NUREG/CR-6672: RESULTS AND CONCLUSIONS

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ABSTRACT

The results of the seven sets of RADTRAN analyses performed for the NUREG/CR-6672 study, using the analysis methods and input data described in the previous five papers, are presented. Specifically, incident-free and accident population dose risks are presented, mainly for rail spent fuel shipments via 200 generic routes that span all regions of the continental United States, but also for four real rail shipment routes, and for the NUREG-0170 rail route. In addition, the accident risks calculated using NUREG-0170 and Modal Study rail accident source terms and source term probabilities are compared to the risks calculated using the 21 rail accident source terms and source term probabilities developed for the NUREG/CR-6672 study. These calculations suggest that spent fuel shipment accident risks are three to four orders of magnitude smaller than the spent fuel accident risk estimates published in NUREG-0170, NRC's EIS for the transport of radioactive materials. Finally, the conclusions suggested by these results are presented.

INTRODUCTION

The preceding five papers [1-5] have described:

- the structure of the NUREG/CR-6672 [6] analyses,
- the four generic spent fuel casks examined by those analyses,
- the response of the four generic casks to fully engulfing optically dense fires,
- the response of the casks, especially the monolithic steel rail cask, to collisions,
- the conversion of unyielding surface cask impact speeds to yielding surface impact speeds that inflict the same damage on the cask as was caused by the unyielding surface impact,
- the scaling of 30-mph impact, rod strain maps to higher impact speeds and the estimation of rod failure fractions by comparison of the scaled strains to a strain failure criterion,
- the models used to estimate source term release fractions and severity fractions,
- the construction of cumulative distributions for important RADTRAN input parameters (e.g., route parameters) that can take on a wide range of values in the real world, and
- the construction of 200 sets of input parameter values for these important RADTRAN input parameters by structured Monte Carlo (LHS) sampling of the cumulative distributions constructed for these parameters.

This paper will summarize the RADTRAN [7,8] calculations performed for the NUREG/CR-6672 study and present the principal results of the study and the conclusions drawn from those results.

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RADTRAN CALCULATIONS

Seven sets of RADTRAN calculations were performed for the NUREG/CR-6672 study. Each calculation developed estimates of the radiological consequences and risks associated with the shipment of a single generic Type B cask that contains power reactor spent fuel. Two types of consequences and risks were estimated—those associated with the occurrence of accidents during the shipment and those associated with shipments that take place without the occurrence of accidents.

The seven sets of RADTRAN calculations examined four cask designs, two shipment modes, two sets of routes, two types of spent fuel, and three sets of accident source terms. The four generic cask designs examined were steel-lead-steel truck and rail casks, a steel-DU-steel truck cask, and a monolithic steel rail cask. The two shipment modes were truck and rail. The two sets of routes were (a) 200 representative routes selected by Latin Hypercube Sampling (LHS) of cumulative route parameter distributions and (b) four illustrative real routes plus the two NUREG-0170 shipment routes. The four real routes were: (1) the Crystal River Nuclear Plant in Florida to Hanford, Washington; (2) the Maine Yankee Nuclear Plant to Skull Valley, Utah; (3) the Maine Yankee Nuclear Plant to the Savannah River Site in South Carolina; and (4) the Kewaunee Nuclear Plant in Wisconsin to the Savannah River Site. The two types of spent fuel were three year cooled, high-burnup PWR and BWR fuel. The three sets of accident source terms were the NUREG-0170 [9] source terms, the Modal Study source terms [10], and the new source terms developed by the NUREG/CR-6672 study.

Accident Risks for Representative Routes. By taking all possible combinations of the single set of central estimate values developed for the RADTRAN input parameters that do not vary widely in the real world, the 200 sets of parameters that do vary widely in the real world, and the 21 sets of representative truck or rail accident severity and release fraction values, input for 4200 single-pass RADTRAN 5 truck or rail spent fuel transportation calculations was developed for each generic truck or rail cask. Because each of the 200 sets of input values for the parameters, that do vary widely in the real world, was run with each of the 21 representative accident source terms, 200 different Complementary Cumulative Distribution Functions (CCDFs) were developed for each combination of a fuel type and a generic cask. Figure 1 presents the 200 CCDFs that were calculated for hypothetical accidents during the transport of 3-year-cooled, high-burnup, PWR spent fuel in the generic steel-lead-steel truck cask.

