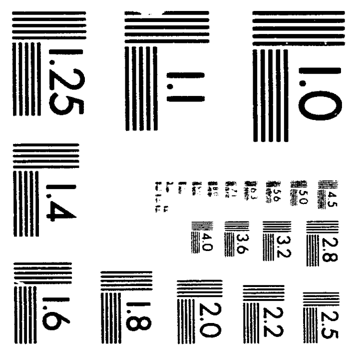

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ANSI N14.5 SOURCE TERM LICENSING OF SPENT-FUEL TRANSPORT CASK CONTAINMENT[1]

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ABSTRACT

American National Standards Institute (ANSI) Standard N14.5 states that "compliance with package containment requirements shall be demonstrated either by determination of the radioactive contents release rate or by measurement of a tracer material leakage rate." The maximum permissible leakage rate from the transport cask is equal to the maximum permissible release rate divided by the time-averaged volumetric concentration of suspended radioactivity within the cask. The development of source term methodologies at Sandia National Laboratories (SNL) provides a means to determine the releasable radionuclide concentrations within spent-fuel transport casks by estimating the probability of cladding breach, quantifying the amount of radioactive material released into the cask interior from the breached fuel rods, and quantifying the amount of radioactive material within the cask due to other sources. These methodologies are implemented in the Source Term Analyses for Containment Evaluations (STACE) software. In this paper, the maximum permissible leakage rates for the normal and hypothetical accident transport conditions defined by 10 CFR 71 are estimated using STACE for a given cask design, fuel assembly, and initial conditions. These calculations are based on defensible analysis techniques that credit multiple release barriers, including the cladding and the internal cask walls.

INTRODUCTION

The containment requirements for the transportation of radioactive material are defined by U.S. Nuclear Regulatory Commission (NRC) regulations (10 CFR 71, 1990). Procedures generally acceptable to the NRC for assessing compliance with these provisions have been identified in Regulatory Guide 7.4 (US NRC, 1975), and containment and leak test procedures are specified in American National Standards Institute (ANSI) Standard N14.5 (ANSI, 1987).

ANSI N14.5 states that "compliance with package containment requirements shall be demonstrated either by determination of the radioactive contents release rate or by measurement of a tracer material leakage rate." The maximum permissible leakage rates from the transport cask L_i (cm^3/s), where i represents either normal (N) or accident (A) transport conditions, can be determined from the maximum permissible release rates R_i (Ci/s) and the time-averaged volumetric concentrations of suspended radioactivity within the cask C_i (Ci/cm^3) by:

$$L_i = R_i / C_i . \qquad (1)$$

The maximum permissible release rates are specified in ANSI N14.5 to be $R_N = A_2 \times 10^{-6}$ per hour and $R_A = A_2$ per week (^{85}Kr is the exception for accident conditions since 10,000 Curies are permitted to be released during the one-week period). The quantity A_2 is an activity limit which, under specific release scenarios, would prevent radiological effects from exceeding a specified level consistent with radiological protection standards of the International Commission on Radiological Protection (ICRP). Values of A_2 (e.g., A_2 = 7 Ci for ^{60}Co; A_2 = 10 Ci for ^{137}Cs) are tabulated in Appendix A of 10 CFR 71.

ANSI N14.5 further states that "C_N and C_A shall be determined by the performance of tests on prototypes or models, reference to previous demonstrations, calculations, or reasoned argument," and that "consideration shall be given to (1) the chemical and physical forms of the materials within the containment system, (2) the possible release modes, such as diffusion of gases,

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airborne transportation of powders or particulates, reactions with water or other materials present in the system, and solubility, and (3) the maximum temperature, pressure, vibration, and the like, to which the contained material would be subjected for normal and accident conditions of transport."

Although ANSI N14.5 is quite prescriptive for determining L_N and L_A, little guidance is given regarding how to utilize the chemical, physical, and particulate nature of the contained radioactive material in determining the activity concentrations C_N and C_A. The quantity of radionuclides such as krypton, xenon and iodine which are released from failed fuel rods into the cask interior are specified in example problems in Appendix B of ANSI N14.5 which conservatively assume that 3% and 100% of the fuel rods fail during regulatory normal and hypothetical accident transport conditions, respectively. However, ANSI N14.5 does not demonstrate how to actually determine the concentration of these released fission gases for regulatory transport conditions, nor does it address the impact of spent-fuel fines or other radionuclides such as CRUD on C_N and C_A.

