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SOURCE TERMS FOR ANALYSIS OF ACCIDENTS AT A HIGH LEVEL WASTE REPOSITORY*

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V. MUBAYI, R.E. DAVIS, and R. YOUNGBLOOD
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, NY 11973
516-282-2056

ABSTRACT

This paper describes an approach to identifying source terms from possible accidents during the preclosure phase of a high-level nuclear waste repository. A review of the literature on repository safety analyses indicated that source term estimation is in a preliminary stage, largely based on judgement-based scoping analyses. The approach developed here was to partition the accident space into domains defined by certain threshold values of temperature and impact energy density which may arise in potential accidents and specify release fractions of various radionuclides, present in the waste form, in each domain. Along with a more quantitative understanding of accident phenomenology, this approach should help in achieving a clearer perspective on scenarios important to preclosure safety assessments of geologic repositories.

INTRODUCTION

In the safety analysis of nuclear facilities, the expression "source term" generally refers to the quantities of various species of radiological materials present at (or emanating from) a particular location (or set of locations) following an accident. The source term serves as an input to some ancillary analysis, for example, a consequence assessment which calculates radiological doses delivered to the public or the facility workers from the event. Since these doses have to be in compliance with guidelines specified in applicable Nuclear Regulatory Commission (NRC) regulations[1] or Department of Energy (DOE) orders[2], the estimation of source terms for the spectrum of accidents deemed credible at the facility is a necessary element in the safety analysis. In the case of facilities which have yet to be constructed, such as the geologic repository for disposing high level nuclear wastes, the accuracy and sophistication of source term estimation depends in part on the level of detail to which the design effort has progressed, including descriptions of the physical layout, equipment and operational procedures. Even in a preliminary analysis, however, it is necessary to have some perspective on the types of accidents which could compromise the retention of radiological materials, in order to be able to screen out those events which have no significant consequences. As the analysis progresses and attention is devoted to more significant initiators, source term estimation should become more rigorous and be able to potentially locate deficiencies, if any, in the design. The iterative process of safety assessment at each stage of the design effort should be capable of producing a technically defensible risk profile of the facility which meets the applicable licensing guidelines. At this time, the repository to be constructed at the candidate site, Yucca Mountain, Nevada, is in the Conceptual Design stage. This will be followed by an Advanced Conceptual Design in 1991 and a final License Application Design by 1995, according to the current plans of DOE.

Performance assessments of the repository are divided into two chronological stages: (1) the preclosure stage, when the facility is actually being constructed and operated to receive, repackage and emplace the waste, and (2) the postclosure stage, which follows the sealing of the underground storage vaults. (Most of the steps in the safety assessment of the preclosure stage of a repository are also applicable to the MRS, monitored retrieval storage, facility). DOE initiated a Preclosure Risk Assessment Methodology (PRAM) program to support repository preclosure safety analyses conducted in support of design and operations assessment, prioritization of research and development, identification of items important to safety, licensing and public communication. The risk categories covered by PRAM are as follows: radiological risk to the public and

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the workers from accidents, radiological risk to the public and the workers from routine operations, and non-radiological risk to the workers from accidents. The work reported in this paper on source term estimation was carried out at Brookhaven National Laboratory (BNL) under the PRAM program.

OVERVIEW OF REPOSITORY FACILITY AND HIGH LEVEL WASTE

Details of the conceptual design of the repository in tuff at Yucca Mountain are provided in the Site Characterization Plan-Conceptual Design Report (SCP-CDR)[3] produced by the Nevada Nuclear Waste Site Investigation (NNWSI) Project. To summarize very briefly, the repository consists of a surface facility and a sub-surface emplacement area. The surface facility is comprised of an access/receiving area where the waste shipments will be received and inspected, several hot cells where the waste will be unloaded, consolidated and repackaged, and buildings housing support facilities such as the radwaste treatment system. The sub-surface facility will comprise several underground tunnels where the wastes will be emplaced. Unlike other repository concepts examined earlier, the transfer of the waste to the sub-surface emplacement vault at Nevada will not be through a vertical elevator shaft but a curving ramp on which diesel-powered transporters will operate to carry the waste packages.