Because the density of the curves in Figure 1 is so great, to better depict the spread of possible consequences and their probabilities of occurrence, four compound CCDFs were constructed for each RADTRAN calculation that used the 200 sets of representative input. These four compound CCDFs were the expected (mean) result, and the 5th, 50th (median), and 95th quantile results, where for any specific single consequence value the corresponding 5th and 95th quantile probabilities are the probabilities of the CCDFs that lie 10 up from the bottom and 10 down from the top of the set of 200 CCDFs, the corresponding median quantile probability is the average of the probability values for CCDF 100 and CCDF 101, and the expected (mean) result is the average of all of the CCDF probability values that correspond to the specified consequence value. Figure 2 presents the results produced by this procedure for the calculation that examined hypothetical accidents during the transport of for 3-year-cooled, high-burnup, PWR spent fuel in the generic steel-lead-steel rail cask.

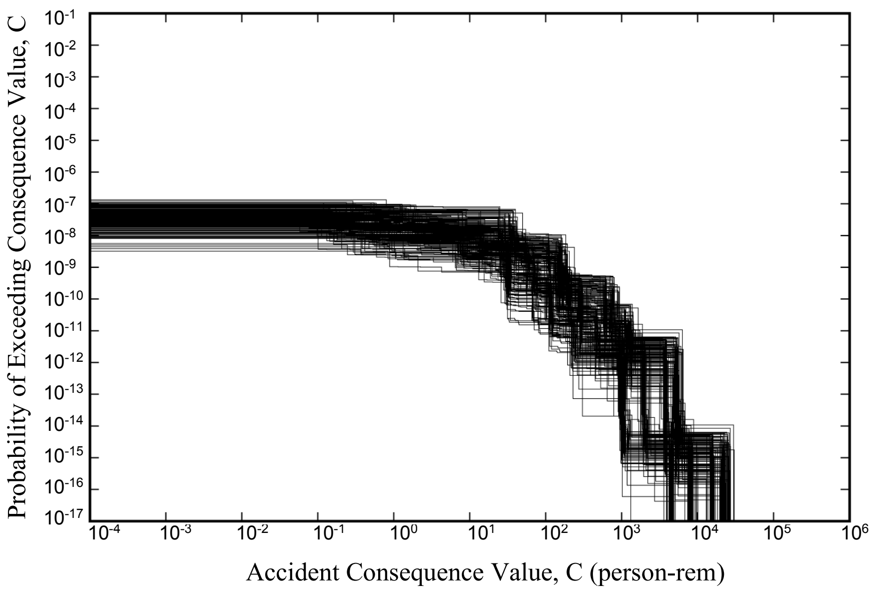

Figure 1. Two hundred truck accident population dose risk CCDFs, one CCDF for each representative truck route. Each RADTRAN 5 calculation examined all 21 representative truck accident source terms and assumed transport of PWR spent fuel in the generic steel-lead-steel truck cask.

