spent fuel. These normalized average values are also shown in Table 3.33, along with the normalized average values for the 3- and 10-year-old fuel. Note that the functional dependence on N is remarkably similar in all three cases.

Table 3.33. Normalized averaged values of $\sqrt{N}\phi(N)$ for Pb, Fe, and U-metal casks containing 1 to 26 1-, 3-, and 10-year-old spent fuel assemblies

N	1-year-old[a]	3-year-old[b]	10-year-old
1	0.2283	0.2287	0.2265
4	0.8434	0.8382	0.8404
8	1.6363	1.6519	1.6492
10	2.0894	---	---
12	2.5813	2.6058	2.5912
13	2.7457	---	---
15	3.1519	3.1762	---
18	---	3.7739	3.7977
21	---	4.5457	4.5826
26	---	---	5.6593

[a]Normalized by dividing all of the actual values by 1.3132.

[b]Normalized by dividing all of the actual values by 1.2194.

To determine the adequacy of the conceptual model described above, the normalized average values of $\sqrt{N}\phi(N)$ given in Table 3.33 were plotted as a function of N and fitted (by eye) with a straight line as shown in Fig. 3.4. The nearly linear character of the data tends to confirm the adequacy of the conceptual model. Using this fit, the following approximation was adopted for $\sqrt{N}\phi(N)$:

$$\sqrt{N}\phi(N) = (0.2086)N + (0.01917) \; , \tag{3.23}$$

and the unshielded gamma dose rate, which is proportional to $\phi(N)$, was assumed to vary as

$$F_{\gamma}(N) = \sqrt{N} + (0.01917/0.2086)/\sqrt{N} \tag{3.24a}$$

$$= \sqrt{N} + (0.0919/\sqrt{N}) \; , \tag{3.24b}$$

where N is the number of spent fuel assemblies in the cask. This geometric factor is used by CAPSIZE to account for the gamma self-shielding provided by the fuel inside a cask.

More recently, the data in Table 3.33 have been approximated using a linear least-squares regression analysis. The result, given by

$$\sqrt{N}\phi(N) = (0.21603)N - (0.032139) \; , \tag{3.25}$$

is shown in Fig. 3.5. This fit is somewhat better and shows that the conceptual model developed is quite good. More importantly, the coefficient corresponding to d in Eq. 3.21 is now negative and therefore more consistent with what one would expect from Eq. 3.20. While a revised formula for $F_{\gamma}(N)$ could and probably should be incorporated in the CAPSIZE program, this improvement has not been made to date. For large casks with more than 21-26 assemblies, the effective gamma source would then be 3-5% higher. Casks with fewer assemblies would be essentially unaffected.

3.2.4.8 Initial determination of $\Sigma_n^{g\text{-shld}}$

Table 3.16 shows the intermediate and final results for a Pb cask carrying 21 10-year-old PWR fuel assemblies. Under accident conditions, a cask with 2.92 in. of gamma shielding and no neutron shield would yield a neutron dose rate of 41.9 mrem/hr (10 ft from the centerline), while a similar cask with 5.15 in. of gamma shielding and no neutron shield would yield a neutron dose rate of 36.4 mrem/hr (10 ft from the centerline). The corresponding cross section is thus given by

$$\Sigma_n^{g\text{-shld}} = [\ln(41.9/36.4)]/(5.15-2.92) = 0.0631/\text{inch} \; . \tag{3.26}$$

Using similar data in Tables 3.10-3.18, initial estimates of $\Sigma_n^{g\text{-shld}}$ were calculated for Pb, Fe, and U-metal casks containing 1-, 3-, and 10-year-old spent fuel. These results are shown in Table 3.34, along with the average value for each type of cask. Note that the actual values at various cooling times never differed by more than 5% from the average values.

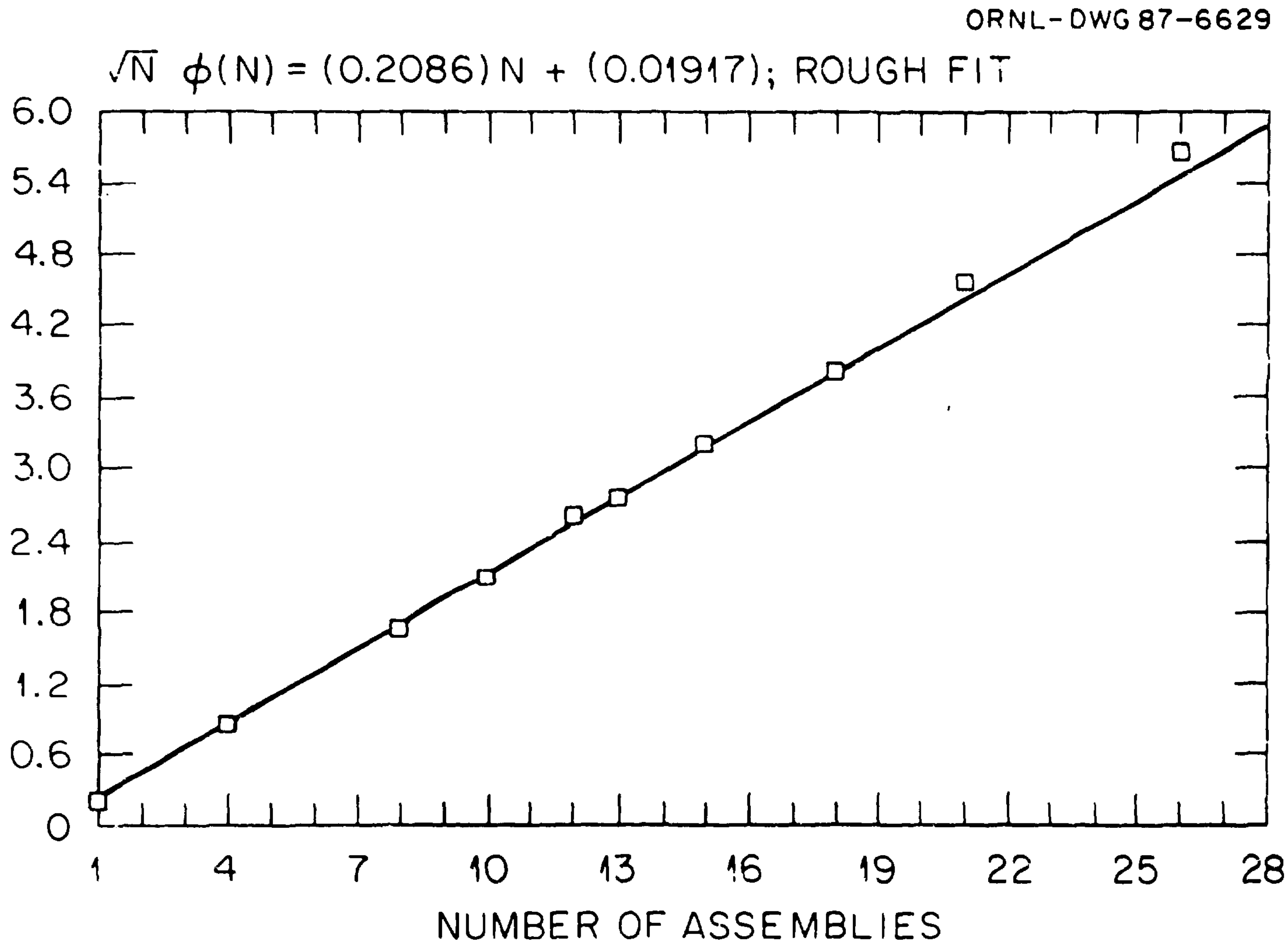

Fig. 3.4. Rough linear fit of $\sqrt{N}\phi(N)$ as a function of the number of assemblies in a cask. Here $\sqrt{N}\phi(N) = (0.2086)N + (0.01917)$. This rough linear fit of the normalized average values of $\sqrt{N}\phi(N)$ is used by CAPSIZE to estimate the parametric variation of the unshielded gamma dose rate as a function of the number of assemblies in a cask.

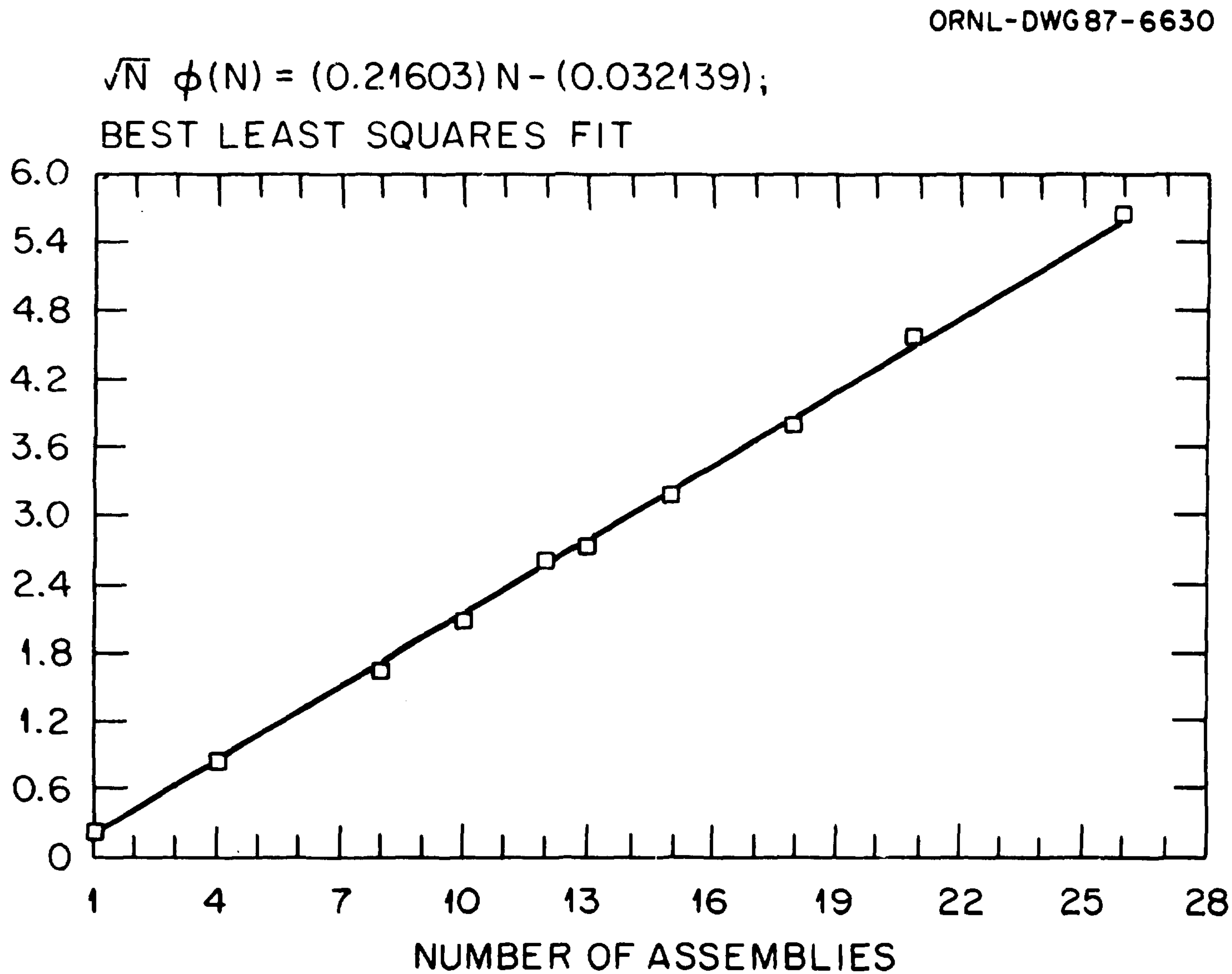

Fig. 3.5. Improved linear least squares fit of $\sqrt{N}\phi(N)$ as a function of the number of assemblies in a cask. Here $\sqrt{N}\phi(N)$ = (0.21603)N - (0.032139). Note that this least squares fit is more consistent with the physics of the conceptual model than the rough linear fit shown in Fig. 3.4.

Table 3.34. Initial estimates of $\Sigma_n^{g\text{-shld}}$ for Pb, Fe, and U-metal casks containing 1-, 3-, and 10-year-old spent fuel

Cooling time	Pb cask	Fe cask	U cask
1 yr	0.0580	0.1582	0.2889
3 yr	0.0595	0.1595	0.2900
10 yr	0.0631	0.1592	0.2900
average values	0.0602	0.1590	0.2896

3.2.4.9 Determination of the neutron source strength, and an auxiliary correlation [$F_n(N)$] for the parametric variation of the unshielded neutron dose rate as a function of the number of assemblies in the cask

Table 3.25 shows the optimized neutron and gamma shield thicknesses for Pb, Fe, and U-metal casks containing 1 to 26 10-year-old spent fuel assemblies. Based on the intermediate (unpublished) results, Table 3.35 shows the corresponding neutron dose rates (D_n^{acc}) 10 ft from the centerline if the neutron shields were suddenly lost. Using the average values of $\Sigma_n^{g\text{-shld}}$ shown in Table 3.34, together with the gamma shield thickness shown in Table 3.25, one can estimate the unshielded neutron dose rates 10 ft from the centerline as:

$$D_n^o = D_n^{acc}\exp\{\Sigma_n^{g\text{-shld}}t_\gamma + \Sigma_n^{steel}(t^{i.shell} + t^{o.shell} + t^{o.liner})\} , \qquad (3.27)$$

where Σ_n^{steel} may (as a first approximation) be assumed to be the same as $\Sigma_n^{g\text{-shld}}$ for the Fe cask, and the thicknesses of the inner and outer steel shells and the outside liner are as shown in Table 2.1. The resulting estimates of the unshielded neutron dose rates (D_n^o) are shown in Table 3.36 as a function of the number of spent fuel assemblies. As expected, results for the Pb, Fe, and U-metal casks are very similar, deviating by no more than 3% from the average values (also shown in Table 3.36).

Table 3.35. Accidental neutron dose rates (D_n^{acc}) 10 ft from the centerline of optimized Pb, Fe, and U-metal casks, as a function of the number of 10-year-old spent fuel assemblies in the cask*

N	Pb cask	Fe cask	U cask
1	4.286	2.465	3.479
4	11.990	6.228	9.327
8	19.031	9.397	14.545
12	27.250	13.003	20.488
18	33.326	15.437	24.921
21	39.175	17.468	29.103
26	45.445	20.429	33.571

*Dose rates are in mrem/hr and assume no neutron shield present.

Table 3.36. Unshielded neutron dose rates (D_n^o) 10 ft from the centerline of Pb, Fe, and U-metal casks, as a function of the number of 10-year-old spent fuel assemblies in the cask*

N	Pb cask	Fe cask	U cask	Average values
1	10.0794	10.9875	10.5224	10.5298
4	28.9016	30.6368	30.1531	29.8972
8	46.4854	48.7159	48.6851	47.9621
12	67.2459	70.1437	70.3884	69.2593
18	82.8361	85.5553	87.3718	85.2544
21	97.7858	98.6199	103.2225	99.8761
26	113.9156	117.2527	120.4569	117.2084

*Dose rates are in mrem/hr and assume no inner steel shell, no gamma shield, no outer steel shell, no neutron shield, and no outside liner.

Intermediate (unpublished) results for the 1- and 3-year-old spent fuel casks described in Tables 3.20 and 3.22 were likewise used to obtain estimates of the unshielded neutron dose rates at these decay times. These results were also very insensitive to the type of cask considered, although the average results for the 1-year-old fuel were 2.2265 times higher, and the average results for the 3-year-old fuel were 1.3289 times higher.

Table 3.37 shows the normalized average values of the unshielded neutron dose rates (D_n^o) for the 1-, 3-, and 10-year-old fuel as a function of the number of assemblies. In all three cases, the functional dependence on the number of assemblies is essentially identical. As noted in Eq. 3.5a of Sect. 3.2.2, a value of 100.0 mrem/hr was adopted as the base value corresponding to 21 10-year-old PWR spent fuel assemblies.

The data in Table 3.37 was also examined to determine how the unshielded neutron dose rate (D_n^o) increased with the number of assemblies. Due to the longer mean free path of neutrons in a dry cask and a small amount of subcritical neutron multiplication (which increases with the number of assemblies present), the conceptual model used for gammas was not considered applicable to neutrons. Moreover, the unshielded neutron dose rates shown in Table 3.37 are obviously increasing faster than $\sqrt{N}$. Curiously, graphical analysis of the data in Table 3.37 shows the unshielded neutron dose rate to be increasing approximately as $(10.25)N^{0.75}$. A slightly better approximation, shown in Fig. 3.6, is given by

$$D_n^o(N) = (10.64)(N^{0.73643}) . \tag{3.28}$$

Within the CAPSIZE program, this functional dependence is represented simply as

$$F_n(N) = N^{0.73643} , \tag{3.29}$$

with D_n^o properly normalized to 100.0 mrem/hr for the case of 21 10-year-old PWR spent fuel assemblies previously irradiated to 33,000 MWD/MTIHM (cf. Eq. 3.5a in Sect. 3.2.2).