The latest projections of the charecteristics and quantities of the high level waste to be stored in the repository are contained in the "Integrated Data Base for 1987: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics".[4] Additional detail on the radiological properties of the different waste forms is provided in DOE/RW-0184.[5] In brief, high level waste consists of spent fuel rods from civilian nuclear power plants and (to be) vitrified high-level waste (VHLW) from: (1) the commercial reprocessing plant at West Valley, N.Y. and (2) reprocessing operations at DOE/Defense reactor sites. The latter is currently stored as liquid, sludge, or powder at the individual sites.

APPROACH TO THE SOURCE TERM

Current estimates of the source term in repository preclosure safety analyses are preliminary and characterized by a high degree of uncertainty. This is due to two reasons: (1) lack of quantification of the phenomenology of potential accidents; at the present stage of the design, estimates of the key parameters such as impact energy, temperature, etc. which characteriza the conditions experienced by the wastes during various postulated accidents, are not available, and (2) paucity of experimental information on releases of radiological materials from the waste forms under conditions appropriate to the repository environment; most of the data on releases pertains to conditions which are representative of accidents at light-water reactors (steam and hydrogen atmospheres) and its validity under repository conditions is unclear.

Published studies of repository safety have generally resorted to informed judgement for accident sequence quantification and simple bounding calculations for estimates of the source term. For example, several studies have examined the "hoist drop" accident (relevant to the vertical elevator shaft design) in which the elevator car carrying the final emplacement waste package, housed in a transfer cask, suffers a free fall into the sub-surface area. Generally, the package is assumed to reach terminal velocity. This assumption is conservative since it maximizes the impact energy. Beyond this point the analysis becomes qualitative. Less attention has been paid to repository accidents involving fires although these are likely to be more serious from the viewpoint of radiological releases.

In the case of waste repositories, the source term at a particular location, which could be in-plant or ex-plant, will depend on:

- the location of the accident
- the quantity of waste material involved
- the inventory of specific species of radionuclides in the waste
- the primary release fractions of each species from the accident
- the retention of specific species by any engineered barriers enclosing the waste
- the attenuation of different species along the release path either due to natural causes, e.g., gravitational settling, or to engineered barriers, e.g., filters.

The approach adopted by PRAM is to model the source term, defined as the release of specific species of radionuclides to the environment, as a simple equation of the form:

$$R(i) = n * I_i * F(i) * A_p * A_f$$

where,

- $R(i)$ is the release of the ith species of radionuclides,
- n is the quantity of waste, in appropriate units, involved in the accident,
- I_i is the inventory of the ith species per unit quantity of waste,
- $F(i)$ is the primary release fraction of the ith species in the accident,
- A_p is the attenuation of the release by the packaging encapsulating the waste,
- A_f is the attenuation of the release by the facility.

The quantity of waste involved in an accident is specific to a particular location within the repository. Different quantities of waste material will be handled in the individual compartments and operations during the processes of unpacking, consolidation, repacking and emplacement. The inventory of various radionuclide species is determined by the type of waste form. In the case of spent fuel rods, for example, the major determining factors are the age of the fuel from time of discharge and its burnup. To facilitate the analysis, certain (conservative) values of these factors were adopted by PRAM; 5-year age, 60000 megawatt-day per metric tonne of uranium (MWD/MTU) burnup for PWR fuel and 40000 MWD/MTU for BWR fuel. The ORIGEN-2 code developed by Oak Ridge National Laboratory[6] was recommended to calculate the inventory for various values of age and burnup.

PRIMARY RELEASE FRACTION, F(i)

The primary release of a specific radionuclide from the waste following an accident is to its local environment, e.g., the cladding on a fuel rod. The release depends on the species and the driving energies characterizing the accident conditions. In the repository, mechanical impacts, e.g., drop of a waste package, and thermal events, e.g., fires leading to elevated temperatures, are considered to be the most likely internal initiators. A combination of mechanical and thermal stress could also occur, e.g., a collision followed by a fire. External initiators include seismic events and aircraft crashes. (While mechanical and thermal stress are very likely to be the most probable causes of waste package disruption and radionuclide dispersal, it should be noted that criticality, through some yet to be defined scenario, has not been excluded so far from the list of possible accidents.) In the absence of quantitative information on accident phenomenology, the approach adopted to formulate estimates of F(i) is to partition the accident space into generic domains defined by selected thresholds of mechanical and thermal stress and associate estimates of release fractions of various radionuclides with each of the domains. A schematic representation is shown in Figure 1. E(I) is a mechanical impact energy density boundary, T(J) is a temperature threshold and D(I,J) denotes a release domain. Specific values of F(i) are associated with each of the domains. As the accident phenomena are more concretely quantified, values of release fractions appropriate to the identified range of accident conditions can be used to calculate the source term.