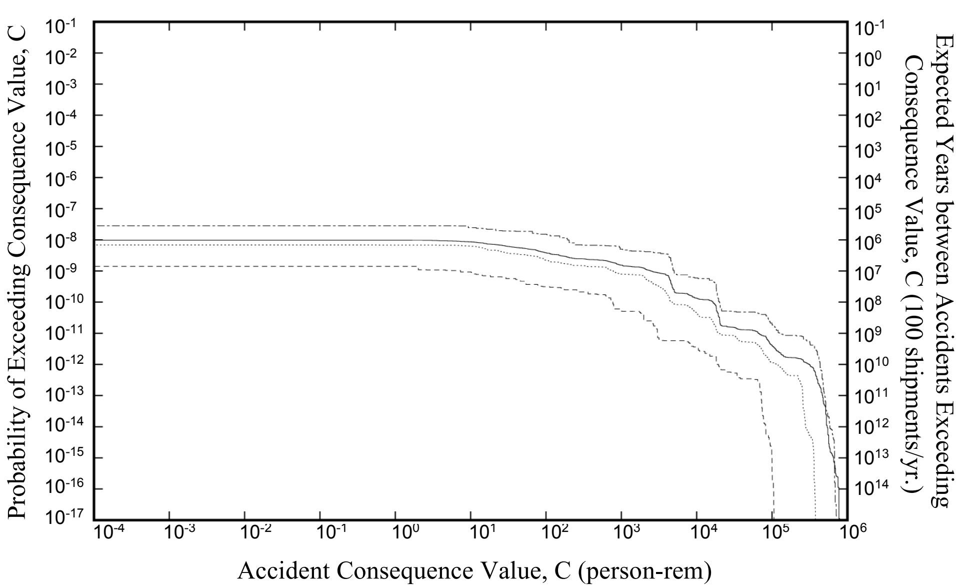

Figure 2. Rail accident population dose risk CCDFs for transport of PWR spent fuel in the generic steel-lead-steel rail cask over the 200 representative rail routes. Each underlying RADTRAN 5 calculation generated results for all of the 21 representative rail accident source terms.

Mean (————) CCDF, and 95th (–·–·–·–·–) , 50th (··········), and 5th (----------) quantiles

Figures 1 and 2 show that the chance of having a truck or rail accident that causes a total population dose of 0.1 person-rem or greater is less than 10^{-7} per shipment. Because the area under a CCDF is the expected result for the calculation that generated the CCDF, the expected accident risks for spent fuel transport by truck or rail can be calculated by averaging the mean results for the 200 truck or rail CCDFs or calculating the area under the mean truck or rail compound CCDF. For transport of 3-year-cooled, high-burnup PWR spent fuel in steel-lead-steel truck or rail casks, the estimated values of the expected accident risk per shipment are respectively 8.0×10^{-7} and 9.4×10^{-6} person-rem. Since the truck cask carries 1 spent fuel assembly while the rail cask carries 24, most of the difference between the expected risks is accounted for by the greater inventory of the rail cask.

Accident Risks for Real Rail Routes. The four illustrative real routes examined in the NUREG/CR-6672 study were chosen for the following reasons. The routes from the Crystal River nuclear plant to Hanford are about the longest routes possible in the continental United States. Because they traverse the Boston-Washington urban corridor, the routes from the Maine Yankee nuclear plant to the Savannah River Site have urban length fractions and population densities that are about as high as is possible in the continental United States. The routes from the Maine Yankee nuclear plant to Skull Valley represent long routes to the Yucca Mountain area that traverse the urban Midwest. And the routes from the Kewaunee nuclear plant to the Savannah River Site have route parameter values (especially the urban parameter values) close to the means of the route parameter distributions used to construct the 200 representative truck and rail routes contained in the LHS sample of size 200. Figure 3 presents the mean CCDFs calculated for the four real rail routes and for the NUREG-0170 representative rail route and compares these CCDFs to the highest 95th and the lowest 5th quantile compound CCDFs

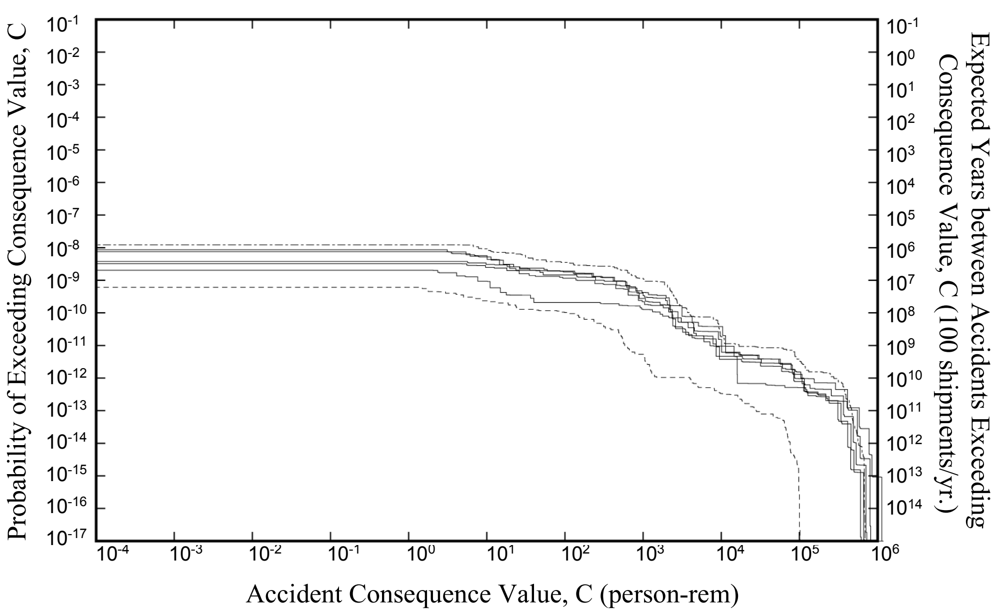

Figure 3. Comparison of rail accident population dose risk CCDFs for transport of PWR spent fuel in the generic monolithic steel cask over four illustrative rail routes and the NUREG-0170 representative rail route.