The reference air leakage rate, L_R (std cm^3/s), is equivalent to the most restrictive of L_N or L_A, and ANSI N14.5 calls for L_R to be established by calculation. If L_R is determined to be greater than or equal to 10 std cm^3/s, the package is exempt from required leak testing; and if L_R is determined to be greater than or equal to 0.1 std cm^3/s, the package is exempt from assembly leak testing. Finally, if L_R is less than or equal to 10^{-7} std cm^3/s, leakage tests must only demonstrate that the measured leakage rate does not exceed 10^{-7} std cm^3/s. ANSI N14.5 defines this as the practical limit of "leak-tightness." If the leak-tight criterion requiring a 10^{-7} std cm^3/s leakage rate is utilized, accurate knowledge of C_N and C_A is not required. However, this approach generally leads to increased cask maintenance costs, personnel exposure, and limited lifetime usage of the casks in the certification and recertification process.

The development of source term methodologies at Sandia National Laboratories (SNL) provides an alternative to the leak-tight approach. These methodologies are implemented in the Source Term Analyses for Containment Evaluations (STACE) software (Seager et al., 1992a) which is a task of the Cask Systems Development Program (CSDP) sponsored by the United States Department of Energy's Office of Civilian Radioactive Waste Management (OCRWM). C_N and C_A are calculated for a given cask design, fuel assembly, and initial conditions by estimating the probability of cladding breach within spent-fuel transport casks, quantifying the amount of radioactive material released into the cask interior from the breached fuel rods, and quantifying the amount of radioactive material within the cask due to other sources. STACE then calculates the corresponding maximum permissible leakage rates, L_N and L_A, for normal and hypothetical accident transport conditions, respectively. These calculations are based on defensible analysis techniques that credit multiple release barriers, including the cladding and the internal cask walls. Use of the source term methodology is expected to result in safety benefits and cost savings by determining maximum permissible leakage rates higher than the leak-tight criterion of 10^{-7} std cm^3/s: (1) occupational exposure can be reduced if the time required to perform containment assessment before transport is reduced, (2) fabrication expenses can be reduced, and (3) maintenance expenses can be reduced and cask service life can be extended.

SOURCE TERM METHODOLOGY

The source term methodology considers the individual contributions from the three distinct sources of radionuclides in a spent-fuel transport cask: (1) radionuclides that can be released through breaches in the spent-fuel cladding (Sanders et al., 1992), (2) activated corrosion and free fission products, referred to as CRUD, adhering to the surface of spent-fuel rods (Sandoval et al., 1991), and (3) residual contamination that may build up in the cavity of a cask over time (Sanders et al., 1991). This section describes the methodologies used to estimate the activity concentrations for each of the three sources.

The concentrations of the individual sources are additive, and the maximum permissible leakage rate for the combined source can be written (Sanders et al., 1993):

$$L = \frac{R}{C_{SF} + C_{CRUD} + C_{RC}} \,. \tag{2}$$

Since the maximum permissible release rate R is dependent upon the value of A_2 , an A_2 value must be determined for the mixture which includes C_{SF}, C_{CRUD}, and C_{RC}. The mixture value for A_2 is calculated by:

$$A_2 \text{ for mixture} = \frac{1}{\sum \frac{f(i)}{A_2(i)}} \tag{3}$$

where f(i) is the fraction of the activity of nuclide i to the total activity of the mixture, and $A_2(i)$ is the A_2 value of nuclide i (ANSI, 1987).

Spent-Fuel Contribution

Spent-fuel contains the largest potential source of releasable radioactivity (Sanders et al., 1992). The contribution of spent-fuel to the overall maximum permissible leakage rate largely depends upon its initial pre-transport condition and on subsequent fuel rod response to transportation conditions. The type and amount of radioactive materials that may be released from the fuel rod to the cask cavity are governed by fuel cladding failure which is a function of cask and assembly designs, transport loading conditions, fuel irradiation histories, and other initial and pre-transport conditions.

Four steps are used to apply the source term methodology to spent-fuel for normal and hypothetical accident transport conditions. These steps are discussed in greater detail in a recent paper (Seager et al., 1992b). The first step involves characterization of the dynamic environment experienced by the cask and its contents. These dynamic environments are defined in 10 CFR 71 and are divided into normal and hypothetical accident transport conditions. The most severe normal and hypothetical accident transport conditions are the 0.3-m and 9-m free drop impacts onto unyielding targets, respectively (Sanders et al., 1992). A rigid-body kinematics model is used to analyze the impact event by defining the center of gravity deceleration load history applied to the fuel assemblies.

