A major purpose of the Technical Information Center is to provide the broadest dissemination possible of information contained in DOE's Research and Development Reports to business, industry, the academic community, and federal, state and local governments.

Although a small portion of this report is not reproducible, it is being made available to expedite the availability of information on the research discussed herein.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

ORNL/CSD/TM-227

PARAMETRIC STUDY OF RADIATION DOSE RATES FROM RAIL AND TRUCK

SPENT FUEL TRANSPORT CASKS

C. V. Parks
O. W. Hermann
J. R. Knight

ORNL/CSD/TM--227

DE85 018146

Date Published - August 1985

COMPUTING AND TELECOMMUNICATIONS DIVISION

at

Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, TN 37830

NOTICE This document contains information of a preliminary nature
It is subject to revision or correction and therefore does not represent a
final report

Martin Marietta Energy Systems, Inc.
operating the

Oak Ridge National Laboratory
Oak Ridge Y-12 Plant

Oak Ridge Gaseous Diffusion Plant
Paducah Gaseous Diffusion Plant

under Contract No. AC05-84OR21400

for the

DEPARTMENT OF ENERGY

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

EP

TABLE OF CONTENTS

	<u>Page</u>
LIST OF TABLES	v
LIST OF FIGURES	vii
ACKNOWLEDGMENTS	ix
ABSTRACT.	xi
1. INTRODUCTION	1
2. DESCRIPTION OF ANALYSES	3
2.1. Fuel Depletion and Cask Shielding Analysis by SAS2	3
2.2. Consolidated Fuel Shipping Cask Analyses by SAS1	5
3. DESCRIPTION OF THE TRANSPORT CASKS AND SPENT FUEL	6
3.1. The IF 300 Transport Cask and Model	6
3.2. The NLI 1/2 Transport Cask and Model	6
3.3. The PWR Spent Fuel Assembly and Reactor Operating History	6
4. DATA INPUT TO CASES.	11
5. RESULTS AND DISCUSSION	15
6. SUMMARY	26
REFERENCES	27

LIST OF TABLES

	<u>Page</u>
Table 3.1. Data for PWR Fuel Inventory Calculation	10
Table 5.1. SAS2 Dose Rates for the IF 300 Rail Cask	16
Table 5.2. SAS2 Dose Rates for the NLI 1/2 Truck Cask.	17
Table 5.3. SAS2 Dose Rates for the IF 300 Rail Cask with Variation in Fuel Cooling Time	18
Table 5.4. SAS2 Dose Rates for the NLI 1/2 Truck Cask with Variation in Fuel Cooling Time	19
Table 5.5. SAS2 Dose Rates for the IF 300 Cask with Maximum ⁵⁹ Co	20
Table 5.6. SAS1 Dose Rates for the IF 300 Rail Cask with Consolidated Fuel	21
Table 5.7. SAS1 Dose Rates for the NLI 1/2 Truck Cask with Consolidated Fuel	22
Table 5.8. Comparison of IF 300 Rail Cask 5-Year-Decay Case with Different Quadrature Sets	23

LIST OF FIGURES

	<u>Page</u>
Figure 2.1. Basic computational flow path invoked by SAS2	4
Figure 3.1. Radial model for the IF 300 rail cask	7
Figure 3.2. Radial model for the NLI 1/2 truck cask	8
Figure 4.1. Sample input data to SAS2 for IF 300 cask under normal dry fuel transport conditions	12
Figure 4.2. Sample input data to SAS1 for IF 300 cask case with consolidated fuel pins (consolidation factor = 1) and normal dry fuel transport conditions.	13
Figure 4.3. Sample ORIGEN-S/SAS2 restart for cases with fuel cooled beyond 5 years	14

ACKNOWLEDGMENTS

The authors would like to acknowledge the Transportation Technology Center at Sandia National Laboratories who requested and financed this work. Appreciation is also expressed to L. B. Shappert who monitored the work and to Jeanette Hamby who prepared the manuscript.

ABSTRACT

Neutron and gamma dose rates from typical rail and truck spent fuel transport casks are reported for a variety of spent PWR fuel sources and cask conditions. The IF 300 rail cask and NLI 1/2 truck cask were selected for use as appropriate cask models. All calculations (cross section preparation, generation of spent fuel source terms, radiation transport calculations, and dose evaluation) were performed using various modules of the SCALE computational system. Conditions or parameters for which there were variations between cases include: detector distance from cask, spent fuel cooling time, the setting of fuel or neutron shielding cavities to either wet or dry, the cobalt content of assembly materials, normal fuel assemblies and consolidated canisters, the geometry mesh interval size, and the order of the angular quadrature set.

**PARAMETRIC STUDY OF RADIATION DOSE RATES FROM RAIL AND TRUCK SPENT FUEL
TRANSPORT CASKS**

C. V. Parks
O. W. Hermann
J. R. Knight

1. INTRODUCTION

At the request of the Transportation Technology Center (TTC) at Sandia National Laboratories, neutron and gamma dose rates were computed for typical rail and truck spent fuel transport casks as a function of various parameters and conditions. The work resulted from a series of requests by TTC over a six-month period from July 1984 to January 1985. The IF 300 and NLI 1/2 casks as described in Refs. 1-2 were selected for use as appropriate cask models. The spent fuel loaded into the casks was characterized by calculated isotopic densities of PWR fuel discharge cooled for several different decay times. The fuel burnup analysis was performed by the SAS2 control module³ of the SCALE computational system.⁴ Spent fuel decay to the various cooling times was simulated with ORIGEN-S.⁵ The dose rates were calculated using the SAS1⁶ and SAS2 control modules in SCALE. For the radiation transport, a one-dimensional radial analysis was performed. The resulting angular fluxes at the cask surface were then integrated over the finite cask surface to obtain the scalar fluxes and dose rates at specified locations.

