LAWRENCE LIVERMORE NATIONAL LABORATORY

Fission Energy and Systems Safety Program

December 15, 1999
NTSS00-24/LEF

Ms. Patricia Eng
NMSS/SFPO/STRD
U.S. Nuclear Regulatory Commission
Two White Flint North MS 013D13
11545 Rockville Pike
North Bethesda, MD 20852

SUBJECT: Memo report on Final Draft Report, "Reexamination of NUREG-0170 Spent Fuel Shipment Risk, Estimates"

Dear Ms. Eng:

Please thank John Cook for his review and comments on our draft memo report documenting our review of the Sandia National Laboratories final draft report "Re-examination of NUREG-0170, Spent Fuel Shipment Risk Estimates". In response to his comments, we have revised our memo to clarify or modify it as appropriate. These are our specific responses to his comments:

1. The inventories in Table 7.9 are conservative and can be used for estimating accident releases but should not be used to calculate the radiation levels and temperatures of the representative casks. As discussed in the radiation and thermal sections of our memo, the radiation levels and fuel rod temperatures would be significantly higher if they were properly calculated and would result in significantly higher consequences and risk.

2. We revised our comments on the comparison of NUREG-0170, Model Study and Sandia Study results in the Summary and Conclusions section.

3. Statements that speculate how the public might react to the study have been removed.

4. By canning the spent fuel another barrier is provided against release but this will not effect the fuel rod failure strain criteria. If credit is taken for the canning then it should be analyzed in the study and be required for spent fuel shipments.

An Equal Opportunity Employer•University of California•P.O. Box 808 Livermore, California 94550•
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Attachment 1

5. The inclusion of the 50-year groundshine dose is overly conservative and has a major impact on the comparison of the NUREG-0170, Modal Study and Sandia Study results as discussed in the Summary and Conclusions.

Attached are two copies of the subject report. The text report was also e-mailed to you. Please do not hesitate to call me at (925) 423-0195 or e-mail me at fischer3@llnl.gov if you have any questions concerning the memo report.

Sincerely,

Larry E. Fischer
NRC Transportation and Storage Projects
Fission Energy and Systems Safety Program

Attachment

LEF/ab

Sheaffer M.	L-638
Smith, C.	L-632
Surles, T.	L-640
Theofanous, T.	UCSB
Cook, J.	(NRC)

Comments on Final Draft Report

Re-examination of NUREG-0170
Spent Fuel Shipment Risk Estimates

LLNL Review Team

Reviewer	Review Areas
Brian Anderson	Containment/Release
Moe Dehghani	Structures/Closure
Larry Fischer	General/Thermal/Fuel
Edwin Jones	Methodology/RADTRAN/Results
Mike Sheaffer	Shielding/Release/Special Topics
Monika Witte	Structures/Fuel
Theo Theofanous (UCSB)	Risk Assessment and Management

Introduction

The Nuclear Regulatory Commission (NRC) contracted Lawrence Livermore National Laboratory (LLNL) to review the report "Re–examination of NUREG-0170 Spent Fuel Shipment Risk Estimates" [1]. The updated NUREG-0170 report was being prepared by Sandia National Laboratories (SNL) and was to be submitted to the NRC and LLNL for review.

NRC, LLNL, and SNL agreed that the technical review would be conducted not as a Safety Analysis Report review but as a Risk Estimate Review that emphasized the use of realistic assumptions and analyses rather than bounding conservative ones. LLNL assembled a team of six members to perform the detailed review: Brian Anderson, Moe Dehghani, Larry Fischer (Project Leader), Ed Jones, Mike Sheaffer and Monika Witte. Professor Theo Theofanous of the University of California, Santa Barbara (UCSB) was contracted by LLNL to assist in the review and provide an independent overview in Risk Assessment and Management.

LLNL met with the NRC and SNL in Albuquerque, New Mexico, January 26-27, 1999, to review the overall methodology and technical approaches used in conducting the SNL Study, which would then be documented in their report. The overall methodology and technical approaches were judged to be reasonable provided that they were properly implemented. Of particular interest were the implementation of details, assumptions, and reasoning in five major areas:

1. The use of RADTRAN and Latin Hypercube Sampling (LHS) to assess risk.
2. The closure/seal failure on the cask.
3. The leakage hole size and release of radioactive material from the cask cavity to the environment.
4. The failure mechanisms for the spent fuel rods.
5. The use of 50-year groundshine assumption for the affected area following an unrestricted release.

These identified areas were later documented for the NRC and SNL as initial comments (Appendix A).

The first draft of the SNL report was received in mid-March 1999. The draft was incomplete and difficult to evaluate. It was decided to delay the detailed review until a second draft was submitted. The major portion of the second draft was received in mid-May followed by the final three sections in mid-June. The detailed review was completed in mid-July. Because of scheduling conflicts, LLNL provided review comments in two meetings: July 21 at Albuquerque on structures (Appendix B) and July 29 in Livermore on all other areas (Appendix C). During the meetings LLNL presented its major technical concerns and in some cases suggested approaches to resolve the concerns. Prior to closing the meeting all parties thought that the concerns could be satisfactorily addressed and resolved.

The third and final draft of the report was received in mid-October. This draft was reviewed with respect to LLNL's comments presented in previous meetings to determine if they had been satisfactorily addressed and resolved. It appears that in some cases the concerns were resolved but in others there are still open issues/items as identified in the rest of this report. To illustrate our concerns in two areas, we have included thermal and release analyses that estimate significantly higher fuel temperatures, and radioactive material releases than those documented in the updated NUREG-0170 report.

At the end of this report we summarize eight major concerns which are presented in detail in the following sections. An overall conclusion is that the report lacks clarity, robustness, and communicability to the public.