The second step involves modeling of the stresses induced in the spent-fuel by the dynamic environment. Detailed geometric and computational models are analyzed to obtain steady-state and transient temperature profiles inside the cask, the range of fuel and cladding temperatures, and the deterministic mechanical response of the fuel rods and assemblies. A response analysis of a loaded transport cask and its contents is performed by isolating smaller substructures from the total system and analyzing them separately. For end drop conditions and the initial impact phase of an oblique drop, a single rod model adequately represents the response of the whole assembly, conservatively assuming that all rods in the assembly have similar deformation patterns. For side drop conditions and the slapdown phase of an oblique drop, a two-dimensional assembly model is utilized to examine individual rods, spacer grids, and end-tie plates. The detailed geometric model of the assembly consists of several hundred beam-column elements that represent individual rods, and special nonlinear hysteretic truss elements that represent spacer grids and rod-to-rod interaction at each interface (Barrett et al., 1992). Deterministic response parameters, cladding stresses and strains, and rod interaction forces are obtained at critical points along each length of the rod.

The third step involves the application of probabilistic methods to determine the likelihood of cladding breach in the spent-fuel (Foadian et al., 1992). Two properties specifically used in this evaluation are the material ductility, which is related to ductile tearing from excessive strain, and fracture toughness, which is used to determine the extension of generated or pre-existing partial (partially through the wall) cracks. Three cladding failure modes which can occur are transverse tearing, rod breakage, and longitudinal tearing. Transverse tearing requires that the strain exceed the material ductility limits. It is assumed that once a crack is initiated, it will immediately extend through the wall, thus forming a pinhole or narrow transverse crack. Rod breakage is the extension of an existing transverse crack, and it requires a bending stress intensity that exceeds the fracture toughness of the intact material. Depending on the amount of available energy, a narrow transverse tear could extend through a large portion of the cladding cross section, or even result in a guillotine break. Longitudinal tearing, the opening of a part-wall longitudinal crack on the inside of the cladding, requires a hoop stress intensity that exceeds the fracture toughness. The driving force for the hoop stress intensity is a pinch load arising from rod-to-rod interaction. The source term methodology determines probabilities for the three different types of cladding breach. This is an essential prerequisite for defining release mechanisms, because the physical composition of fuel rod contents that could be released through a cladding breach is strongly dependent on the geometry of the cladding breach. A pinhole failure, for example, could result in the release of fission gases, volatile species, and finely dispersed fuel, whereas a guillotine break could further permit the release of fuel fragments.

The fourth step concerns the prediction of the activity concentration in the cask cavity using knowledge of the cask void volume, the inventory of radionuclides residing in fuel-cladding gaps, and estimates of the fraction of gases, volatile species, and fuel fines released. The spent-fuel source term includes radionuclides released from the fuel matrix to the fuel-cladding gap in gaseous and vapor form, as well as gas-borne particulate fines. A model for the gap inventory has been developed to account for the buildup of xenon and krypton isotopes in the fuel-cladding gap (Sanders et al., 1992). To determine the buildup of moderately volatile species (iodine, cesium, and tellurium) in the gap, it is assumed that they have the same mobility and diffusion characteristics as the noble gases. An upper bound for the quantity of fission gas released into the gap based on the maximum burnup of the spent-fuel is 25% for BWR fuel and 16% for PWR fuel (Barrett et al., 1993). The entire gap inventory is then conservatively assumed to be readily available for release into the cask cavity in the event of cladding breach, irrespective of breach location or size. Once the gases and volatile species escape from the fuel rod, they should disperse homogeneously throughout the free volume of the cask.

Nonvolatile radioactive species in the form of fuel fines can also escape from fuel rods through cladding breaches. The fines are ejected from the fuel rods at the point of cladding failure by the purging actions of fill and fission product gases escaping into the cask cavity. Based on limited data available for fines release at the time of cladding failure (Lorenz et al., 1980; Lorenz et al., 1981; Sandoval et al., 1986), Sanders et al. (1992) have recommended that 0.003% of the mass of fuel in a rod should be considered released from the rod after a cladding breach, regardless of the location or temperature. Unfortunately, there is no clear indication of the particle size distribution of the ejected fines. Unlike the gases and volatile species, the dispersal of particulate material (e.g., fuel fines) is subject to attenuation processes, such as gravitational settling and deposition on surfaces. These processes tend to reduce the airborne radionuclide concentration significantly. In the experimental investigations, ~90% of the fines ejected from the rods were deposited only a short distance from the cladding breach. Therefore, it has been recommended (Fischer, 1990; Sanders et al., 1992) that only 10% of the fuel fines ejected from the fuel rod are assumed to remain airborne. However, if the size distribution of the ejected fuel fines is known, this attenuation can be more accurately determined.

Finally, the previously mentioned experimental data do not account for new fines, if any, produced due to crushing of the fuel pellets due to regulatory transport conditions. Published data for fuel fines size distributions due to crushing of spent-fuel are limited to sizes greater than 5 μm; moreover, their interpretation has not been consistent with regard to the production of particles smaller than the original grain size. It is such small particles, from 0.09 μm to 3 μm in diameter, which can remain suspended for the longest times within a cask (Reardon et al., 1992). Thus, they are the most likely to be released from a cask through a breach in the cask containment system.