The approach of defining a limited number of domains in accident space to characterize primary release fractions is similar to that adopted in NUREG/CR-4829,[7] a study of spent fuel transportation accidents. However, the boundaries defining the domains for repository accidents are more arbitrary than those of NUREG/CR-4829, since a detailed structural analysis of the repository waste package, including the transfer cask, during potential accidents, has not been carried out so far. (The term "repository waste package" is used here loosely to describe any engineered barrier enclosing the waste which may differ from location to location as the waste moves through the various compartments of the repository; it should not be confused with the term "waste package" as defined in 10 CFR 60). The approach to establishing the impact energy and temperature thresholds was to review the applicable literature on releases, both experimental and theoretical, and look for natural thresholds in the release mechanisms, provided either by experimental data or by empirical models, of the various radionuclide groups. (Table 1 shows the classification of different radionuclides into groups for the purpose of identifying release fractions.)

Details of this literature review, which includes experimental data, analytical estimates and recommendations from earlier studies on radionuclide releases, are provided in Table B.1 of the draft PRAM Procedures Guide.[8] Much of the information originated from previous and ongoing assessments of releases from reactor accidents and fuel transportation accidents. Thus, with only a few exceptions, the information on release fractions pertains largely to reactor spent fuel.

Releases Due to Thermal Stress, $F_{th}(i)$

In reactor fuel rods, during normal operation, a portion of the inventory of fission products resides in the gap between the fuel pellets and the cladding and is commonly referred to as the gap inventory. If the clad is breached, a significant fraction of this inventory can be released promptly depending on the accident conditions. For temperatures to about 1000°C, an empirical model of the release of the fission products, noble gases (NG), cesium (Cs) and iodine (I), consisting of a prompt "burst" release followed by a slower "diffusion" release, was proposed by Lorenz.[9] Data on the gap inventory of NGs obtained at room temperature were reported in *NUREG/CR-2928*, NUREG/CR-3600, NUREG/CR-4043, and NUREG-0418. Analytical estimates of the gap inventory were reported in WASH-1400 and NUREG-0559.[10] In NUREG-0418, an analytical method of predicting NG release at high burnup was proposed. This method was applied in NUREG-0559 to obtain NG gap releases for a range of reactor fuel types, burnups, and power ratings.

In past safety evaluations, the fraction of fission gases promptly releasable was generally assumed to be less than or equal to 0.3. This value is fairly conservative for burnups in the range of 30000 MWd/MTU, typical of past and

recent practice. However, there are indications that future levels of burnup at power plants will be significantly higher. Larger gap inventories are associated with higher burnup.

The results of NUREG-0559 cover both boiling and pressurized water reactors (BWR and PWR) and were judged to be the most appropriate source to estimate the NG gap inventory. Since the NGs do not react with the cladding, this inventory is available for prompt release in the event of a clad breach and constitutes the release fraction for NGs in the lowest temperature range. The value recommended for PRAM is based on the NUREG-0559 predictions associated with the highest linear power rating, 15 kW/ft, and peak burnups of 40000 MWd/MTU for BWR fuel and 60000 MWd/MTU for PWR fuel. (The latter are assumed to serve as a basis for estimating NG gap inventory in spent fuel with the same burnups on a batch-average basis, conforming to the recommended values of burnup chosen for PRAM). Based on these predictions, a value of 0.5 for NG gap inventory was selected. Although there are small predicted differences in the NG gap inventories for different reactor types, the estimated uncertainty in the value itself is significantly larger than the difference. Thus 0.5 can be regarded as the generic value of the gap inventory (and the release fraction F_{th}) for NG in the lowest temperature regime. It is possible that releases of NG in the lowest temperature range could be higher if impact events pulverize the fuel. This would lead to production of new surface area and rupture of microbubbles of NG within the fuel. No attempt seems to have been made so far to quantify this effect.