Methodology/RADTRAN (Sections 2.0, 3.0 and 7.0)

Within the Probabilistic Risk Analysis (PRA) framework the report does a good job of laying out the RADTRAN inputs by comparing RADTRAN 1 and RADTRAN 5 variables and by explaining the fixed and sampled input variables for this study. However, for the non-expert, the RADTRAN calculus that utilizes the inputs is not explicitly stated nor is it clear. For someone comfortable with the use of RADTRAN, the nature of the inputs is of great interest; for the unfamiliar, the nature of the inputs does not mean much without an understanding of the role they play in determining risk. The nature of the risk equation used in the SNL Study is not mentioned until Section 7.2; it required an experienced reviewer a thorough reading of all of Sections 3 and 7, and sojourns into Appendix E, before understanding the probable risk formulation. There remained some uncertainty about how the risk was calculated by RADTRAN until questioning the report authors.

Examples follow of modeling assumptions that are important to the analysis but are implicit in the discussion and not explicitly presented for the reader in a single place. The risk per shipment is the product of the magnitude of an accident consequence and its probability of occurrence. For the probability of occurrence, as in the Modal study, it is assumed that the probability of an accident is unrelated to the type of accident. Also, route segments are assumed to be uncorrelated or statistically disjoint. Given these assumptions and the Latin Hypercube Sampling (LHS) approach to representing the potential plethora of transportation routes, each representative 'route' (of the 200 sample size) is specified by: the route length; the fractions for rural, suburban, and urban segments; the accident frequencies for each type of link (rural, suburban, and urban); sampled parameters for each type of link, (e.g., population density, weather conditions, etc.); and the fixed input variables. These assumptions and the LHS approach of this study have important implications which are not clearly explained.

Given the approach used by SNL, the risk of an accident occurrence is not determined on a route-specific basis, but is rather an 'aggregate' analysis because the route segments are simply frequency counted. Thus, route data are homogenized in the process of forming distributions for purposes of applying

LHS to a selected set of RADTRAN variables. Such a statistical process will intrinsically cause the results to tend toward the mean, and may miss outlier routes of both low and high risk. This approach also defeats, in part, the value of using RADTRAN in the first place. As discussed in the SNL report, RADTRAN was developed to perform route-specific analyses for the transportation of radioactive materials, i.e., to advance the risk analysis art beyond simple statistical arguments.

It is not clear how the magnitude of consequence is calculated in the aggregate analysis approach of this study. The reader is told the magnitude is calculated using a transportation consequence code, like RADTRAN and is a strong function of accident source term, concurrent meteorology, affected population, and emergency response. It would be instructive to include a clear description of the RADTRAN calculus for magnitude of consequence or example algebraic combinations of input parameters or variables. Accepting the statement at face value that RADTRAN appropriately calculates the consequence is not satisfactory.

Of particular interest to an analyst is how the consequence calculation is related to the link types and probability of accident occurrence. The RADTRAN output in Appendix E indicates that the risk is calculated for each type of link (rural, suburban, and urban), and thus one may assume these risks are added for a particular route. It is not clear how the probability of an accident is encoded or related to its consequences by the RADTRAN calculation. Table 7.31 provides the relation among accident cases between severity fraction, a factor in the probability calculus, and release fractions, a factor in the source term. This leads one to speculate that the RADTRAN calculation loops over accident cases for each link type and then loops or sums risks over link types. The report presentation should not leave room for speculation.

RADTRAN INPUT Radiation Levels (Section 3.0)

Our previously stated concerns over inappropriate transport indexes (TIs) in Tables 3.10 and 3.11 of the SNL radiation assessment have been partially addressed. Rather than directly using the radiation levels in the referenced ORNL report, which were calculated for heavily-shielded casks designed for very short-cooled fuel, new TIs were estimated using the equations in RADTRAN and an assumed radiation level of 10 mrem/h at two meters from the surface of the spent fuel cask. (The regulatory limit is 10 mrem/h at two meters from the side of the vehicle, and current casks are typically designed so that the design-basis fuel results in a radiation level slightly lower than this limit.)

Although the selection of five-year-cooled fuel as the design basis for truck casks is probably reasonable, the use of three-year-cooled fuel as the design basis for rail casks does not appear realistic and results in essentially the same issue raised in our earlier comments. Current practice is to base the cask design on five- or even ten-year-cooled fuel (depending on burnup) in order to reduce the required shielding (and weight) and to decrease the heat load to a manageable level.

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Consequently, the assumption that the cask meets the regulatory radiation limits for three-year-cooled fuel is nonconservative in estimating external radiation levels. In general, the TIs in Table 3.11 should shifted down one or two rows to match the applicable cooling time, or, alternatively, the analysis should use the same fuel-assembly/cooling-time distribution for both the rail and truck casks.

Even though a TI of 13 for the design basis fuel appears to be somewhat low, this seems to be an artifact of how RADTRAN calculates radiation levels as a function of distance from the cask. Because the exposures used in the study are really based on an assumed level of 10 mrem/h at two meters from the cask, this appears not to affect the results. Similarly, the effect of using a radiation level at two meters from the cask rather than at two meters from the vehicle is probably small since the actual levels will not be exactly at the regulatory limit.

No change bars were used in the revised draft, and no attempt was made to review material not addressed in initial comments. It was noticed, however, that additional information on neutron emission from spontaneous fission was added to Section 3.3.3.1, and this information was apparently used in the loss-of-shielding analysis in Section 9. The statement that Cf-252 is the only nuclide in spent fuel of sufficient abundance to constitute a neutron source is not correct. For example, for typical cooling times and burnups of current fuel, the neutron source (n/s) from Cm-244 is many orders of magnitude larger. Furthermore, even though the neutron emission rate noted in the report for Cf-252 appears to be approximately 500 times larger than its actual value (perhaps due to incorrect units), the rate calculated is not sufficient to conclude that Cf-252 is the largest source of spontaneous fission neutrons.

Generic Casks (Section 4.0)

Our previously stated concerns regarding the selection of specific casks and materials of construction have been satisfactorily resolved. The discussions on the use of the face seal design and the differentiation between storage and transport cask have been added as requested.