CRUD Contribution

The methodology for modeling the CRUD source term differs from that for spent-fuel due to the wider range and better quality of available data (Sandoval et al., 1991). There are two types of CRUD: a fluffy, easily removed CRUD composed mostly of hematite that is usually found on BWR rods; and a tenacious type composed of nickel-substituted spinel occurring on PWR rods. In a few BWR reactors, copper is also an important constituent. Along individual rod cladding, the average-to-peak observed specific activity of CRUD is approximately two, independent of the radionuclide. The nuclides which are important in the CRUD total activity depend on the time since discharge from the reactor; for shipments of five-year or older fuel, ^{60}Co accounts for over 92% of the activity in PWR fuel and 98% of the activity in BWR fuel.

The concentration of CRUD suspended in the cavity of a loaded spent-fuel transport cask depends on: (1) the amount of CRUD initially adhering to the transported assemblies, (2) the fraction and size distribution of CRUD spalled due to normal and hypothetical accident transport conditions, and (3) depletion and resuspension mechanisms acting on the suspended particles. CRUD aerosols have a time-dependent concentration after a spallation-inducing event. An expected particle size distribution for CRUD has been developed based on one sample of fuel that is believed to be representative of BWR fuel. The distribution has a precise log-normal shape with a mean number diameter equal to 3 μm and a geometric standard deviation of 1.87. Since a detailed particle size distribution is available, it is possible to account for the behavior of aerosols inside the cask cavity.

The average CRUD concentrations in a cask cavity can be expressed as the concentration immediately after spallation and initial mixing, multiplied by a Release Reduction Factor that incorporates all geometrical information on the cask volume, settling and collection areas, and the aerosols' time-varying size distribution. Values for the Release Reduction Factor for a few existing cask designs range from 4.7×10^{-6} to 2.3×10^{-3}, depending on the cask type, the fuel type, and the orientation of the cask (Sandoval et al., 1991). However, these values are all much lower than the 1×10^{-1} value utilized by Fischer (1990) for CRUD.

Residual Contamination Contribution

After casks have been used to transport spent-fuel, their interior surfaces (especially the bottom) accumulate a residual contamination from CRUD spalled off the transported assemblies, or from immersion in storage pool water during loading and unloading of the assemblies. This residual contamination remains even following decontamination. The residual contamination report (Sanders et al., 1991) discusses the mechanisms leading to spallation but does not quantify the adhesion forces themselves, and it presents previously unpublished data that clarify the amount of residual contamination present.

The largest amount of residual contamination reported is approximately 1 Ci. This amount is conservatively assumed to be present in the transport cask, and all of it is assumed to spall in both normal and hypothetical accident conditions of transport. The same particle size distribution used for CRUD is used for residual contamination, and identical Release Reduction Factors result when the cask and fuel loading are the same.

EXAMPLE SOURCE TERM ANALYSES

Source term analyses of a lead-shielded truck cask representative of typical cask designs (Fischer et al., 1987) were performed using STACE. The internal cask cavity has a diameter and length of 0.343 meters and 4.54 meters, respectively. The cask contains one Babcock & Wilcox 17x17 Mark C PWR assembly with an average burnup of 30 GWd/MTU (neutron fluence of 5.0×10^{25} n/m^2) which is transported 10 years following reactor discharge. The assembly and basket have a volume of 0.08 m^3 and 0.20 m^3, respectively, giving a cask void volume of 0.14 m^3.

The normal and hypothetical accident transport conditions of a 0.3-m and 9-m side drop onto an unyielding target, respectively, were analyzed for the lead-shielded truck cask with impact limiters. The impact limiters had an assumed crush strength of 6.9 MPa. Using an ambient temperature of 27°C, fuel temperatures were found to range from 220°C to 267°C. The mechanical and cladding breach analyses used the lower fuel rod temperature of 220°C for the cladding, since the cladding breach probabilities increase with decreasing temperatures (Barrett et al., 1993). The radionuclide release analyses utilized the higher fuel rod temperature of 267°C, since the volatile species of radionuclides have increased volatility at higher temperatures.

The releasable source term for the specified normal and hypothetical accident transport conditions are given in Table 1 and Table 2, respectively. The radionuclides are separated into three areas within each table. The first area includes radionuclides which are external to the spent-fuel, such as CRUD and residual contamination. The second area includes fission gases and volatile species which reside in the fuel-cladding gap (also includes radionuclides contained in the fuel rod plenum). Finally, the third area includes radionuclides which are contained in the spent-fuel.