The lower range of the temperature threshold has been derived from a series of gap release calculations using the CORSOR code.[11] One natural breakpoint for this empirical release model is at about 1100°C. Isothermal heating of spent fuel at this temperature for 70 minutes does not cause a significant transient release of any fission products compared to the gap release. The heating duration of 70 minutes is moderately conservative. (For example, the test conditions specified for fuel shipping casks in 10 CFR 71.73 are 800°C for 30 minutes.) Hence, for PRAM accident analyses, 1100°C was adopted as the lower temperature threshold and the gap release of NGs below this threshold is believed to adequately represent the total release fraction of NGs in this domain.

The CORSOR model has no other natural breakpoint except that beyond 2000°C there will be virtually complete release of the NGs in 10 to 20 minutes. To subdivide the temperature space between 1100 and 2000°C, it is useful to look beyond the model to anticipated accident conditions. Since a potential source of thermal energy in repository accidents could arise from fires involving transportation fuels, it is reasonable to consider the time-temperature relationship associated with transportation accidents. In NUREG/CR-4829, a temperature range of 1400 to 2400°F (760 to 1315°C) was cited for the open burning of hydrocarbon fuels and durations up to 5 hours were considered. CORSOR calculations at 1315°C indicate that about 1, 2, and 10 percent of the remaining NG inventory will be released for heating durations of 30, 60, and 300 minutes respectively. While a duration of 300 minutes may be overly conservative, it leads only to a 10 percent increase in the NG release fraction, from the value of 0.5 estimated above for the gap release (below 1100°C) to 0.55. However, the temperature-time boundary of 1315°C for 5 hours is a reasonable threshold for the release of other fission products.

The last temperature regime arises from considerations of extreme accidents. For example, the possibility that the zircaloy clad on the spent fuel rods could be ignited leading to a highly exothermic zirconium fire remains undetermined. Severe reactor accidents have been modeled extensively, but it does not seem reasonable to adopt limiting conditions associated with reactor core-melt accidents. Given this uncertainty, the choice of a generic extreme thermal condition is based on considerations of bounding estimates. The limiting temperature of hydrocarbon fires in air was reported to be 1315°C. In the analysis of reactor core melting and zirconium fires in spent fuel pools, peak temperatures have been predicted to exceed 2500°C. The spent fuel received at the repository will be at least five years out-of-reactor and the decay heat will have been reduced considerably. The quantity of spent fuel in a repository accident, at least in the surface facility, is likely to be a small fraction of the amount considered in a core melt accident. Thus it is very unlikely that the waste could reach a temperature of 2500°C. What the peak temperature could be is unknown. To estimate a generic release fraction for extreme accidents, the waste is assumed to be isothermally heated at a temperature of 2000°C for 5 hours. The temperature is intermediate between 1300°C and 2500°C and the duration is the same as in the previous regime. These values are viewed as conservative. Under these conditions, there will be a complete release of the volatile fission products from the fuel. Thus under accident conditions where the temperature may exceed 1315°C, a release fraction of 1.0 was adopted for NGs.

To establish release fractions for other fission products under thermally-induced accidents, values derived from sources reviewed in Ref. 8 were supplemented with calculations made with the CORSOR code. Below 1100°C, a number of studies indicate that the volatiles, Cs and I, group release will be dominated by the gap rather than the transient release. (These

studies, however, were limited to fuel with a lower burnup than that assumed for PRAM). Lorenz's model for the gap release process (burst plus diffusion), referred to earlier, produces a release fraction of 3E-4 to 5E-3 for Cs and around 5E-4 for I at temperatures of about 850°C; the coefficients in this model were empirically estimated from end-of-cycle LWR fuel elements with burnups in the range of 30000 MWd/MTU. Application of this model to 5-year out-of-reactor spent fuel rods in the same temperature range leads to release fractions of 5E-3 for Cs and 1E-2 for I. An attempt was made to correct these numbers for higher burnup, based on some limited evidence that the gap inventory of Cs and I increases with burnup as is the case for noble gases, although it is not evident that the Lorenz model can be extrapolated. This effort led to the assumption that the total inventory of Cs and I resides in the gap; however, unlike the NGs, Cs and I are not gaseous under all conditions. The burst component of the Lorenz model was used to calculate the fractional release of Cs and I in the lowest temperature range, based on their inventories estimated in ORIGEN-2 for 5-year out-of-reactor spent fuel with the burnups specified earlier. The contribution due to the diffusion portion of the Lorenz model was ignored since the burst process contributes about 70 to 80 percent of the total gap release. The release fractions estimated by this method are 0.03 for Cs and 0.04 for I. These fractions were adopted as the F_{th} values for Cs and I for temperatures below 1100°C.