Structures/Closure/Fuel (Section 5.0)

In the SNL Study (accident analysis) the local calculated deformation in the bolt region will not be correct using the model described (p. 5-5). Bolt failure is determined by percent strain in the bolt shank, which is modeled by a single element (per Fig. 5-4 p. 5-5). Because the shank is a single element the boundary conditions which tie the shank to the lid and base will over-constrain the boundary of the bolt. This results in an incorrect calculation of the true strain and in an underestimation of the opening dimensions (p. 5-9). The model described is far too simple to provide the openings listed in Table 5.7 with the accuracy predicted (p. 5-12); hence the leakage rate will likely be underestimated for impacts above 45 mph.

The equivalent velocity method used (p. 5-16) is described as follows: "For an impact onto a real target to be as damaging to the cask as the impact onto the rigid target, the target must be able to impart a force equal to this peak force to the cask." Because the objective of the study is to provide a realistic estimate of risk why is an equivalent velocity method used? It seems more reasonable to calculate impacts between real casks and real surfaces. This capability currently exists and should be used rather than an approximate equivalent velocity method, which likely underestimates the damage.

The percentage of fuel rods damaged for each impact is estimated based on the peak rigid-body acceleration and uses an average failure strain of 4%. The average failure strain of 4% is justified by referencing a personal communication and the STACE report [2]. The justification is weak considering the importance in using the 4% strain. In reviewing the STACE report and other fuel failure [3,4] reports it appears that the 2% average strain failure is more reasonable. The fuel failure rates and associated risks are significantly underestimated in the SNL Study when the 4% strain criterion is used.

Thermal (Section 6.0)

Casks are not designed to transport three year cooled fuel with 60 GWD/MT burnup because the radiation levels and heat load would be too high to allow efficient packaging. The argument that the cask inner wall temperature can be used as a surrogate for average spent fuel rod temperatures (p. 6-6) is not correct. The average temperatures of the spent fuel rods will be significantly higher than the cask inner wall temperature. A rail cask design to transport 24 fuel assemblies with 2.8 kW heat load per assembly as described in the report is not realistic because some fuel rods would likely fail during normal transport conditions due to overheating.

A brief analysis performed at LLNL using a 24 bundle rail cask (2.8 kW assembly) indicates that the 4 inner assemblies in a rail cask would likely undergo rod burst due to creep rupture within a few days following loading. The brief analysis is summarized in the following paragraphs.

The temperature of the fuel rods is estimated using the inner wall temperature of 215°C (Table 6.4). The temperature increase from the cask inner wall to the central four assemblies is estimated using reference 5 (p.136-140). The temperature increase to the four fuel assemblies with 852 watt average heat load for all assemblies in cask is 210-90 =120°C. For an increased heat load from 852 W to 2796 W (Table 6.3) the temperature increase is 120 x 2796/952 = 393°C. However, the higher temperatures for the hotter assemblies will increase the effective thermal conductivity by ~ 4/3 resulting in an estimated temperature for the four center assemblies T = 393 (3/4) + 215 = 510°C (950°F).

At the end of life the pressure in a fuel rod is near but below the PWR operating pressure of 2000 psi and is estimated to be 1600-1800 psi at 600°F. The rod pressure at 950°F will be ~ 2000 psi. As shown in Figure 1, the rod will likely rupture after 100 hours of creep at 950°F [6].

The SNL Study significantly underestimates the temperatures of the fuel rods, hence their time of failure, radioactive material releases, and the associated risks. As recommended at the July 29 meeting, the fuel burnup/cooling times should be specified such that the fuel assembly heat load is ≤ 1 kW so that the fuel rod temperatures are more realistic and the fuel failure rates are more reasonable.

Containment/Release (Section 7.0)

Releasable Source Term

The analysis of the "volatile portion" of the releasable source term (the compounds in the releasable source term considered volatile under accident conditions) is not well supported. For an accident without a fire, some of the condensable vapors will condense on the relatively cooler cask interior walls; however, an analysis that includes the vaporization temperatures of the various condensable vapors and the cask wall temperature was not included.

The Sandia Study assumes that the settling of particles is fast relative to the time for depressurization. The assertion that the settling of particles greatly reduces the particle contribution to the releasable source term and the amount of particles released from the cask seems to be based on the assumption that the cavity environment is stagnant. The particles are a major driver in the calculation of the effective A_2 of the releasable source term. Neglecting much of the particulate material in the releasable source term, the calculation of the effective A_2 of the releasable source term needs to be better supported.

The assumption that larger particles will congregate near the opening of a breached rod cladding and effectively filter out much of the smaller respirable particles that might be in the fill gas escaping the breached rod does not seem applicable for an accident analysis. In an accident the breach in a spent fuel rod cladding may be much larger than a pin hole leak or a hairline crack. In an accident, a spent fuel rod could be fractured and/or fragmented, which would allow little if any filtering of fuel fine particles by other fuel fine particles.

Leakage

For a 1 mm^2 leak hole in a containment vessel initially at 5 atm with helium fill gas at an average temperature of 640° K, our analysis indicates that it takes about 30 minutes for the cask to depressurize to near ambient pressure. This result agrees with the MELCORE calculations given in the report for the time to depressurize a rail cask. The initial leakage rate for this depressurization is over 800 cc/s. For comparison, ANSI N14.5 considers a leak of 0.1 cc/s to be a large leak.

We also performed analyses on the time to depressurize a rail cask for leakage holes with cross-sectional areas of 0.1 mm^2 and 0.01 mm^2. When the leak hole area is 0.1 mm^2 the cask takes less than 9 hours to depressurize, and when the leak hole area is 0.01 mm^2 the cask takes less than 4 days to depressurize. (See Figure 2.) The leakage rates for these two cases are 80 cc/s and 8 cc/s, respectively which are significantly higher than .1 cc/s.