Five Mean CCDFs (———), and Highest 95th (----------) and Lowest 5th (----------) quantiles

generated by the four calculations that examined each possible combination of a type of spent fuel (PWR or BWR) and a generic rail cask (steel-lead-steel or monolithic steel). Figure 3 shows that the hypothetical accident CCDFs, calculated for transport of 3-year-cooled, high-burnup, spent fuel in a monolithic steel rail cask along the four real rail routes and the NUREG-0170 representative rail route, all lie within the range of the results obtained for the 200 representative rail routes that were generated by LHS sampling of the distributions of RADTRAN input parameters that have values that vary widely in the real world.

Comparison of Accident and Incident-Free Dose Risks. Table 1 compares the accident and incident free dose risks developed by the NUREG/CR-6672 study.

Table 1. Comparison of Accident and Incident-Free Dose Risks (person-rem)

Dose Risk	Cask	Spent Fuel (No. Assemblies)	Truck	Rail
Incident-Free	All	All	$(4.4 \pm 2.6) \times 10^{-2}$	$(2.0 \pm 1.4) \times 10^{-2}$
Accident	Steel-Lead-Steel	PWR (T=1; R=24)	$(8.0 \pm 8.5) \times 10^{-7}$	$(9.4 \pm 12) \times 10^{-6}$
		BWR(T=3; R=52)	$(3.3 \pm 3.6) \times 10^{-7}$	$(9.2 \pm 12) \times 10^{-6}$
	Steel-DU-Steel	PWR (T=1)	$(2.3 \pm 2.4) \times 10^{-6}$	
		BWR (T=3)	$(1.1 \pm 1.2) \times 10^{-6}$	
	Monolithic Steel	PWR (R=24)		$(2.0 \pm 2.5) \times 10^{-6}$
		BWR (R=52)		$(1.5 \pm 1.9) \times 10^{-6}$

Because the probability of the incident-free dose is unity (i.e., if the shipment is completed without the occurrence of a severe accident, then the calculated incident-free dose will be incurred), incident-free doses and dose-risks have the same values. Moreover, because the probabilities of severe accidents are so small ($< 10^{-7}$ per shipment), as Table 1 shows, incident-free dose risks are about four orders-of-magnitude larger than accident dose risks. Because incident-free dose depends only on cask surface dose rate and on-route and along route population densities, as Table 1 shows, incident-free dose risks do not depend on spent fuel type or on the type of cask in which the spent fuel is transported. Finally, Table 2 compares the single shipment, NUREG/CR-6672, incident free doses to those presented in NUREG-0170.

Table 2. Comparison of NUREG-0170 Incident-Free Doses to the Incident-Free Doses Developed by the NUREG/CR-6672 Study

	Truck		Rail	
	NUREG-0170	**NUREG/CR-6672**	**NUREG-0170**	**NUREG/CR-6672**
Stops	0.0190	0.0153	0.0052	0.0044
Other[a]	0.145	0.0288	0.02729	0.0159
Stops + Other	0.164	0.0441	0.0325	0.0203

a. Sum of crew, on-route, and off-route doses.

Table 2 shows that for single shipments the sum of the other incident-free doses (i.e., crew, on-route, off-route, and stop doses) developed by the NUREG/CR-6672 study for spent fuel transport by trucks with two-person crews is about one-fourth of the sum of the corresponding NUREG-0170 truck doses. It also shows that the sum of the "Stops + Other" NUREG/CR-6672 incident-free doses for transport by rail is about two-thirds of the sum of the corresponding

NUREG-0170 rail doses. The similarity of these incident-free results is not surprising, because both studies assume that the surface dose rates of spent fuel transportation casks are somewhat below the regulatory limit and both use on-route and along-route population densities and population densities at rest stops that are not very different.