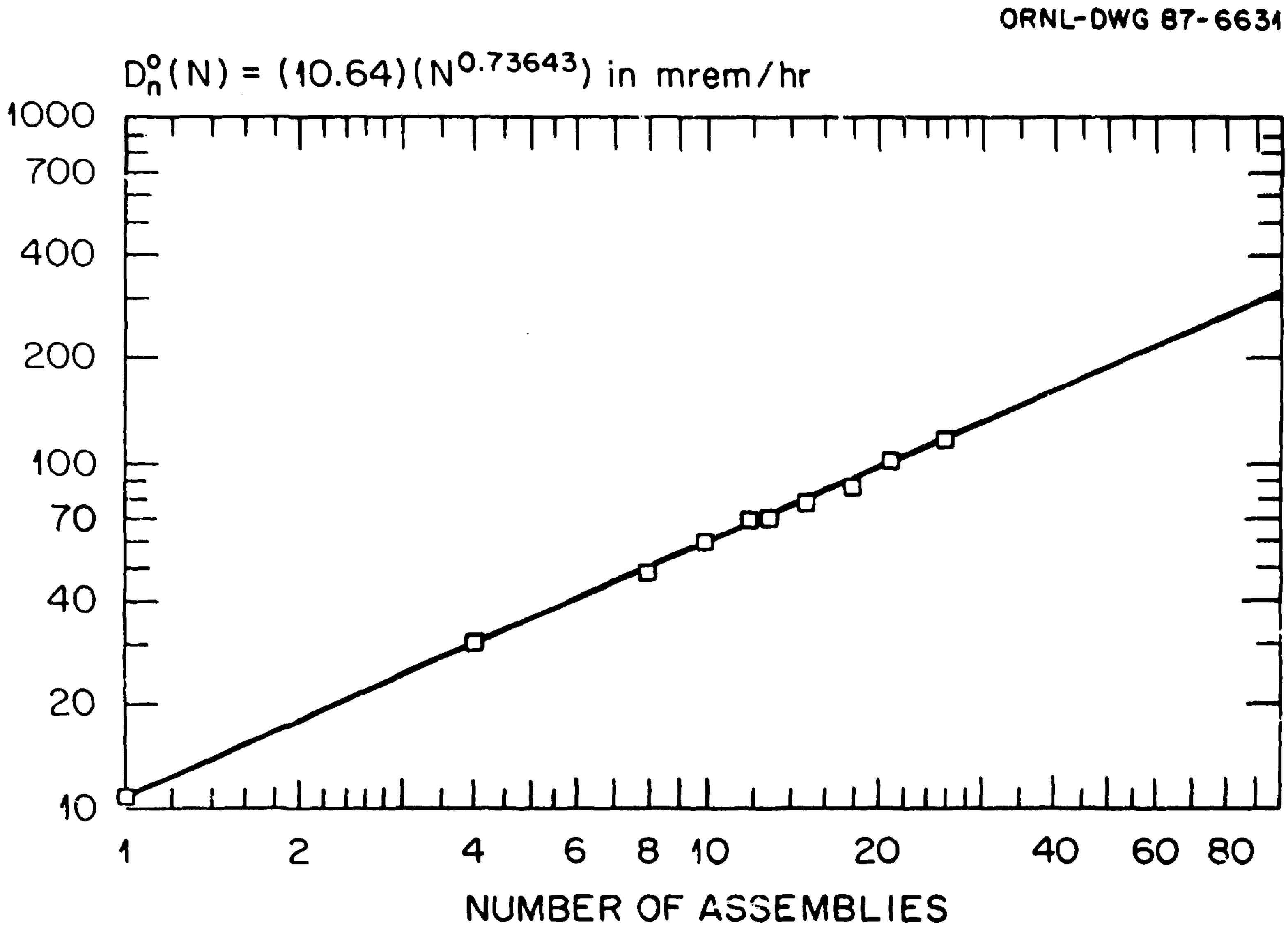

Fig. 3.6. Least squares fit of the unshielded neutron dose rate [$D_n^o(N)$] as a function of the number of assemblies in a cask. Data points correspond to the normalized average values shown in Table 3.37. Here $D_n^o(N) = (10.64)N^{0.73643}$.

Table 3.37. Normalized averaged values of the unshielded neutron dose rate (D_n^o) for Pb, Fe, and U-metal casks containing 1 to 26 1-, 3-, and 10-year-old spent fuel assemblies

N	1-year-old[a]	3-year-old[b]	10-year-old
1	10.7216	10.6776	10.5298
4	29.9174	29.9617	29.8972
8	47.5215	47.8026	47.9621
10	57.8701	---	---
12	68.5799	68.9875	69.2593
13	70.2630	---	---
15	76.4030	77.0451	---
18	---	84.5840	85.2544
21	---	99.7384	99.8761
26	---	---	117.2084

[a]Normalized by dividing the actual average values (for Pb, Fe, and U-metal casks) by 2.2265.

[b]Normalized by dividing the actual average values (for Pb, Fe, and U-metal casks) by 1.3289.

3.2.4.10 Final determination of $\Sigma_n^{g\text{-shld}}$

The initial estimates of $\Sigma_n^{g\text{-shld}}$ given in Sect. 3.2.4.8 were based on calculated dose rates for gamma shields of different thicknesses. Since the differences (Δt) were only a fraction of the total shield thickness, the calculated values were more typical of the differential cross sections. To be consistent with its intended use, however, the quantity that is needed is the average cross section that is applicable across the entire shield.

Given the unshielded neutron dose rate shown in Table 3.37, one can use the initial values of $\Sigma_n^{g\text{-shld}}$ to calculate the approximate neutron dose rate at 10 ft from the centerline if the neutron shield were suddenly lost. This quantity is given by

$$D_n^{acc}(\text{appx}) = D_n^o \exp\{-[\Sigma_n^{g\text{-shld}} t_\gamma + \Sigma_n^{steel}(t^{i.shell} + t^{o.shell} + t^{o.liner})]\} , \tag{3.30}$$

where Σ_n^{steel} is assumed to be the same as $\Sigma_n^{g\text{-shld}}$ for the Fe cask. This quantity [D_n^{acc}(appx)] was calculated as a function of the number of spent fuel assemblies, for Pb, Fe, and U-metal casks containing 1-, 3-, and 10-year-old spent fuel. Comparisons were then made with more exact results based on the S_8P_3 discrete ordinates shielding calculations. The average value of [D_n^{acc}(exact)/D_n^{acc}(appx)] was 1.0846 for Fe casks carrying 1-year-old fuel, 1.0536 for Fe casks carrying 3-year-old fuel, and 1.0221 for Fe casks carrying 10-year-old fuel. Other values for Pb and U-metal casks are shown in Table 3.38.

Table 3.38. Average values of $[D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx})]$ for Pb, Fe, and U-metal casks containing 1-, 3-, and 10-year-old spent fuel*

Cooling time	Pb cask	Fe cask	U cask
1 yr	1.0173	1.0846	1.0778
3 yr	0.9992	1.0536	1.0495
10 yr	0.9675	1.0221	1.0171

*Based on initial estimates of $\Sigma_n^{g\text{-}shld}$

Using typical gamma shield thicknesses of 12.0, 10.5, and 9.0 in. for Fe casks containing 1-, 3-, and 10-year-old spent fuel, the initial value of $\Sigma_n^{g\text{-}shld}$ (0.1590/inch) was adjusted until the new ratio of $[D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx})]$ was precisely 1.0. If, for example, one considers an Fe cask with 1-year-old fuel this procedure would yield

$$\frac{D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx}_1)}{D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx}_2)} = \frac{1.0846}{1.0} = \frac{D_n^{acc}(\text{appx}_2)}{D_n^{acc}(\text{appx}_1)}$$

$$= \frac{\exp\{-[\Sigma_n^{g\text{-}shld}(12.0) + \Sigma_n^{steel}(0.375 + 0.375 + 0.75)]\}}{\exp\{-[(0.1590)(12.0) + (0.1590)(1.5)]\}} \; . \tag{3.31}$$

In the case of the Fe cask, $\Sigma_n^{g\text{-}shld} = \Sigma_n^{steel}$, and Eq. 3.31 may be solved to yield

$$\Sigma_n^{g\text{-}shld} = \Sigma_n^{steel} = 0.1530/\text{inch}, \tag{3.32}$$

which is a somewhat better estimate of the average value of $\Sigma_n^{g\text{-}shld}$ across the entire gamma shield. As shown in Table 3.8 of Sect. 3.2.1, the corresponding values for 3- and 10-year-old fuel are 0.1546/inch and 0.1565/inch, respectively. These same values were also assumed to apply to the thick inner and outer steel shells (and the outside liner) of the Pb and U-metal casks. Using typical gamma shield thicknesses of 5.7, 4.75, and 3.6 in. for Pb casks containing 1-, 3-, and 10-year-old fuel, the initial value of $\Sigma_n^{g\text{-}shld}$ (0.0602/inch) was then adjusted until the new ratio of $[D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx})]$ was precisely 1.0. If one considers a Pb cask with 1-year-old fuel the above procedure would yield

$$\frac{D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx}_1)}{D_n^{acc}(\text{exact})/D_n^{acc}(\text{appx}_2)} = \frac{1.0173}{1.0} = \frac{D_n^{acc}(\text{appx}_2)}{D_n^{acc}(\text{appx}_1)}$$

$$= \frac{\exp\{-[\Sigma_n^{g\text{-}shld}(5.7) + (0.1530)(1.5 + 2.0 + 0.75)]\}}{\exp\{-[(0.0602)(5.7) + (0.1590)(4.25)]\}} \; . \tag{3.33}$$

In this case, Eq. 3.33 may be solved to yield

$$\Sigma_n^{g\text{-}shld} = 0.0617/\text{inch}, \tag{3.34}$$

which is a somewhat better estimate of the average value of $\Sigma_n^{g\text{-shld}}$ across the entire thickness of the Pb gamma shield. As shown in Table 3.8 of Sect. 3.2.1, the corresponding values for 3- and 10-year-old fuel are 0.0643/inch and 0.0723/inch, respectively. Using typical gamma shield thicknesses of 3.35, 2.85, and 2.30 in. for U-metal casks containing 1-, 3-, and 10-year-old spent fuel, the corresponding revised values of $\Sigma_n^{g\text{-shld}}$ (0.2735, 0.2781, and 0.2860/inch) were all calculated in this same fashion. A summary of the revised data is given in Table 3.39.

Table 3.39. Final (best) estimates of $\Sigma_n^{g\text{-shld}}$ for Pb, Fe, and U-metal casks containing 1-, 3-, and 10-year-old spent fuel*

Cooling time	Pb cask	Fe cask	U cask
1 yr	0.0617	0.1530	0.2735
3 yr	0.0643	0.1546	0.2781
10 yr	0.0723	0.1565	0.2860

*Where $\Sigma_n^{g\text{-shld}}$ is given in units of inch^{-1}.

3.2.4.11 Determination of $\Sigma_n^{\text{i.shell}}$, $\Sigma_n^{\text{o.shell}}$, and $\Sigma_n^{\text{o.liner}}$

The inner and outer steel shells and the outside liner are all made of stainless steel. The corresponding neutron dose rate attenuation cross sections were therefore assumed to be the same as that for the carbon steel gamma shield in the Fe cask (cf. Table 3.39). Indeed, this assumption was already used in the final determination of $\Sigma_n^{g\text{-shld}}$ for the Pb and U-metal casks.

3.2.4.12 Determination of $\Sigma_{n,dif}^{\text{n.shld}}$ and $\Sigma_{n,avg}^{\text{n.shld}}$

The effective dose rate attenuation cross section to be used in simple exponential calculations of attenuation must depend on the total thickness of the shield, with the "effective" cross section being slightly less for thicker shields. [This reduced effective cross section accounts for neutrons that escape through the neutron shield after being scattered several times.] The net attenuation through the shield is therefore represented by the following expression:

$$D_{outer}^{n} = D_{inner}^{n}\exp\{-[\Sigma_{n,avg}^{\text{n.shld}} t_{ref}^{\text{n.shld}} + \Sigma_{n,dif}^{\text{n.shld}}(t_n - t_{ref}^{\text{n.shld}})]\} \qquad (3.35)$$

where $\Sigma_{n,avg}^{\text{n.shld}}$ is the average cross section over some typical reference thickness, and $\Sigma_{n,dif}^{\text{n.shld}}$ is the differential cross section applied to each additional increment of neutron shielding. Alternately, one could use a constant cross section for the entire shield and then multiply the resulting dose rate by a correction factor that increases with the thickness of the shield. Since the neutron shield thicknesses of most shipping casks vary over a relatively narrow and well-known range, Eq. 3.35 was deemed to be a more desirable approach. Because the effect of scattered radiation only becomes noticeable in highly absorbing shields of significant thickness, this approach was restricted to (1) attenuation of the neutron dose rate in the neutron shield and (2) attenuation of the gamma dose rate in the gamma shield.

3.2.4.12.1 Determination of $\Sigma_{n,dif}^{n.shld}$

Table 3.9 in Sect. 3.2.3 shows the intermediate results from a typical series of 1-D shielding calculations for a Pb cask containing 21 10-year-old PWR fuel assemblies. Iterations 6-10 show the neutron dose rate at 10 ft from the centerline as the neutron shield thickness is varied from 0.0010 to 4.1304 in. Using these two extremes, the average value of $\Sigma_n^{n.shld}$ may be estimated as

$$\Sigma_{n,avg}^{n.shld} = [\ln(39.1746/0.96354)]/(4.1304 - 0.0010) = 0.8973/\text{inch}, \tag{3.36}$$

while, using the results from iterations 8 and 10 where the neutron shield thickness varies from 2.9780 in. to 4.1304 in., the differential value of $\Sigma_n^{n.shld}$ may be estimated as

$$\Sigma_{n,dif}^{n.shld} = [\ln(2.33834/0.96354)]/(4.1304 - 2.9780) = 0.7693/\text{inch}. \tag{3.37}$$

As already noted, it should not be surprising that the differential cross section applicable to the outermost portion of the neutron shield is significantly less than the average cross section. For that reason, all estimates of the differential cross section [$\Sigma_{n,dif}^{n.shld}$] were based on intermediate data where, for a given cask, the neutron shield thickness was within an inch or so of the final thickness and the neutron dose rates were within a factor of 2 or 3 of the final dose rate. The final determination process was based on a more complete set of intermediate results (like those shown in Table 3.9) for Pb, Fe, and U-metal casks optimized for 1-, 3-, 5-, and 10-year-old spent fuel. In each case, estimated values of $\Sigma_{n,dif}^{n.shld}$ were calculated for casks containing 1 assembly, 12 assemblies, and 15, 21, 24, or 26 assemblies (depending on the age of the spent fuel and the maximum capacity of a legal weight cask containing such fuel). While there was some slight variation with the number of assemblies, the values of $\Sigma_{n,dif}^{n.shld}$ based on casks with 12 assemblies were considered typical. Indeed, these are the values reported in Table 3.8. Values of $\Sigma_{n,dif}^{n.shld}$ for the highest capacity casks were generally about 1% higher, while values of $\Sigma_{n,dif}^{n.shld}$ for casks with only one assembly were generally 2-5% lower. (Given a maximum error of 5% and a typical neutron shield thickness of 4.0 in., one would then have a maximum uncertainty of 0.2 in. in the thickness of the relatively lightweight neutron shield. For larger casks, the uncertainty would be less.)

From Table 3.8 it can be seen that the values of $\Sigma_{n,dif}^{n.shld}$ for the Fe casks are significantly greater than the corresponding values for the Pb or U-metal casks. This is primarily a spectral effect. Because the Fe gamma shields are much thicker than the Pb or U-metal gamma shields, and because the Fe atoms are significantly lighter, the neutrons entering the neutron shield of an Fe cask have a softer energy spectrum. Because of the softer spectrum they are then more readily absorbed by the boron in the neutron shield.

3.2.4.12.2 Determination of $\Sigma_{n,avg}^{n.shld}$

The intermediate results shown in Tables 3.10-3.18 may be used to obtain reasonable estimates of $\Sigma_{n,avg}^{n.shld}$ for Pb, Fe, and U-metal casks containing 1-, 3-, and 10-year-old spent fuel. To illustrate, consider the results in Table 3.16 for a Pb cask containing 21 10-year-old PWR fuel assemblies. For the optimal case (D'_γ = 30 mrem/hr), a 4.13-in.-thick neutron shield will yield a neutron dose rate of 0.964 mrem/hr 10 ft from the centerline; under accident conditions where the neutron shield is lost (t_n = 0.001 in.), the neutron dose rate there will rise to 39.2 mrem/hr. The average cross section may then be estimated as

$$\Sigma_{n,avg}^{n.shld} = [\ln(39.2/0.964)]/(4.13 - 0.001) = 0.8974/\text{inch}. \tag{3.38}$$

Data in the other tables may be used in similar fashion to obtain estimates of $\Sigma_{n,avg}^{n.shld}$ for the other cases of interest. Unfortunately, the reference thickness ($t_{ref}^{n.shld}$) of the neutron shield would be different in each case To simplify the problem, a slightly different approach was taken. After surveying a large number of casks (cf. Table IV.19 in ref. 4), it was found that optimized neutron shield thicknesses typically vary from 3.6 to 4.4 in. for Pb casks, from 3.5 to 4.0 in. for Fe casks, and from 2.7 to 3.8 in. for U-metal casks. Subsequent to that survey, constant reference thicknesses of 4.0, 3.75, and 3.25 were selected for Pb, Fe, and U-metal casks. Average neutron dose rate attenuation cross sections were then defined in terms of those reference thicknesses. Using D_n^{nom} and D_n^{acc} to represent the nominal and accidental neutron dose rate 10 ft from the centerline, $\Sigma_{n,avg}^{n.shld}$ may be written as

$$\Sigma_{n,avg}^{n.shld} = \{\ln(D_n^{acc}/D_n^{nom}) - \Sigma_{n,dif}^{n.shld}(t_n - t_{ref}^{n.shld})\}/t_{ref}^{n.shld} , \tag{3.39}$$

where $\Sigma_{n,dif}^{n.shld}$ is the differential neutron dose rate attenuation cross section described in Sect. 3.2.4.12.1, $t_{ref}^{n.shld}$ is the reference shield thickness described above, and t_n is the actual neutron shield thickness. To be entirely consistent with $\Sigma_{n,dif}^{n.shld}$, the final estimates of $\Sigma_{n,avg}^{n.shld}$ shown in Table 3.8 were based on values of D_n^{acc}, D_n^{nom}, and t_n obtained for optimized Pb, Fe, and U-metal casks carrying 12 1-, 3-, 5-, and 10-year-old PWR spent fuel assemblies. These intermediate parameters are given in Tables 3.40-3.42. Very similar results could have been obtained, however, using the less precise data in Tables 3.10-3.18.

Table 3.40. Optimized neutron shield thicknesses (inches) for Pb, Fe, and U-metal casks containing 12 1-, 3-, 5-, and 10-year-old PWR spent fuel assemblies

Cooling time	Pb cask	Fe cask	U cask
1 yr	4.3842	3.9735	3.7160
3 yr	4.1886	3.7868	3.4835
5 yr	4.0668	3.5730	3.2969
10 yr	4.0407	3.6843	3.2600

Table 3.41. Nominal neutron dose rates (D_n^{nom} in mrem/hr) 10 ft from the centerline of optimally designed Pb, Fe, and U-metal casks containing 12 1-, 3-, 5-, and 10-year-old PWR spent fuel assemblies*

Cooling time	Pb cask	Fe cask	U cask
1 yr	0.95398	0.21198	1.15929
3 yr	0.74891	0.21800	1.00605
5 yr	0.81008	0.29571	1.18545
10 yr	0.76003	0.28400	1.16346

*Based on multigroup S_8P_3 discrete ordinates shielding calculations with neutron shield thicknesses (t_n) as shown in Table 3.40.

Table 3.42. Accidental neutron dose rates (D_n^{acc} in mrem/hr) 10 ft from the centerline of optimally designed Pb, Fe, and U-metal casks containing 12 1-, 3-, 5-, and 10-year-old PWR spent fuel assemblies*

Cooling time	Pb cask	Fe cask	U cask
1 yr	52.5824	17.2289	31.8697
3 yr	33.6108	13.1074	22.4863
5 yr	31.4684	13.4626	22.2998
10 yr	27.2499	13.0027	20.4880

*Based on multigroup S_8P_3 discrete ordinates shielding calculations for casks in which the loss of the neutron shield was simulated by setting t_n = 0.001 in.