Release

An analysis was performed to determine the number of A_2's released from a rail cask holding 24 spent fuel assemblies for various leakage hole diameters. From NUREG/CR-6487[6], the releasable source term for a rail cask holding 24 spent fuel PWR assemblies would have a total activity of 28,982 Ci. With an effective A_2 for the releasable source term for PWR spent fuel under hypothetical accident conditions of 27.9 Ci, the releasable source term in the cask is equivalent to about 1040 A_2's. Using the following parameters, the number of A_2's released, as a function of time for a rail cask is calculated for various leakage hole diameters: T=640 K, Molecular weight of fill gas = 4 g/gmol, viscosity of fill gas = 0.031 cP, volume = 2.68×10^6 cc, and leakage path length = 10 cm. The results for the number of A_2's released as a function of time are shown in Figure 3. From Figure 3, it is clear that for the case when the leakage hole has an area of 1 mm^2, there are almost 400 A_2's released in less than 17 minutes. For hypothetical accident conditions, it is useful to determine the time that is takes to release one A_2 of releasable material. Figure 4 shows the relevant region of Figure 3 to make this determination. From Figure 4, we see that the cask with a leak hole of 1 mm^2 an A_2 of materials is released in less than 10 seconds. For a cask with a leakage hole with an area of 0.1 mm^2 it takes less than 100 seconds to release an A_2, and for the cask with a leakage hole with an area of 0.01 mm^2 it takes less than 1000 seconds (17 minutes).

From these analyses it is clear that the hole sizes less than 1 mm should be evaluated for leakage flows and radioactive material releases or the radioactive material releases and associate risks will be significantly underestimated.

Comments on Results (Section 8.0)

The primary interests of the reader regarding the RADTRAN calculations are what routes were assessed, what are the expected population doses, and what is the interpretation of the results. With respect to the set of representative routes selected by LHS, the actual set of parameters should be included, perhaps as an appendix, in both distributional and tabular forms. This would make the report self-contained in terms of the information needed to reproduce or compare the calculations in the future. Also, the discussion on the sensitivity of the LHS sample size is unusual. A convergence or a stability criterion usually determines the size of a Monte Carlo sample, and such a criterion is not specified in this study. In addition, in LHS techniques, the relation between the number of equal probability sections/areas and the sample number should be explicitly stated.

The SNL report compares incident-free doses to those calculated in the NUREG-0170 study (Section 8.9). The new incident-free doses are presented for RADTRAN 5 calculations that examined PWR spent fuel in a steel-lead-steel spent fuel cask and that used the representative set of 200 truck or 200 rail routes generated by LHS.

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The SNL incident-free doses were limited to those doses incurred while en route; and do not include the storage and handler doses that were included in the original NUREG-0170 study.

On a per shipment basis, the sum of the en route incident-free doses developed by the SNL study for spent fuel transport by truck is about 3 times larger than the sum of the corresponding NUREG-0170 truck doses. Even this number may be an underestimation for routes of most concern to the public, as the average population density over the entire NUREG-0170 actual truck route exceeds the average population density of the set of representative 200 truck routes used by this study by about a factor of 2.5. The sum of this study's en route incident free doses for transport by rail, on a per shipment basis, is about two-thirds of the sum of the corresponding NUREG-0170 rail doses. Both studies assume that the spent fuel transportation casks' surface dose rates are below but near the regulatory limit.

On a campaign basis, the shipment of the 1994 spent fuel inventory (over 30 years) leads to average yearly population doses for transport by truck and rail that are, respectively, 4.5 times larger and nearly identical to the NUREG-0170 estimate for 1985. It should be noted that the calculated radiation levels were not done correctly which results in lower estimated doses, especially for the rail cask.

Since the mean population dose is important to the public, the section should provide a summary description of the nature of the dose calculated, e.g., is it whole body dose or CEDE, what are the exposure modalities (pathways and timeframes). It might also provide a sample calculation in order to demonstrate the dose algorithm. Section 9.1 does address some noteworthy dose considerations, but the comments are rather ad hoc.

The relatively small impact in Calculation Number 21 of Table 8.15 of the inclusion of groundshine (a factor of about 30) over the representative route set is noteworthy. In Section 8.13.2 of the report, a comparison between a (normalized) calculation using RADTRAN 1 vs. RADTRAN 4/5 of the same calculation case (Number 21), but using the original NUREG-0170 truck routes, show that groundshine increases the LCFs by a factor of 700 (Table 8.14), although without groundshine RADTRAN 4/5 agrees with RADTRAN 1 (within a factor of 3.3) over the same route set. This may cause one to speculate that the risk (i.e., effect of groundshine) is greater for actual routes than for the statistical routes of the so-called representative set, or reinforces the notion that the statistical approach will make results tend towards the mean.

Also, in several of the CCDFs, the 'shin' of the mean curve is comparable to or greater than the 95th percentile curve, implying there are cases with extreme consequences. Such extreme consequences are of interest, but the statistical approach taken in this study will not highlight those cases.

Although not explicitly stated until Section 9.1.4, the dose calculation subsumes a 50-year dose from groundshine. NRC should consider whether it really wants to be subjected to accounting for radiation in groundshine for 50 years.

Groundshine is certainly important in the short term after an accident until people are segregated from a contaminated area or until the contamination is cleaned up below (EPA) regulatory limits. Once segregation or regulatory limits are achieved, there should not be a penalty for accounting for residual radioactivity below general use maximum contamination levels (MCL's). It is not realistic that persons be allowed to be exposed to above-regulatory radioactive contamination limits for 50 years.

The SNL Study contains a comparison of the NUREG-0170 and Modal Study calculations with the SNL Study (Sec. 8.14). Each of the calculations for the NUREG-0170 Model II and Modal Study source terms examined transport of PWR spent fuel in a steel-lead-steel spent fuel cask and used the LHS sample of 200 truck or rail routes. The NUREG-0170 calculation used only inhalation pathways (compared to all exposure pathways in Fig. 8.24), while for the Modal Study source terms all exposure pathways (i.e., including groundshine) were utilized. The Modal Study results for inhalation pathways only (i.e., no groundshine) are not provided.

