Review of Criticality Safety and Shielding Analysis Issues for Transportation Packages*

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INTRODUCTION

The staff of the Nuclear Engineering Applications Section (NEAS) at Oak Ridge National Laboratory (ORNL) have been involved for over 25 years with the development and application of computational tools for use in analyzing the criticality safety and shielding features of transportation packages carrying radioactive material (RAM). The majority of the computational tools developed by ORNL/NEAS have been included within the SCALE modular code system (SCALE 1995). This code system has been used throughout the world for the evaluation of nuclear facility and package designs. With this development and application experience as a basis, this paper will present a perspective on important issues related to nuclear safety analyses for a package design.

CRITICALITY SAFETY ANALYSES

Transportation packages that contain fissile material in amounts or configurations that will not allow a fissile exception must be analyzed to ensure that the package will remain sufficiently subcritical under normal and accident conditions specified by the regulations. Monte Carlo codes operating on modern, high-performance personal computers and workstations provide package designers and technical reviewers with efficient tools for calculating the neutron multiplication factor, k_{eff}, for a wide range of packaging conditions and contingencies. However, in recent years these tools and the methods applied by criticality safety analysts have become a more important part of the design process as package development moves towards increased fissile material payload and/or provides for added diversity of material form and content. Such design criteria cause the margin between the established subcritical limit and the package design basis analyses to become diminishingly small and can alter the traditional design basis such that a more rigorous

*This work was performed by Oak Ridge National Laboratory, managed by Lockheed Martin Energy Research Corp. for the U.S. Department of Energy under contract number DE-AC05-96OR22464.

evaluation and analysis of the package, its contents, and potential contingencies are required. Designs that approach the subcritical limit and changes to the traditional design basis nearly always lead to added scrutiny in the review process and challenge the designer to ensure (1) an acceptable subcritical limit has been established and (2) all credible package configuration and conditions have been bound by the safety analyses.

The k_{eff} of the package can be maintained by ensuring adequate control of the parameters that affect the neutron balance. These parameters are the (1) type, mass, and form of the fissile material; (2) moderator-to-fissile material ratio (degree of moderation); (3) amount and distribution of absorber materials; (4) internal and external package geometry; and (5) reflector effectiveness (both package and external). For any particular package design, it is important not only to **identify** these parameters, but to **understand the interrelationship** of the parameters and their effects on k_{eff}. The criticality safety analyst must ensure that all credible package conditions have been explored and that unknown or uncertain parameter conditions have been assumed such that the maximum k_{eff} value is obtained.

The use of burnup credit in a spent-fuel packaging design is an example of a strategy that seeks to change the traditional design basis (i.e., fresh-fuel assumption) by providing a bounding specification of the fuel isotopics that is closer to reality than the fresh-fuel assumption. Many of the analysis issues related to specifying spent-fuel isotopics that yield a bounding k_{eff} value have been identified and discussed in several papers (DeHart and Parks 1995). Strategies that demand a more rigorous analysis of the actual package contents or configuration can provide a challenge to the criticality safety analyst and may be limited by the computational capabilities or nuclear data that are available and validated. Again, burnup credit provides an example of a challenging computational issue in the effort that has been expended to demonstrate that Monte Carlo codes can provide an accurate, converged solution when applying a realistic axial profile of the spent-fuel isotopics (DeHart and Parks 1995).

Packages are now often designed to have predicted k_{eff} values that are just under the accepted subcritical limit. A subcritical limit for a package should be established by an effective consideration of the bias (average deviation between the calculation and measurement) and uncertainty (derived from the spread in the calculational results) of the computational method observed from the analysis of critical experiments as well as an added margin of safety that is intended for assurance of subcriticality. Given that sufficient exper-iments are analyzed, the bias and uncertainty can be estimated using established statistical techniques. Although methods for estimating a safety margin have been proposed, the safety margin remains a value that should be set by the consideration of such issues as the number and applicability of the critical experiments used in the validation and the experience of the industry with the package design and contents. The safety margin should never be less than 2% in k_{eff}, and the typical value used in transportation packaging is 5% (thus leading to the traditional design limit of $k_{eff} = 0.95$).

The ANSI/ANS-8.1 Standard calls for validation of the analysis methods and data by comparison against pertinent critical experiments that have the same characteristics as the system being evaluated. When the characteristics of the packaging system move beyond the parameter range of the available experiments, the standard indicates that the validation can be extended by observing trends in the bias and uncertainty with system parameters and by using independent calculational methods. Unfortunately, the Standard provides no guidance on limiting the extensions of the bias and uncertainty, and, as scientists continue to discover, extrapolation can often lead to a poor prediction of actual behavior. Even interpolation over large ranges with no experimental data can be misleading. An example of this latter situation has been demonstrated in Figure 1, where a selected cross-section library is shown to perform well for high (fast) and low (thermal) neutron energy systems but has a substantial bias at intermediate energies where the ^{235}U resonance data (missing in this library) become important (Parks et al. 1995). Comparison with other computational methods may illuminate the deficiency with a data library or code; however, given discrepant results from independent methods, it is not always a simple matter to determine which result is "correct" in the absence of experimental data.