4. ALGORITHM USED BY CAPSIZE FOR DETERMINING NEAR OPTIMAL NEUTRON AND GAMMA SHIELD THICKNESSES

In the CAPSIZE program, the thickness of the neutron and gamma shields will be calculated so that, when used in conjunction with the inner and outer steel shells and the outside liner described in Table 2.1, the combined neutron and gamma dose rates 10 ft from the centerline of the cask will be reduced to some desired dose rate (D_d) set by the user. The corresponding neutron and gamma dose rates with none of these five components present (D_n^o and D_γ^o) are given by Eqs. 3.5a and 3.5b. As noted in Sects. 3.2.2, 3.2.4.4, 3.2.4.7, and 3.2.4.9, these values depend on the burnup of the fuel, the cooling time, the number of assemblies in the cask, and the corresponding SAS2/ORIGEN-S results described in Sect. 3.1.

The procedure used by CAPSIZE to determine the neutron and gamma shield thicknesses is essentially the same as that outlined in Sect. 3.2.3 except that: (1) simple exponential shielding formulae using the one-group data derived in Sect. 3.2.4 will be used in place of the more rigorous multigroup S_8P_3 discrete ordinates shielding calculations; and (2) the initial design point (D_γ') used to establish the thickness of the gamma shield will now be proportional to the final [combined] dose rate (D_d) specified by the user--that is,

$$D_\gamma'(\text{new}) = D_\gamma'(\text{old}) \times [D_d/(10.0 \text{ mrem/hr})] \,, \tag{4.1}$$

where D_γ'(old) corresponds to the optimal initial design points given in Table 3.19 for Pb, Fe, or U-metal casks containing 1-, 2-, 3-, 5-, 7-, or 10-year-old spent fuel. Given the type of cask and the age of the spent fuel, the data points in Tables 3.7 and 3.8 are then interpolated to obtain the most appropriate set of cross sections for the cooling time of interest. [For cooling times in excess of 10 years, the code uses the tabulated values at 10 years; for cooling times less than 1 year, it uses the values at 1 and 3 years to extrapolate back to the time of interest. This same procedure is also used for the initial design point D_γ'(old).] The thickness of the gamma shield is then calculated as

$$t_\gamma = t_{ref}^{g.shld} + \Delta t_\gamma \,, \tag{4.2}$$

where

$$t_{ref}^{g.shld} = 4.5 \text{ in. for Pb casks,} \tag{4.3a}$$

$$= 10.0 \text{ in. for Fe casks,} \tag{4.3b}$$

$$= 2.75 \text{ in. for U-metal casks,} \tag{4.3c}$$

and

$$\Delta t_\gamma = [\ln(D_\gamma^o/D_\gamma') - \Sigma t]/\Sigma_{\gamma,dif}^{g.shld} , \tag{4.4}$$

where D_γ^o (given by Eq. 3.5b) is the unshielded gamma dose rate 10 ft from the centerline, D_γ' is the new initial design point given by Eq. 4.1, and

$$\Sigma t = \Sigma_\gamma^{i.shell} t^{i.shell} + \Sigma_{\gamma,avg}^{g.shld} t_{ref}^{g.shld} + \Sigma_\gamma^{o.shell} t^{o.shell} . \tag{4.5}$$

Determining the neutron shield thickness that yields the total desired dose rate (D_d) specified by the user is an iterative process. If D_n^{10} and D_γ^{10} represent the neutron and gamma dose rates 10 ft from the centerline, then the total dose rate there is given by

$$D^{10} = D_n^{10} + D_\gamma^{10} , \tag{4.6}$$

where

$$D_\gamma^{10} = D_\gamma' \exp\{-[\Sigma_\gamma^{n.shld}(t_{ref}^{n.shld} + \Delta t_n) + \Sigma_\gamma^{o.liner} t^{o.liner}]\} , \tag{4.7}$$

and

$$D_n^{10} = D_n^o \exp\{-[\Sigma_n^{i.shell} t^{i.shell} + \Sigma_n^{g.shld} t_\gamma + \Sigma_n^{o.shell} t^{o.shell}$$

$$+ (\Sigma_{n,avg}^{n.shld} t_{ref}^{n.shld} + \Sigma_{n,dif}^{n.shld} \Delta t_n) + \Sigma_n^{o.liner} t^{o.liner}]\} , \tag{4.8}$$

where

$$t_{ref}^{n.shld} = 4.00 \text{ in. for Pb casks,} \tag{4.9a}$$

$$= 3.75 \text{ in. for Fe casks,} \tag{4.9b}$$

$$= 3.25 \text{ in. for U-metal casks,} \tag{4.9c}$$

D_γ' is the new initial design point given by Eq. 4.1, and D_n^o (given by Eq. 3.5a) is the unshielded neutron dose rate 10 ft from the centerline. To determine the required amount of neutron shielding,

$$t_n = t_{ref}^{n.shld} + \Delta t_n , \tag{4.10}$$

the value of Δt_n is varied until the value of D^{10} given by Eq. 4.6 is the same as the total desired dose rate (D_d) specified by the user. Because D^{10} varies in a monotonic fashion, a very efficient binary search procedure is used to determine Δt_n.

5. DETAILS OF MISCELLANEOUS CALCULATIONS PERFORMED BY CAPSIZE

5.1 DETERMINING THE SIZE OF THE REMOVABLE ALUMINUM BASKET AND THE INSIDE DIAMETER OF THE CASK

The inner diameter of the cask depends primarily on the number of assemblies (N) in the cask and the thickness (t) of the aluminum basket between the fuel assemblies. It also depends on the minimum thickness (w) of the aluminum basket between the outermost fuel assemblies and the inner wall of the cask. Here, it is assumed that w = 1.0 in. in all cases. It is also assumed that a clearance (t_g) of 0.125 in. is provided between the spent fuel assembly and the aluminum basket, and between the aluminum basket and the inner wall of the cask. If the width of a typical PWR fuel assembly (W_a) is assumed to be 8.445 in., then the width of the basket cavity receiving the fuel assembly is given by

$$W_c = W_a + 2t_g , \tag{5.1}$$

the effective pitch between fuel assemblies is given by

$$P = W_c + \theta t \quad (\text{where } \theta=0 \text{ for } N=1, \text{ and } \theta=1 \text{ for } N>1), \tag{5.2}$$

and the inner diameter of the cask is calculated as

$$D_i = P\sqrt{(D/P)^2} - \theta t + 2(w + t_g) , \tag{5.3}$$

where $(D/P)^2$ is given in Table 5.1, and (D/P) is the minimum diameter-to-pitch ratio for a dense array of square assemblies inside a cylindrical container (cf. Fig. 2.4a-e). Single assembly casks are assumed to contain no aluminum basket, and θ is set equal to zero. For casks with more assemblies, the outer radius of the basket is calculated as

$$R_b = 0.5D_i - t_g , \tag{5.4}$$

and the cross-sectional area of the 13.8-ft-long basket is calculated as

$$A_b = \pi R_b^2 - NW_c^2 . \tag{5.5}$$

The aluminum basket (also called an insert or fuel assembly separator) is assumed to weigh 168.49 lb/ft^3.

Table 5.1. Minimum[a] diameter-to-pitch ratios for dense arrays of square assemblies in a cylindrical container (see Figs. 2.4a-e)

No. of assemblies	$(D/P)^2$	No. of assemblies	$(D/P)^2$
1	2	23-24	40
2	5	25-26	41
3	6.65	27-31	50
4	8	32	52
5	10	33-34	53
6-7	13	35-37	58
8	15.68	38-39	61
8	17	40-42	65
9-10	18	43-44	68
11-12	20	45-46	72
13	22.60	47-48	74
14-15	26	49-52	80
16-18	32	53-56	85
19-21	34	57-58	89
22	37	59-61	90

[a]These values were used in the original version of the CAPSIZE program. A newer and more complete list used in recent versions of the program is given in Appendix B.

5.2 OVERALL LOADED WEIGHT OF A CASK

Each PWR fuel assembly is assumed to weigh 1509.8 lb. The cross-sectional area (A_b) of the removable aluminum basket, also called an insert or fuel assembly separator, is given by Eq. 5.5. A density of 168.49 lb/ft^3 is assumed when calculating the weight of the 13.8-ft-long removable aluminum basket.

The inner diameter of the cask cavity (D_i) is given by Eq. 5.3, while the length of the cavity inside the cask (L_i) is assumed to be 14.3 ft. This provides a 6-in. space in the axial direction for a set of lightweight internal shock absorbers which are otherwise ignored in this analysis.

The volume of the inner steel shell, the gamma shield, the outer steel shell, the neutron shield, and the outside liner (sometimes called the outer barrel) are each calculated as

$$V_j = \pi R_j^2 L_j - \pi R_{j-1}^2 L_{j-1} . \tag{5.6}$$

where

$$R_j = R_{j-1} + t_j . \tag{5.7}$$

$$L_j = L_{j-1} + 2t_j . \tag{5.8}$$

and t_j is the thickness of the particular component. The thickness of the inner steel shell, the outer steel shell, and the outside liner will depend on the type of cask, as shown in Table 2.1. The density of these stainless steels is assumed to be 494.43 lb/ft^3. The neutron and gamma shield thicknesses are calculated as shown in Sect. 4. The Pb gamma shield weighs 708.56 lb/ft^3, the carbon-steel (Fe) gamma shield weighs 488.26 lb/ft^3, and the depleted U-metal gamma shield weighs 1189.25 lb/ft^3. The neutron shield described in Sect. 2 is assumed to weigh 62.43 lb/ft^3 in all cases.

External cooling fins, when required, are assumed to be made of stainless steel weighing 494.43 lb/ft^3. The external volume of each fin is given by

$$V_f^{ext} = \pi(R_6^2 - R_5^2)t_f \ , \tag{5.9}$$

where t_f is optimal thickness of the fins, $\ell_f = R_6 - R_5$ is the optimal length of the fins, $D_5 = 2R_5$ is the outside diameter of the outside liner, and $D_6 = 2R_6$ is the outside diameter of the cask (including fins) as given by the CAPSIZE program. The optimal length and thickness of the fins will depend on the internal decay heat load, the ambient temperature, and the maximum allowable surface temperature. When required, these dimensions will be calculated as noted in Sect. 5.4. The total number of fins (N_f) will generally vary from 46 to 50, depending on the final length of the cask. In all cases, these circumferential fins are assumed to be spaced every 4 in. along the length of the cask.

When calculating the weight of a cask with cooling fins, each fin is actually assumed to extend outward from the outer steel shell, through the neutron shield, through the outside liner, and out into the surrounding air. The volume given by Eq. 5.9 accounts only for the portion of the fin beyond the outside liner. That portion of the fin inside the cask is given by

$$V_f^{int} = \pi(R_4^2 - R_3^2)t_f \ , \tag{5.10}$$

where the neutron shield extends from R_3 to R_4. When calculating the weight of the cask, the code therefore diminishes the volume of the neutron shield by $N_f V_f^{int}$, and increases the volume ascribed to the stainless steel fins by the same amount.

5.3 CRITERIA FOR USING EXTERNAL COOLING FINS

For each cask considered, the CAPSIZE program will perform the same steady-state thermal analysis previously incorporated in the SCOPE Shipping Cask Optimization and Parametric Evaluation code (cf. Sect. V of ref. 4). Because the CAPSIZE program was intended as a simple desktop tool for interactively determining the size and capacity of casks meeting certain constraints, the more voluminous output associated with the thermal analysis has been suppressed. Some of that information, however, is both useful and necessary. The size and weight of any cask will, for example, depend on the presence or absence of external cooling fins and, ultimately, on their design. In general, the CAPSIZE program assumes that external cooling fins will not be used on a cask if it can dissipate the internal decay heat load to the environment (at 130°F) while maintaining an outside surface temperature less than 250°F. Assuming a cask has no fins, the temperature on the outside surface can be calculated in an iterative fashion using the following expression:

$$Q_{leak} = A_{cask}\{\sigma\epsilon_c(T_{surf}^4 - T_{amb}^4) + C(T_{surf} - T_{amb})^{4/3}\} \ , \tag{5.11}$$

where

T_{surf} and T_{amb} are the surface and ambient temperatures in degrees Rankine (°R),

σ is the Stefan-Boltzmann constant [0.1714×10^{-8} (Btu/hr)/(ft^2)(°R)^4],

ϵ_c is the surface emissivity of the cask; typically 0.587 for stainless steel (dimensionless),

C is the constant (0.18) used in the McAdams correlation[11] for natural convection heat transfer when one has a horizontal cylinder in air. [Note: for large rail and truck casks at these elevated temperatures (T_{amb} = 130°F), the Grashof-Prandtl number product shows the boundary layer to be in the turbulent regime, thus dictating the present choice for C.]

A_{cask} is the outer surface area (ft^2) of the outside liner, not including the two ends,

Q is the total decay heat load (Btu/hr) imposed by the spent fuel [based on ORIGEN results and the number of fuel assemblies]*,

and

Q_{leak} is the total amount of heat dissipated by the cask per unit time, as given by Eq. 5.11 (Btu/hr).

To solve Eq. 5.1 for the cask surface temperature, the program uses a "binary-split" search procedure in which it first assumes a very high value for $T_{surf}^{(0)}$ (3460°R), calculates a value for $Q_{leak}^{(0)}$, picks $T_{surf}^{(1)}$ to be midway between T_{amb} and $T_{surf}^{(0)}$, and calculates another value for $Q_{leak}^{(1)}$; depending on whether $Q_{leak}^{(1)}$ is higher or lower than the known decay heat load (Q), the code will then pick $T_{surf}^{(2)}$ so as to be midway between T_{amb} and $T_{surf}^{(1)}$, or midway between $T_{surf}^{(1)}$ and $T_{surf}^{(0)}$. In just a few iterations, T_{surf} can be calculated to any desired degree of accuracy (typically ± 0.05°R). If the outside surface temperature is found to be less than 250°F (710°R), the program assumes that external cooling fins will not be used. In practice, fins are seldom required for casks carrying spent fuel that has been out of the reactor for more than 2 or 3 years.

5.4 DETERMINATION OF THE OPTIMAL FIN DIMENSIONS

In those cases where external cooling fins are necessary, the CAPSIZE program will perform a numerical search to determine the optimal fin dimensions (ℓ_f and t_f) that will minimize the weight of the loaded cask while keeping the outside surface temperature at (or slightly below) 250°F. The circumferential fins previously described are assumed to be made of stainless steel and spaced every 4 in. along the length of the cask. Typically, this will yield 46-50 fins per cask. In the search for the optimal fin dimensions, the fin thickness is varied from 0.25 in. to 2.0 in. in increments of 0.0625 in., while the fin length ($\ell_f = R_6 - R_5$) is varied from 2 in. to 12 in. in increments of 1.0 in.

The actual search procedure for determining the optimal fin dimensions is described in detail in Sect. V.B of ref. 4. For each proposed set of fin dimensions the program will:

1. calculate the various geometric view factors (fin → fin, fin → cask, etc.),

*Direct solar heating of the larger rail casks may raise the outside surface temperature by 20-40°F. The CAPSIZE program currently neglects the effect of solar heating on the assumption that the cask would normally be sheltered from the direct rays of the sun by an opaque covering over a large, lightweight frame structure surrounding the entire cask.

2. calculate the effective emissivities of the cask and the fins (which, for long thick fins, may be quite different from the tabulated material properties),
3. calculate the heat transfer coefficients for the cask and the fins using expressions similar to that shown in Eq. 5.11 which accounts for thermal radiation as well as natural convection,
4. calculate the fin effectiveness (η), and adjust h_f accordingly,
5. calculate the total heat dissipated (Q_{leak}) by the fins and the cask assuming the surface temperature of the outside liner is 250°F,
6. if Q_{leak} is greater than the initial decay heat load and the weight of the cask with these fins is less than the previous minimum based on other dimensions, save ℓ_f and t_f for future reference.

After finding the optimal fin dimensions, the code will then calculate the actual temperature on the outer surface of the outside liner.

6. A TYPICAL INTERACTIVE SESSION USING CAPSIZE

6.1 USING THE CAPSIZE PROGRAM IN THE NORMAL INTERACTIVE MODE

The CAPSIZE program will run on IBM-PC, XT, or AT personal computers or on IBM compatibles. The presence of an 8087 or 80287 math coprocessor chip is highly desirable but not necessary. With such a chip installed, the program will run an order of magnitude faster than without one. (Actual running times are described below.) No other special equipment is necessary. The executable file, CAPSIZE.EXE, is similar to a load module on large mainframe computers. It is ready to go and fully self-contained. It does not need to read or access any other files, and it does not create or write any other files. Assuming this file is on a floppy disk in drive A, one need only type

A:CAPSIZE

(followed by a return) to begin execution. After the preliminary title page (i.e., screen), the following information will be displayed:

```
--------------------------------------------------------------------------------
BU=33000 MWD/MT    Time=10 yrs    Dose=10.0 mrem/hr    Weight Limit=190000 lbs

Separator Thickness= 2.2500 in
================================================================================

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb   CT=yy   DD=dd.d   WL=wwwwww   ST=s.ssss        [enter BU=0 to stop]
```

This shows the initial default values assigned to the various parameters [fuel burnup, cooling time, desired dose rate 10 ft from the centerline of the cask, the weight limit for the loaded cask, and the thickness of the basket (or insert) between the fuel assemblies inside the cask]. As shown below, one may change one or more of these values in any order. Alternately, one may simply press the return key, in which case the following information will be generated:

```
------------------------------------------------------------------------------

Capacities are as follows:

  Fe cask = 13 assys,  Wt=174141 lbs,  G-shld= 9.14"  N-shld= 3.71"  OD= 80.7"
  Fe cask = 15 assys,  Wt=190355 lbs,  G-shld= 9.21"  N-shld= 3.73"  OD= 84.7"

  Pb cask = 15 assys,  Wt=171718 lbs,  G-shld= 3.85"  N-shld= 4.10"  OD= 80.2"
  Pb cask = 18 assys,  Wt=197171 lbs,  G-shld= 3.91"  N-shld= 4.16"  OD= 86.6"

  U  cask = 21 assys,  Wt=188821 lbs,  G-shld= 2.44"  N-shld= 3.46"  OD= 82.6"
  U  cask = 22 assys,  Wt=200059 lbs,  G-shld= 2.45"  N-shld= 3.48"  OD= 85.4"

------------------------------------------------------------------------------
BU=33000 MWD/MT     Time=10 yrs     Dose=10.0 mrem/hr     Weight Limit=190000 lbs

Separator Thickness= 2.2500 in
==============================================================================

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb   CT=yy   DD=dd.d   WL=wwwwww   ST=s.ssss        (enter BU=0 to stop)
```

This shows that an Fe cask carrying 13 10-year-old fuel assemblies (previously irradiated to 33,000 MWD/MT) with a 2.25-in.-thick fuel assembly separator and enough neutron and gamma shielding to reduce the dose rate down to 10 mrem/hr 10 ft from the centerline will weigh somewhat less than the 190,000-lb weight limit, while an Fe cask carrying 15 such assemblies will be slightly over the 190,000-lb weight limit. It also shows the actual weight of each cask, the amount of neutron and gamma shielding required in each case, and the outside diameter of each cask. All dimensions are in inches. Similar output for a cask with 14 assemblies is not given since, as shown in Fig. 2.4b, a cask with 14 assemblies would have to use the same internal packing arrangement and have the same inside diameter as a cask with 15 assemblies.[a] As such, a cask with 14 assemblies would weigh essentially the same as a cask with 15 assemblies. In the case of the Pb casks, those with 16 or 17 assemblies would weigh essentially the same as the one with 18 assemblies.[b] The CAPSIZE program therefore filters out these extraneous cases. In the case of the U-metal casks, one containing 21 assemblies is just under the 190,000-lb weight limit, while one large enough for 22 assemblies is significantly over the specified weight limit.