The CRUD is assumed to be composed entirely of ^{60}Co, and the activity due to CRUD on the PWR rods is conservatively encompassed by the assumption that the entire assembly surface area of 34 m^2 has a ^{60}Co specific activity of 220 μCi/cm^2 (Sandoval et al., 1991). One additional Curie of ^{60}Co residual contamination is also assumed to be present. Since both the CRUD and residual contamination are conservatively assumed to completely spall during both the normal and hypothetical accident transport conditions, the initial ^{60}Co release to the cask cavity is equal to the total inventory of ^{60}Co in the CRUD and residual contamination. The initial activity density is determined by dividing the initial release quantity by the cask void volume of

TABLE 1: RELEASABLE SOURCE TERM FOR 0.3-M SIDE DROP NORMAL TRANSPORT CONDITION

Nuclide	Total Inventory (Ci)	Initial Release to Cask Cavity (Ci)	Initial Activity Density (Ci/m^3)	Time-Averaged Activity Density (Ci/m^3)	Activity Fraction f(i)	Release Limit A_2 (Ci)	f(i) / A_2
External to Fuel							
$^{60}Co^a$	7.57×10^1	7.57×10^1	5.40×10^2	9.94×10^{-1}	7.97×10^{-2}	7	1.14×10^{-2}
Fuel-Cladding Gap							
$^3H^b$	9.68×10^1	3.67×10^{-1}	2.62×10^0	2.62×10^0	2.10×10^{-1}	1×10^3	2.10×10^{-4}
$^{85}Kr^c$	3.27×10^2	1.24×10^0	8.85×10^0	8.85×10^0	7.10×10^{-1}	1×10^3	7.10×10^{-4}
$^{134}Cs^d$	1.48×10^{-3}	5.60×10^{-6}	4.00×10^{-5}	4.00×10^{-5}	3.20×10^{-6}	10	3.20×10^{-7}
$^{137}Cs^d$	2.41×10^{-2}	9.13×10^{-5}	6.52×10^{-4}	6.52×10^{-4}	5.22×10^{-5}	10	5.22×10^{-6}
Fuel							
$^3H^b$	1.94×10^1	2.20×10^{-6}	1.57×10^{-5}	1.57×10^{-6}	1.26×10^{-7}	1×10^3	1.26×10^{-10}
^{55}Fe	1.04×10^2	1.18×10^{-5}	8.40×10^{-5}	8.40×10^{-6}	6.74×10^{-7}	1×10^3	6.74×10^{-10}
^{60}Co	1.02×10^3	1.16×10^{-4}	8.26×10^{-4}	8.26×10^{-5}	6.62×10^{-6}	7	9.46×10^{-7}
^{63}Ni	1.76×10^2	2.00×10^{-5}	1.42×10^{-4}	1.42×10^{-5}	1.14×10^{-6}	1×10^2	1.14×10^{-8}
$^{85}Kr^c$	1.72×10^3	1.95×10^{-4}	1.39×10^{-3}	1.39×10^{-4}	1.12×10^{-5}	1×10^3	1.12×10^{-8}
^{90}Sr	2.41×10^4	2.74×10^{-3}	1.96×10^{-2}	1.96×10^{-3}	1.57×10^{-4}	4×10^{-1}	3.92×10^{-4}
^{90}Y	2.41×10^4	2.74×10^{-3}	1.96×10^{-2}	1.96×10^{-3}	1.57×10^{-4}	10	1.57×10^{-5}
^{106}Ru	2.21×10^2	2.51×10^{-5}	1.79×10^{-4}	1.79×10^{-5}	1.44×10^{-6}	7	2.05×10^{-7}
^{125}Sb	5.26×10^2	5.97×10^{-5}	4.26×10^{-4}	4.26×10^{-5}	3.42×10^{-6}	2.5×10^1	1.37×10^{-7}
^{125m}Te	1.28×10^2	1.46×10^{-5}	1.04×10^{-4}	1.04×10^{-5}	8.34×10^{-7}	1×10^2	8.34×10^{-9}
$^{134}Cs^d$	2.09×10^3	2.37×10^{-4}	1.69×10^{-3}	1.69×10^{-4}	1.36×10^{-5}	10	1.36×10^{-6}
$^{137}Cs^d$	3.41×10^4	3.87×10^{-3}	2.76×10^{-2}	2.76×10^{-3}	2.21×10^{-4}	10	2.21×10^{-5}
^{144}Ce	6.31×10^1	7.17×10^{-6}	5.12×10^{-5}	5.12×10^{-6}	4.10×10^{-7}	7	5.86×10^{-8}
^{147}Pm	4.18×10^3	4.75×10^{-4}	3.39×10^{-3}	3.39×10^{-4}	2.72×10^{-5}	2.5×10^1	1.09×10^{-6}
^{151}Sm	1.51×10^2	1.71×10^{-5}	1.22×10^{-4}	1.22×10^{-5}	9.81×10^{-7}	9×10^1	1.09×10^{-8}
^{154}Eu	1.55×10^3	1.76×10^{-4}	1.26×10^{-3}	1.26×10^{-4}	1.01×10^{-5}	5	2.02×10^{-6}
^{155}Eu	5.58×10^2	6.34×10^{-5}	4.53×10^{-4}	4.53×10^{-5}	3.63×10^{-6}	6×10^1	6.05×10^{-8}
^{238}Pu	8.91×10^2	1.01×10^{-4}	7.23×10^{-4}	7.23×10^{-5}	5.80×10^{-6}	3×10^{-3}	1.93×10^{-3}
^{239}Pu	1.46×10^2	1.66×10^{-5}	1.18×10^{-4}	1.18×10^{-5}	9.50×10^{-7}	2×10^{-3}	4.75×10^{-4}
^{240}Pu	2.04×10^2	2.32×10^{-5}	1.65×10^{-4}	1.65×10^{-5}	1.32×10^{-6}	2×10^{-3}	6.62×10^{-4}
^{241}Pu	3.34×10^4	3.79×10^{-3}	2.71×10^{-2}	2.71×10^{-3}	2.17×10^{-4}	1×10^{-1}	2.17×10^{-3}
^{241}Am	7.36×10^2	8.37×10^{-5}	5.97×10^{-4}	5.97×10^{-5}	4.79×10^{-6}	8×10^{-3}	5.98×10^{-4}
^{244}Cm	4.60×10^2	5.23×10^{-5}	3.73×10^{-4}	3.73×10^{-5}	2.99×10^{-6}	1×10^{-2}	2.99×10^{-4}
Total	**1.31×10^5**	**7.73×10^1**	**5.52×10^2**	**1.25×10^1**	**1.00**	**A_2 (mixture) = 53.0 Ci**	