Unfortunately, the demand for the shipment of fissile materials that have design basis analyses beyond the area where experiments are available is an established fact. Examples of situations where little measured data exist include low-enriched uranium with greater than 5 wt % ^{235}U, low-moderation conditions, and large arrays of interacting units. In the absence of sufficient experimental data, the criticality safety analyst must attempt to (1) use rigorous and defensible analysis methods and data that provide an accurate representation of the physics of neutron interaction, (2) seek an understanding of the parameters that can or may affect the k_{eff} of the packaging system, and (3) apply independent methods and data for corroboration and added insight into the physics of the packaging system. Analysts must always remember that computational methods are tools that aid the criticality assessment. Methods developers must strive to ensure that these tools provide accurate and meaningful information which can help the analyst effectively interpret the physics of the packaging system as well as defend the validity of the results.

RADIATION SHIELDING ANALYSES

A complete radiation shielding analysis of a RAM package requires a determination of the radiation source term, a simulation of the radiation transport through the packaging, and a calculation of the dose at points exterior to the package. Depending on the package design and the intended contents, the model and associated analyses can range from very simple to very complex. At the same time that certification authorities have increased their requests for clear, defensible analyses, package designers have sought to increase the payload or source intensity of RAM. Thus the shielding analyses of RAM packages are increasing in rigor in an effort to ensure that accurate, bounding doses are obtained without being overly conservative. To ensure this result, the analyst must use adequate methods and data in both the source term and radiation transport analysis.

Cross-Section Effects

Uncertainties in cross sections can be especially evident in RAM transportation package analyses where particle attenuation of 3 to 4 orders of magnitude are common. In such cases, errors of only a few percent can result in dose-rate errors of 25 to 35%. This general observation is consistent with the results of a study (Broadhead et al. 1995a) that analyzed several simple geometry benchmark shielding problems. This series of benchmarks demonstrates the attenuation of neutrons and gamma rays through various thicknesses of standard cask materials. A summary of the results as analyzed with the SCALE system is shown in Figure 2. The study indicates that other analysis methods and data yield similar error bounds in the dose results.

Analyses involving spent nuclear fuel can be particularly difficult because of the additional uncertainties arising from the characterization of the spent-fuel source. The source magnitude, its variation over time and space, and the resulting source spectrum are all key components of the shielding analysis. The characterization of actinide and fission-product nuclide concentrations, and hence the neutron/gamma sources are, in principle, highly dependent on the cross-section data. This dependency is particularly true for the neutron radiation source, which is derived mostly from actinides that are very sensitive to both cross sections and flux spectra. In practice, the total inventory of fission products is directly proportional to the total power produced, and hence becomes less sensitive to cross-section data. However, for shielding calculations, several less-abundant fission-product nuclides typically dominate the contributions to the overall dose rate (Broadhead et al. 1995b). The most important fission products are ^{144}Pr at early decay times (<2-4 years), and ^{90}Y, ^{134}Cs, ^{137}Cs, ^{106}Rh, and ^{154}Eu. Of these, only ^{137}Cs and ^{90}Y are considered high-abundance fission products. The ^{154}Eu and ^{134}Cs have very low fission yields, and their presence is largely due to neutron capture by ^{153}Eu and ^{133}Cs, making them highly dependent on cross-section data. Validation studies (DeHart et al. 1995) for the spent-fuel isotopic prediction code ORIGEN-S have recently shown that for a fairly broad range of PWR spent-fuel types, actinide and fission-product inventories of importance to criticality and shielding applications can be typically predicted to within 20%. This study shows that ^{244}Cm, the primary contributor to the neutron source, is underpredicted by 20% using ENDF/B-IV data but only 7% using ENDF/B-V data. Gamma sources such as ^{60}Co found in activated material can be difficult to estimate with this accuracy unless both the initial concentration and time-dependent flux spectrum are known.

Flux-to-Dose Conversion Factors

The fluence-to-dose conversion factors found in ANSI/ANS-6.1.1-1977 (ANSI77) are based on a dose-equivalent (DE) formalism where the absorbed dose (the mean energy absorbed by a layer of tissue per unit mass of tissue) is modified by a quality factor, Q, which is equal to 1 for gamma rays and typically varies from 1 to 10 for neutrons. These dose factors were derived from calculations of the maximum dose equivalent (MDE) in slab and cylindrical phantoms. Because of the conservative approach taken in their derivation, these dose factors were recommended and have been used for a number of years in the design of radiation shields. The ANSI/ANS-6.1.1-1991 standard (ANSI91) uses an effective dose-equivalent (EDE) formalism for the fluence-to-dose factors. This EDE formalism uses the evaluated DE in each specific organ tissue of anthropomorphic phantoms, then sums all organs together with a specified weight function to obtain the effective whole-body dose or EDE. This effective dose takes into account the body shielding of internal organs since the DE is evaluated within each organ. The net effect of including body shielding as compared with the conservative approach used in the ANSI77 standard is a lowering of predicted dose rates based on the EDE formalism.