The results described in this report show comparisons of dose rates for different conditions of the transport cask and analytic model. The parametric changes included:

- . variations in spent fuel cooling times;
- . variations in detector distances from the cask;
- . the setting of fuel cavities to either wet or dry;
- . the setting of neutron shielding cavities to either wet or dry;
- . variations in the Co-59 content of the clad and structural materials;
- . the fuel pins either spaced as in the design of the assembly, or, consolidated, as in a packed form;

- . variations in spatial interval sizes of the mesh representing the cask geometry;
- . and, variations in the order of angular quadrature used in the shielding analyses.

2. DESCRIPTION OF ANALYSES

Computation of the dose rates involved a fuel depletion and decay calculation, development of the radiation sources, and shielding analyses of the transport casks. Isotopic densities of the spent fuel were computed by a point depletion model, using cross sections dependent on burnup and fuel assembly design. One-dimensional radiation transport models which account for resonance self-shielding were used in both the burnup dependent cross section preparation and the shielding analysis of the cask. Scalar fluxes and subsequent dose rates at external detector locations are obtained by integrating angular fluxes over the finite cask surface. The time dependent isotopic densities computed during fuel irradiation and decay periods used a matrix exponential expansion model, supplemented by Bateman equations in the treatment of short-lived isotopes. The SCALE 27-neutron-group ENDF/B-IV shielding library was used in the depletion analysis and the SCALE 27-neutron/18-gamma-group coupled ENDF/B-IV library was used in the shielding analysis.

2.1. Fuel Depletion and Cask Shielding Analysis by SAS2

The Shielding Analytical Sequence No. 2 (SAS2) control module in SCALE was used to compute the fuel depletion, the spent fuel decay, and the shielding analyses for all cases except those using consolidated fuel. Resonance self-shielding of cross sections is achieved in SAS2 using the BONAMI-S⁷ and NITAWL-S⁸ codes while XSDRNP-S⁹ is used for cell weighting of cross sections and the shielding analysis. These codes along with another data processing code called COUPLE¹⁰ are used to produce the burnup-dependent cross sections used in conjunction with the ORIGEN-S depletion analysis. The XSDOSE code¹¹ is used to integrate the angular fluxes over the finite cask surface and evaluate the scalar fluxes and dose rates at the detector locations. For this work the detector locations were along the extended axial midplane of the cask, although XSDOSE can consider detector locations at any radial and axial position. Figure 2.1 shows the basic computational flow path of the SAS2 control module.

A SAS2 case may be halted at several points during the execution in order that intermediate data interfaces may be saved and altered before the case is restarted and completed. The first SAS2 case was halted at the end of the fuel depletion and radiation source computation in order to save the significant cost of repeating the same fuel depletion case many times. The ORIGEN-S module was used to decay the discharged fuel (from data saved when the SAS2 case was halted) to produce the neutron and gamma sources at a specified set of cooling times. Then, the radiation sources were input to either SAS1 or a restart of SAS2. The methods and models applied by SAS2 are documented in greater detail elsewhere.^{3,12}

Figure 2.1. Basic computational flow path invoked by SAS2

2.2. Consolidated Fuel Shipping Cask Analyses by SAS1

The Shielding Analytical Sequence No. 1 (SAS1) control module⁶ in SCALE was used to compute dose rates for casks where fuel pins are assumed to be removed from the assembly and packed into the fuel cavities of the cask as consolidated fuel. SAS1 permits the consolidated fuel to be represented by appropriate densities and volume fractions, while SAS2 requires the pins to be spaced in accordance with the assembly specified for the depletion calculation. Like SAS2, the SAS1 control module also uses NITAWL-S and BONAMI-S for resonance self-shielding, XSDRNPM-S for the shielding analysis and XSDOSE for the dose calculations.

3. DESCRIPTION OF THE TRANSPORT CASKS AND SPENT FUEL

3.1. The IF 300 Transport Cask and Model¹

The IF 300 is a cask designed primarily for transport by railroad. Up to seven PWR fuel assemblies or eighteen BWR fuel assemblies can be shipped in the IF 300. The IF 300 cask body is a depleted uranium-shielded, stainless-steel-clad annular cylinder, closed at one end. After fuel loading the open end is closed with a bolted and sealed head. Fuel is held in the cavity by a stainless steel basket which is varied depending on the specific type fuel being shipped. Although designed originally for shipping fuel with a water-filled cavity, the IF 300 is currently restricted by NRC to shipments with dry cask cavities. Masking the active fuel zone on the outer cask surface is a water-filled corrugated containment which acts as a neutron shield. However, during accident conditions the containment is assumed to rupture and the neutron shielding water lost to the environment. For purposes of this study, a one-dimensional (radial) shielding model of the IF 300 was used. The model is schematically shown in Figure 3.1 and is nearly identical to that used in Ref. 1.

3.2. The NLI 1/2 Transport Cask and Model²

The NLI 1/2 is a cask designed primarily for transport by truck. The cask can carry one PWR spent fuel assembly or two BWR spent fuel assemblies. The fuel is shipped dry in an interchangeable aluminum basket which fills the fuel cavity. Depleted uranium and lead is used as the gamma shielding material while a neutron shield of water in a steel jacket is provided on the outside of the cask body. As noted above, during accident conditions the shielding water is assumed to be lost to the environment. The one-dimensional radial shielding model used for the analysis is shown in Figure 3.2 and was taken from Ref. 2. As noted in Ref. 2 the design of the NLI 1/2 indicates that at least a two-dimensional shielding analysis needs to be performed for the most accurate results. However, due to time and funding constraints the model of Figure 3.2 was employed. The reported results are expected to be as accurate as needed for the intended use.