The results are summarized in Table 8.17 and Figures 8.25 and 8.26. Under these circumstances, the maximum consequences for the Modal Study are higher than the other studies (NUREG-0170 Model II and SNL Study) as are the mean accident population dose risks (see Table 8.17). By contrast, the SNL Study has comparable maximum consequences to the Modal Study (as 'explained' by the last paragraph on page 8-60), but the expected risk for the SNL Study is about 3 orders of magnitude smaller than that for the Modal Study.

It would be interesting to know the Modal Study result without groundshine. Using the comparison of NUREG-0170 Model II with and without groundshine to provide a factor of about 30 (see Table 8.15 and Figure 8.24) as a guideline, one might expect the maximum Modal Study consequence to be similar to that for NUREG-0170 Model II, while the Modal Study expected risk may be an order of magnitude lower. It is hard to draw any conclusions without a consistent basis for comparison.

As the report points out, the decreasing risks estimated from the original NUREG-0170 study, through the Modal Study, to this latest study correlate with an evolution in sophistication of analysis of the release of radionuclides during putative transportation accidents. Because both Model I and Model II in NUREG-0170 assumed the spent fuel casks might fail when subjected to the loads that characterize minor accidents, the fraction of all truck and train accidents estimated to lead to cask failure by these models is very large and extremely conservative. When, as was done by the Modal Study, cask failure and thus 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 and, consequently, mean accident population dose risks decrease by one or two orders of magnitude. In this 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 due to heating by fires, such that radionuclide releases decrease further, thereby leading to a reduction in mean accident population dose by a factor of 100-1000 times lower than the Modal study.

It must be emphasized that this lowering of risk estimates with more sophisticated analyses does not necessarily mean one has a better handle on the risks, and that they are necessarily lower than previously thought. The result may be an artifact of the number of subdivisions of phenomena for which independent probabilistic values are obtained and then compounded. To avoid this inherent pathology, it is important to differentiate between independent, dependent, and common mode failures.

The Sandia Study results depend heavily on the severity and release fractions developed in this report. Unfortunately, these developments are unsatisfactory in the respect that there does not appear to be any consistent or balanced treatment of physical phenomena (causal or otherwise). For instance, the issue of seal failure is settled by a simplifying assumption while the deposition of vapors onto the cask interior surfaces is treated apparently in detail beyond the plausibility of MELCOR, and both approaches support the conclusion of limited radionuclide releases. Furthermore, there are no stated criteria or methodological framework for deciding what level of phenomena should be studied and why, or on how to maintain causal or intellectual consistency.

Section 9 Special Topics

Our previous comment questioning the rationale for discussing certain topics in this chapter has been addressed in general terms by a brief explanation in the introduction. The decision on the appropriateness and usefulness of this chapter and its topics should be left to NRC.

An evaluation of loss-of-shielding accidents has been added. The details of the method and its overall effect on the consequences are rather difficult to follow. A more straightforward approach might have been to use the results of the structural/thermal evaluations and calculate the radiation levels directly. In addition to increased radiation levels from lead slump and localized melting/loss of lead shielding, which were addressed in the report, spent fuel casks typically have an external neutron shield that is susceptible to both structural and fire damage over a large portion of the cask surface area. This can result in a substantial increase in neutron radiation and a modest increase in gamma radiation in the vicinity of the cask, neither of which are addressed. As noted in comments on Section 3, the data on spontaneous fission neutrons from spent fuel was also applied incorrectly.

The discussion on RADTRAN's treatment of population migration in Section 9.1.1 is not clear. However, the main point of this section appears to be that an average individual dose cannot be determined by dividing the population dose for the entire campaign by the total population because 80% of the population moves every three years. In Section 9.1.4, the rationale for excluding ingestion dose during a 50-year period appears to based on the fact that the population exposed to such dose would not necessarily be the same as those exposed to other doses and that the linear-threshold assumption would be overly conservative. Although both of these reasons are valid considerations, they are also applicable to groundshine. Consequently, the rationale for considering the long-term effect of groundshine but neglecting that of ingestion is not clear.

Summary and Conclusions

Eight major concerns are summarized as follows:

1. The incident-free dose is directly proportional to the TI input used in RADTRAN. Although improvements have been implemented in the revised draft, the revised assumptions still underestimate radiation doses, especially for rail casks. In addition, the assumption that Cf-252 is the only significant neutron emitter in spent fuel is not correct. The neutron source from the Cm-244 is many orders of magnitude larger.

2. The choice of representative routes selected for the study appears to have no logical justification. After presenting a rationale for selecting 474 specific routes based on the current distribution of spent fuel and hypothetical interim and final storage sites, another 274 routes are mixed with this sample, apparently only because data on these additional routes were readily available. The length of a route and its population density are significant variables in the risk analysis. If this study is intended to look at national averages and extremes, other methods appear much more suitable than Latin Hypercube Sampling (LHS). If it is to be based on representative spent-fuel-shipment routes, sampling from a mixture of unrelated routes appears misleading. If the current approach has merit, its justification should be explained in more detail.

3. The NRC should consider whether it really wants to be subjected to accounting for radiation in groundshine for 50 years. Groundshine is certainly important in the short term after an accident until people are segregated from a contaminated patch or until the contamination is cleaned up below (EPA) regulatory limits. Once segregation or regulatory limits are achieved, there should not be a penalty for accounting for residual radioactivity below general use maximum contamination levels. It appears that if this groundshine is included it inappropriately increases the consequence by a factor of 30.

4. Although many values for RADTRAN input are specifically identified, the lack of information on input values selected by the LHS routine (which are by definition important) inhibits an interested party from reproducing the study

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results. Furthermore, for the reader unfamiliar with the details of RADTRAN, the lack of explanation on how the input is used is likely to increase doubt on the credibility of the calculations. Sufficient information should be included in the main report and appendices to enable a knowledgeable reader to reproduce the results.

5. The radiological consequence of an accident is essentially proportional to the release of radionuclides from the cask. The explanation of radionuclides available for release and the quantity released under various accident conditions is lengthy and unclear. The one element bolt model on the closure will not correctly calculate the opening dimensions or leakage area. Also, leakage areas much less than 1 mm^2 (down to .01 mm^2) can release a significant amount of radioactive material.