[a]From CRUD on fuel rods and 1 Ci of residual contamination.
[b]50% of ^3H is assumed released to fuel-cladding gap, 10% is retained with fuel, 40% is retained with cladding.
[c]16% of ^{85}Kr is assumed released to fuel-cladding gap, 84% is retained with fuel.
[d]Very small percentages of ^{134}Cs and ^{137}Cs are released to fuel-cladding gap, remainder is retained with fuel.

TABLE 2: RELEASABLE SOURCE TERM FOR 9-M SIDE DROP ACCIDENT TRANSPORT CONDITION

Nuclide	Total Inventory (Ci)	Initial Release to Cask Cavity (Ci)	Initial Activity Density (Ci/m^3)	Time-Averaged Activity Density (Ci/m^3)	Activity Fraction f(i)	Release Limit A_2 (Ci)	f(i) / A_2
			External to Fuel				
$^{60}Co^a$	7.57×10^1	7.57×10^1	5.40×10^2	5.92×10^{-3}	2.25×10^{-3}	7	3.21×10^{-4}
			Fuel-Cladding Gap				
$^3H^b$	9.68×10^1	3.67×10^{-1}	2.62×10^0	2.62×10^0	9.94×10^{-1}	1×10^3	9.94×10^{-4}
$^{85}Kr^c$	3.27×10^2	1.24×10^0	8.85×10^0	8.85×10^0	(e)	1×10^3	(e)
$^{134}Cs^d$	1.48×10^{-3}	5.60×10^{-6}	4.00×10^{-5}	4.00×10^{-5}	1.52×10^{-5}	10	1.52×10^{-6}
$^{137}Cs^d$	2.41×10^{-2}	9.13×10^{-5}	6.52×10^{-4}	6.52×10^{-4}	2.47×10^{-4}	10	2.47×10^{-5}
			Fuel				
$^3H^b$	1.94×10^1	2.20×10^{-6}	1.57×10^{-5}	1.57×10^{-6}	5.96×10^{-7}	1×10^3	5.96×10^{-10}
^{55}Fe	1.04×10^2	1.18×10^{-5}	8.40×10^{-5}	8.40×10^{-6}	3.19×10^{-6}	1×10^3	3.19×10^{-9}
^{60}Co	1.02×10^3	1.16×10^{-4}	8.26×10^{-4}	8.26×10^{-5}	3.13×10^{-5}	7	4.48×10^{-6}
^{63}Ni	1.76×10^2	2.00×10^{-5}	1.42×10^{-4}	1.42×10^{-5}	5.41×10^{-6}	1×10^2	5.41×10^{-8}
$^{85}Kr^c$	1.72×10^3	1.95×10^{-4}	1.39×10^{-3}	1.39×10^{-4}	(e)	1×10^3	(e)
^{90}Sr	2.41×10^4	2.74×10^{-3}	1.96×10^{-2}	1.96×10^{-3}	7.43×10^{-4}	4×10^{-1}	1.86×10^{-3}
^{90}Y	2.41×10^4	2.74×10^{-3}	1.96×10^{-2}	1.96×10^{-3}	7.43×10^{-4}	10	7.43×10^{-5}
^{106}Ru	2.21×10^2	2.51×10^{-5}	1.79×10^{-4}	1.79×10^{-5}	6.80×10^{-6}	7	9.72×10^{-7}
^{125}Sb	5.26×10^2	5.97×10^{-5}	4.26×10^{-4}	4.26×10^{-5}	1.62×10^{-5}	2.5×10^1	6.47×10^{-7}
^{125m}Te	1.28×10^2	1.46×10^{-5}	1.04×10^{-4}	1.04×10^{-5}	3.95×10^{-6}	1×10^2	3.95×10^{-8}
$^{134}Cs^d$	2.09×10^3	2.37×10^{-4}	1.69×10^{-3}	1.69×10^{-4}	6.43×10^{-5}	10	6.43×10^{-6}