3.3. The PWR Spent Fuel Assembly and Reactor Operating History

Spent fuel sources vary as a function of reactor design, assembly type within a reactor, operating power history of the assembly, and the cooling time after discharge. A specific assembly design and power history was selected for use in this study. A "typical" power history,

ORNL-DWG 85-8326

Figure 3.1. Radial model for the IF 300 rail cask

ORNL-DWG 85-8325

Figure 3.2. Radial model for the NLI 1/2 truck cask

using 20% downtime, was employed. Had the most conservative history (no downtime) been used, there would have been only a few percent increase in the dose rates at the shorter cooling times and an even smaller increase at longer times. A completely filled lattice is assumed by SAS2 for the fuel assemblies, i.e., no water holes are present in the assembly. This model has been shown¹² to be significantly conservative for the assembly type (17x17) used in this study when water holes are present. Use of the 17x17 type assembly is also conservative in comparison with other types of PWR assemblies, e.g., 15x15, 14x14. However, many assemblies currently used in reactors often do not have many, if any, water holes. Thus, the typical assembly history and model used probably combine to provide a case study which is at some point between typical and highly conservative depending on the actual fuel assembly to which the calculations are compared. The Westinghouse 17x17 fuel assembly loaded with a 3.3 wt. % U-235 enrichment, operated at a specific power of 37.5 MW/MTU for 880 full power days, and discharged at a burnup of 33 GWD/MTU was specified for the cases reported here. The detailed data for the fuel assembly and power history can be seen in Table 3.1.

Table 3.1. Data for PWR Fuel Inventory Calculation

Percent theoretical density^a for UO₂: 90.172
 Initial uranium composition (wt. %):

²³⁴ U	= 0.029
²³⁵ U	= 3.200
²³⁶ U	= 0.016
²³⁸ U	= 96.755

Initial uranium loading (kg/assembly): 461.4

Fuel rods per assembly: 264 with Zircaloy-4 clad
 Rod pitch (cm): 1.25984
 Pellet O.D. (cm): 0.81915
 Gap O.D. (cm): 0.83566
 Clad O.D. (cm): 0.94996
 Active Fuel Length (cm): 365.76

Fuel temperature (K): 1000
 Clad temperature (K): 605
 Moderator temperature (K): 583
 Moderator density (g cm⁻³): 0.706575
 VF^b for moderator: 0.707850
 Average boron concentration (wt. ppm): 550
 VF for ¹⁰B: 4.04045×10^{-6}
 VF for ¹¹B: 1.75841×10^{-5}

---

^aTheoretical UO₂ density in SAS2 is 10.96 g cm⁻³. This percent theoretical density value is based on smearing the fuel mass in the pellet and gap region.

^bVF is a SAS2 parameter related to volume fraction and/or percent theoretical density (see Ref. 3).

4. DATA INPUT TO CASES

Examples of input data to the SAS2 cases are shown in Figures 4.1 and 4.2. Data used in Figure 4.1 include: the description of the IF 300 shipping cask, the fuel assembly and operating history described in Section 3.3, a 5-year cooling time, the mesh interval size set to one-half the default size (SZFCASK = 0.5), an angular quadrature order of 16 (ISNS = 16) for the final shielding case, and a set of four dose detector distances of 0, 100, 200, and 300 cm from the cask radial surface. Input for the NLI 1/2 differs only in the cask zone descriptions (see Figure 3.2). The SAS1 input for the IF 300 loaded with fuel pins from seven assemblies is shown in Figure 4.2. The fuel source is read from an output file generated by SAS2 or ORIGEN-S.

As noted above, a SAS2 case was halted after computing the discharged densities and the spent fuel radiation source at a 5-year cooling time. Then, either the 5-year source case was continued in a SAS2 restart case, or ORIGEN-S was run first to calculate sources at other cooling times before restarting SAS2. An example of the ORIGEN-S/SAS2 run is shown in Figure 4.3.

```

= SAS2
PVR TYPICAL IRRADIATION HISTORY - 3.2% ENRICH - 33GWD/MTU
27GROUPSHLD LATTICECELL
UO2 1 0.9017246 1000 92234 0.029 92235 3.2 92236 0.016 92238 96.755 END
SR-90 1 0 1-20 1000 END
Y-89 1 0 1-20 1000 END
ZR-94 1 0 1-20 1000 END
ZR-95 1 0 1-20 1000 END
NB-94 1 0 1-20 1000 END
TC-99 1 0 1-20 1000 END
RH-103 1 0 1-20 1000 END
RH-105 1 0 1-20 1000 END
XE-131 1 0 1-20 1000 END
XE-135 1 0 8.72-9 1000 END
CS-134 1 0 1-20 1000 END
CS-137 1 0 1-20 1000 END
BA-136 1 0 1-20 1000 END
LA-139 1 0 1-20 1000 END
PR-143 1 0 1-20 1000 END
ND-143 1 0 1-20 1000 END
ND-145 1 0 1-20 1000 END
PM-147 1 0 1-20 1000 END
SM-149 1 0 1-20 1000 END
SM-151 1 0 1-20 1000 END
SM-152 1 0 1-20 1000 END
EU-153 1 0 1-20 1000 END
EU-154 1 0 1-20 1000 END
ZIRCALLOY 2 1 605 END
B-10 3 0 4.04045-6 583 END
B-11 3 0 1.75841-5 583 END
H2O 3 0.7078495 583 END
CO-59 3 0 1-20 583 END
SS304 4 2.1375-2 END
SS304 5 2.1375-2 END
SS304 6 2.17124-2 END
B4C 6 7.7066-2 END
H2O 7 1D-12 END
SS304 8 1.0 END
U(.27)METAL 9 1.0 END
H2O 10 .944 END
END COMP
SQUAREPITCH 1.25984 0.83566 1 3 0.94996 2 END
NPIN/ASSM=264 FUELNGTH=365.76 NCYCLES=3 NLIB/CYC=1
LIGHTEL=16
POWER=17.3025 BURN=293.3333 DOWN=73.33333 END
POWER=17.3025 BURN=293.3333 DOWN=73.33333 END
POWER=17.3025 BURN=293.3333 DOWN=180 END
C 0.059993 N 0.033765 O 62.098 AL 0.045678
SI 0.065863 P 0.14216 TI 0.049832 CR 2.3398
MN 0.10963 FE 4.5991 CO 0.033443 NI 4.4021
ZR 100.83 NB 0.32753 MO 0.18156 SN 1.6518
27N-18COUPLE TEMPCASK(K)=394.27 NUMZONES=9 DRYFUEL=YES DETECTORS=4 END
0 100 200 300
5 12.7532 6 21.723 4 38.0493 7 47.625 8 48.895 9 59.06 8 63.01 10 74.44 8 74.88
ZONE=1 FUELBNL=1 ZONE=3 FUELBNL=6
PRESSURE=50 SZFCASK=.5 ISNS=16
END