The thickness of the fuel assembly separator was assumed to be 2.25 in. for all of the casks described above. This corresponds to the inherently subcritical fuel assembly separator described in Sect. III of ref. 4, where it is also called a removable aluminum basket or insert. By taking credit for the reduced fissile inventory in burned fuel or by developing a more advanced basket design that still ensures the criticality safety of the system, it may be possible to use a thinner fuel assembly separator to increase the capacity of a cask. While CAPSIZE makes no check on the neutronic acceptability (i.e., criticality safety) of the system, it will allow the user to change the separator thickness and determine what the effect would be on the capacity of the cask. Assume that one could independently show that a separator thickness of 1.5 in. was safely subcritical and that he was interested in determining

[a]A newer version of the program using an enhanced set of optimal packing configurations shows that an Fe cask with 14 assemblies would weigh 185,289 lbs (cf. Appendix C).

[b]The newer version of the program also shows that a Pb cask with 17 assemblies would weigh 184,955 lbs (cf. Appendix C).

the potential capacity of casks carrying 5-year-old spent fuel with a desired dose rate of just 2 mrem/hr 10 ft from the centerline. To perform the necessary CAPSIZE analysis, one would then enter

```
ST=1.5    CT=5    DD=2.0
```

on a single line, followed by a return. While the keywords (like ST=) may be typed in either upper or lower case, there must not be any embedded blanks between the three characters (i.e., between the ST and the =). After the numeric value, which may or may not contain a decimal point, one must have at least one blank space before the next keyword. CAPSIZE would then generate the following information[a] for the conditions specified by the user:

```
--------------------------------------------------------------------------------

Capacities are as follows:

  Fe cask = 10 assys,  Wt=179615 lbs,  G-shld=11.58"  N-shld= 4.07"  OD= 78.3"
  Fe cask = 12 assys,  Wt=191330 lbs,  G-shld=11.68"  N-shld= 4.11"  OD= 80.9"

  Pb cask = 13 assys,  Wt=180957 lbs,  G-shld= 5.48"  N-shld= 5.09"  OD= 78.8"
  Pb cask = 15 assys,  Wt=197103 lbs,  G-shld= 5.53"  N-shld= 5.16"  OD= 82.6"

  U  cask = 15 assys,  Wt=174604 lbs,  G-shld= 3.31"  N-shld= 4.53"  OD= 75.4"
  U  cask = 18 assys,  Wt=199658 lbs,  G-shld= 3.34"  N-shld= 4.64"  OD= 81.4"

--------------------------------------------------------------------------------
BU=33000 MWD/MT    Time= 5 yrs    Dose= 2.0 mrem/hr    Weight Limit=190000 lbs
Separator Thickness= 1.5000 in
================================================================================

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb   CT=yy   DD=dd.d   WL=wwwwww   ST=s.ssss        [enter BU=0 to stop]
```

Although the thinner fuel assembly separator would have reduced the size and weight of the casks, the higher source terms associated with the 5-year-old spent fuel and the lower desired dose rate of 2 mrem/hr tend to increase the shielding requirements, thus forcing a net reduction in the number of assemblies that can be carried in casks meeting the prescribed weight limit.

Assume that one next wanted to look at 58,000-lb truck casks optimized for fuel burned to 45,000 MWD/MT and, since criticality safety is generally not a problem for smaller truck casks, that the fuel assembly separator could be reduced to 0.5 in. The user would then enter

```
BU=45000    WL=58000    ST=0.5
```

on a single line followed by a return, and CAPSIZE would then generate the following field of information:

[a]The new version of the program shows that a Pb cask with 14 assemblies will weigh 192,058 lbs, and that a U-metal cask with 17 assemblies would weigh 187,634 lbs (cf. Appendix C).

```
--------------------------------------------------------------------------------

Capacities are as follows:

  Fe cask =  1 assys,  Wt= 67950 lbs,  G-shld=10.88"  N-shld= 3.97"  OD= 47.3"

  Pb cask =  1 assys,  Wt= 55011 lbs,  G-shld= 4.94"  N-shld= 4.69"  OD= 42.3"
  Pb cask =  2 assys,  Wt= 79150 lbs,  G-shld= 5.14"  N-shld= 5.03"  OD= 51.1"

  U  cask =  1 assys,  Wt= 43577 lbs,  G-shld= 2.98"  N-shld= 4.04"  OD= 35.6"
  U  cask =  2 assys,  Wt= 65428 lbs,  G-shld= 3.09"  N-shld= 4.41"  OD= 44.3"

--------------------------------------------------------------------------------
BU=45000 MWD/MT    Time= 5 yrs    Dose= 2.0 mrem/hr    Weight Limit= 58000 lbs

Separator Thickness=  .5000 in
================================================================================

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb   CT=yy   DD=dd.d   WL=vvvvvv   ST=s.ssss        [enter BU=0 to stop]
```

As noted in Sect. 5.1, casks with a single assembly will have an inside diameter of 14.55 in. and will not use an internal aluminum basket to hold the fuel assembly. Casks with more than one assembly will be assumed to have a removable aluminum basket (which is referred to here as the fuel assembly separator). Because the Pb and U-metal casks with one assembly were under the 58,000-lb weight limit specified by the user, results will be displayed for casks with one and two assemblies. Because the Fe cask with one assembly was already over the specified weight limit, the results with two assemblies are not shown (or even calculated). If this was the last case of interest, the user would enter

```
BU=0
```

(followed by a return) to terminate execution of the CAPSIZE program.

Running time on an IBM-PC, XT, or AT with an 8087 or 80287 math coprocessor chip is almost negligible. All three output edits shown above required a total of 28 seconds on a 4.77 MHz IBM-XT with the 8087 math coprocessor. The first case, corresponding to a 190,000-lb cask with 10-year-old fuel required 12 seconds to generate all the data shown; the second case, corresponding to a 190,000-lb cask with 5-year-old fuel, required 11 seconds; and the third case, corresponding to a 58,000-lb cask with 5-year-old fuel, required 5 seconds. To determine that a cask with 21 assemblies is just under the specified weight limit and that a cask with 22 assemblies is just over the specified weight limit, the CAPSIZE program will first determine the shielding requirements and overall weight for casks with one assembly, then two assemblies, then three assemblies, etc., until the specified weight limit is exceeded. As the specified weight limit is reduced, fewer internal calculations will be required and the program will appear to run somewhat faster. Casks with high internal heat loads requiring external cooling fins will take a few additional seconds as the algorithm for optimizing the fin design is envoked. Calculations for a 190,000-lb cask carrying 1-year-old spent fuel may, for example, take up to 44 seconds. While the CAPSIZE program will work on personal computers without an 8087 or 80287 math coprocessor chip, these systems will take about ten times longer to complete the same set of calculations. Since the math coprocessor chips are relatively inexpensive, they are highly recommended.

When running the CAPSIZE program in the interactive mode, one may always press the Shift-Print Screen keys on the IBM-PC, XT, or AT to print whatever happens to be on the screen at that time. Each of the three examples shown above represent a single screen of information. Alternately, if one wants a continuous printout of all the information generated during an interactive session, the Control-Print Screen keys may be pressed at the beginning and end of the session. In this mode, all of the information that appears on the screen will also be printed as it is generated. If one wishes to save all of the information in a file that can later be edited, then the CAPSIZE program should be run in batch mode as described in Sect. 6.2.

Lastly, it should be pointed out that the CAPSIZE program was designed to be a fast interactive desktop tool for estimating the size and capacity of Pb, Fe, and U-metal casks optimized to meet certain constraints (external dose rate, overall weight limit, etc.) while carrying a specific type of spent fuel (PWR assemblies having a specified burnup and cooling time). Unlike the earlier SCOPE code for Shipping Cask Optimization and Parametric Evaluation,[4] the CAPSIZE program will calculate the amount of neutron and gamma shielding required to meet the prescribed constraints for the type of fuel specified. Because of its interactive nature, however, little additional information is given. Once a range of interesting cask designs have been identified using the interactive CAPSIZE program, the SCOPE code may then be used to develop more detailed information. Using neutron and gamma shield thicknesses generated by CAPSIZE, the SCOPE code may be used to: (1) determine the cask dimensions, including the length and thickness of any external cooling fins that might be required; (2) calculate and list the steady-state temperatures at different points in the cask, given an arbitrary ambient temperature specified by the user; and (3) calculate and list the maximum transient temperatures in key components during and after the postulated 30-min fire. A very fast-running, inexpensive mainframe version of the SCOPE code is currently available free of charge from the Radiation Shielding Information Center in Oak Ridge, and a version for the IBM-PC may be available in the near future. Together, CAPSIZE and SCOPE may be used for a broad range of scoping analyses. Ultimately, however, detailed shielding, criticality safety, heat transfer, and stress analyses will have to be performed using more advanced codes as found in the NRC-sponsored SCALE system for Standardized Computational Analyses for Licensing Evaluation.[12]

6.2 USING THE CAPSIZE PROGRAM IN BATCH MODE SO AS TO CAPTURE THE OUTPUT IN A FILE FOR LATER USE

Section 6.1 described the necessary input for the CAPSIZE program, its use in an interactive mode, and how to make a printed copy of the output as it is generated. There are times, however, when one might like to save the output in a file which could be edited later for reporting purposes. To do this, one could initiate execution of the program by entering

A:CAPSIZE >d:output.fil

where "output.fil" is any arbitrary name for the file in which the output is to be stored, and "d" is the disk drive on which the file is to be written. While this procedure will work and the user may still enter input from the keyboard as before, this procedure is not recommended since all of the output (including prompts for the user to enter more data) will be routed to the output file, thus leaving the user "flying blind." A much better procedure is to initiate execution of the program by entering

A:CAPSIZE <c:input.fil >d:output.fil

where "input.fil" is an arbitrary name for a file containing the input for all of the cases of interest and "c" is the disk drive on which the input file is to be found. The program will then proceed with no additional input from the user, and save all of the output produced in the output file which may then be printed or edited at some later time. Nothing will appear on the screen until the program has terminated. To save the output for the cases described in Sect. 6.1, the corresponding input file would look like:

```
TITLE CARD (MUST BE INCLUDED; WILL BE IGNORED)
BU=33000  CT=10  DD=10.0  WL=190000  ST=2.25
BU=33000  CT= 5  DD= 2.0  WL=190000  ST=1.50
BU=45000  CT= 5  DD= 2.0  WL= 58000  ST=0.5
BU=0
```

While a dummy title card is required, it will be ignored by the program. Each additional line will define a particular case of interest. While it is not necessary to enter all five input parameters on each line, this practice is highly recommended when running in batch mode. If a parameter is not defined on a given line, the program will use the last value that was assigned to that parameter. Long input files with lots of permutations may therefore be difficult to interpret at some later time if those parameters subject to change are not defined on each line. (Interactive users don't have that problem since the program always displays the old values before prompting the user for new ones.) As always, the last line of the input file should say BU=0. This tells the program to stop execution. Without this line the system would get hung up, requiring the user to reboot.

7. COMPARISON OF CAPSIZE RESULTS WITH EARLIER, MORE EXACT RESULTS FOR LARGE AND SMALL CASKS OPTIMIZED FOR 1-, 2-, 3-, 5-, 7-, AND 10-YEAR-OLD SPENT FUEL

Table 7.1 shows CAPSIZE estimates of the neutron and gamma shield thicknesses, the overall weight, and the outside diameter of small Pb, Fe, and U-metal casks carrying 1-, 2-, 3-, 5-, 7-, or 10-year-old PWR spent fuel assemblies. These casks contain 2, 3, or 4 assemblies, depending on the age of the spent fuel. In all cases, the PWR fuel was irradiated to 33,000 MWD/MT, the desired dose rate 10 ft from the centerline of the cask was 10 mrem/hr, and the fuel assembly separator was assumed to be 2.25 in. thick. These smaller casks all weigh 70,000-100,000 lb.[a] Table 7.2 shows similar CAPSIZE results for larger casks containing 10-22 assemblies. These casks all weigh 190,000-220,000 lb.[b] Tables 7.1 and 7.2 also show a set of highly optimized neutron and gamma shield thicknesses which, when modeled using multigroup S_8P_3 discrete ordinates calculations, have been found to yield dose rates of 10 mrem/hr (± 1%) 10 ft from the centerline. Using these predetermined optimized shield thicknesses, the overall weight of each cask was calculated using the SCOPE code. (Indeed, all of the results for the larger rail casks are taken directly from Table VII.3 of ref. 4.) *Comparisons of the shield thicknesses show that the CAPSIZE program can generally estimate the necessary neutron and gamma shield thicknesses to within 0.16 and 0.08 in., respectively, while comparisons of the overall weights show the CAPSIZE results to generally be within 1000 lb of the previously reported results.* The only differences worth mentioning relate to the large rail casks carrying 10 or 12 1-year-old fuel assemblies. Although the neutron and gamma shield thicknesses calculated by CAPSIZE differ by less than 0.02 in. from the earlier, more exact results, the outside cask dimensions and the overall weights do differ by a significant amount. This is because the CAPSIZE program assumes the external cooling fins to be made of stainless steel while, in this particular set of SCOPE analyses, the external cooling fins were assumed to be made of copper, which has a much higher thermal conductivity and which can therefore dissipate the same amount of decay heat with much thinner and much shorter cooling fins that weigh considerably less. In the case of the U-metal

[a]While a weight limit of 70,000 lb was specified, the results shown in Table 7.1 correspond to those casks that were just over that weight limit.

[b]While a weight limit of 190,000 lb was specified, the results shown in Table 7.2 correspond to those casks that were just over that weight limit.

cask with 12 assemblies, for example, the optimal stainless-steel fins would have to be 10 in. long, whereas the optimal copper fins would only have to be 6 in. long. When the SCOPE analysis was repeated using stainless-steel fins (and the same neutron and gamma shield thicknesses), the results were the same as those generated using the CAPSIZE program.

Tables 7.1 and 7.2 indicate that the estimates of neutron and gamma shield thicknesses and overall cask weights obtained using the CAPSIZE program are quite realistic and compare rather well with the more rigorous results that one would obtain if a complete set of optimization studies were to be conducted using multigroup discrete ordinates codes for the shielding analysis. Several points and counterpoints should be noted, however. (1) The argument has been made that the one-group cross-section data used by CAPSIZE was based on data in ref. 4, that use of that data should reproduce the data in ref. 4, and that the comparisons in Tables 7.1 and 7.2 are therefore meaningless. While this is a legitimate concern, the good comparisons shown in Tables 7.1 and 7.2 should not be dismissed as totally meaningless since they tend to confirm the adequacy of the many approximations found in the program. Note, for example, that: (a) The source terms used in CAPSIZE are based on a 2-D interpolation of the SAS2/ORIGEN-S data in Tables 3.1-3.3, not on the original ORIGEN-2 data in ref. 4. (b) The unshielded neutron and gamma source terms used by CAPSIZE also depend on very simple correlations that relate the relative strengths to the number of assemblies in the cask. These account for spatial self-shielding by the fuel itself. The data in Tables 7.1 and 7.2 tend to confirm the adequacy of these simple correlations. (c) CAPSIZE employs a very simple algorithm using both an average cross section and a differential (incremental) cross section to account for the attenuation of neutrons in the neutron shield and the attenuation of photons in the gamma shield. Based on the data in Tables 7.1 and 7.2, it can now be said that these simple algorithms appear to adequately represent the net attenuation through a broad range of shield thicknesses typically found in spent fuel casks. (d) To account for the effect of spectral changes in the source terms over time, the one-group cross-section data used by the CAPSIZE program is primarily based on more rigorous results at cooling times of 1, 3, and 10 years. The data used at other decay times is simply interpolated. Based on the tabulated results at 2, 5, and 7 years, this simple approach appears to be adequate. (e) The availability of accurate source terms and good cross-section data do not, by themselves, guarantee that one will accurately predict the optimal amount of neutron and gamma shielding that will minimize the weight of a cask. In the present case, at least, the simple shielding design algorithm used by CAPSIZE appears to yield results that are very close to the known minima. Taken collectively, these findings (a-e) are not insignificant. (2) The criticism noted above does have some merit, however. In particular, the calculated capacity of casks carrying extended burnup fuels may not be quite optimal since the neutron source grows exponentially with burnup whereas the gamma source grows linearly with burnup. While the neutron and gamma shield thicknesses calculated by CAPSIZE should reduce the total dose rate down to the desired value 10 ft from the centerline, and while the shield thicknesses calculated by CAPSIZE appear to be reasonably balanced (i.e., not lopsided), there is no guarantee that the shielding design algorithm currently used by CAPSIZE will give thicknesses that actually correspond to the minimum overall cask weight for that type of fuel. Moreover, as indicated in Sect. 3.1, there are also some grounds for speculating that the ORIGEN-based neutron source terms used by CAPSIZE may be somewhat conservative (i.e., high) relative to more realistic values that one might obtain for higher-enriched fuels specifically designed for extended burnup. These uncertainties should be investigated when and if funding becomes available.

While further comparisons would still be desirable for optimized casks carrying extended burnup fuels, the degree of accuracy demonstrated to date suggests that the CAPSIZE program will be a valuable desktop tool for evaluating the likely impact of proposed cask design specifications.

Table 7.1. Comparison of CAPSIZE results with earlier, more exact SCOPE results for small 70,000- to 100,000-lb casks optimized for 1-, 2-, 3-, 5-, 7-, and 10-year-old PWR spent fuel

	Type cask	No. of assemblies	Weight (lbs)	Thickness of Shield G-shield	N-shield	Outside diameter
CT= 1 Year						
CAPSIZE	Fe	2	101849	11.30 in.	3.86 in.	57.8 in.
SCOPE	Fe	2	100400	11.15 in.	3.85 in.	57.5 in.
CAPSIZE	Pb	2	85013	5.27 in.	3.84 in.	51.2 in.
SCOPE	Pb	2	83800	5.16 in.	3.86 in.	51.0 in.
CAPSIZE	U	2	70921	3.19 in.	3.11 in.	44.1 in.
SCOPE	U	2	69800	3.12 in.	3.10 in.	43.9 in.
CT= 2 Y						
CAPSIZE	Fe	2	92356	10.36 in.	3.77 in.	55.7 in.
SCOPE	Fe	2	92300	10.36 in.	3.83 in.	55.8 in.
CAPSIZE	Pb	2	77752	4.63 in.	3.81 in.	49.9 in.
SCOPE	Pb	2	77600	4.62 in.	3.80 in.	49.8 in.
CAPSIZE	U	3	77132	2.91 in.	3.11 in.	47.2 in.
SCOPE	U	3	77200	2.90 in.	3.25 in.	47.5 in.
CT= 3 Y						
CAPSIZE	Fe	2	87777	9.89 in.	3.71 in.	54.7 in.
SCOPE	Fe	2	86400	9.74 in.	3.71 in.	54.4 in.
CAPSIZE	Pb	2	74242	4.31 in.	3.82 in.	49.2 in.
SCOPE	Pb	2	73000	4.20 in.	3.81 in.	49.0 in.
CAPSIZE	U	3	74075	2.72 in.	3.07 in.	46.8 in.
SCOPE	U	3	73000	2.66 in.	3.07 in.	46.7 in.