$^{137}Cs^d$	3.41×10^4	3.87×10^{-3}	2.76×10^{-2}	2.76×10^{-3}	1.05×10^{-3}	10	1.05×10^{-4}
^{144}Ce	6.31×10^1	7.17×10^{-6}	5.12×10^{-5}	5.12×10^{-6}	1.94×10^{-6}	7	2.78×10^{-7}
^{147}Pm	4.18×10^3	4.75×10^{-4}	3.39×10^{-3}	3.39×10^{-4}	1.29×10^{-4}	2.5×10^1	5.15×10^{-6}
^{151}Sm	1.51×10^2	1.71×10^{-5}	1.22×10^{-4}	1.22×10^{-5}	4.64×10^{-6}	9×10^1	5.16×10^{-8}
^{154}Eu	1.55×10^3	1.76×10^{-4}	1.26×10^{-3}	1.26×10^{-4}	4.78×10^{-5}	5	9.56×10^{-6}
^{155}Eu	5.58×10^2	6.34×10^{-5}	4.53×10^{-4}	4.53×10^{-5}	1.72×10^{-5}	6×10^1	2.86×10^{-7}
^{238}Pu	8.91×10^2	1.01×10^{-4}	7.23×10^{-4}	7.23×10^{-5}	2.74×10^{-5}	3×10^{-3}	9.15×10^{-3}
^{239}Pu	1.46×10^2	1.66×10^{-5}	1.18×10^{-4}	1.18×10^{-5}	4.50×10^{-6}	2×10^{-3}	2.25×10^{-3}
^{240}Pu	2.04×10^2	2.32×10^{-5}	1.65×10^{-4}	1.65×10^{-5}	6.27×10^{-6}	2×10^{-3}	3.14×10^{-3}
^{241}Pu	3.34×10^4	3.79×10^{-3}	2.71×10^{-2}	2.71×10^{-3}	1.03×10^{-3}	1×10^{-1}	1.03×10^{-2}
^{241}Am	7.36×10^2	8.37×10^{-5}	5.97×10^{-4}	5.97×10^{-5}	2.27×10^{-5}	8×10^{-3}	2.83×10^{-3}
^{244}Cm	4.60×10^2	5.23×10^{-5}	3.73×10^{-4}	3.73×10^{-5}	1.42×10^{-5}	1×10^{-2}	1.42×10^{-3}
Total	$\mathbf{1.31 \times 10^5}$	$\mathbf{7.73 \times 10^1}$	$\mathbf{5.52 \times 10^2}$	$\mathbf{2.64 \times 10^0}$ **(e)**	**1.00**	**A_2 (mixture) = 30.8 Ci**	

[a]From CRUD on fuel rods and 1 Ci of residual contamination.
[b]50% of ^3H is assumed released to fuel-cladding gap, 10% is retained with fuel, 40% is retained with cladding.
[c]16% of ^{85}Kr is assumed released to fuel-cladding gap, 84% is retained with fuel.
[d]Very small percentages of ^{134}Cs and ^{137}Cs are released to fuel-cladding gap, remainder is retained with fuel.
[e]Does not include contribution from ^{85}Kr, since ^{85}Kr is evaluated separately against a 10 A_2 release limit.

0.14 m^3. The time-averaged activity density is determined by multiplying the initial activity density by the Release Reduction Factor for the CRUD and residual contamination. The Release Reduction Factor was calculated to be equal to 1.84×10^{-3} and 1.10×10^{-5} for the normal and hypothetical accident transport conditions, respectively.