```

Figure 4.1. Sample input data to SAS2 for IF 300 cask under normal dry fuel transport conditions

```

=SAS1
RAIL CASK IF-300 SASRAIL
27N-18COUPLE MULTIREGION
UO2      1 .69779 1000 92234 .029 92235 3.2 92236 .016 92238 96.755 END
ZIRCALLOY 1 .22616 605 END
N        3 1.2-3 END
SS304    5 2.1712-2 END
B4C      5 7.7066-2 END
SS304    6 END
U(.27)METAL 7 END
H2O      8 .944 END
SS304    9 3.3725-2 END
SS304   10 4.7427-2 END
SS304   11 7.9875-2 END
END COMP
CYLINDRICAL END
1 12.75 ONEEXTERMOD 5 21.72 NOEXTERMOD 1 38.05 ONEEXTERMOD
3 47.63 NOEXTERMOD 6 48.90 NOEXTERMOD 7 59.06 NOEXTERMOD
6 63.01 NOEXTERMOD 8 74.44 NOEXTERMOD 6 74.88 NOEXTERMOD END ZONE
END

RAIL CASK IF-300 CASE 1
CYLINDRICAL
1 12.75 18 1 9 .63380 60 2 1.8689+5 1
5 21.72 8 0
  38.05 25 1 0 0. 60 2 1.1213+6 6
  47.63 5 0 6 48.90 2 0 7 59.06 45 0
6 63.01 6 0 8 74.44 16 0 6 74.88 1 0 END ZONE
END
365.76
END

```

Figure 4.2. Sample input data to SAS1 for IF 300 cask case with consolidated fuel pins (consolidation factor = 1) and normal dry fuel transport conditions.

=ORIGENS

0\$\$\$ A8 26 A11 -71 E T

COOLING CASE

3\$\$\$ 21 0 1 A33 18 4## A4 1-40 T

35\$\$\$ 0 T

56\$\$\$ A14 5 3 74 E T

PWR 3CYC-3.2%-880D-33GWD/MTU-1BNDL, COOLED 5 YR, IF300(DRYDRY) CASK

PWR ASSEMBLY

60## 0.25 1 2 3 5 7 10 15 20 25

65\$\$\$ 1 8Z 1 2Z 1 8Z 2Q21 61## F1-3

81\$\$\$ 2 0 26 1 E

82\$\$\$ A5 2 E

83## 1+7 8+6 6.5+6 5+6 4+6 3+6 2.5+6 2+6 1.66+6 1.33+6 1+6

8+5 6+5 4+5 3+5 2+5 1+5 5+4 1+4

6T

PWR ASSEMBLY FOR SHIPPING CASK

56\$\$\$ 2Z A10 5 E T

56\$\$\$ F0 T

END

=SAS2 PARM=RESTARTS

PWR 3CYC-3.2%-880D-33GWD/MTU-1BNDL, COOLED 5 YR, IF300-DRYDRY, ISNS=16

END

Figure 4.3. Sample ORIGIN-S/SAS2 restart for cases with fuel cooled beyond 5 years

5.0. RESULTS AND DISCUSSION

Tables 5.1-5.8 provide the shielding analysis results for both casks in the basic chronological order in which the analyses were performed. Tables 5.1 and 5.2 primarily provide SAS2 dose rates from spent PWR fuel assemblies with a 5 year cooling time. Both normal conditions of transport (water-filled neutron shield) and accident conditions (dry neutron shield) are considered for the IF 300 and NLI 1/2 casks. Table 5.1 also provides dose rates from the IF 300 for normal dry transport of fuel cooled three years from discharge (minimum cool time required for the specified fuel to meet heat load requirements). As expected, the results of Tables 5.1-5.2 indicate a large increase in the neutron dose under accident conditions when the neutron shield is lost. All of the results in Tables 5.1-5.2 were obtained using an S_8 angular quadrature set in the calculations. For casks with a dry fuel cavity and water-filled neutron shield, analyses were done both with the SAS2 default mesh size (SZFCASK = 1.0) and with a reduced mesh size (SZFCASK = 0.5). Cases with a reduced mesh size generated a 15-20% increase in the neutron dose rate and consequently a 7-10% increase in the total dose rate. These changes were considered significant and so future SAS2 cases were performed with the reduced mesh size.

Tables 5.3-5.4 provide dose rates from SAS2 for spent PWR fuel assemblies with varying cooling times from 5 years to 25 years. Normal conditions of transport (dry cavity, water-filled neutron shield) and accident conditions (dry cavity, dry neutron shield) were considered. The smaller mesh size generated with SZFCASK = 0.5 was used for all the analyses. Note, however, that surface neutron and total dose rate results from the "benchmark" case with 5 year-old fuel and accident conditions did not significantly change from that obtained in Tables 5.1-5.2 with SZFCASK = 1.0. Thus, it appears from these results that the analyses performed for accident conditions do not require the smaller SAS2 mesh required for analyses with a water-filled neutron shield. The difference is probably caused by the reduced mesh size needed for the water shield where a lot of neutron scattering takes place. Note, also, that the accident analysis cases of Tables 5.3-5.4 were performed with an S_{16} quadrature set. Thus, based on the above comparison with Tables 5.1-5.2, a quadrature set higher than S_8 also appears unnecessary for analyzing accident conditions.