In the next temperature regime, 1100°C < T < 1315°C, CORSOR predicts a fractional release of 0.1 for both Cs and I due to the transient release process. Hence the total fractional releases of Cs and I in this temperature range are estimated to be 0.13 and 0.14, respectively. The release fraction of tellurium (Te) in this temperature range, using CORSOR, is estimated to be 3E-3. The release of Te is affected by the degree of oxidation of cladding. If more than 70 percent of the clad is oxidized in the accident, a release fraction of 0.1 for Te is more appropriate.

In the highest temperature regime, T > 1315°C, the CORSOR model predicts nearly complete release of Cs, I, and Te for the assumed conditions, which include complete oxidation of the clad. Thus these groups are assigned a release fraction of 1.0 in this temperature domain.

For the other radionuclide groups, which include the barium (Ba), ruthenium (Ru), lanthanum (La), and cerium (Ce) groups, CORSOR model calculations indicate that the fractional releases predicted for all temperature ranges are, with one exception, lower than the value of the pre-existing fuel fines residing in the gap. The fractional value of pre-existing fines is estimated at 3E-5, based on the work of Lorenz who reported the fractional release of 3E-5 of the fuel as particulate via the burst release mechanism. The exception for the refractory releases, which exceed the pre-existing fines, is the Ba group (barium and strontium) in the highest temperature range where CORSOR calculations predict a value of 5E-3. Hence a value of 3E-5 for F was adopted for the Ru, La, and Ce groups for all temperature ranges. For the Ba group, F was set equal to 3E-5 for T < 1315°C, and 5E-3 for T above 1315°C.

The literature review carried out in Ref. 8 revealed a large uncertainty in the release fractions for tritium and carbon-14. Values quoted in the literature include: 0.1 for H^3 (Jardine[12]), 0.01 for both H^3 and C^{14} (DOE/ET-0029), 0.012 for H^3 and 0.027 for C^{14} (DOE/EIS-0046F), and 0.1, 0.4, and 0.57 for both H^3 and C^{14} releases from reactor spent fuel with burnups ranging from 33000 MWd/MTU to 55000 (PWR) and 55000 (BWR), respectively (DOE/RW-0035/1). At this time, any choice of F(i) for these isotopes is arbitrary. A pragmatic (and conservative) assignment is to place H^3 and C^{14} along with the NG group and assume the same value of release fractions as the noble gases.

Table 2 shows the estimated value of F_{th} in the specified temperature ranges for the various radionuclide groups defined in Table 1.

There does not appear to be any experimental data on the release of fission products or other radionuclides from "reheated" high-level vitrified waste. (Data on impact releases from this waste form are analyzed below.) In general, releases from vitrified waste should be much smaller than releases from spent fuel since much of the release would have occurred during the vitrification process itself, i.e., calcination and glassification. Additionally, the noble gas inventory would have been effectively removed even earlier during fuel dissolution. Thus it is plausible that the spent fuel radionuclide release fractions for thermal accidents adopted are conservative when applied to vitrified waste. The values may, in fact, be overly conservative; the gap inventory which contributes to spent fuel releases at lower temperatures will be absent from vitrified waste and alterations of the chemical form of some of the volatiles, e.g., the formation of cesium silicates, are likely to decrease their volatility. In the absence of relevant data, the adoption of the spent fuel F_{th} values for vitrified waste can be justified on grounds of conservatism.