Figure 3 shows the ratio of the ANSI91-to-ANSI77 dose factors. For energies below 2 MeV, the ANSI91 neutron fluence-to-dose factors predict doses a factor of 2 to 3 lower than the ANSI77 dose factors. For energies above 2 MeV, the effects range from 8 to 34% lower. For light-water-reactor (LWR) spent fuel, doses from neutrons having energies above 2 MeV should be of secondary importance. Thus if ANSI91 is used in conjunction with a possible doubling of the neutron quality factor as recommended by the International Commission on Radiological Protection (ICRP), the result would be neutron doses nearly equal to or smaller than those from ANSI77 (Broadhead et al. 1992). Without a doubling of the neutron quality factor, results based on ANSI91 only would be much lower than those based on ANSI77 for neutron dose rates. For photons, the ratios of dose factors shown in Figure 3 indicate lower predicted doses of some 7 to 40% (excluding the very bottom group) from the ANSI91 proposed set. Typical effects should be 10 to 20% because most of the dose contribution from LWR fuel is dominated by photon energies between 1 and 3 MeV.

Based on these observations, immediate utilization of ANSI91 in cask design studies is not recommended until resulting uncertainties in dose-measurement techniques and the adoption of ICRP recommendations are resolved. A potential problem that could arise as a result of using the ANSI91 dose factors is that the preshipment package dose measurements will probably measure a different physical quantity (DE) than the calculations predict (EDE). Calculations performed with ANSI91 dose factors would predict the dose with body shielding taken into account, while the measurement would not, unless the instruments have been recalibrated to obtain an EDE. The current thinking is that new, more sophisticated measurement techniques need to be developed to accurately measure the EDE defined by ICRP. Also, the immediate use of the new ANSI dose factors would make neutron dose estimates nonconservative if ICRP recommendations on doubling the neutron quality factor are subsequently adopted.

Validation Using Prototypical Cask Environments

The most difficult situation in which to quantify expected dose rates is in an actual or prototypic configuration. In such a situation all the various uncertainty contributions interact simultaneously. For this reason it is important to have high-quality measurements of actual cask configurations in order to provide a means for validating the analysis methods used in a package design.

Source generation analyses and multidimensional dose-rate analyses have been performed (Broadhead et al. 1995a) with selected computer codes in order to compare with measured doses from a number of spent-fuel metal storage casks and concrete storage cask/ modules. This work not only provides a valuable reference for the performance of the SCALE shielding analysis methods but also provides a concise description of the packages, contents, and measured data that can be used by analysts to provide a validation of their method applicable to spent-fuel transportation packages. Measurements were made using a variety of measurement techniques, with the participation of several laboratories. The individual casks that were measured and the characteristics of the fuel contents are shown in Table 1. General trends from the study indicate agreement to within 30% with neutron dose-rate measurements on the cask side, lid, and bottom; and agreement to within 30% for gamma-ray doses on the cask lid, bottom, and sides below and above the active fuel. For gamma-ray doses on the cask side corresponding to the active fuel, predictions were up to a factor of 2 higher than the measurements. This surprising result is consistent for a variety of cases and points to a potential problem in the calculated radiation source.

Table 1. Characteristics of Fuel Assemblies Loaded Into Storage Casks.

Cask	Fuel Type	Cooling Time, y	Burnup GWd/MTU	Enrichment wt %	Source of Assemblies
CASTOR-V/21	PWR	2–4	30–36	2.9–3.1	Surry 2
MC-10	PWR	4–10	24–35	1.9–3.2	Surry 2, Surry 1
TN-24P	PWR	4	29–31	2.9–3.2	Surry 2
TN-24P[a]	PWR	6–12	24–35	1.9–3.2	Surry 2, Turkey Point
VSC[a]	PWR	9–14	27–35	2.6–3.2	Surry 2, Turkey Point

[a]Consolidated fuel canisters.

Radiation Transport Recommendations/Guidelines

Many times the most geometrically complex region is that of the package cavity or basket. Simplified one-dimensional (1-D) geometries are generally accurate in the radial direction if the cavity region can be smeared without significant conservatism in the cask design. For example, advantages can be gained from the prudent placement of a cask package with strong azimuthal asymmetry. However, this scenario is much more difficult to model with simplified geometries. An alternative to a full three-dimensional (3-D) analysis is a two-dimensional (2-D) radial slice model if azimuthal variations are important or a 2-D axial slice model if axial variations (e.g., shine from the end-fitting regions) are important. Full 3-D Monte Carlo models simplify many of the modeling decisions required for other geometries, but the setup and analysis phases are usually much more complex.