The results of Table 5.5 are presented for the IF 300 cask to show the effect of the Co-59 content. The fuel pin cladding and assembly structural material contain a small content of Co-59. However, even small quantities of Co-59 can produce a significant Co-60 gamma source in spent fuel. All the tabular results except those of Table 5.5 were obtained using the Co-59 content of the clad and structural material equivalent to 1/2 the values specified as maximum in the handbook.¹³ Table 5.5 shows cases similar to three of those in Table 5.3, except

Table 5.1. SAS2 Dose Rates for the IF 300 Rail Cask*

Decay Time	Fuel Cavity	Neutron Shield	Dose Type	Dose Rate in mr/hr				SZFCASK**
				Surface	1 Meter	2 Meter	3 Meter	
5 yr.	Dry	Wet	Neutron	7.175	2.211	1.191	.7025	1.0
			Gamma	8.976	2.718	1.406	.8291	
			Total	16.151	4.929	2.596	1.5316	
5 yr.	Wet	Wet	Neutron	1.958	.6114	.3034	.1950	1.0
			Gamma	2.421	.8092	.4454	.2710	
			Total	4.379	1.4206	.7488	.4660	
5 yr.	Dry	Dry	Neutron	255.27	78.90	40.08	22.96	1.0
			Gamma	5.34	1.97	1.14	.71	
			Total	260.61	80.87	41.22	23.67	
5 yr.	Dry	Wet	Neutron	8.278	2.625	1.415	.8335	0.5
			Gamma	9.084	2.776	1.443	.8525	
			Total	17.362	5.401	2.858	1.6860	
3 yr.	Dry	Wet	Neutron	9.030	2.863	1.543	.9091	0.5
			Gamma	16.193	5.362	2.934	1.7782	
			Total	25.223	8.225	4.478	2.6873	

* Notes: (1) For specified fuel a 3 yr. decay time is the minimum required to transport with a dry cavity, viz. the decay heat is less than ~40,000 BTU/hr.

(2) Analyses were done with an S₈ quadrature set.

** SZFCASK < 1 provides a finer spatial mesh than that automatically generated by SAS2.

Table 5.2. SAS2 Dose Rates for the NLI 1/2 Truck Cask^{*}

Decay Time	Fuel Cavity	Neutron Shield	Dose Type	Dose Rate in mr/hr				SZFCASK ^{**}
				Surface	1 Meter	2 Meter	3 Meter	
5 yr.	Dry	Wet	Neutron	1.743	.4191	.2148	.1239	1.0
			Gamma	3.565	.9206	.4724	.2776	
			Total	5.308	1.3397	.6872	.4015	
5 yr.	Dry	Dry	Neutron	77.19	18.92	9.148	5.112	1.0
			Gamma	3.28	.96	.530	.324	
			Total	80.47	19.88	9.678	5.436	
5 yr.	Dry	Wet	Neutron	2.093	.5191	.2665	.1534	0.5
			Gamma	3.690	.9589	.4942	.2910	
			Total	5.783	1.4780	.7607	.4444	

^{*} Analyses were performed with an S_8 quadrature set.

^{**} SZFCASK < 1 provides a finer spatial mesh than that automatically generated by SAS2.

Table 5.3. SAS2 Dose Rates for the IF 300 Rail Cask with Variation in Fuel Cooling Time*

Decay Time	Fuel Cavity	Neutron Shield	Dose Type	Dose Rate in mr/hr			
				Surface	1 Meter	2 Meter	3 Meter
5 yr	Dry	Wet	Neutron	8.273	2.623	1.414	8.329×10^{-1}
			Gamma	9.065	2.769	1.439	8.502×10^{-1}
			Total	1.734×10^1	5.392	2.853	1.683
10 yr	Dry	Wet	Neutron	6.896	2.187	1.179	6.944×10^{-1}
			Gamma	5.726	1.633	8.078×10^{-1}	4.646×10^{-1}
			Total	1.262×10^1	3.820	1.987	1.159
15 yr	Dry	Wet	Neutron	5.766	1.829	9.854×10^{-1}	5.806×10^{-1}
			Gamma	4.480	1.254	6.109×10^{-1}	3.483×10^{-1}
			Total	1.025×10^1	3.083	1.596	9.289×10^{-1}
20 yr	Dry	Wet	Neutron	4.827	1.531	8.249×10^{-1}	4.861×10^{-1}
			Gamma	3.615	1.001	4.834×10^{-1}	2.742×10^{-1}
			Total	8.442	2.532	1.308	7.603×10^{-1}
25 yr	Dry	Wet	Neutron	4.057	1.287	6.934×10^{-1}	4.085×10^{-1}
			Gamma	2.968	8.158×10^{-1}	3.917×10^{-1}	2.214×10^{-1}
			Total	7.025	2.103	1.085	6.299×10^{-1}
5 yr	Dry	Dry	Neutron	2.551×10^2	7.961×10^1	4.023×10^1	2.328×10^1
			Gamma	5.586	2.085	1.199	7.674×10^{-1}
			Total	2.607×10^2	8.170×10^1	4.143×10^1	2.405×10^1
10 yr	Dry	Dry	Neutron	2.126×10^2	6.634×10^1	3.352×10^1	1.940×10^1
			Gamma	1.843	6.689×10^{-1}	3.785×10^{-1}	2.397×10^{-1}
			Total	2.144×10^2	6.701×10^1	3.390×10^1	1.964×10^1
15 yr	Dry	Dry	Neutron	1.777×10^2	5.544×10^1	2.802×10^1	1.621×10^1
			Gamma	1.055	3.767×10^{-1}	2.109×10^{-1}	1.326×10^{-1}
			Total	1.788×10^2	5.582×10^1	2.823×10^1	1.634×10^1
20 yr	Dry	Dry	Neutron	1.488×10^2	4.644×10^1	2.347×10^1	1.358×10^1
			Gamma	6.722×10^{-1}	2.360×10^{-1}	1.307×10^{-1}	8.149×10^{-2}
			Total	1.495×10^2	4.668×10^1	2.360×10^1	1.366×10^1
25 yr	Dry	Dry	Neutron	1.250×10^2	3.900×10^1	1.971×10^1	1.141×10^1
			Gamma	4.564×10^{-1}	1.575×10^{-1}	8.626×10^{-2}	5.334×10^{-2}
			Total	1.255×10^2	3.916×10^1	1.980×10^1	1.146×10^1

* Analyses for normal transport conditions (wet neutron shield) were performed with an S_8 quadrature set. Analyses for accident conditions (dry neutron shield) were performed with an S_{16} quadrature set. All analyses used SZFCASK = 0.5.