6. The shielding and thermal analyses are not correct for the representative cask, especially the rail cask. Both the external radiation levels and the fuel rod temperature are too low; hence the radiological consequence and risks are underestimated for normal transport and accident conditions.

7. The Sandia Study still uses a 4% fuel rod failure strain criterion. We discussed this criterion at our July 29 meeting and thought a 2% failure strain was to be used as an average. An average 4% failure strain appears to not be realistic for fuel burnups from 16 GWD/MT to 60 GWD/MT based on the information documented in references 3 and 4. Using the 4% failure strain results in underestimating the radiological consequences and risks for transport accident conditions.

8. The study presents a lengthy hypothesis and analysis of selected phenomena (i.e., release of nuclides from damaged fuel rods) while for other equally important phenomena it appears to assume simplistic conclusions (i.e., size of seal failure and resultant release). Either the level of detail should be more appropriately balanced or additional justification should be presented for the assumptions used in this report.

The updated NUREG-0170 report concludes that the calculated incident-free truck dose value is 3 times higher than that calculated in the original NUREG-0170 primarily because of the higher estimated (based on experience) truck stop dose (p. 8-23). It also concludes that the calculated incident-free rail dose value to be 1.6 times lower than the original NUREG-0170 one. However, we think that the estimated incident-free doses in the updated NUREG-0170 are low because the radiation levels were not correctly calculated for the representative casks as noted in concern 1. Also concern 2 could impact the incident-free doses.

The updated NUREG-0170 concludes that calculated accident consequences and risks of the Modal Study are higher than the original NUREG-0170 by factors of 30 and 1.2 respectively. The original NUREG-0170 estimate did not

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include 50-year groundshine doses whereas they were included in the Modal Study calculation. We think that the inclusion of groundshine is overly conservative and not reasonable as noted in concern 4. The Sandia Study (accident analysis) significantly underestimates the consequences and risks except for the inclusion of 50-year groundshine because of concerns 5, 6 and 7.

Overall the report lacks clarity, robustness and communicability to the public.

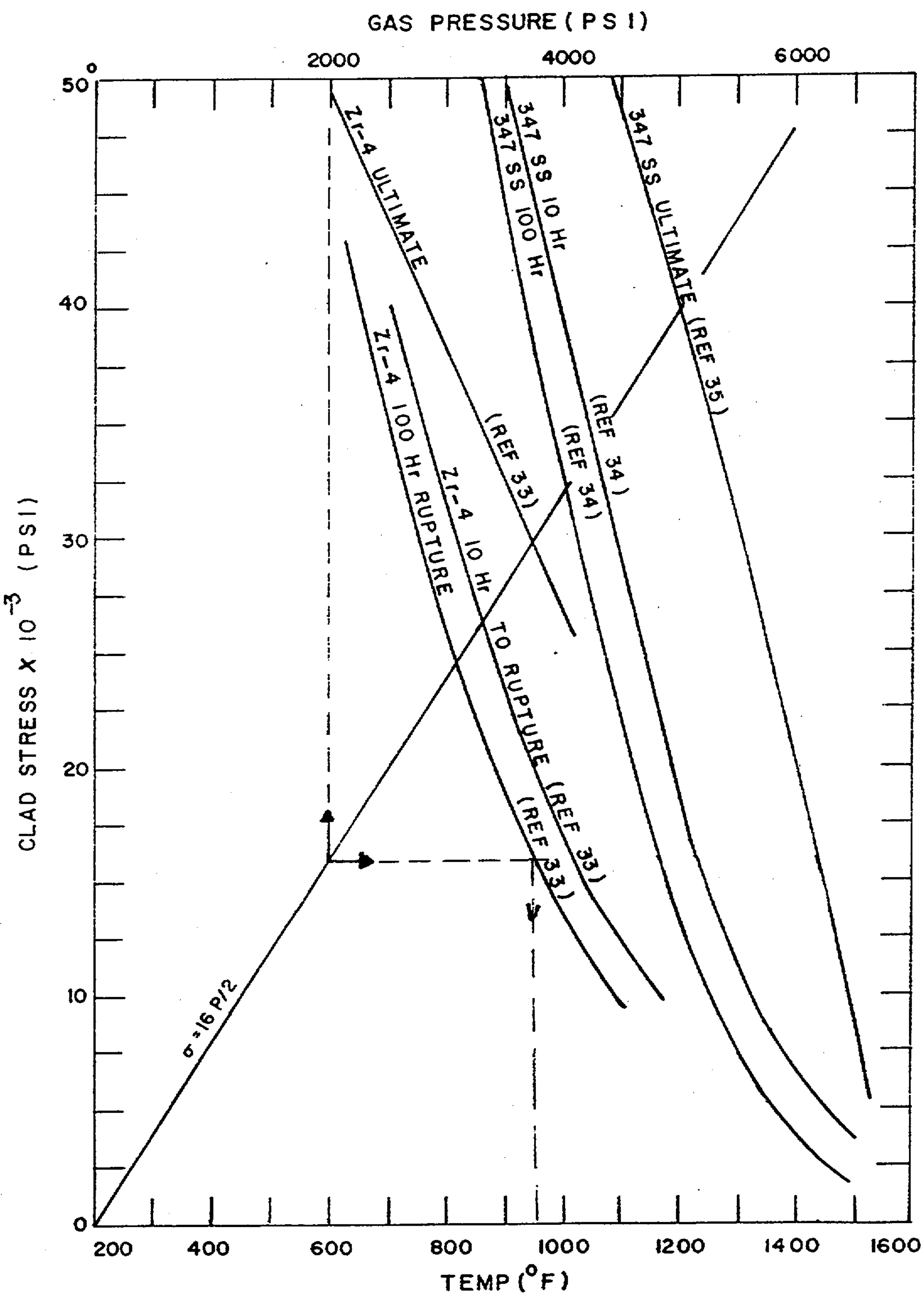

Figure 1 Creep-Rupture Data for Stainless Steel and Zircaloy Tubing.