Table 7.1 (continued)

	Type cask	No. of assemblies	Weight (lbs)	Thickness of Shield G-shield	N-shield	Outside diameter
CT= 5 Years						
CAPSIZE	Fe	2	80660	9.13 in.	3.66 in.	53.0 in.
SCOPE	Fe	2	79200	9.00 in.	3.46 in.	52.4 in.
CAPSIZE	Pb	3	80248	3.92 in.	3.82 in.	52.2 in.
SCOPE	Pb	3	79100	3.83 in.	3.77 in.	51.9 in.
CAPSIZE	U	4	77429	2.47 in.	3.03 in.	49.0 in.
SCOPE	U	4	76800	2.45 in.	2.85 in.	48.6 in.
CT= 7 Years						
CAPSIZE	Fe	2	76502	8.67 in.	3.62 in.	52.1 in.
SCOPE	Fe	2	75300	8.56 in.	3.49 in.	51.6 in.
CAPSIZE	Pb	3	76688	3.62 in.	3.78 in.	51.5 in.
SCOPE	Pb	3	75700	3.55 in.	3.65 in.	51.1 in.
CAPSIZE	U	4	74436	2.31 in.	2.95 in.	48.5 in.
SCOPE	U	4	73900	2.28 in.	2.93 in.	48.4 in.
CT=10 Years						
CAPSIZE	Fe	2	72707	8.24 in.	3.58 in.	51.1 in.
SCOPE	Fe	2	71700	8.11 in.	3.63 in.	50.9 in.
CAPSIZE	Pb	3	73428	3.34 in.	3.72 in.	50.8 in.
SCOPE	Pb	3	72500	3.26 in.	3.78 in.	50.8 in.
CAPSIZE	U	4	71648	2.15 in.	2.85 in.	48.0 in.
SCOPE	U	4	71400	2.13 in.	2.86 in.	47.9 in.

Table 7.2. Comparison of CAPSIZE results with earlier, more exact SCOPE results for large 190,000- to 220,000-lb casks optimized for 1-, 2-, 3-, 5-, 7-, and 10-year-old PWR spent fuel

	Type cask	No. of assemblies	Weight (lbs)	Thickness of Shield G-shield	N-shield	Outside diameter
CT= 1 Year						
CAPSIZE	Fe	10	209054	12.19 in.	3.96 in.	91.7 in.[a]
SCOPE	Fe	10	205400	12.17 in.	3.96 in.	87.7 in.[b]
CAPSIZE	Pb	12	219601	5.92 in.	4.36 in.	94.0 in.[c]
SCOPE	Pb	12	195700	5.92 in.	4.38 in.	88.1 in.[d]
CAPSIZE	U	12	217377	3.55 in.	3.72 in.	90.5 in.[e]
SCOPE	U	12	172700	3.56 in.	3.72 in.	82.5 in.[f]
CT= 2 Years						
CAPSIZE	Fe	12	200953	11.32 in.	3.87 in.	86.3 in.
SCOPE	Fe	12	203500	11.46 in.	3.92 in.	86.7 in.
CAPSIZE	Pb	13	191255	5.29 in.	4.27 in.	83.6 in.
SCOPE	Pb	13	193300	5.39 in.	4.25 in.	83.8 in.
CAPSIZE	U	18	217352	3.26 in.	3.71 in.	90.9 in.[g]
SCOPE	U	18	220200	3.31 in.	3.67 in.	92.4 in.[h]
CT= 3 Years						
CAPSIZE	Fe	13	201722	10.87 in.	3.83 in.	84.4 in.
SCOPE	Fe	13	201300	10.85 in.	3.79 in.	84.3 in.
CAPSIZE	Pb	15	196321	5.00 in.	4.28 in.	82.9 in.
SCOPE	Pb	15	195800	4.99 in.	4.21 in.	82.7 in.
CAPSIZE	U	18	206994	3.07 in.	3.66 in.	86.4 in.
SCOPE	U	18	206200	3.06 in.	3.58 in.	86.2 in.

Table 7.2 (continued)

	Type cask	No. of assemblies	Weight (lbs)	Thickness of Shield G-shield	Thickness of Shield N-shield	Outside diameter
CT = 5 Years						
CAPSIZE	Fe	15	205669	10.13 in.	3.80 in.	86.7 in.
SCOPE	Fe	15	204700	10.11 in.	3.56 in.	86.2 in.
CAPSIZE	Pb	18	211122	4.51 in.	4.28 in.	88.0 in.
SCOPE	Pb	18	210100	4.49 in.	4.07 in.	87.5 in.
CAPSIZE	U	18	191120	2.74 in.	3.56 in.	81.5 in.
SCOPE	U	18	190600	2.74 in.	3.35 in.	81.1 in.
CT = 7 Years						
CAPSIZE	Fe	15	197709	9.66 in.	3.77 in.	85.7 in.
SCOPE	Fe	15	197200	9.65 in.	3.59 in.	85.3 in.
CAPSIZE	Pb	18	203877	4.20 in.	4.23 in.	87.3 in.
SCOPE	Pb	18	203300	4.20 in.	4.01 in.	86.8 in.
CAPSIZE	U	21	194548	2.60 in.	3.56 in.	83.1 in.
SCOPE	U	21	194500	2.61 in.	3.49 in.	83.0 in.
CT = 10 Years						
CAPSIZE	Fe	15	190355	9.21 in.	3.73 in.	84.7 in.
SCOPE	Fe	15	189800	9.19 in.	3.68 in.	84.5 in.
CAPSIZE	Pb	18	197171	3.91 in.	4.16 in.	86.6 in.
SCOPE	Pb	18	196600	3.90 in.	4.02 in.	86.3 in.
CAPSIZE	U	22	200059	2.45 in.	3.48 in.	85.4 in.
SCOPE	U	22	200200	2.46 in.	3.43 in.	85.4 in.

[a] Based on using 5 in. long external cooling fins made of stainless steel.

[b] Based on using 3 in. long external cooling fins made of copper.

[c] Based on using 8 in. long external cooling fins made of stainless steel.

[d] Based on using 5 in. long external cooling fins made of copper.

[e] Based on using 10 in. long external cooling fins made of stainless steel.

[f] Based on using 6 in. long external cooling fins made of copper.

[g] Based on using 4 in. long external cooling fins made of stainless steel.

[h] Based on using 5 in. long external cooling fins made of stainless steel.

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Appendix A

SHIELDING METHODS FOR EVALUATING THE VERSATILITY OF PARTICULAR SHIPPING CASKS

Appendix A.1 THE KWIKDOSE PROGRAM FOR THE IBM-PC

Appendix A.2 MULTIGROUP DISCRETE ORDINATES ADJOINT CALCULATIONS

Appendix A.1

THE KWIKDOSE PROGRAM FOR THE IBM-PC

After a shipping cask has been designed for a certain type of spent fuel, the number of assemblies it can carry is fixed, as are the thicknesses of the various steel shells, the neutron shield, and the gamma shield. The question then becomes "What other types of spent fuel may be shipped in the cask?"

Using the same neutron and gamma source terms, miscellaneous correlations, and one-group cross sections found in the CAPSIZE program, a fast new interactive shielding program called KWIKDOSE has been written to compute and display a 2-D table showing the total dose rate 10 ft from the centerline of a specific (user defined) cask, as a function of the spent fuel's burnup and cooling time. Table A.1 shows a typical interactive session using the KWIKDOSE program on an IBM-PC. The program is completely self-prompting and needs no additional explanation.* It is also quite fast. To generate the tabulated results shown in Table A.1, for example, required a total of 15 sec on an IBM-XT with an 8087 math coprocessor. Although approximate, and subject to all of the caveats described in Sect. 3, this information is useful in estimating what types of fuel may or may not be shipped in a particular cask.

I. Limitations of the KWIKDOSE Program

The most obvious limitation of the KWIKDOSE program is that it always assumes the neutron shield to be 1/3 water and 2/3 ethylene glycol, containing a total of 1% natural boron by weight, as described in Sect. 2. Other approximations, limitations, or uncertainties stem from a) the assumptions used in generating the neutron and gamma source terms, b) the limited ability of the code to account for spatial self-shielding of the gamma flux in the central fuel zone, c) the inability to account for subcritical neutron multiplication in the fuel, d) the inability to accurately model secondary gamma production and subsequent attenuation, e) the inability of the program to accurately model neutron transmission through very thin or nonexistent neutron shields, and f) the inability of the program to account for radically different source spectra. Each of these points are amplified below:

The total neutron and gamma source terms for PWR spent fuel at various burnups and cooling times were generated using the SAS2/ORIGEN-S code package, as noted in Sect. 3.1. For similar fuel and similar operating histories, these ORIGEN-S source terms will be essentially the same as those one would obtain using the ORIGEN2 code system. In general, however, the actual source terms will depend on the initial enrichment of the fuel, the presence of water holes in the assembly, the amount of boron in the coolant, the specific power density, and the entire operating history--as well as the overall burnup and cooling time. Obviously the source terms used by KWIKDOSE (cf. Tables 3.1-3.4) are tied to a very specific set of assumptions, some of which could be made more realistic.

Sections 3.2.2 and 3.2.4.7 describe a simple correlation which attempts to account for the spatial self-shielding of the gamma flux by the fuel. This correlation gives the effective gamma dose rate at the inner wall of the cask as a function of the number of assemblies in the cask. The supporting data, however, are based on calculations in which the fuel assemblies were homogenized with the removable aluminum basket inside the central cavity of the cask. The walls forming the compartments of the basket were assumed to be 1 in. thick, and the central cavity of the cask was assumed to be cylindrical

*To print the output as it is being generated, the user should press the Control-PrintScreen keys at the beginning and end of the interactive session.

Table A.1. Typical Interactive Session Using the KWIKDOSE Program on an IBM-PC

A>KWIKDOSE

Do you wish to continue (Y/N)? y

Enter number of PWR assemblies in cask: 21

Enter type of cask (Pb, Fe, U): Pb

Enter thickness of components in inches.
Press return to get default values ( ).

Inner steel shell (1.500):

Gamma shield (4.500): 3.97

Outer steel shell (2.000):

Neutron shield (4.000): 4.13

Outside liner (.750):

Table ___: Nominal neutron plus gamma dose rate, 10 ft from the centerline of the cask, as a function of the burnup and cooling time of the spent fuel

Cooling Time	Burnup (GWD/MT)											
	5	10	15	20	25	30	35	40	45	50	55	60
1 YRS	19.7	39.5	63.2	87.1	114.4	142.0	172.6	203.7	237.4	271.7	308.0	345.4
2 YRS	8.3	16.7	27.4	38.3	51.2	64.5	79.6	95.1	112.3	130.2	149.4	169.5
3 YRS	5.4	10.7	17.8	24.9	33.5	42.4	52.7	63.4	75.4	88.0	101.7	116.1
4 YRS	3.6	7.2	1[illegible].8	16.6	22.4	28.4	35.4	42.8	51.2	60.2	70.0	80.6
5 YRS	2.6	5.3	8.6	12.0	16.1	20.4	25.4	30.9	37.1	43.9	51.4	59.6
6 YRS	2.2	4.3	6.9	9.6	12.8	16.1	20.1	24.5	29.4	34.9	41.1	47.9
7 YRS	1.8	3.6	5.7	7.9	10.5	13.2	16.4	20.0	24.2	28.8	34.0	39.9
8 YRS	1.6	3.2	5.0	6.8	9.0	11.3	14.1	17.2	20.7	24.8	29.4	34.6
9 YRS	1.4	2.8	4.4	6.0	7.9	10.0	12.3	15.1	18.2	21.9	26.0	30.8
10 YRS	1.3	2.5	3.9	5.3	7.0	8.8	10.9	13.3	16.1	19.5	23.2	27.5
12 YRS	1.2	2.4	3.7	5.0	6.4	8.1	10.0	12.2	14.7	17.7	21.0	24.9
14 YRS	1.1	2.3	3.5	4.7	6.1	7.5	9.3	11.3	13.6	16.3	19.4	22.9
16 YRS	1.1	2.2	3.3	4.5	5.7	7.1	8.7	10.6	12.7	15.2	18.0	21.3
18 YRS	1.0	2.1	3.2	4.3	5.4	6.7	8.2	10.0	11.9	14.3	16.9	19.9
20 YRS	1.0	2.0	3.0	4.1	5.2	6.4	7.8	9.4	11.3	13.4	15.9	18.7
25 YRS	.9	1.8	2.7	3.6	4.5	5.6	6.8	8.2	9.7	11.6	13.6	16.0
30 YRS	.8	1.6	2.4	3.2	4.0	4.9	5.9	7.1	8.4	10.0	11.7	13.7

Do you wish to continue (Y/N)? n
Stop - Program terminated.

A>

(cf. Sect. 5.1). For a small truck cask with three assemblies and a thin-walled basket in a tight-fitting rectangular-shaped shield, the smeared UO_2 density may be more than twice as great as that in the reference calculations. In such a case, the fuel itself would actually provide more self-shielding than provided by the current correlation, and the KWIKDOSE values 10 ft from the centerline would appear to be too high. For very thick-walled baskets, the converse is also true. Certainly these correlations could be improved by making them dependent on the average density of the heavy metal (or UO_2) inside the central fuel zone.

Sections 3.2.2 and 3.2.4.9 describe a simple correlation which shows the parametric variation of the unshielded neutron dose rate as a function of the number of assemblies in a cask. The original discrete ordinates shielding calculations upon which the supporting data are based assumed that the PWR assemblies in the (dry, unpoisoned) casks had initially been enriched to 3.2 wt % ^{235}U, burned to 33,000 MWD/MTIHM, and cooled for ten years. Fissioning of the depleted ^{235}U and residual ^{239}Pu was included in these original calculations. Although KWIKDOSE does attempt to correlate the unshielded neutron dose rate with the number of assemblies in the cask, and although it does increase or decrease this in proportion to the total spontaneous fission plus (α,n) neutron source in the spent fuel at various burnups and cooling times, it cannot account for the somewhat higher subcritical neutron multiplication that might exist if the cask were filled with water or contained fuel that was less depleted. Likewise, it cannot account for fixed neutron poisons in the fuel assembly separator basket.

Although (n,γ) reactions produce secondary gammas throughout the entire cask, a large portion of these are produced in the borated neutron shield. The current version of the KWIKDOSE program cannot account for the production of secondary gammas, and makes a feeble attempt to account for their presence by using an artificially low gamma attenuation cross section in the outside liner (cf. Table 3.7). These are, however, two distinct physical processes and should be modeled separately. If, for example, the primary gamma shield were very thick or the ratio of the neutron-to-gamma source terms was significantly higher (as in the case of extended burnup fuel), then the current accounting mechanism would certainly underestimate the gamma dose rate outside the cask. A better approach would be to couple the secondary gamma production rate in certain zones with the neutron attenuation rate, and to restore the gamma attenuation cross section in the outside liner to a more realistic value.

The algorithm for accounting for neutron attenuation through the neutron shield is based on the use of an average cross section typical of some reference thickness together with a differential cross section that is applicable to each additional increment of neutron shielding (cf. Sect. 3.2.2 or 3.2.4.12). For that particular borated mixture of water and ethylene glycol, the algorithm is quite satisfactory over a wide range of typical shield thicknesses. The approach does, however, have one fundamental flaw. While the neutron dose rates on either side of an infinitely thin neutron shield should be identical, the present algorithm will show a reduction of 30-40% (see Eq. 3.4 and the data in Table 3.8). Thus, for casks with no neutron shield or very thin neutron shields, the KWIKDOSE program will underestimate the neutron dose rate outside the cask by a similar amount. (Indeed, such casks should probably be considered outside the program's intended range of applicability.) In retrospect, a better approach might have been to use an algorithm with a constant attenuation cross section and a gradually increasing, multiplicative dose buildup factor that accounts for the diminished effectiveness of additional neutron shielding but approaches 1.0 in the limit of a very thin neutron shield.

Lastly, it should be noted that the one-group cross-sections used by the program are based on shipping cask shielding calculations for PWR spent fuel burned to 33 GWD/MTIHM and cooled for 1-10 years. Spectral differences in the neutron or gamma source terms at shorter or longer cooling times, or at much lower or higher burnups, may cause the effective cross-section data to be slightly different. Moreover, if the cask nominally attenuates the gamma dose rate by five orders of magnitude, then a mere 3% reduction in the effective attenuation cross section could cause a 41% change in the calculated external dose rate. The applicability of this data to fuel at different burnups and cooling times is addressed further in Sect. A.1.II. Obviously it should not be used in grotesquely different applications.

II. Comparison of KWIKDOSE Results With More Exact Results for a Typical Cask Where KWIKDOSE Should be Applicable

The cask described in Table A.1 corresponds to a hypothetical rail cask containing 21 PWR fuel assemblies in a 56.54-in.-diam cavity surrounded by a 1.5-in.-thick inner steel shell, a 3.97-in.-thick Pb gamma shield, a 2.0-in.-thick outer steel shell, a 4.13-in.-thick neutron shield, and a 0.75-in.-thick outer steel barrel. Table A.2a shows the KWIKDOSE results for the total neutron plus gamma dose rate 10 ft from the centerline as the burnup was varied from 20 to 50 GWD/MTIHM and the cooling time was varied from one to ten years. Table A.2a also shows a set of more exact results obtained for the same cask using the XSDRNPM multigroup discrete ordinates code, the DLC-23/CASK coupled cross-section library having 22 neutron groups and 18 gamma groups, and the multigroup neutron and gamma source terms produced by the SAS2/ORIGEN-S calculations described in Sect. 3.1.* In the worst case, corresponding to a burnup of 20 GWD/MTIHM and a decay time of ten years, the KWIKDOSE result was 56% greater than the XSDRNPM results. This discrepancy is substantially less at higher burnups and/or shorter cooling times, with the two sets of results generally agreeing to within 10-20%.