Table 5.4. SAS2 Dose Rates for the NLI 1/2 Truck Cask with Variation in Fuel Cooling Time

Decay Time	Fuel Cavity	Neutron Shield	Dose Type	Dose Rate in mr/hr			
				Surface	1 Meter	2 Meter	3 Meter
5 yr	Dry	Wet	Neutron	2.092	5.188×10^{-1}	2.664×10^{-1}	1.534×10^{-1}
			Gamma	3.681	9.565×10^{-1}	4.928×10^{-1}	2.901×10^{-1}
			Total	5.773	1.475	7.592×10^{-1}	4.435×10^{-1}
10 yr	Dry	Wet	Neutron	1.745	4.329×10^{-1}	2.223×10^{-1}	1.280×10^{-1}
			Gamma	1.938	4.699×10^{-1}	2.297×10^{-1}	1.316×10^{-1}
			Total	3.683	9.028×10^{-1}	4.520×10^{-1}	2.596×10^{-1}
15 yr	Dry	Wet	Neutron	1.460	3.622×10^{-1}	1.860×10^{-1}	1.071×10^{-1}
			Gamma	1.423	3.363×10^{-1}	1.610×10^{-1}	9.109×10^{-2}
			Total	2.883	6.985×10^{-1}	3.470×10^{-1}	1.982×10^{-1}
20 yr	Dry	Wet	Neutron	1.225	3.033×10^{-1}	1.560×10^{-1}	8.980×10^{-2}
			Gamma	1.105	2.570×10^{-1}	1.213×10^{-1}	6.835×10^{-2}
			Total	2.330	5.608×10^{-1}	2.773×10^{-1}	1.579×10^{-1}
25 yr	Dry	Wet	Neutron	1.030	2.554×10^{-1}	1.312×10^{-1}	7.551×10^{-2}
			Gamma	8.840×10^{-1}	2.030×10^{-1}	9.489×10^{-2}	5.292×10^{-2}
			Total	1.914	4.586×10^{-1}	2.261×10^{-1}	1.284×10^{-1}
5 yr	Dry	Dry	Neutron	7.708×10^1	1.912×10^1	9.202	5.200
			Gamma	3.671	1.094	6.015×10^{-1}	3.785×10^{-1}
			Total	8.075×10^1	2.021×10^1	9.803	5.579
10 yr	Dry	Dry	Neutron	6.426×10^1	1.594×10^1	7.672	4.335
			Gamma	1.140	3.342×10^{-1}	1.825×10^{-1}	1.144×10^{-1}
			Total	6.540	1.627×10^1	7.855	4.449
15 yr	Dry	Dry	Neutron	5.372×10^1	1.333×10^1	6.414	3.624
			Gamma	6.060×10^{-1}	1.758×10^{-1}	9.558×10^{-2}	5.977×10^{-2}
			Total	5.433×10^1	1.351×10^1	6.510	3.684
20 yr	Dry	Dry	Neutron	4.501×10^1	1.117×10^1	5.374	3.037
			Gamma	3.545×10^{-1}	1.017×10^{-1}	5.501×10^{-2}	3.429×10^{-2}
			Total	4.536×10^1	1.127×10^1	5.429	3.071
25 yr	Dry	Dry	Neutron	3.781×10^1	9.381	4.514	2.551
			Gamma	2.192×10^{-1}	6.207×10^{-2}	3.336×10^{-2}	2.071×10^{-2}
			Total	3.803×10^1	9.443	4.547	2.572

* Analyses for normal transport conditions (wet neutron shield) were performed with an S_8 quadrature set. Analyses for accident conditions (dry neutron shield) were performed with an S_{16} quadrature set. All analyses used SZFCASK = 0.5.

Table 5.5. SAS2 Dose Rates for the IF 300 Cask with Maximum ^{59}Co *

Decay Time	Fuel Cavity	Neutron Shield	Dose Type	Dose Rate in mr/hr			
				Surface	1 Meter	2 Meter	3 Meter
5 yr	Dry	Wet	Neutron	8.273	2.623	1.414	8.329×10^{-1}
			Gamma	1.028×10^1	3.210	1.693	1.008
			Total	1.855×10^1	5.833	3.107	1.841
15 yr	Dry	Wet	Neutron	5.766	1.829	9.854×10^{-1}	5.806×10^{-1}
			Gamma	4.805	1.372	6.792×10^{-1}	3.908×10^{-1}
			Total	1.057×10^1	3.201	1.665	9.714×10^{-1}
25 yr	Dry	Wet	Neutron	4.057	1.287	6.934×10^{-1}	4.085×10^{-1}
			Gamma	3.055	8.476×10^{-1}	4.101×10^{-1}	2.328×10^{-1}
			Total	7.112	2.135	1.104	6.413×10^{-1}

* Analyses were performed with an S_8 quadrature set and SZFCASK = 0.5.

Table 5.6. SAS1 Dose Rates for the IF 300 Rail Cask with Consolidated Fuel*

Decay Time	Consolidation Factor**	Dose Type	Dose Rate in mr/hr			
			Surface	1 Meter	2 Meter	4 Meter
5 yr.	1	Neutron	8.299	2.648	1.417	.556
		Gamma	7.710	2.317	1.172	.449
		Total	16.009	4.915	2.589	1.005
5 yr.	1.5	Neutron	10.355	3.301	1.766	.693
		Gamma	9.532	2.824	1.412	.534
		Total	19.887	6.125	3.178	1.227
5 yr.	2	Neutron	11.748	3.743	2.002	.785
		Gamma	10.838	3.186	1.584	.594
		Total	22.586	6.929	3.586	1.379
10 yr.	1	Neutron	6.926	2.210	1.182	.464
		Gamma	5.043	1.412	.676	.243
		Total	11.969	3.622	1.858	.707
10 yr.	1.5	Neutron	8.641	2.755	1.473	.578
		Gamma	6.531	1.823	.870	.310
		Total	15.172	4.578	2.343	.888
10 yr.	2	Neutron	9.803	3.124	1.670	.655
		Gamma	7.607	2.120	1.010	.360
		Total	17.410	5.244	2.680	1.015

*Analyses were performed for normal dry fuel transport conditions with an S_{16} quadrature set and a mesh size comparable to SZFCASK = 1.0.