Releases Due to Mechanical Impacts, F_{imp}

Mechanical impacts which are likely to breach the waste package will release the waste material in the form of particulate which could then be transported as an aerosol (in the case

of spent fuel, there will be, in addition, the release of the gap inventory if the clad is breached). From the standpoint of safety analysis, it is also important to estimate the fraction of particulate whose size is in the respirable range, approximately 10 microns or less. Studies on releases from impact events were reviewed in Ref. 8. These included the results of experiments on impacts of vitrified material at velocities of 20 and 40 m/sec (corresponding to free fall drop heights of about 4.5 and 9 m respectively), reported by Walker[13] which produced fractional releases of particulate in the ranges of 2E-5 to 3E-4 and 4E-4 to 3E-3, respectively. Walker did not recommend a value or a range of values to use in accident analyses. However, these values were recommended in other studies, e.g., in NUREG/CR-4303,[14] a study of preclosure risk for the basalt repository. Wilmot[15] (SAND-80-2124) reported the consensus of an expert panel which assessed the risks associated with spent fuel transportation. The panel argued that although any impact event would generate a larger fraction of fuel fines than that reported by Lorenz for a thermally induced rod burst, the cladding would retard the release of particulate from the fuel. A value of 2E-6 for the fractional release of particulate was assumed; however, the primary release fraction upon impact and the retention due to the cladding were not separately defined. This impact source term has been used in the regulatory area, for example, the Surry Dry Cask Storage Study.[16]

During several PRAM meetings, a consensus developed that the experiments performed by Jardine[17] and collaborators at Argonne National Laboratories represented a reasonable basis for estimating the respirable fraction of aerosols that might be generated during repository impact accidents. These experiments consisted of dropping cylinders of pyrex glass and other materials from specified heights onto an unyielding surface and measuring the size and quantity of the resulting particulates. Summaries of the glass data and a plot of the respirable particulate (<10 microns) fraction versus impact energy density are contained in Table B.3 and Figure B.1 of Ref. 8. The impact energy densities covered a range from 0.41 J/cm3 to 140 J/cm3, and the respirable fraction by weight ranged from 3E-5 to 3E-2. A simple bounding equation summarizing this data over the range from 1 to 140 J/cm3 is:

$$\log_{10} F_{imp} = 0.8 \log_{10} E - 3.2$$

where F_{imp} is the fractional release of respirable size particulates and E is the energy density (in J/cm3) incident at the waste. (The lower limit of fractional release, in the case of spent fuel, due to impacts should not be assumed to be less than 3E-5, the fractional value of the pre-existing fines assumed to density required to form this amount of particulate, lower than the range used to derive the above equation, the limit is probably academic for any impact accident of practical concern).

Several key areas of uncertainty in this estimation of the particulate release fractions need to be noted. Firstly, the data were obtained from specimens of pristine pyrex, which is different from the vitrified waste and spent fuel that will be received at the repository. Secondly, there is considerable uncertainty in prescribing the incident energy deposited in the waste. An attempt can be made to apply this data to repository impact accidents by assuming reasonable fall heights which allow estimation of the total impact energy available and arbitrarily assuming a fractional deposition of this energy to the waste. However a substantial analytical and experimental effort is needed to establish the fraction of total package energy which would be deposited in the waste itself for the range of anticipated conditions. Thirdly, there is the purely statistical uncertainty associated with the pyrex data itself. Additional research is needed to resolve these uncertainties in the application of this data to repository safety analyses.

Releases Due to Combined Thermal and Impact Accidents

Releases due to combined thermal and impact events are calculated from a simple convolution of the thermally-induced releases with the impact-generated release fractions. Since the impact releases were estimated with a continuous function, the selection of the impact energy density thresholds is arbitrary (although the limits are based on the experimental data source mentioned above); for convenience, the thresholds selected are $1 < E < 10$, $10 \leq E < 100$, and $100 < E < 140$, where E is in J/cm^3. The total release due to a combined event is given by the equation :

$$F(i) = F_{th}(i) + [1 - F_{th}(i)] F_{imp}.$$

Table 3 provides estimates of the release fractions F(i), calculted on the basis of this relation, for the specified ranges of temperature and impact energy density. (In Table 3, where a range of values is quoted for a F(i) in a particular cell, the lower value of the range corresponds to a value associated with the lower end of E for that cell and the upper value to the higher end of E.)

ATTENUATION DUE TO THE PACKAGE AND THE FACILITY

As the waste moves through the repository, one or more packaging components will be present, which include fuel cladding (for spent fuel), canister (for VHLW), waste container,

transfer cask, etc. Each component can be considered as a barrier which has the potential to retard the release of radiological material released in an accident. If a barrier survives an accident, the waste would remain localized within the component. Even if a barrier is breached, it seems reasonable, intuitively, to expect some degree of attenuation.