Effect of Source Terms on Accident Risks. Because the spent fuel risk assessment methodology developed by the Modal Study [10] was the basis for all of the analyses conducted for the NUREG/CR-6672 study, it was of interest to compare accident population dose risk CCDFs and mean accident population doses calculated by RADTRAN 5 using NUREG-0170 Model I and Model II source terms, Modal Study source terms, and the source terms developed for the NUREG/CR-6672 study. Each of these calculations examined transport of PWR spent fuel in a steel-lead-steel spent fuel cask and used the LHS sample of size 200 that contained the representative set of 200 truck or rail routes. Thus, the calculations differ only in the sets of source terms used and in their treatments of exposure pathways (the NUREG-0170 calculations only modeled inhalation dose pathways, while the Modal Study calculation and the calculation that used the source terms developed for NUREG/CR-6672 modeled all exposure pathways). Accordingly, these calculations compare the NUREG-0170 result to the Modal Study result and to the result developed by the NUREG/CR-6672 study. Figure 4 presents the mean accident CCDFs obtained using each of these sets of source terms for transport of 3-year-cooled, high-burnup, PWR spent fuel in a steel-lead-steel cask over the set of 200 representative rail routes.

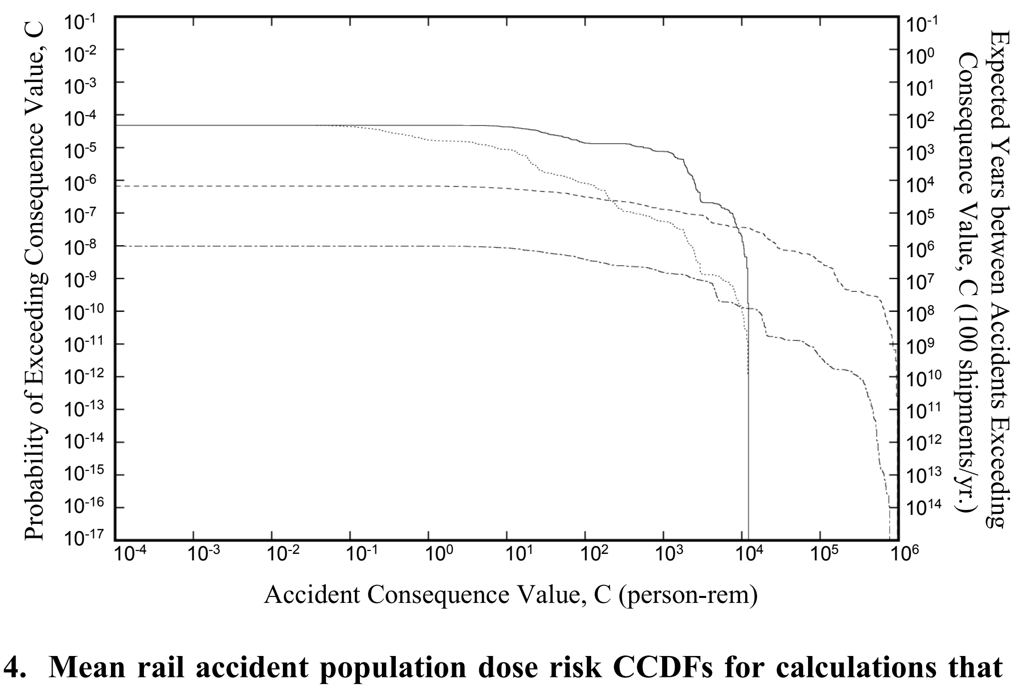

Figure 4. Mean rail accident population dose risk CCDFs for calculations that compared the source terms developed by NUREG-0170, the Modal Study, and the NUREG/CR-6672 study.

NUREG-0170 source term, NUREG-0170 Model I release fractions, only inhalation pathways
NUREG-0170 source term, NUREG-0170 Model II release fractions, only inhalation pathways
PWR inventory, 20 Modal Study rail source terms, all exposure pathways
PWR inventory, 21 NUREG/CR-6672 rail accident source terms, all exposure pathways

The areas under the CCDFs presented in Figure 4 give the expected values of the rail accident dose-risks calculated by each study. Table 3 presents the absolute and relative values of these rail accident dose-risks and also the corresponding absolute and relative truck accident dose-risk values.