Point-kernel techniques enjoy the geometric accuracy of 3-D methods, typically without the computational complexity associated with Monte Carlo methods. The major drawbacks to point-kernel techniques are the 1-D nature and multilayer effects of the buildup factors and the fact that typically only gamma-ray calculations are possible. Still these methods can be quite useful if an effective buildup factor method is used and the analyst is primarily interested only in gamma rays.

The general recommendations for radial studies also apply to axial analyses; however, the use of 1-D approximations are typically more difficult along the cask lid/bottom. Deterministic 2-D methods in RZ geometry are quite useful since in theory the lid and bottom can be modeled simultaneously. For very tall packages, such as spent-fuel casks, Monte Carlo studies must typically use an axial biasing scheme for efficiency because of the large number of mean-free paths the particles would travel from the top to the bottom of the model.

SUMMARY

This paper has highlighted a number of criticality safety and shielding analysis issues that confront the designer and reviewer of a new RAM package. Changes in the types and quantities of material that need to be shipped will keep these issues before the technical community and will provide challenges to future package design and certification.

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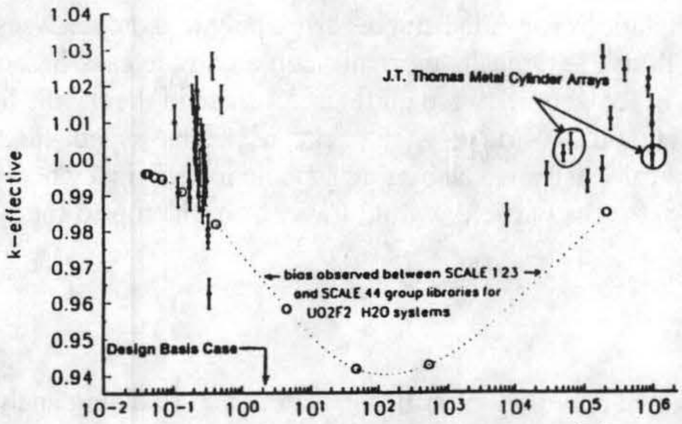

Figure 1. SCALE 123-group validation results for highly enriched uranium cases with bias observed against more accurate library.

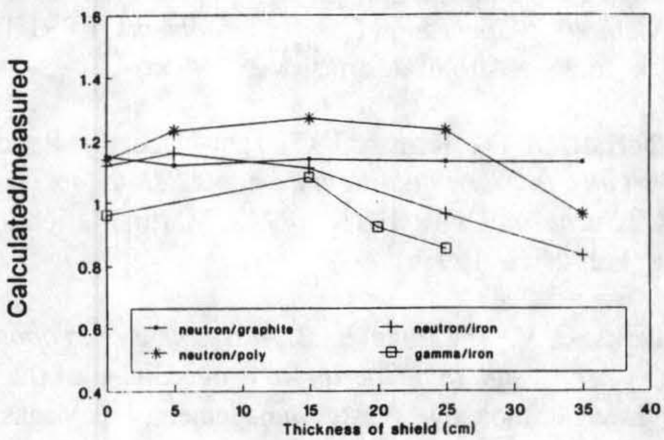

Figure 2. Calculated-vs-measured dose-rate trends for various cask-type materials.

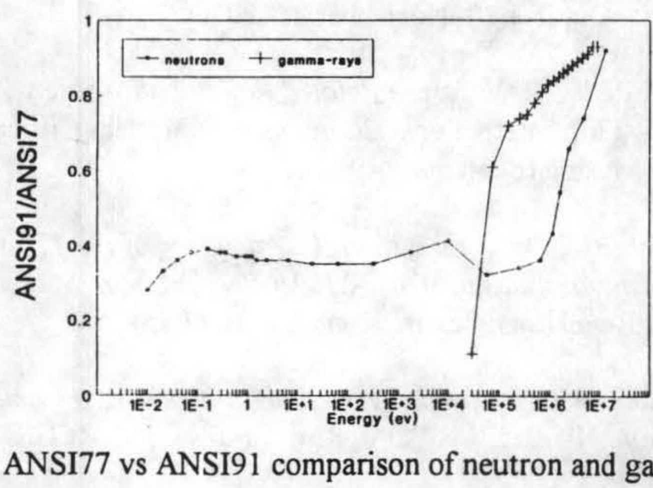

Figure 3. ANSI77 vs ANSI91 comparison of neutron and gamma-ray flux-to-dose conversion factors.