**Consolidation Factor = $\frac{\text{number of fuel pins placed in an assembly-equivalent area}}{\text{number of fuel pins per assembly}}$

Table 5.7. SAS1 Dose Rates for the NLI 1/2 Truck Cask with Consolidated Fuel^{*}

Decay Time	Consolidation Factor ^{**}	Dose Type	Dose Rate in mr/hr			
			Surface	1 Meter	2 Meter	4 Meter
5 yr.	1	Neutron	2.126	.531	.270	.102
		Gamma	2.917	.749	.377	.146
		Total	5.043	1.280	.647	.248
5 yr.	1.5	Neutron	2.878	.718	.366	.139
		Gamma	3.656	.919	.455	.173
		Total	6.534	1.637	.821	.312
5 yr.	2	Neutron	3.481	.868	.442	.167
		Gamma	4.287	1.062	.520	.195
		Total	7.768	1.930	.962	.362
10 yr.	1	Neutron	1.774	.443	.226	.086
		Gamma	1.536	.359	.168	.060
		Total	3.310	.802	.394	.146
10 yr.	1.5	Neutron	2.402	.599	.305	.116
		Gamma	2.119	.490	.228	.080
		Total	4.521	1.089	.533	.196
10 yr.	2	Neutron	2.905	.724	.369	.140
		Gamma	2.635	.607	.280	.098
		Total	5.540	1.331	.649	.238

*Analyses were performed for normal dry fuel transport conditions with an S₁₆ quadrature set and a mesh size comparable to SZFCASK = 1.0.

**Consolidation Factor = $\frac{\text{number of fuel pins placed in an assembly-equivalent area}}{\text{number of fuel pins per assembly}}$

Table 5.8. Comparison of IF 300 Rail Cask 5-Year-Decay Case with Different Quadrature Sets*

Detector Location	Dose Type	Dose Rate in mr/hr	
		S ₁₆ Quadrature Set	S ₈ Quadrature Set
Surface	Neutron	8.347	8.273
	Gamma	8.988	8.505
	Total	17.335	17.338
1 Meter	Neutron	2.661	2.623
	Gamma	2.786	2.769
	Total	5.447	5.392
2 Meters	Neutron	1.423	1.414
	Gamma	1.440	1.439
	Total	2.863	2.853
3 Meters	Neutron	0.856	0.833
	Gamma	.863	0.850
	Total	1.719	1.683

* Analyses were performed for normal dry fuel transport conditions with SZFCASK = 0.5.

that the maximum Co-59 content is specified. Thus, the difference between the corresponding dose rates of Table 5.3 and Table 5.5 equals the dose rates that are contributed by 1/2 the maximum Co-59 contents. Note that in all of these cases the Co-59 contribution appears rather small. However, the reader should keep in mind that the Co-59 effect is dependent both on the fuel type and design and on the specified burnup.^{12,13}

Tables 5.6-5.7 present results from SAS1 for the IF 300 and NLI 1/2 cask assuming normal (wet neutron shield) dry-cavity transport conditions and consolidated fuel loading, i.e., the fuel pins are removed from the assemblies and consolidated into a close-packed arrangement in a canister. The packing or consolidation factor is the number of pins per assembly equivalent area divided by the number of pins per assembly. Tables 5.6-5.7 provide results for consolidation factors of 1, 1.5 and 2 and cooling times of 5 years and 10 years. Since the assembly structure is not loaded in the cask, the spent fuel source used was altered to remove the gamma source contribution from Co-60 in the assembly structural materials.

Results for a consolidation factor of 1 in Tables 5.6-5.7 are presented to show differences between the results of SAS1 and the corresponding results of SAS2 as presented in Tables 5.3-5.4. The gamma dose rate results of Tables 5.6-5.7 are significantly smaller (13-25%) than the corresponding doses shown in Tables 5.3-5.4 because of the reduced gamma source noted above. The neutron dose rate differences are small and are due to minor changes in the calculational techniques: 1) SAS2 performed a cell-weighted calculation for the assembly cross-section while SAS1 did not and 2) the corresponding SAS2 results of Tables 5.3-5.4 were done with an S_8 quadrature set and SZFCASK = 0.5 while the results of Tables 5.6-5.7 were obtained with an S_{16} quadrature set and a mesh size comparable to SZFCASK = 1.0. The relatively good agreement in the corresponding neutron dose rate results indicate that the mesh spacing can be relaxed back to the SAS2 default if an S_{16} quadrature set is used for the water-filled neutron shield cases.

Finally, Table 5.8 provides a comparison of IF 300 results from SAS2 for normal dry fuel transport conditions using an S_8 and S_{16} quadrature set and a mesh size generated by SZFCASK = 0.5. The source was 5 year cooled spent fuel. Indications from previous results of this case was that an S_8 quadrature set required SZFCASK = 0.5 (see Table 5.1) but that using an S_{16} quadrature set allowed a relaxation to SZFCASK = 1.0 (compare Table 5.3 and Table 5.6). As a further test an S_{16} quadrature set was used with SZFCASK = 0.5 to see if the answers changed appreciably. The conclusion is that the surface dose rate difference is insignificant, indicating that the reduced mesh is not needed with an S_{16} quadrature set.

Note that all of the analyses presented use a one-dimensional (radial) transport treatment and the fuel sources are therefore assumed to be distributed uniformly in the axial direction. Thus, the computed dose rates are more correctly described as an average over the nominal fuel length since a typical fuel rod or assembly has an axial flux profile with some type of peaking factor. Therefore, at the axial peak, the actual dose rate (particularly at or close to the surface) would most likely be higher than that reported here. The peaking effect tends to die out as the detector distance from the surface is increased. The safety analysis reports^{1,2} for the subject casks used a peaking factor value of 1.2. So if maximum dose rates are desired, use of a similar peaking factor value should be considered.