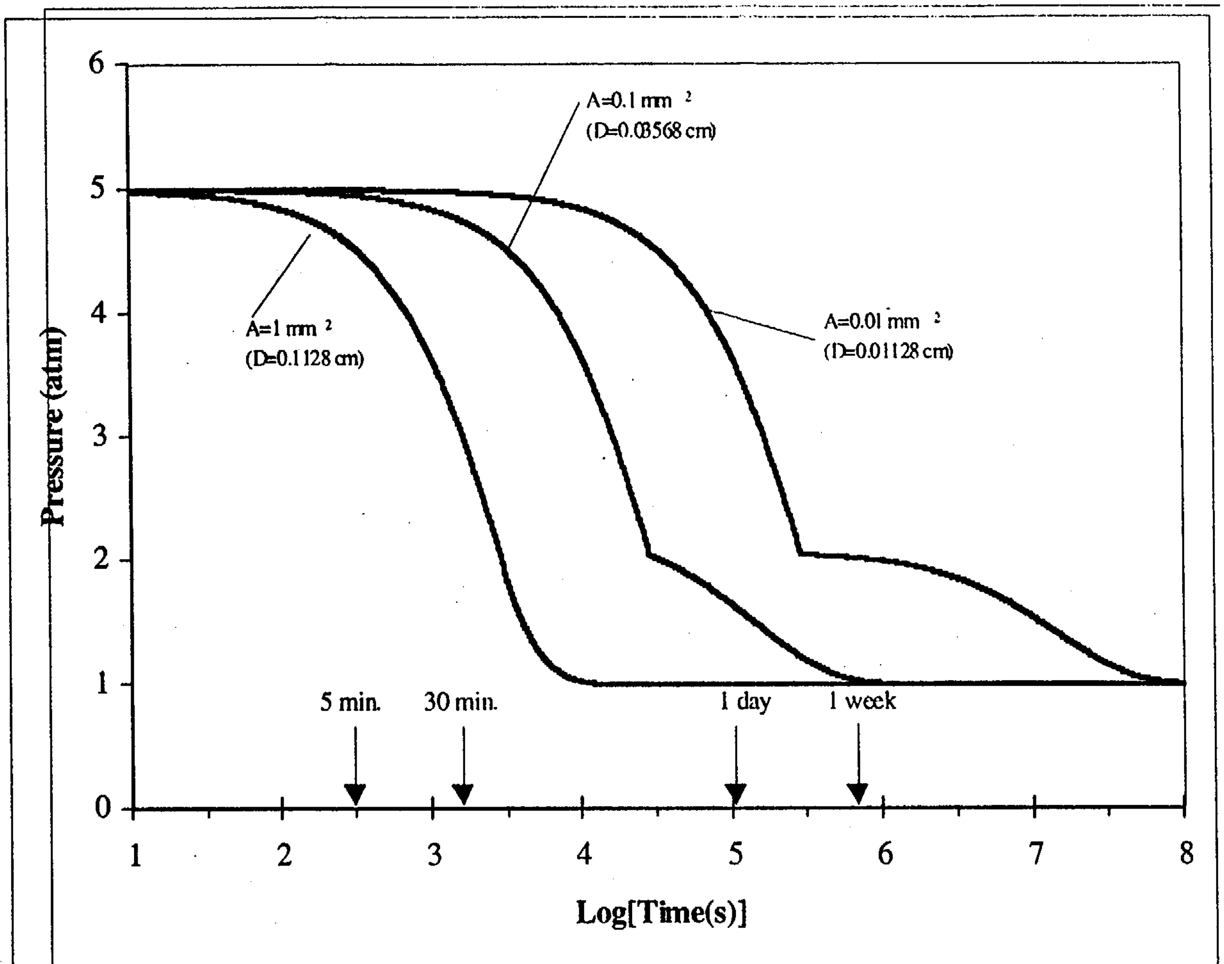

Figure 2. Pressure vs. Log[Time]

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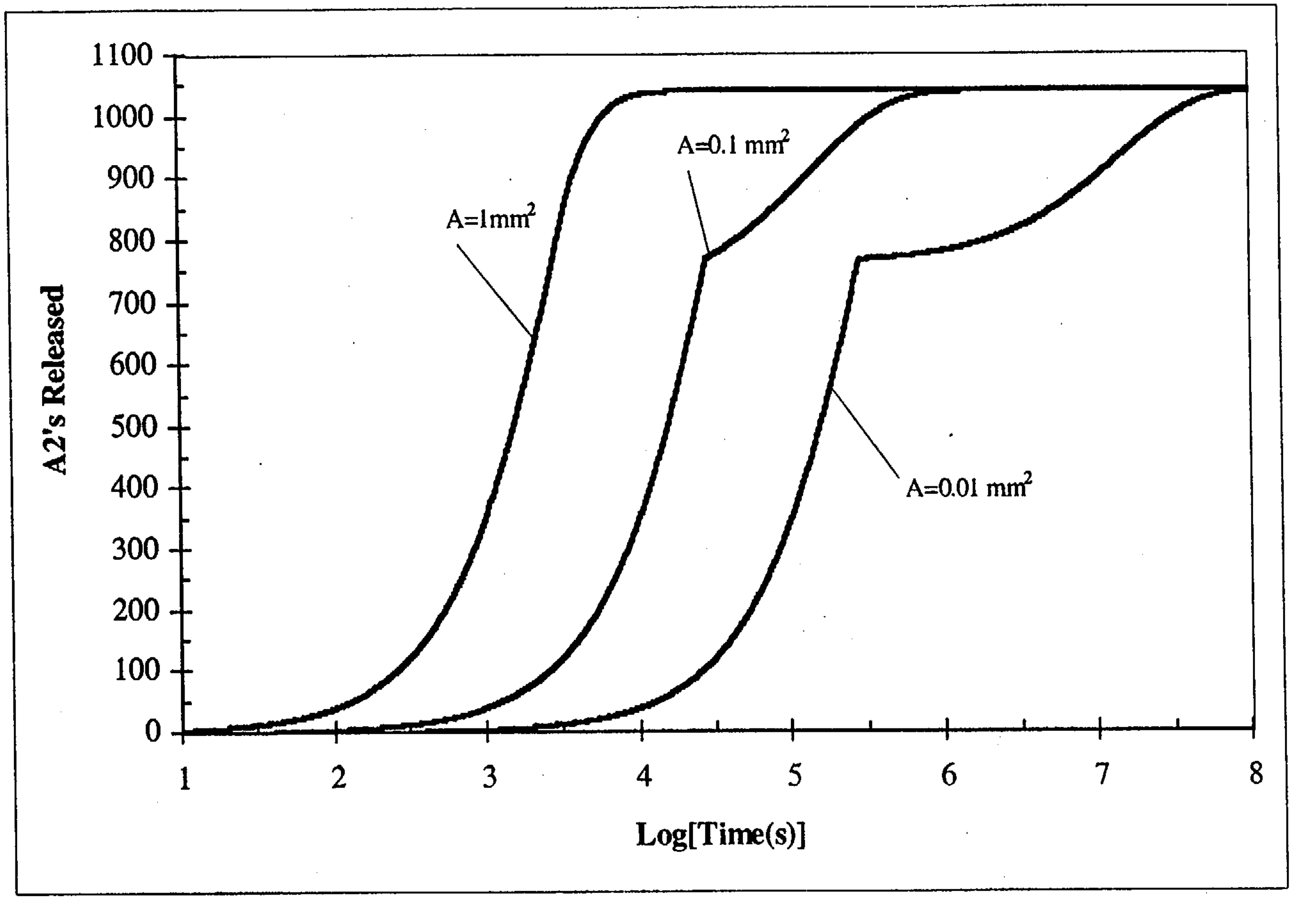

