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RADIATION DOSE RATES FROM CONSOLIDATED FUEL IN CURRENT GENERATION SHIPPING CASKS

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As part of a DOE study of public risk from cask transport of consolidated and unconsolidated spent fuel cooled beyond five years, radiation dose rates from two current generation shipping casks were evaluated. The IF300 rail shipping cask[1] (with a capacity of seven PWR assemblies) and NLI 1/2 truck shipping cask[2] (with a capacity of one PWR assembly) were selected as appropriate cask models. A Westinghouse 17x17 fuel assembly with a 3.3 wt % ^{235}U enrichment, operated at a specific power of 37.5 MW/MTU for 880 full-power days (20% downtime in history), and discharged at a burnup of 33 GWD/MTU was specified. Results for spent fuel cooling times of 5, 10, 15, and 25 years are reported here.

All calculations were performed using the SCALE computational system.[3] The SAS2 control module was used for the fuel depletion and the shielding analyses for unconsolidated fuel (normal fuel assemblies). The execution includes: repeated passes through BONAMI-S, NITAWLS, XSDRNPM-S, COUPLE and ORIGEN-S for cross-section processing and fuel burnup; radiation source computations by ORIGEN-S; a one-dimensional radial shielding analysis of the cask via XSDRNPM-S; and the final determination of the dose rates by XSDOSE using the surface angular flux from the finite cask. The SAS1 module was used for the shielding analyses involving consolidated fuel because appropriate fuel number densities and volume fractions can be input whereas the SAS2 shielding portion automatically uses the assembly data from the depletion analysis. The SAS1 module also employs NITAWL-S, XSDRNPM-S, and XSDOSE for resonance self-shielding, radiation transport, and dose evaluation, respectively

The SCALE 27-group ENDF/B-IV neutron cross sections were used in the depletion analysis and the SCALE 27n-18γ coupled ENDF/B-IV library was used in the shielding analysis.

The calculations were performed by stopping SAS2 following the depletion analysis, running ORIGEN-S stand-alone to decay the fuel assembly to the specified cooling times and generate radiation sources, and finally using these sources as input to a SAS1 case or a SAS2 restart case with the appropriate cask model specifications. ORIGEN-S generates an output file with both the neutron and gamma source spectrum in the specified 27n-18γ energy group structure at the required decay times. The ORIGEN-S output files differed for the consolidated and unconsolidated fuel in that Co-60 in the structural hardware of the assembly was removed for the consolidated fuel because the hardware is not present in the cask and Co-60 contributes almost all of the hardware's gamma source.

The results obtained with the consolidated fuel are shown in Table 1 where a consolidation of fuel pins from two assemblies is assumed to be placed in an assembly-equivalent area. Dose rates for the casks containing unconsolidated fuel assemblies are provided in Table 2. All of the results are for normal conditions of transport; i.e., dry fuel cavity and water-filled neutron shield. As expected the doubling of the fuel capacity by consolidation of fuel pins does not double the dose rate because of increased self-shielding and loss of the Co-60 hardware source. In general the results indicate the amount of extra shielding available (beyond that needed to meet NRC and DOT regulations) in these casks for long-term cooled fuel in both a consolidated and unconsolidated form. However, note that the dose rates are based on a uniform axial distribution of the source and so represent an "average" radial dose rate emitted from the cask. If a maximum dose rate is desired, multiplication of the results by an axial peaking factor is recommended. These results provide useful information for evaluating public exposure for present generation shipping casks and give a basis for comparison during the design of new generation casks developed specifically for long-term cooled fuel.

Table 1. Dose Rates from PWR Consolidated Fuel* Using the IF300 Rail Cask and NLI 1/2 Truck Cask Models

Cask	Fuel Cooling Time	Dose Type	Dose Rate in mr/hr** Surface	1 Meter	2 Meter
IF300	5 yr	Neutron	11.748	3.743	2.002
		Gamma	10.838	3.186	1.584
		Total	22.586	6.929	3.586
IF300	10 yr	Neutron	9.803	3.124	1.670
		Gamma	7.607	2.120	1.010
		Total	17.410	5.244	2.680
IF300	15 yr	Neutron	8.203	2.614	1.398
		Gamma	6.253	1.733	0.822
		Total	14.456	4.347	2.220
IF300	25 yr	Neutron	5.784	1.843	0.986
		Gamma	4.361	1.205	0.570
		Total	10.145	3.048	1.556
NLI 1/2	5 yr	Neutron	3.481	0.868	0.442
		Gamma	4.287	1.062	0.520
		Total	7.768	1.930	0.962
NLI 1/2	10 yr	Neutron	2.905	0.724	0.369
		Gamma	2.635	0.607	0.280
		Total	5.540	1.331	0.649
NLI 1/2	15 yr	Neutron	2.431	0.606	0.309
		Gamma	2.129	0.485	0.223
		Total	4.560	1.091	0.532
NLI 1/2	25 yr	Neutron	1.715	0.428	0.218
		Gamma	1.470	0.333	0.152
		Total	3.185	0.761	0.370

*Fuel pins from two fuel assemblies consolidated into one assembly - equivalent area.

**ANSI dose conversion factors used to obtain the dose rates.

Table 2. Dose Rates from PWR Fuel Assemblies Using the IF300 Rail Cask and NLI 1/2 Truck Cask

Cask	Fuel Cooling Time	Dose Type	Dose Rate in mr/hr*		
			Surface	1 Meter	2 Meter
IF300	5 yr	Neutron	8.273	2.623	1.414
		Gamma	9.065	2.769	1.439
		Total	17.338	5.392	2.853
IF300	10 yr	Neutron	6.896	2.187	1.179
		Gamma	5.726	1.633	0.808
		Total	12.622	3.820	1.987
IF300	15 yr	Neutron	5.766	1.829	0.985
		Gamma	4.480	1.254	0.611
		Total	10.246	3.083	1.596
IF300	25 yr	Neutron	4.057	1.287	0.693
		Gamma	2.968	0.816	0.392
		Total	7.025	2.103	1.085
NLI 1/2	5 yr	Neutron	2.092	0.518	0.266
		Gamma	3.681	0.956	0.493
		Total	5.773	1.475	0.759
NLI 1/2	10 yr	Neutron	1.745	0.433	0.222
		Gamma	1.938	0.470	0.230
		Total	3.683	0.903	0.452
NLI 1/2	15 yr	Neutron	1.460	0.362	0.186
		Gamma	1.423	0.336	0.161
		Total	2.883	0.698	0.347
NLI 1/2	25 yr	Neutron	1.030	0.256	0.131
		Gamma	0.884	0.203	0.095
		Total	1.914	0.459	0.226

*ANSI Standard dose conversion factors were used.

References

1. IF300 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. 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.