Table A.2b shows a more complete set of results for this same cask, where the spent fuel burnup now ranges from 5 to 60 GWD/MTIHM and the cooling time ranges from 1 to 30 years. Agreement remains quite good as long as the burnup is high enough or the cooling time is short enough to yield dose rates above 8 to 10 mrem/hr; agreement deteriorates markedly, however, whenever the burnup is low enough and the cooling time is long enough to yield significantly lower dose rates. The good agreement at high burnup (50-60 GWD/MTIHM) and long cooling times (20-25 years) was particularly remarkable given the uncertainties associated with KWIKDOSE's inability to properly account for the secondary gammas produced by the significantly higher neutron source at these burnups. At low burnups and long cooling times, the external dose rates calculated by KWIKDOSE are significantly higher than the more exact XSDRNPM results. For this cask, the effect of the neutron source relative to the gamma source is insignificant at very low burnups. The lack of good, one-group, gamma attenuation data corresponding to the somewhat softer gamma spectrum from this older, low-burnup spent fuel is therefore believed to be the primary cause of the disagreement. [Due to the lack of an available data base, KWIKDOSE is forced to use one-group cross-section data based on source spectra from standard (33 GWD/MTIHM), 10-year-cooled spent fuel whenever the actual cooling time exceeds ten years. Data in Table 3.7 suggest that the actual gamma attenuation cross sections for 20 or 25-year-old spent fuel may be significantly higher.] The use of an artificially low gamma attenuation cross section in the outside liner to account for the now insignificant production of secondary gammas at low burnups is a second, less important cause of the discrepancy.

*XSDRNPM results at 33 GWD/MTIHM were calculated using the multigroup neutron and gamma source terms listed in Appendix C of Ref.4.

Table A.2a. Neutron plus gamma dose rate (mrem/hr), 10 ft from the centerline of the cask, as a function of the burnup and cooling time of the spent fuel

Cooling Time	Calc Type	Burnup (GWD/MT) 20	25	30	33	35	40	45	50
1 yr	(a)	87.1	114.4	142.0	160.3	172.6	203.7	237.4	271.7
	(b)	80.7	-----	127.3	165.1c	-----	179.2	-----	237.2
2 yr	(a)	38.3	51.2	64.5	73.5	79.6	95.1	112.3	130.2
	(b)	39.6	----	64.4	85.2c	----	93.7	----	128.5
3 yr	(a)	24.9	33.5	42.4	48.5	52.7	63.4	75.4	88.0
	(b)	21.4	----	36.2	49.9c	----	55.0	----	78.7
4 yr	(a)	16.6	22.4	28.4	32.5	35.4	42.8	51.2	60.2
	(b)	13.0	----	23.0	33.0c	----	36.4	----	54.4
5 yr	(a)	12.0	16.1	20.4	23.4	25.4	30.9	37.1	43.9
	(b)	8.9	----	16.3	24.3c	----	26.8	----	41.6
6 yr	(a)	9.6	12.8	16.1	18.5	20.1	24.5	29.4	34.9
	(b)	---	----	----	----	----	----	----	----
7 yr	(a)	7.9	10.5	13.2	15.1	16.4	20.0	24.2	28.8
	(b)	5.4	----	10.3	16.2c	----	18.0	----	29.2
8 yr	(a)	6.8	9.0	11.3	12.9	14.1	17.2	20.7	24.8
	(b)	4.5	---	8.8	14.0c	----	15.6	----	25.8
9 yr	(a)	6.0	7.9	10.0	11.3	12.3	15.1	18.2	21.9
	(b)	---	---	----	----	----	----	----	----
10 yr	(a)	5.3	7.0	8.8	10.0	10.9	13.3	16.1	19.5
	(b)	3.4	---	6.7	10.9c	----	12.4	----	21.1

(a) Results obtained using the KWIKDOSE program and the one-group source terms and cross sections contained therein (cf. Sect. 3).

(b) Results obtained by folding the results of a single multigroup XSDRNPM adjoint calculation with the 22-group neutron source terms and 18-group gamma source terms obtained from a series of SAS2/ORIGEN-S calculations performed as described in Sect. 3.1.

(c) Results obtained by folding the results of the same multigroup XSDRNPM adjoint calculation with the 22-group neutron source terms and 18-group gamma source terms listed in Appendix C of Ref. 4. These source terms were based on ORIGEN-2 analyses of a typical PWR fuel assembly.

Table A.2b. Neutron plus gamma dose rate (mrem/hr), 10 ft from the centerline of the cask, as a function of the burnup and cooling time of the spent fuel

Cooling time	Calc type	Burnup (GWD/MT)												
		5	10	15	20	25	30	33	35	40	45	50	55	60
1 yr	(a)	19.7	39.5	63.2	87.1	114.4	142.0	160.3	172.6	203.7	237.4	271.7	308.0	345.4
	(b)	---	38.5	---	80.7	---	127.3	165.1c	---	179.2	---	237.2	---	302.5
2 yr	(a)	8.3	16.7	27.4	38.3	51.2	64.5	73.5	79.6	95.1	112.3	130.2	149.4	169.5
	(b)	---	18.3	---	39.6	---	64.4	85.2c	---	93.7	---	128.5	---	169.9
3 yr	(a)	5.4	10.7	17.8	24.9	33.5	42.4	48.5	52.7	63.4	75.4	88.0	101.7	116.1
	(b)	---	9.6	---	21.4	---	36.2	49.9c	---	55.0	---	78.7	---	108.4
4 yr	(a)	3.6	7.2	11.8	16.6	22.4	28.4	32.5	35.4	42.8	51.2	60.2	70.0	80.6
	(b)	---	5.6	---	13.0	---	23.0	33.0c	---	36.4	---	54.4	---	77.8
5 yr	(a)	2.6	5.3	8.6	12.0	16.1	20.4	23.4	25.4	30.9	37.1	43.9	51.4	59.6
	(b)	---	3.8	---	8.9	---	16.3	24.3c	---	26.8	---	41.6	---	61.3
6 yr	(a)	2.2	4.3	6.9	9.6	12.8	16.1	18.5	20.1	24.5	29.4	34.9	41.1	47.9
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---
7 yr	(a)	1.8	3.6	5.7	7.9	10.5	13.2	15.1	16.4	20.0	24.2	28.8	34.0	39.9
	(b)	---	2.2	---	5.4	---	10.3	16.2c	---	18.0	---	29.2	---	44.9
8 yr	(a)	1.6	3.2	5.0	6.8	9.0	11.3	12.9	14.1	17.2	20.7	24.8	29.4	34.6
	(b)	---	1.8	---	4.5	---	8.8	14.0c	---	15.6	---	25.8	---	40.2
9 yr	(a)	1.4	2.8	[illegible]	6.0	7.9	10.0	11.3	12.3	15.1	18.2	21.9	26.0	30.8
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---
10 yr	(a)	1.3	2.5	3.9	5.3	7.0	8.8	10.0	10.9	13.3	16.1	19.5	23.2	27.5
	(b)	---	1.4	---	3.4	---	6.7	10.9c	---	12.4	---	21.1	---	33.6
11 yr	(a)	1.2	2.5	3.8	5.2	6.7	8.4	9.6	10.4	12.7	15.4	18.5	22.1	26.2
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---
12 yr	(a)	1.2	2.4	3.7	5.0	6.4	8.1	9.2	10.0	12.2	14.7	17.7	21.0	24.9
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---
13 yr	(a)	1.2	2.4	3.6	4.8	6.2	7.8	8.8	9.6	11.7	14.1	16.9	20.1	23.8
	(b)	---	.9	---	2.4	---	4.9	8.0c	---	9.3	---	16.5	---	26.9
14 yr	(a)	1.1	2.3	3.5	4.7	6.1	7.5	8.5	9.3	11.3	13.6	16.3	19.4	22.9
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---
15 yr	(a)	1.1	2.2	3.4	4.6	5.9	7.3	8.3	9.0	10.9	13.1	15.7	18.7	22.1
	(b)	---	.7	---	1.9	---	4.0	6.6c	---	7.9	---	14.3	---	23.7
16 yr	(a)	1.1	2.2	3.3	4.5	5.7	7.1	8.0	8.7	10.6	12.7	15.2	18.0	21.3
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---
18 yr	(a)	1.0	2.1	3.2	4.3	5.4	6.7	7.6	8.2	10.0	11.9	14.3	16.9	19.9
	(b)	---	.5	---	1.4	---	3.1	5.0c	---	6.3	---	11.8	---	19.9
20 yr	(a)	1.0	2.0	3.0	4.1	5.2	6.4	7.2	7.8	9.4	11.3	13.4	15.9	18.7
	(b)	---	.4	---	1.1	---	2.6	4.3c	---	5.5	---	10.4	---	17.9
23 yr	(a)	.9	1.9	2.8	3.8	4.8	5.9	6.7	7.2	8.6	10.3	12.3	14.5	17.0
	(b)	---	.3	---	.9	---	2.1	3.3c	---	4.5	---	8.8	---	15.3
25 yr	(a)	.9	1.8	2.7	3.6	4.5	5.6	6.3	6.8	8.2	9.7	11.6	13.6	16.0
	(b)	---	.3	---	.7	---	1.8	2.9c	---	4.0	---	7.9	---	13.9
30 yr	(a)	.8	1.6	2.4	3.2	4.0	4.9	5.5	5.9	7.1	8.4	10.0	11.7	13.7
	(b)	---	---	---	---	---	---	---	---	---	---	---	---	---

[a] Results obtained using the KWIKDOSE program and the one-group source terms and cross sections contained therein (cf. Sect. 3).

[b] Results obtained by folding the results of a single multigroup XSDRNPM adjoint calculation with the 22-group neutron source terms and 18-group gamma source terms obtained from a series of SAS2/ORIGEN-S calculations performed as described in Sect. 3.1.

[c] Results obtained by folding the results of the same multigroup XSDRNPM adjoint calculation with the 22-group neutron source terms and 18-group gamma source terms listed in Appendix C of Ref. 4. These source terms were based on ORIGEN-2 analyses of a typical PWR fuel assembly.

III. Comparisons of KWIKDOSE Results With More Exact Results for a Wide Variety of Hypothetical Shipping Casks

Unlike CAPSIZE, the KWIKDOSE program can be easily abused by users who may not be familiar with the many limitations of the program or the types of casks or spent fuels for which the program is applicable. Even among those who are aware of its many limitations, there is frequently a temptation to use the program beyond those limits when performing quick, first-cut scoping analyses. The inevitable question then is "How credible are the subsequent results?" Although such a question does not merit a detailed response, some insights may be provided by postulating hypothetical cask models composed of various materials and then comparing the external dose rates calculated by KWIKDOSE with those calculated using more rigorous methods and data. To that end, a wide variety of hypothetical cask designs were considered. The external dose rates used for reference were calculated using the 1-D XSDRNPM discrete ordinates tranpsort code, the SCALE ENDF-IV 27n-18g coupled cross-section library, and multigroup source terms based on an independent SAS2H/ORIGEN-S analysis of a Westinghouse 17 x 17 PWR fuel assembly burned to 35 GWD/MTIHM and cooled for ten years.

The ratio (K/X) of the dose rate calculated using KWIKDOSE to that calculated using the more exact XSDRNPM code was the parameter of interest in the subsequent comparisons. For large and small Pb, Fe, and U-metal casks similar to those described in Sect. 2, the KWIKDOSE dose rates were generally within 10 to 20% of those calculated by XSDRNPM. For large rail casks with neutron shields made of solid borated hydrocarbons, the KWIKDOSE program gave results that were 1.1 to 1.9 times higher than the XSDRNPM results, while for small truck casks with neutron shields made of solid borated hydrocarbons, the KWIKDOSE program gave results that were 1.4 to 3.0 times higher than the XSDRNPM results. For casks with neutron shields made of solid borated inorganic mixtures containing 5 to 7 wt% water, the KWIKDOSE program gave results that were 2.5 to 3.5 times higher than the XSDRNPM results, while for casks with no neutron shield or outside liner, the KWIKDOSE program gave results that were 1.1 to 2.0 times lower than the XSDRNPM results.

Too much should not be said or inferred based on loosely defined comparisons of this nature. It may, however, be helpful to keep other uncertainties in proper perspective. (1) Based on XSDRNPM results using the SAS2/ORIGEN-S source terms at 30 and 40 GWD/MTIHM, the corresponding dose rate at 33 GWD/MTIHM would be about 8.6 mrem/hr; using the ORIGEN2 source terms based on a slightly different initial enrichment, operating history, and power density, the same XSDRNPM calculation would yield a dose rate of 10.9 mrem/hr (cf. Table A.2a). While ORIGEN-S and ORIGEN2 will generally give the same results for the same burnup scenario, the point to be made here is that differences in the initial enrichment of the fuel or operating conditions in the reactor can easily cause an uncertainty of up to 27% in the dose rate outside a shipping cask. (2) Seemingly small uncertainties in the cross-section data can, when compounded over many mean free paths, lead to large uncertainties in the dose rate outside a cask. If a cask nominally attenuates the gamma dose rate by six orders of magnitude, then a 2% uncertainty in the cross-section data may cause the calculated external dose rate to be 32% too high or 24% too low, while a 3% uncertainty may cause the calculated external dose rate to be 51% too high or 34% too low.* Compounding these with a 27% uncertainty due to the source term could easily give uncertainties of up to 70% in the external dose rate. For an analyst to claim much greater accuracy using even high-order transport codes would require that he have detailed knowledge of the fuel being placed in the cask and a high degree of confidence in the cross-section data being used.

*Intercomparisons of cross-section libraries used for spent fuel cask shielding analyses have shown such variations in the calculated external dose rate to be common among the 17 multigroup libraries studied (cf. Table 2 of Ref. 13).

Appendix A.2

MULTIGROUP DISCRETE ORDINATES ADJOINT CALCULATIONS

After a shipping cask has been designed for a certain type of spent fuel, the number of assemblies it can carry is fixed, as are the thicknesses of the various steel shells, the neutron shield, and the gamma shield. The question then becomes "What other types of spent fuel may be shipped in the cask?"

Tabulated results showing the external dose rate as a function of the burnup and cooling time of the spent fuel (cf. Table A.1) are very useful in deciding what types of spent fuel may or may not be shipped in a particular cask. Ideally, however, it would be nice if this informaton could be generated in a more rigorous fashon for any arbitrary type of cask. Given the actual neutron and gamma source terms and the corresponding spectra for various types of fuel, one could, of course, perform an entire set of 1-D multigroup discrete ordinates shielding calculations to generate a similar set of results for any particular cask of interest. While the results would be useful, the cost associated with such a large number of cases may be prohibitive if done in the normal "forward" mode. Fortunately, the same field of information may be generated inexpensively with no loss of rigor by using the results of a single multigroup adjoint calculation.

I. Description of the Methodology

In the adjoint XSDRNPM calculation, the actual source in the homogenized fuel zone is set to zero, the multigroup **adjoint** source terms (S^{g*}) in a thin mesh interval at the point of interest (10 ft from the centerline) are defined to be proportional to the flux-to-dose conversion factors for the respective energy groups, and the total adjoint source per cm of length in the system (a parameter called XNF) is set equal to the sum of all the flux-to-dose conversion factors. The volume-integrated adjoint flux (Φ^{g*}) in what is normally the homogenized fuel zone is then the result of interest. [This volume-integrated quantity is explicitly listed as the "total flux" in the fine-group summary table for zone 1 in the XSDRNPM adjoint calculation.] Using the results of this single adjoint calculation, the dose rate at the point of interest may be easily obtained for spent fuel at any burnup and/or cooling time. If S^g represents the homogenized neutron and gamma source density (n/s/cm^3 and p/s/cm^3) in zone 1 for any arbitrary burnup and cooling time, then the dose rate at the point of interest is given by

$$D = \sum_g \Phi^{g*} S^g \ . \qquad \text{(A.1)}$$

Moreover, the different dose rates obtained by substituting different source terms in Eq. A.1 will be identical[†] to those that would be obtained if the normal (forward mode) shielding calculations were actually performed for each set of spent fuel source terms.

[†]While this is certainly true for any arbitrary set of source terms, it does assume that the isotopic composition of the spent fuel to be shipped is identical (or at least similar) to that of the spent fuel used in the adjoint calculation. Obviously if one went to ship 93% enriched fuel that had been burned down to 91%, the subcritical neutron multiplication factor and (perhaps) the resulting external dose rate would be different than what one would calculate if the adjoint calculation had been based on 3.2% enriched fuel that had been burned down to 1.2%. For low burnup fuels where the effect of the neutron source is negligible, or for heavily poisoned dry casks where the subcritical neutron multiplication factor is quite small, or for a broad range of PWR spent fuels where the fissile content of the spent fuel to be shipped is similar to that used in the adjoint calculation, the approach described above is quite good.

II. Mathematical Basis

While the well-known technique described above can (and has) been verified by direct numerical comparisons, the mathematical proof is quite straightforward. If S and R are vectors representing the multigroup source terms and the flux-to-dose response function, then the forward and adjoint problems may be posed in matrix notation as

$$B\phi = S \quad \text{and} \quad B^*\phi^* = R . \tag{A.2a,b}$$

Multiplying the first equation by $\tilde{\phi}^*$ and the second by $\tilde{\phi}$ [where $(\tilde{\ })$ denotes the transpose], we have

$$\tilde{\phi}^* B\phi = \tilde{\phi}^* S \quad \text{and} \quad \tilde{\phi} B^*\phi^* = \tilde{\phi} R = D . \tag{A.3a,b}$$

But, since the transpose of a product equals the product of the transposed components in reverse order,

$$\tilde{\phi}^* B\phi = \tilde{\phi}\tilde{B}\phi^* . \tag{A.4}$$

If the adjoint operator (B^*) is *defined* as the transpose of the forward operator ($\tilde{B}$), which is the case in multigroup transport codes, then B^* may be substituted for $\tilde{B}$ and Eq. A.4 may be written as

$$\tilde{\phi}^* B\phi = \tilde{\phi}\tilde{B}\phi^* = \tilde{\phi} B^*\phi^* . \tag{A.5}$$

Recognizing that the left-most term of Eq. A.5 is given by Eq. A.3a and that the right-most term of Eq. A.5 is given by Eq. A.3b, it follows that

$$D = \tilde{\phi} R = \tilde{\phi}^* S , \tag{A.6}$$

which, of course, is equivalent to Eq. A.1.

III. Additional Advantages

The attractive feature of this more exact approach is that the volume-integrated multigroup adjoint flux inside the central fuel zone of any arbitrary cask may be calculated in a rigorous fashion, tabulated on a single spec sheet, and treated essentially as a characteristic function of the cask itself. Likewise, multigroup source terms for spent fuel at various burnups, decay times, initial enrichments, power histories, etc., may be tabulated for future reference by other independent groups or organizations. (One such tabulation for standard PWR fuel at 33 GWD/MTIHM may be found in Appendix C of Ref. 4.) DOE personnel trying to evaluate the versatility of a proposed cask design, vendors trying to market a given cask to different customers, or regulatory personnel trying to evaluate the adequacy of an existing cask for somewhat different conditions, could then fold the volume-integrated multigroup adjoint flux with the accepted standardized source terms to quickly and easily generate a table of external dose rates for a wide variety of spent fuels.