The attenuation due to the packaging, A_p, can be defined as the product of factors appropriate to each of the components or barriers present, for example,

$$A_p = (A_{clad}) * (A_{container}) * (A_{cask}).$$

For an accident involving bare fuel assemblies (which may occur, for example in one of the hot cells), the first factor on the right hand side of the above equation would alone define A_p. For an accident that occurs during emplacement, A_p would presumably have contributions from each factor in the equation.

The packaging components which are generic to the repository are the spent fuel cladding and the canisters which hold the processed VHLW. Other components, e.g., the container, transfer cask, etc. are site specific. It is not possible, at this time, to estimate their survivability in potential repository preclosure accidents as it is, for example, with a fuel shipping cask which is a licensable item. A review of various studies in this area reveals that attenuation factors have been estimated judgementally without detailed analytical or experimental back up to the estimates; in fact, a majority of the studies lean to the side of conservatism by assuming no credit for attenuation by the package. Prototypic testing of full-scale transportation packages by Sandia National Laboratories and simulated waste glass containers by the Savannah River Laboratory indicate that credit can be taken in some cases. The problem for the repository package is how much credit and where it is warranted. This area generally requires a substantial additional effort if it is to be resolved in a technically defensible manner. In the interim, only bounding assumptions appear to be warranted. If accident quantification and structural analysis convincingly support survival of the barrier, complete credit for retention should be given. Conversely, no credit should be taken where failure of the barrier is likely. Hence, at this time, A_p should be assigned bounding values of 0 or 1 depending on whether barrier survival is likely or not likely.

If radionuclide release in a particular area of the facility occurs following an accident, the next stage of the analysis addresses the transport of the released material. With the exception of the NGs and tritium, other materials will be releases as particulate, or, if released in a volatile form at an elevated temperature, should quickly condense into or onto particulate as the release migrates from the accident location. (This is somewhat oversimplified, as there is uncertainty in the actual chemical form of the iodine released.) The question then is mainly one of aerosol transport.

The range of approaches to this problem is wide, from a simple approach of using a decontamination factor (DF) to describe the attenuation along the path of the release, to a complex approach which treats aerosol physics realistically. The simple application of a DF is only useful in a very limited context, e.g., HEPA filtration. If a DF is used to describe the integral loss of material along the path of release to the environment (through plateout, deposition, etc.), one has a simple means of estimating the release to the environment but no information on the location of the retained material within the facility. Since consequences to the facility personnel are required in the safety analysis, the approach selected must be able to provide both the quantities and locations of the radiological releases within the facility.

Among the complex approaches, very detailed models of aerosol transport, encoded in computer codes such as CONTAIN,[18] can provide very extensive information, including the temporal and spatial distribution of released material within the facility as well as the release to the environment. However, many of these codes have formidable input requirements which would be difficult to satisfy given the current status of the repository design, the technical data bases and the lack of accident parameter quantification.

At the present stage of repository safety assessment, it is clear that a more detailed treatment than a simple "DF" approach is needed to estimate the transport of released material within the facility. However, given the large uncertainties, the complex aerosol transport models appear to be too detailed, data and resource intensive to be useful at this stage of the design. The use of a simplified, empirically based transport code is recommended as a means of estimating A_f. However, it needs to be pointed out that no code in this field has been specifically developed for a waste repository, and without experience and a verification effort it is difficult to assert that any particular transport code is applicable to, and adequate for, repository accident analysis.

CONCLUSIONS

The above review indicates that a significant additional effort is required to narrow down the uncertainties in the key parameters

which affect the source term in repository preclosure safety analyses. The first step is to arrive at a more concrete quantification of the phenomenology of potential preclosure accidents to gain a better understanding of the driving energies involved. This depends, in part, on the level of detail to which the design effort has progressed. The approach outlined in this paper, of specifying radiological release fractions from the waste as a function of accident driving energy, offers a "first-cut" characterization of the magnitude of potential releases. In the context of a quantitative analysis of accident conditions, it will allow for a useful interaction between facility designers and safety analysts in an initial assessment of scenarios important to safety. As pointed out above, there is a large uncertainty in assigning credit for retention of radionuclides by the engineered barriers enclosing the waste during waste handling operations. This applies not only to accidents but also to off-normal events or anticipated operational occurrences. A more definitive understanding of retention factors will be necessary both in assessing public and worker dose as well as in evaluating procedures important to operational efficiency. Ongoing efforts by DOE and its contractors in the PRAM program are devoted to resolving some of these uncertainties in preparation for the Advanced Conceptual Design phase of the repository.