6.0. SUMMARY

The casks analyzed in the study were the IF 300 rail cask and the NLI 1/2 truck cask. Dose rate results for these typical casks under normal and accident conditions have been presented for PWR spent fuel with cooling times of 5 years, 10 years, 15 years, 20 years and 25 years. The effect of fuel consolidation was also considered for cooling times of 5 and 10 years. The analyses were performed with the SAS1 and SAS2 control modules of the SCALE system.

The analyses were performed to support the Transportation Technology Center at Sandia National Laboratories in their study of public risks from transportation of spent fuel. However, the number of analyses performed allowed some preliminary conclusions to be reached concerning the preferred input to be used for the various cask conditions.

REFERENCES

1. IF 300 Shipping Cask: Consolidated Safety Analysis Report, NEDO-10084-2, Nuclear Fuel and Services Division, General Electric Corporation (October, 1979).
2. Safety Analysis Report: NLI-1/2 Spent Fuel Shipping Cask, National Lead Industries, Inc., Nuclear Division (1974).
3. O. W. Hermann, "SAS2: A Coupled One-Dimensional Depletion and Shielding Analysis Module," as published in Section S2 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 1 (as revised December 1984).
4. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vols. 1-3, Computing and Telecommunications Division at Oak Ridge National Laboratory (as revised December 1984). Original publication was January 1982 followed by major revisions and additions in June 1983 and December 1984.
5. O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," as published in Section F7 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 2 (as revised December 1984).
6. J. R. Knight, "SAS1: A One-Dimensional Shielding Analysis Module," Published as Section S1 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol.1 (to be published).
7. N. M. Greene, "BONAMI-S: Resonance Self Shielding by the Bondarenko Method," as published in Section F1 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 2 (as revised December 1984).
8. R. M. Westfall, L. M. Petrie, N. M. Greene, and J. L. Lucius, "NITAWL-S: Resonance Self-Shielding by the Nordheim Method," as published in Section F2 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 2 (as revised December 1984).

9. N. M. Greene and L. M. Petrie, "XSDRNPM-S: A One-Dimensional Discrete-Ordinates Program for Transport Analysis," as published in Section F3 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 2 (as revised December 1984).
10. O. W. Hermann, "COUPLE: SCALE System Module to Process Problem-Dependent Cross Sections and Neutron Spectral Data for ORIGEN-S Analyses," as published in Section F6 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 2 (as revised December 1984).
11. J. A. Bucholz, "XSDOSE: A Module for Calculating Fluxes and Dose Rates at Points Outside a Shield," as published in Section F4 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (ORNL/NUREG/CSD-2), Vol. 2 (as revised December 1984).
12. J. C. Ryman et al., Fuel Inventory and Afterheat Studies of Uranium Fueled Pressurized Water Reactor Fuel Assemblies Using the SAS2 and ORIGEN-S Modules of SCALE with an ENDF/B-V Updated Cross Section Library, NUREG/CR-2397 (ORNL/CSD-90) (September 1982).
13. A. G. Croff et al., Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, ORNL/TM-6051 (September 1978).

INTERNAL DISTRIBUTION

1. G. E. Whitesides/R. P. Leinius/ C&TD Library	25. T. H. Row
2. J. A. Bucholz	26. J. C. Ryman
3. D. L. Greene	27. L. B. Shappert
4. N. H. Greene	28. J. S. Tang
5-9. O. W. Hermann	29. V. C. A. Vaughen
10. P. E. Johnson	30. R. M. Westfall
11. D. S. Joy	31. R. G. Wymer
12-16. J. R. Knight	32. C. R. Library
17. J. V. Pace	33. Document Reference Section - Y-12
18-22. C. V. Parks	34-35. Laboratory Records
23. L. M. Petrie	36. Laboratory Records - RC
24. J. P. Renier	37. ORNL Patent Section

EXTERNAL DISTRIBUTION

38. J. N. Rogers, Div. 8474, Sandia National Laboratories, Livermore, CA 94550
39. Division of Engineering, Mathematics and Geosciences, U. S. Department of Energy, Washington, DC 20545
40. Office of Assistant Manager for Energy Research and Development, U. S. Department of Energy, Oak Ridge Operations, Oak Ridge, TN 37830
41. L. G. Blalock, Chief, Transportation Branch, U. S. Department of Energy, Oak Ridge Operations, Oak Ridge, TN 37830
42. K. Golliher, U. S. Department of Energy, Albuquerque Operations Office, P. O. Box 5400, Albuquerque, NM 87115
43. J. W. Cashwell, Sandia Laboratories, Nuclear Materials Transportation Technology Department, P. O. Box 5800, Albuquerque, NM 87185
44. S. Neuhauser, Sandia Laboratories, Nuclear Materials Transportation Technology Department, P. O. Box 5800, Albuquerque, NM 87185

45. R. E. Luna, Sandia Laboratories, Nuclear Materials Transportation Technology Department, P. O. Box 5800, Albuquerque, NM 87185
46. J. E. Diekhoner, Acting Director, Transportation Operations and Traffic Division Office, Defense Programs, U. S. Department of Energy, Germantown, MD 20545
47. F. P. Falci, Manager, Transportation R&D Program, U. S. Department of Energy, Washington, DC 20545
48. R. F. Garrison, Manager, Transportation Operations and Traffic Division Office, U. S. Department of Energy, Washington, DC 20545
49. J. F. Ney, Manager, Nuclear Materials Transportation Technology Department 6320, P. O. Box 5800, Albuquerque, NM 87185
50. D. S. Delmastro, P. O. Box 286, Madison, PA 15663
51. R. H. Odegarden, Transportation and Certification Branch, Office of Nuclear Material Safety & Safeguards, Nuclear Regulatory Commission, Mail Stop 396-SS, Washington, DC 20555
- 52-78. Technical Information Center, U. S. Department of Energy, Oak Ridge, TN 37830