The mechanical response and cladding breach analyses predict a single rod failure probability of 1.4×10^{-5} for the 0.3-m normal transport event in which peak cask accelerations of 30 g are experienced. Since the assembly contains 264 fuel-bearing rods, 0.0037 fuel rods are expected to fail due to this normal transport condition. For the 9-m side drop hypothetical accident condition, the peak cask accelerations were calculated to be 100 g, and the analyses predict a single rod failure probability of 1.1×10^{-3}. Therefore, 0.29 fuel rods are expected to fail in the 17x17 assembly due to the hypothetical accident condition. However, these analyses conservatively assume that one fuel rod (out of the 264 rods in the assembly) fails for both normal and hypothetical accident transport conditions.

Radionuclides contained in the fuel-cladding gap and the fuel are assumed to be released to the cask cavity only in the event of a fuel rod failure. Since only one fuel rod is expected to fail for both the normal and hypothetical accident transport conditions, the initial radionuclide release to the cask cavity of a specific radionuclide is determined by (1) dividing the total inventory of that radionuclide in the cask by the total number of fuel rods within the cask (264 in these example analyses) and (2) multiplying by the percentage of that radionuclide which is released from the failed fuel rods. For the case of the fission gases and volatile species which are contained within the fuel-cladding gap, 100% of the radionuclides are assumed to be released to the cask cavity. For the case of the radionuclides contained in the fuel, 0.003% of the mass in the fuel rod is assumed to be released into the cask cavity in the form of fines. As was the case for the ^{60}Co due to CRUD and residual contamination, the initial activity densities of the radionuclides contained in the fuel-cladding gap and the fuel are determined by dividing the initial release quantity by the cask void volume. All of the fission gases and volatile species are assumed to remain airborne so the time-averaged activity density is equal to the initial activity density for those radionuclides. However, the time-averaged activity density for the radionuclides released from the fuel is assumed to be 10% of the initial activity density.

The activity concentrations C_N and C_A were calculated to be 1.25×10^1 Ci/m^3 and 2.64 Ci/m^3, respectively. The values of A_2 for the mixture of radionuclides were calculated using Equation 3 to be 53.0 Ci and 30.8 Ci for the normal and hypothetical accident transport condition, respectively. Using Equation 1, maximum permissible leakage rates of $L_N = 1.18 \times 10^{-3}$ cm^3/s and $L_A = 19.3$ cm^3/s have been calculated.

Based on the above values of L_N and L_A, and using the approach outlined in Appendix B of ANSI N14.5, a reference air leakage rate of $L_R = 1.80 \times 10^{-3}$ std cm^3/s was calculated. The cask was assumed to be helium-filled with no water vapor present. The cask pressure under the normal transport conditions was determined to be 1.815 atm. The calculations utilized a leakage length of 0.5 cm across the O-ring seal, a gas mixture viscosity of 0.0246 cP, and a calculated seal flange temperature of 170°C. The equivalent hole diameter was determined to be equal to 1.569×10^{-3} cm for unchoked flow.

CONCLUSIONS

Following the guidance of ANSI N14.5, the STACE methodology provides a technically defensible means for estimating maximum permissible leakage rates. These containment criteria attempt to reflect the true radiological hazard by performing a detailed examination of the spent-fuel, CRUD, and residual contamination contributions to the releasable source term.

The evaluation of the spent-fuel contribution to the source term has been accurately modeled. The structural model predicts the cask drop load history, the mechanical response of the fuel assembly, and the probability of cladding breach. These data are then used to predict the amount of fission gas, volatile species, and fuel fines that are releasable from the cask. There are some areas where data are sparse or lacking (e.g., the quantity and size distribution of fines released from fuel rod breaches) in which experimental validation is planned. In addition, the CRUD spallation fraction is another area where few data have been found; therefore, this also requires experimental validation. In the interim, a 100% spallation fraction is conservatively assumed for computing the releasable activity due to CRUD. The source term methodology conservatively assumes that there is 1 Ci of residual contamination available for release in the transport cask, although residual contamination is still by far the smallest contributor to the source term activity.

The ANSI N14.5 assumption that 3% and 100% of the fuel rods fail during normal and hypothetical accident transport conditions, respectively, has been shown to be overly conservative by several orders of magnitude for these example analyses. STACE calculated maximum permissible leakage rates for this example assembly under the specified normal and hypothetical accident transport conditions of 1.18×10^{-3} cm^3/s and 19.3 cm^3/s, respectively. A reference air leakage rate of 1.80×10^{-3} std cm^3/s was also calculated.

By determining maximum permissible leakage rates higher than the leak-tight criterion of 10^{-7} std cm^3/s, the source term methodology is expected to result in safety benefits and cost savings, compared to using a leak-tight approach, by reducing occupational exposures if the time required to perform containment assessment before transport is lessened, and by reducing fabrication and maintenance expenses.

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