REFERENCES

1. NRC regulations are specified in 10 CFR Part 60, "Disposal of High-level Radioactive Wastes in Geological Repositories."
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		Temperature		
		T(1)	T(2)	
Impact Energy Density	E(1)	D(1,1)	D(1,2)	D(1,3)
	E(2)	D(2,1)	D(2,2)	D(2,3)
		D(3,1)	D(3,2)	D(3,3)

Figure 1 Schematic form of release domains.

Table 1 Radionuclide Groups

Group	Elements
NG	Xe, Kr, H^3, C^{14}
I	I, Br
Cs	Cs, Rb
Te	Te, Sb, Se
Ba	Ba, Sr
Ru	Ru, Rh, Pd, Mo, Tc
La	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
Ce	Ce, Pu, Np, Am

Table 2 Release Fractions F_{th} for the Radionuclide Groups under Various Thermal Conditions

TEMPERATURE	RELEASE GROUP							
	NG	I	Cs	Te	Ba	Ru	La	Ce
T < 1100 C	0.5	0.04	0.03	◁	.....	3E-5	.....	▷
1100 C < T < 1315 C	0.55	0.14	0.13	3E-3	◁	.....	3E-5	..... ▷
T > 1315 C	1.0	1.0	1.0	1.0	5.E-3	◁	..... 3E-5	..... ▷

* If extensive cladding oxidation occurs during heating, use 0.1 .

Table 3 Release Fractions by Radionuclide Group

IMPACT ENERGY DENSITY RANGE (JOULES / CM^3)

TEMPERATURE (DEG C)

	E = 1	E = 10	E = 100	E = 140
	NG 0.5	NG 0.5	NG 0.5	
	I 4.1E-2 — 4.4E-2	I 4.4E-2 — 6.4E-2	I 6.4E-2 — 7.2E-2	
	Cs 3.1E-2 — 3.4E-2	Cs 5.4E-2 — 5.4E-2	Cs 5.4E-2 — 6.2E-2	
	Te 6.6E-4 — 4.0E-3	Te 5.0E-3 — 2.5E-2	Te 2.5E-2 — 3.3E-2	
	Ba 6.6E-4 — 4.0E-3	Ba 4.0E-3 — 2.5E-2	Ba 2.5E-2 — 3.3E-2	
	Ru, La, Ce 6.6E-4 — 4.0E-3	Ru, La, Ce 4.0E-3 — 2.5E-2	Ru, La, Ce 2.5E-2 — 3.3E-2	
T = 1100	NG 0.55	NG 0.55	NG 0.55	
	I 0.14	I 0.14 — 0.16	I 0.16 — 0.17	
	Cs 0.13	Cs 0.13 — 0.15	Cs 0.15 — 0.16	
	Te* 3.6E-3 — 7.0E-3	Te* 7.0E-3 — 2.8E-2	Te* 2.8E-2 — 3.6E-2	
	Ba 6.6E-4 — 4.0E-3	Ba 4.0E-3 — 2.5E-2	Ba 2.5E-2 — 3.3E-2	
	Ru, La, Ce 6.6E-4 — 4.0E-3	Ru, La, Ce 4.0E-3 — 2.5E-2	Ru, La, Ce 2.5E-2 — 3.3E-2	
T = 1315	NG 1.0	NG 1.0	NG 1.0	
	I 1.0	I 1.0	I 1.0	
	Cs 1.0	Cs 1.0	Cs 1.0	
	Te 1.0	Te 1.0	Te 1.0	
	Ba 5.6E-3 — 9.0E-3	Ba 9.0E-3 — 3.0E-2	Ba 3.0E-2 — 3.6E-2	
	Ru, La, Ce 6.6E-4 — 4.0E-3	Ru, La, Ce 4.0E-3 — 2.5E-2	Ru, La, Ce 2.5E-2 — 3.3E-2	

* These values for Te apply only in the absence of extensive oxidation of the cladding.