Figure 3. A_2's Released vs. Log[Time]

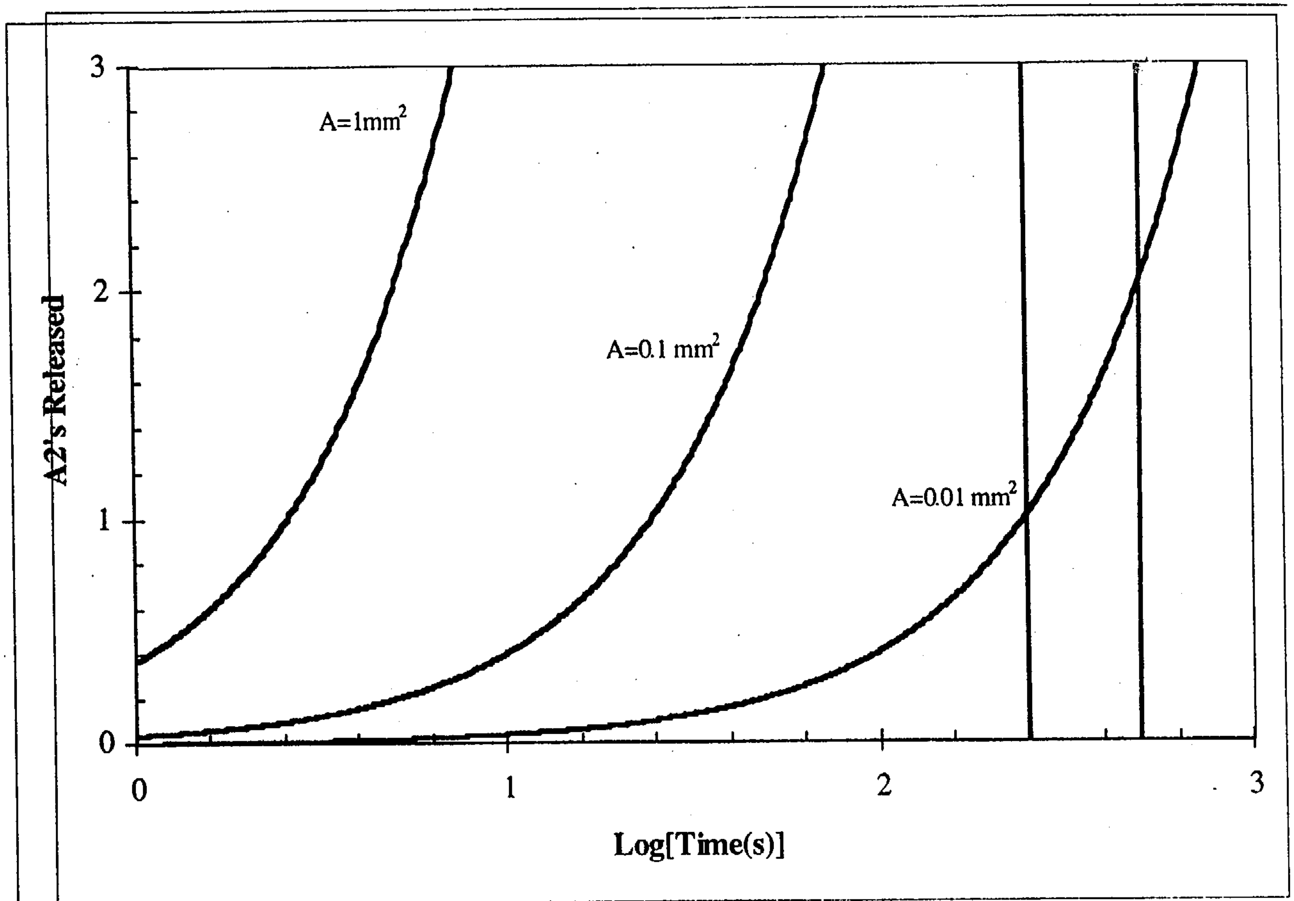

Figure 4. A_2's Released (up to 3 A2's) vs. Log[Time]

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Initial Comments

Appendix A