IV. A Simple Example

A simple 1-D XSDRNPM model was created for the hypothetical shipping cask described in Sect. A.1.II. It consisted of a homogenized fuel zone (56.54 in. in diameter) containing 21 PWR spent fuel assemblies, a 1.5-in.-thick steel shell, a 3.97-in.-thick Pb gamma shield, a 2.0-in.-thick outer steel shell, a 4.13-in.-thick neutron shield, and a 0.75-in.-thick outer steel barrel. Number densities for the nuclides in these materials are shown in Table A.3. The model included 100 mesh intervals extending from the centerline to the surface of the cask ($R = 103.1748$ cm), plus another 151 mesh intervals extending from the surface of the cask out to a point 10 ft from the centerline. The last mesh interval was arbitrarily made 0.1 cm thick, and a vacuum boundary condition was applied along its outer surface. In the adjoint calculation, the volumetric source was set to 0.0 everywhere except in the last mesh interval where the volumetric sources for energy groups 1 to 22 were set equal to the neutron flux-to-dose conversion factors shown in Table 3.5, and the volumetric sources for energy groups 23 to 40 were set equal to the photon flux-to-dose conversion factors shown in Table 3.6. When added together, these 40 factors total 1.88137 (mrem/hr)/(particle/sec/cm^2). The source normalization factor (XNF), which represents the total adjoint source per unit length axially along the cask, was therefore set equal to 1.88137. Using an S_{12} quadrature set, the 1-D adjoint calculation was found to converge in only 11.2 minutes on the IBM-3033. The volume-integrated adjoint flux in what is normally the homogenized fuel zone was listed as the "total flux" in the fine-group summary table for zone 1 of the XSDRNPM output. These results are shown in Table A.4.

Assuming the cask is full of spent fuel having roughly the same amount of depleted fissile material, the dose rate at the point of interest (10 ft from the centerline) may be obtained by folding the volume-integrated adjoint fluxes in Table A.4 with the actual multigroup neutron and gamma source densities. Table A.5, taken from Table C.4 of Ref. 4, shows the multigroup neutron and gamma source spectra for 10-year-cooled PWR spent fuel that had been burned to 33 GWD/MTIHM. As per Table C.1 of Ref. 4, the total neutron source for that specific fuel was 7.374×10^7 n/s/assembly, while the total gamma source was 4.900×10^{15} p/s/assembly. Recognizing that the active fuel region is 12 ft (365.76 cm) long, that the cask holds 21 assemblies, and that the radius of the homogenized fuel zone in this particular cask is 71.8 cm, the neutron spectrum in Table A.5 may be multiplied by

$$(21)(7.374\times10^7)/[\pi(71.8)^2(365.76)]$$

to get the actual neutron source density (n/s/cm^3) in the homogenized fuel zone, while the gamma spectrum shown in Table A.5 may be multiplied by

$$(21)(4.900\times10^{15})/[\pi(71.8)^2(365.76)]$$

to get the actual gamma source density (p/s/cm^3) in the homogenized fuel zone. These source densities are shown in Table A.6. Applying Eq. A.1 to the volume-integrated adjoint fluxes in Table A.4 and the properly normalized source terms in Table A.6 yields a dose rate of 10.9 mrem/hr for 10-year-cooled fuel burned to 33 GWD/MTIHM. Indeed, all of the XSDRNPM results shown in Tables A.2a and A.2b were obtained in this fashion--i.e., by folding the adjoint results in Table A.4 with the appropriate multigroup source terms.

Table A.3. Composition of materials used in the shipping cask

Mixture	Mixture number	Nuclide	Nuclide ID no.	No. density atoms/(barn-cm)
Fuel/Basket	1	U-235	92235	3.33E-5
Fuel/Basket	1	U-238	92238	3.91E-3
Fuel/Basket	1	Pu-239	94239	2.08E-5
Fuel/Basket	1	Pu-240	94240	9.46E-6
Fuel/Basket	1	O	8016	7.93E-3
Fuel/Basket	1	Zr	40000	2.27E-3
Fuel/Basket	1	Al	13027	2.21E-2
SS-304	2	Fe	26000	5.94E-2
SS-304	2	Cr	24000	1.74E-2
SS-304	2	Ni	28000	7.72E-3
SS-304	2	Mn	25055	1.74E-3
Gamma Shield	3	Pb	82000	3.30E-2
Neutron Shield	4	H	1001	6.41E-2
Neutron Shield	4	C	6012	1.42E-2
Neutron Shield	4	O	8016	2.60E-2
Neutron Shield	4	B-10	5010	1.15E-4
Neutron Shield	4	B-11	5011	4.96E-4
Neutron Shield	4	K	19039	3.05E-4
External Void	5	O	8016	1.00E-20

Table A.4. Volume-integrated adjoint flux in the homogenized fuel zone (listed as the "Total Flux" in the XSDRNPM fine-group summary table for zone 1)

Energy group	Adjoint flux[a]	Energy group	Adjoint flux[b]
1 (1)	1.8094E-02	23 (1)	1.2957E-06
2 (2)	1.5357E-02	24 (2)	1.5734E-06
3 (3)	1.3890E-02	25 (3)	1.6038E-06
4 (4)	1.1960E-02	26 (4)	1.4223E-06
5 (5)	1.0200E-02	27 (5)	1.0820E-06
6 (6)	8.5607E-03	28 (6)	6.6380E-07
7 (7)	8.0982E-03	29 (7)	3.5655E-07
8 (8)	8.2416E-03	30 (8)	1.4237E-07
9 (9)	7.2192E-03	31 (9)	4.5258E-08
10 (10)	6.3043E-03	32 (10)	6.7611E-09
11 (11)	4.3719E-03	33 (11)	3.6205E-10
12 (12)	2.9978E-03	34 (12)	9.8422E-12
13 (13)	1.4830E-03	35 (13)	2.9939E-15
14 (14)	8.6932E-04	36 (14)	1.2891E-23
15 (15)	5.6675E-04	37 (15)	2.1791E-51
16 (16)	3.5855E-04	38 (16)	0.0000
17 (17)	2.8325E-04	39 (17)	0.0000
18 (18)	4.2493E-04	40 (18)	0.0000
19 (19)	2.6482E-04		
20 (20)	1.4597E-03		
21 (21)	1.2448E-03		
22 (22)	7.0138E-03		

[a]Given as (mrem/hr)/(neutron/sec/cm^3)

[b]Given as (mrem/hr)/(photon/sec/cm^3)

Table A.5. Neutron and gamma spectra for standard PWR spent fuel burned to 33 GWD/MTIHM and cooled 10 years

Energy group	Neutron spectrum	Energy group	Gamma spectrum
1 (1)	1.253E-04	23 (1)	8.102E-12
2 (2)	1.066E-03	24 (2)	5.291E-11
3 (3)	2.930E-03	25 (3)	3.237E-10
4 (4)	1.461E-02	26 (4)	3.060E-10
5 (5)	3.699E-02	27 (5)	6.021E-08
6 (6)	4.890E-02	28 (6)	4.624E-07
7 (7)	1.229E-01	29 (7)	7.689E-06
8 (8)	1.009E-01	30 (8)	2.189E-04
9 (9)	2.466E-02	31 (9)	8.650E-03
10 (10)	1.274E-01	32 (10)	1.659E-02
11 (11)	2.266E-01	33 (11)	1.771E-02
12 (12)	2.006E-01	34 (12)	1.354E-01
13 (13)	9.239E-02	35 (13)	1.949E-01
14 (14)	4.133E-06	36 (14)	1.012E-02
15 (15)	0.000	37 (15)	2.151E-02
16 (16)	0.000	38 (16)	4.160E-02
17 (17)	0.000	39 (17)	8.853E-02
18 (18)	0.000	40 (18)	4.648E-01
19 (19)	0.000		
20 (20)	0.000		1.000
21 (21)	0.000		
22 (22)	0.000		
	1.000		

Table A.6. Average neutron and gamma source densities inside the homogenized fuel zone (R=71.8 cm) of a cask containing 21 10-year-cooled PWR assemblies previously burned to 33 GWD/MTIHM

Energy group	Neutron source[a]	Energy group	Gamma source[b]
1 (1)	3.2752E-02	23 (1)	1.4073E-01
2 (2)	2.7865E-01	24 (2)	9.1907E-01
3 (3)	7.6588E-01	25 (3)	5.6227E+00
4 (4)	3.8189E+00	26 (4)	5.3153E+00
5 (5)	9.6689E+00	27 (5)	1.0459E+03
6 (6)	1.2782E+01	28 (6)	8.0320E+03
7 (7)	3.2126E+01	29 (7)	1.3356E+05
8 (8)	2.6375E+01	30 (8)	3.8024E+06
9 (9)	6.4459E+00	31 (9)	1.5025E+08
10 (10)	3.3302E+01	32 (10)	2.8817E+08
11 (11)	5.9232E+01	33 (11)	3.0763E+08
12 (12)	5.2434E+01	34 (12)	2.3519E+09
13 (13)	2.4151E+01	35 (13)	3.3854E+09
14 (14)	1.0803E-03	36 (14)	1.7578E+08
15 (15)	0.0000	37 (15)	3.7363E+08
16 (16)	0.0000	38 (16)	7.2261E+08
17 (17)	0.0000	39 (17)	1.5378E+09
18 (18)	0.0000	40 (18)	8.0737E+09
19 (19)	0.0000		
20 (20)	0.0000		
21 (21)	0.0000		
22 (22)	0.0000		

[a]neutrons/sec/cm^3

[b]photons/sec/cm^3

Appendix B

AN ENHANCED SET OF OPTIMAL PACKING ARRANGEMENTS FOR SQUARE FUEL ASSEMBLIES INSIDE A CYLINDRICAL SHIPPING CASK

The list of minimum diameter-to-pitch ratios for dense arrays of square assemblies in a cylindrical container given in Table 5.1, and the optimal packing arrangements illustrated in Figs. 2.4a-e, were taken from Appendix B of Ref. 4 and implemented in the original version of the CAPSIZE program dated June 1985. Thanks to R. I. Smith of Battelle Pacific Northwest Laboratories, nine more compact configurations have recently been added to the list. These include eight new configurations tailored for casks with 6, 14, 16, 17, 25, 27, 28, or 29 assemblies, and an improved configuration for casks with 47-48 assemblies. The enhanced list of minimum diameter-to-pitch ratios for dense arrays of square assemblies in a cylindrical container is given in Table B.1, and the complete (enhanced) set of optimal packing arrangements is illustrated in Figs. B.1a-f. The enhanced data in Table B.1 has been used in all versions of the CAPSIZE program released after March 1987.

Table B.1. Minimum diameter-to-pitch ratios for dense arrays of square assemblies in a cylindrical container (see Figs. B.1a-f)

No. of assemblies	$(D/P)^2$	No. of assemblies	$(D/P)^2$
1	2	23-24	40
2	5	25	40.61
3	6.65	26	41
4	8	27	42.64
5	10	28	45
6	11.57	29	48.59
7	13	30-31	50
8	15.68	32	52
9-10	18	33-34	53
11-12	20	35-37	58
13	22.60	38-39	61
14	25	40-42	65
15	26	43-44	68
16	27.53	45-46	72
17	29	47-48	73
18	32	49-52	80
19-21	34	53-56	85
22	37	57-58	89
		59-61	90

ORNL-DWG 87M-9151

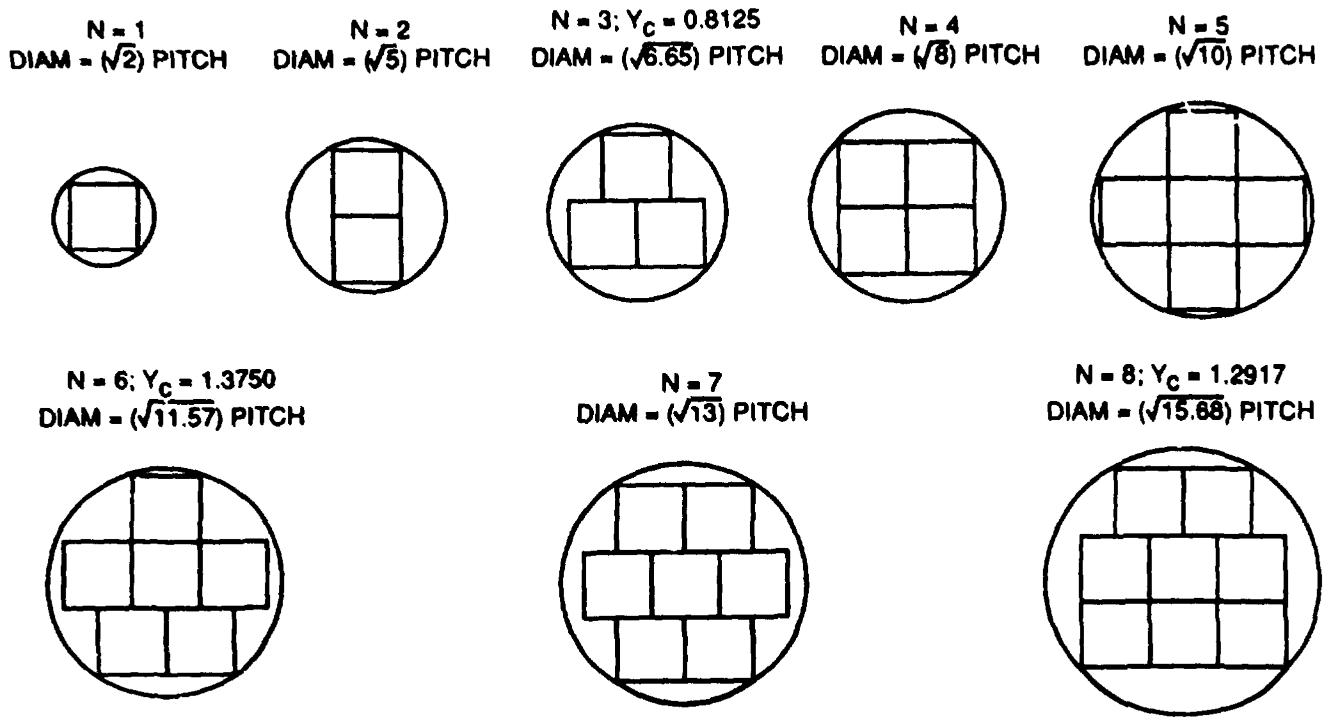

Fig. B.1a. Optimal packing arrangements for 1-8 square assemblies in a cylindrical cask

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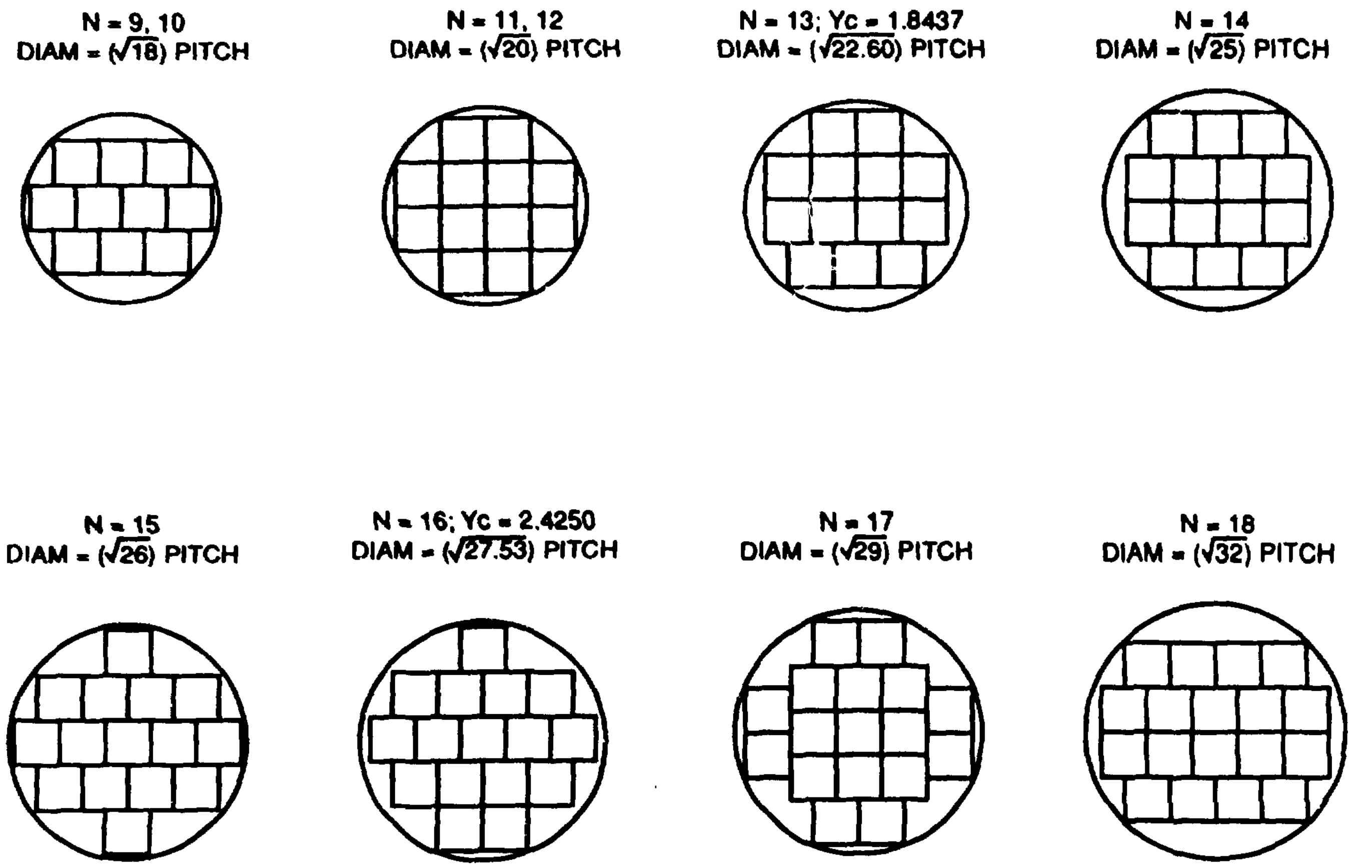

Fig. B.1b. Otpimal packing arrangements for 9-18 square assemblies in a cylindrical cask