Table 3. Comparison of Expected Values of NUREG-0170, Modal Study, and NUREG/CR-6672 Accident Dose-Risks

Study	Truck Accidents		Train Accidents	
	Absolute Value	Relative Value	Absolute Value	Relative Value
NUREG-0170 Model I	1.3×10^{-2}	1.6×10^{4}	1.9×10^{-2}	2.0×10^{3}
NUREG-0170 Model II	7.7×10^{-4}	9.6×10^{2}	4.9×10^{-4}	5.2×10^{1}
Modal Study	1.3×10^{-4}	1.6×10^{2}	1.9×10^{-3}	2.0×10^{2}
NUREG/CR-6672	8.0×10^{-7}	1.0	9.4×10^{-6}	1.0

The relative orderings of the accident CCDFs presented in Figure 4 and the expected values of accident dose-risks presented in Table 3 are entirely consistent with the assumptions made by each study regarding the probability of radionuclide leakage from a spent fuel cask during transportation accidents and the magnitude of the source terms generated by accidents of differing severities. Because both Model I and Model II in NUREG-0170 assumed that spent fuel casks might release a portion of their contents during relatively minor accidents, the fraction of all truck or train accidents predicted by these models to cause releases is very large and extremely conservative. Similarly, because the NUREG-0170 Model I assumed that all cask leaks lead to the release of the entire NUREG-0170 accident source term, the mean population doses calculated using that accident source term are quite large. When, as was done by the Modal Study, cask failure and source term probabilities and magnitudes are estimated from the response of the cask shell to mechanical and thermal loads, both source term probabilities and most source term magnitudes decrease. Consequently, mean accident population dose risks decrease by one or two orders of magnitude. When, as was done by the NUREG/CR-6672 study, cask failure and source term probabilities and magnitudes are estimated by examining the response of cask closures and spent fuel rods to impact loads, and the burst rupture of spent fuel rods caused by heating by fires, then cask release is found to be even less likely and retention of particles and condensable vapors by deposition onto cask interior surfaces is found to be substantial. Accordingly, except for the most severe accidents examined (the double failure accidents for which deposition was assumed not to occur), source term probabilities and most source term magnitudes, decrease even further. Therefore, expected accident population dose risks are further decreased by factors of 10 to 100.

Finally, source term magnitudes for the most severe accidents examined by the Modal Study and the NUREG/CR-6672 study are larger than the largest source term magnitude postulated in NUREG-0170. They are larger because, for both the Modal Study and the NUREG/CR-6672 study, the product of the cask inventory and the largest accident release fractions developed by each of these studies is larger than the largest source term examined by NUREG-0170. Nevertheless, although the largest source terms developed by the analyses performed by the Modal Study and the NUREG/CR-6672 study are larger than the largest NUREG-0170 source term, the accident risks posed by these source terms are substantially smaller because these source terms are so very improbable.

CONCLUSIONS

The results described in detail in the NUREG/CR-6672 report lead to the following conclusions:

- The single cask, truck shipment, expected incident-free population doses developed by the NUREG/CR-6672 study are about one-quarter of those in NUREG-0170.
- The single cask, rail shipment, expected incident-free population doses developed by the NUREG/CR-6672 study are about two-thirds of those in NUREG-0170.
- The use of very conservative cask failure criteria in NUREG-0170 caused its estimates of the fraction of all accidents that release radioactive materials to be much too large and thus very conservative.
- The NUREG-0170 estimate of the largest source term that might be released from a failed spent fuel cask during an unusually severe transportation accident is significantly lower than the largest source terms calculated using Modal Study release fractions or the release fractions developed by the NUREG/CR-6672 study. However, the risks associated with these very large Modal Study or NUREG/CR-6672 source terms are lower than the risk of the largest NUREG-0170 source term because these very large source terms are so very improbable.
- The source terms developed by the Modal Study and by the NUREG/CR-6672 study, which reflect the complexities of rod failure and cask response to transportation accident impact and thermal loads, yield estimates of expected (mean) spent fuel transportation accident population doses that are orders of magnitude smaller than those developed by the NUREG-0170 study.

Overall, the results of the NUREG/CR-6672 study confirm the validity of the NUREG-0170 estimates of spent fuel incident-free population doses. The results also show that the NUREG-0170 estimates of spent fuel accident population dose risks were very conservative, as was believed to be true when NUREG-0170 was published [11].

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