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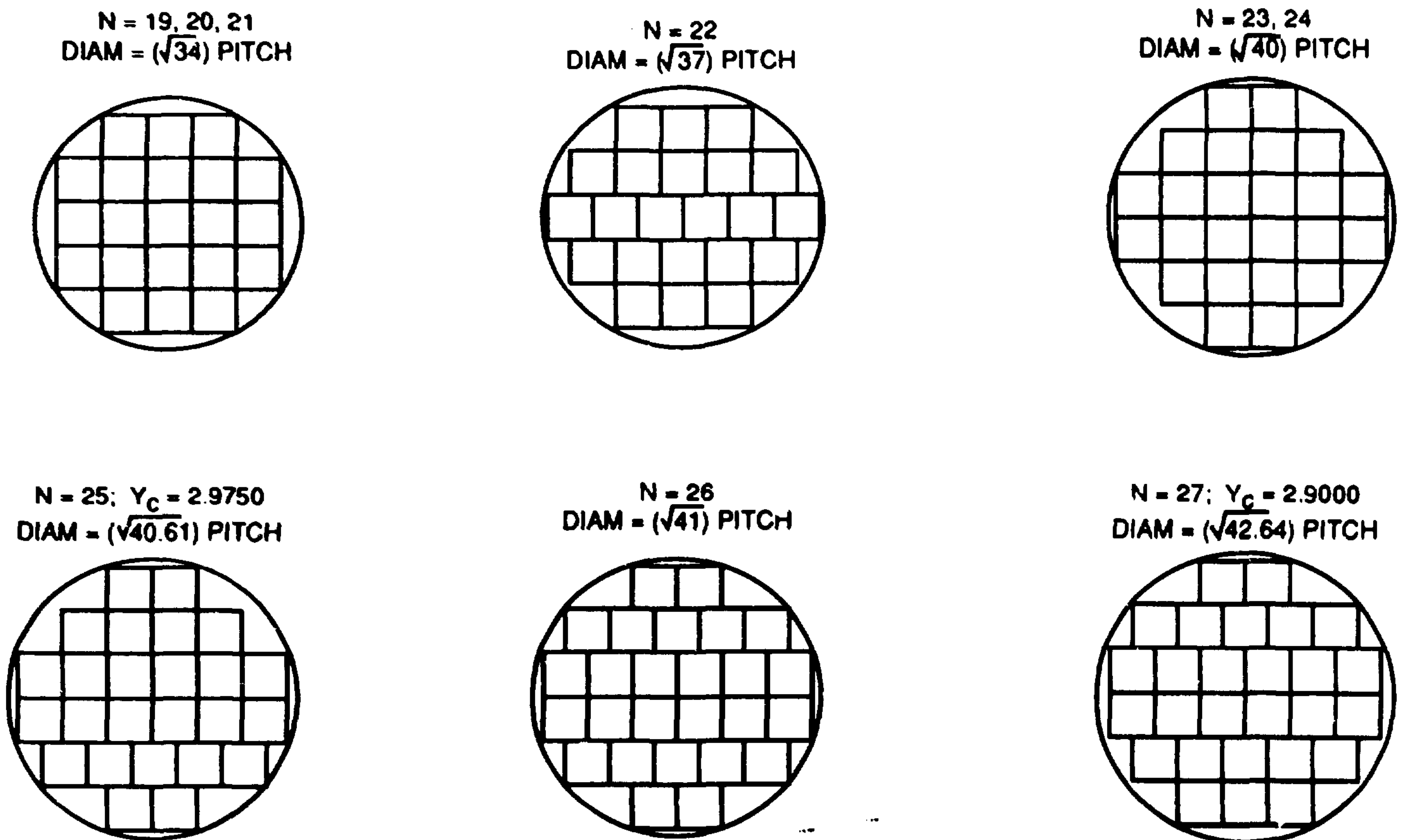

Fig. B.1c. Optimal packing arrangements for 19-27 square assemblies in a cylindrical cask

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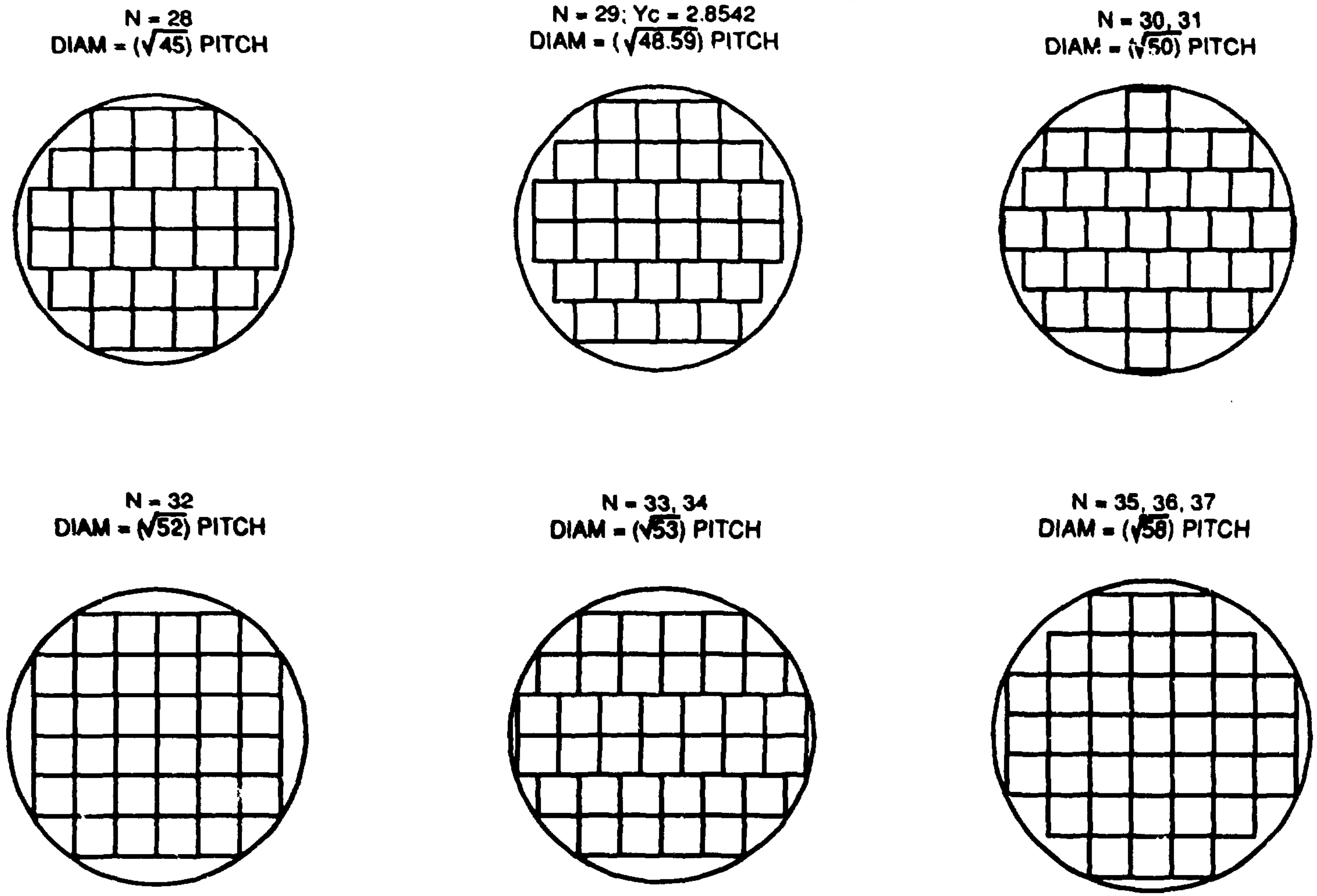

Fig. B.1d. Optimal packing arrangements for 28-37 square assemblies in a cylindrical cask

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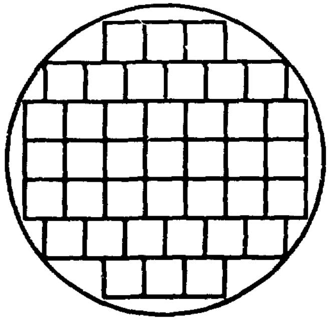
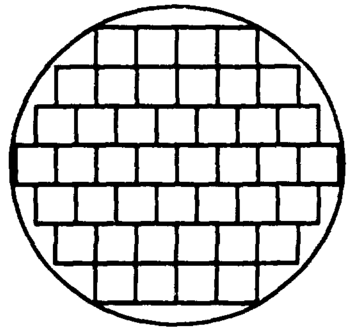
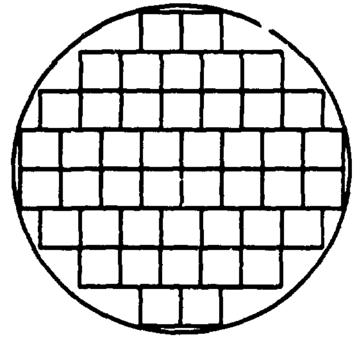
N = 38, 39
DIAM = ($\sqrt{61}$) PITCH



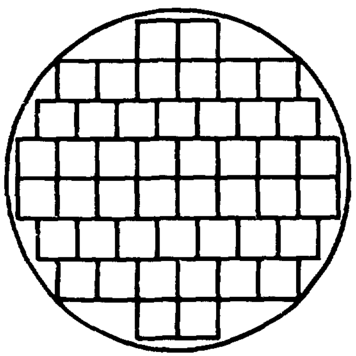
N = 40, 41, 42
DIAM = ($\sqrt{65}$) PITCH



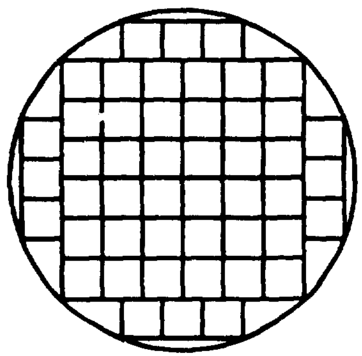
N = 43, 44
DIAM = ($\sqrt{68}$) PITCH



N = 45, 46
DIAM = ($\sqrt{72}$) PITCH



N = 47, 48
DIAM = ($\sqrt{73}$) PITCH



N = 49, 50, 51, 52
DIAM = ($\sqrt{80}$) PITCH

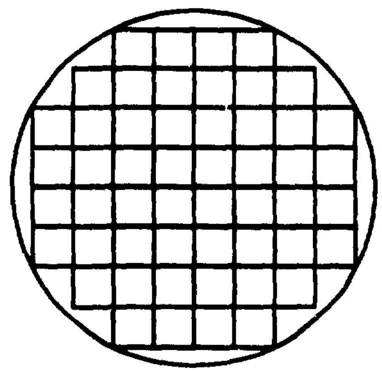


Fig. B.1e. Optimal packing arrangements for 38-52 square assemblies in a cylindrical cask

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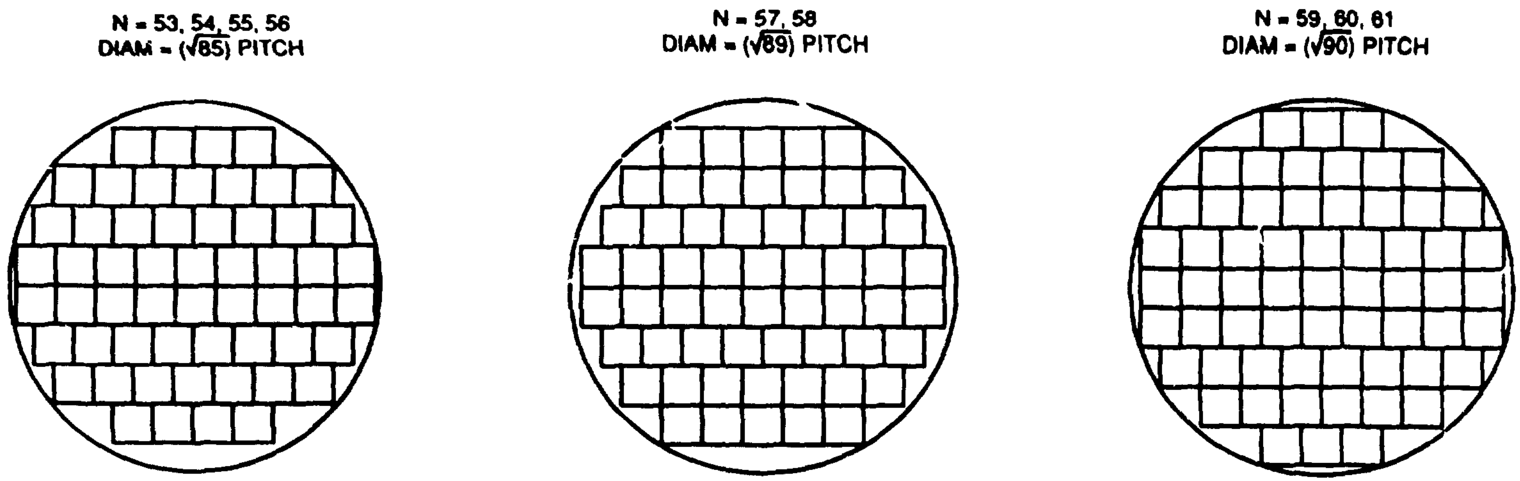

Fig. B.1f. Optimal packing arrangements for 53-61 square assemblies in a cylindrical cask

Appendix C

SAMPLE OUTPUT PRODUCED BY A NEW VERSION OF THE CAPSIZE PROGRAM THAT USES THE ENHANCED SET OF OPTIMAL PACKING CONFIGURATIONS DESCRIBED IN APPENDIX B

Section 6 describes a typical interactive session using the original version of the CAPSIZE program developed in June 1985. The results shown there are based on the original set of optimal packing configurations illustrated in Figs. 2.4a-e. In March 1987, nine more compact configurations were added to the list. These included eight new configurations tailored for casks with 6, 14, 16, 17, 25, 27, 28, or 29 assemblies, and an improved configuration for casks with 47-48 assemblies. The complete enhanced set of optimal packing configurations shown in Appendix B has since incorporated a new version of the CAPSIZE program, first released in March 1987. This appendix shows the revised output that would now be generated, given the three sets of input data described in Sect. 6. Because of the greater continuity afforded by these additional configurations, the number of assemblies in casks just under the specified weight limit may be increased (sometimes by as many as 2 or 3 assemblies) and/or the number of assemblies in casks just over the specified weight limit may be reduced (sometimes by as many as 2 or 3 assemblies). In either case, the range of possible cask capacities is frequently much better defined. For a given type of cask and a fixed number of assemblies, the calculated neutron and gamma shield thicknesses will be the same as in the earlier version of the program.

```
************************************************************************
************************************************************************
************************************************************************
*                                                                   ****
* CAPSIZE -- A PROGRAM WRITTEN BY J. A. BUCHOLZ OF ORNL TO CALCULATE THE  ****
*            CAPACITY & SIZE OF (PB, FE, AND U-METAL) SHIPPING CASKS AS A ****
*            FUNCTION OF THE BURNUP & COOLING TIME OF THE SPENT FUEL, THE ****
*            DESIRED DOSE 10-FT FROM THE CENTERLINE, AND THE SPECIFIED    ****
*            WEIGHT LIMIT FOR THE LOADED CASK.                           ***
*                                                                   **
************************************************************************

 1) The source terms, cross section data, and methods used in the CAPSIZE
    program are documented in ORNL/CSD/TM-248.  Section 6 also serves as a
    users guide.  This version of the program utilizes the enhanced set of
    optimal packing configurations given in Appendix B.  While the program
    may be copied, users are asked not to distribute modified versions of it.

 2) This program will run on PC's with or without an 8087 math coprocessor.
    With an 8087 chip installed, a typical inquiry will take 2-10 seconds.
    Without an 8087, a typical inquiry will take 0.5-2.0 minutes.

                                          (press the RETURN key to continue)
--------------------------------------------------------------------------
BU=33000 MWD/MT     Time=10 yrs     Dose=10.0 mrem/hr     Weight Limit=190000 lbs

Separator Thickness= 2.2500 in
==========================================================================
```

```
To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb    CT=yy    DD=dd.d    WL=wwwwww    ST=s.ssss        [enter BU=0 to stop]
```

a simple "return" then yields:

```
Capacities are as follows:

  Fe cask = 14 assys,  Wt=185289 lbs,  G-shld= 9.18"  N-shld= 3.72"  OD= 83.5"
  Fe cask = 15 assys,  Wt=190355 lbs,  G-shld= 9.21"  N-shld= 3.73"  OD= 84.7"

  Pb cask = 17 assys,  Wt=184955 lbs,  G-shld= 3.89"  N-shld= 4.14"  OD= 83.5"
  Pb cask = 18 assys,  Wt=197171 lbs,  G-shld= 3.91"  N-shld= 4.16"  OD= 86.6"

  U  cask = 21 assys,  Wt=188821 lbs,  G-shld= 2.44"  N-shld= 3.46"  OD= 82.6"
  U  cask = 22 assys,  Wt=200059 lbs,  G-shld= 2.45"  N-shld= 3.48"  OD= 85.4"

BU=33000 MWD/MT    Time=10 yrs    Dose=10.0 mrem/hr    Weight Limit=190000 lbs

Separator Thickness= 2.2500 in

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb    CT=yy    DD=dd.d    WL=wwwwww    ST=s.ssss        [enter BU=0 to stop]
```

st=1.5 ct=5 dd=2.0 followed by a "return" then yields:

```
Capacities are as follows:

  Fe cask = 10 assys,  Wt=179615 lbs,  G-shld=11.58"  N-shld= 4.07"  OD= 78.3"
  Fe cask = 12 assys,  Wt=191330 lbs,  G-shld=11.68"  N-shld= 4.11"  OD= 80.9"

  Pb cask = 13 assys,  Wt=180957 lbs,  G-shld= 5.48"  N-shld= 5.09"  OD= 78.8"
  Pb cask = 14 assys,  Wt=192058 lbs,  G-shld= 5.50"  N-shld= 5.12"  OD= 81.5"

  U  cask = 17 assys,  Wt=187634 lbs,  G-shld= 3.33"  N-shld= 4.61"  OD= 78.5"
  U  cask = 18 assys,  Wt=199658 lbs,  G-shld= 3.34"  N-shld= 4.64"  OD= 81.4"

BU=33000 MWD/MT    Time= 5 yrs    Dose= 2.0 mrem/hr    Weight Limit=190000 lbs

Separator Thickness= 1.5000 in
```

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb   CT=yy   DD=dd.d   WL=wwwww   ST=s.ssss   [enter BU=0 to stop]

bu=45000 wl=58000 st=0.5 followed by a "return" then yields:

Capacities are as follows:

```
  Fe cask =  1 assys,  Wt= 67950 lbs,  G-shld=10.88"  N-shld= 3.97"  OD= 47.3"

  Pb cask =  1 assys,  Wt= 55011 lbs,  G-shld= 4.94"  N-shld= 4.69"  OD= 42.3"
  Pb cask =  2 assys,  Wt= 79150 lbs,  G-shld= 5.14"  N-shld= 5.03"  OD= 51.1"

  U  cask =  1 assys,  Wt= 43577 lbs,  G-shld= 2.98"  N-shld= 4.04"  OD= 35.6"
  U  cask =  2 assys,  Wt= 65428 lbs,  G-shld= 3.09"  N-shld= 4.41"  OD= 44.3"
```

BU=45000 MWD/MT   Time= 5 yrs   Dose= 2.0 mrem/hr   Weight Limit= 58000 lbs

Separator Thickness= .5000 in

===

To change, enter one or more of the following items (in any order) on one line:

BU=bbbbb   CT=yy   DD=dd.d   WL=wwwww   ST=s.ssss   [enter BU=0 to stop]

bu=0 followed by a "return" then yields:

Stop - Program terminated.

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