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WASTE PACKAGE RELIABILITY ANALYSIS

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ABSTRACT

Proof of future performance of a complex system such as a high-level nuclear waste package over a period of hundreds to thousands of years cannot be had in the ordinary sense of the word. The general method of probabilistic reliability analysis could provide an acceptable framework to identify, organize, and convey the information necessary to satisfy the criterion of reasonable assurance of waste package performance according to the regulatory requirements set forth in 10 CFR 60. General principles which may be used to evaluate the qualitative and quantitative reliability of a waste package design are indicated and illustrated with a sample calculation of a repository concept in basalt.

INTRODUCTION

The code of Federal Regulations in its Title 10 Part 60 [1] requires that the applicant for a license to operate a repository demonstrate, among other requirements, that the waste package will contain the waste for 300 to 1,000 years (depending on the thermal load of the geologic repository) and that the engineered barrier system (the waste package and the underground facility) will control the subsequent annual release of any radionuclide to no more than 1 part in 10^5 of the amount present after 1,000 years. Although the controlled release requirement is on the engineered barrier system, the applicant will need to demonstrate substantial contribution by the waste package, unless it can be shown that this requirement can be met by the underground facility alone. In practice, proof of future performance of a complex system, such as a high level nuclear waste package, over a period of hundreds to thousands of years cannot be had in the ordinary sense of the word. According to the code of Federal Regulations in its Title 10 Part 60, the NRC will not require absolute proof of zero release during the containment period or of a yearly controlled release of 1 part in 10^5 thereafter; it shall be demonstrated, however, that the proposed waste package design provides 'reasonable assurance' of compliance with both performance criteria.

This paper proposes a general method of probabilistic reliability analysis (PRA) as a useful means to identify, organize, and convey the information necessary to satisfy the criterion of reasonable assurance of waste package performance, according to the regulatory requirements, during the containment and controlled release periods. Furthermore, much of the value received from a reliability analysis is derived from the act of doing it. This is particularly true for the qualitative elements of reliability analysis. General principles which may be used to evaluate the qualitative and quantitative reliability of a waste package design are hereby indicated and illustrated with a sample calculation for a repository concept in basalt.

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PROPOSED APPROACH TO WASTE PACKAGE PRA

Major components of the waste package system are the primary waste form, the waste form container, and packing materials. Ideally, it would be desirable to predict the performance of such a system through the aid of comprehensive, fully deterministic models which span all possible failure modes in the presence of the evolving near-field environment during the operational life of a repository. The use of such models would be warranted if an adequate data base were available which provides values of the relevant model parameters with a sufficient degree of accuracy. In practice however, only a few simplified models have been presented in the literature, and the relevant data have a great degree of uncertainty. Therefore, it seems more appropriate, at present, to resort to a scheme to predict failure probabilities based on the application of simple phenomenological models. In this scheme, one (1) identifies a radionuclide release scenario, (2) formulates and justifies the relevant models, (3) determines ranges and distributions of the associated parameters viewed as random variables, (4) samples among these according to a probabilistic technique, and (5) determines the predicted failure times. Reliability is then calculated as the probability of the waste package system to meet the regulatory requirements under repository conditions. The associated reliability figure will provide a measure of 'assurance' that the system will perform according to the criteria established in the Rule. The above steps 1 through 5 are accomplished classically by dividing the reliability analysis into two separate parts encompassing, respectively, qualitative and quantitative elements.[2,3]

Qualitative Reliability Analysis

Qualitative reliability analysis provides the design reviewer an identification of the various failure modes which contribute to overall system unreliability. Two general steps have been suggested for performing a waste package reliability analysis for compliance with the regulatory criteria. [4] These are: (1) identification of significant failures and their consequences (generally called a Failure Mode and Effect Analysis - FMEA); and (2) presentation of the above information in a table, chart, fault tree, or other format. An example of a fault tree for waste package analysis is provided in Fig. 1. In particular, the FMEA lists all possible, identified failure modes for waste components and the rationale for their dismissal or retention for further analysis. Standard procedures are available in the literature for performing a FMEA (see, for example, Refs. [2], [3], and [5]). Since none was devised for nuclear waste package analysis, they constitute only useful references, and further work should be needed in this area. Generally speaking, the acceptability of a FMEA depends on the comprehensiveness of the phenomena considered in its preparation. There are no practical methods to prove completeness other than a documented record of search and analysis of alternative failure modes such that repeated, detailed review by competent people fails to produce new credible failure modes.

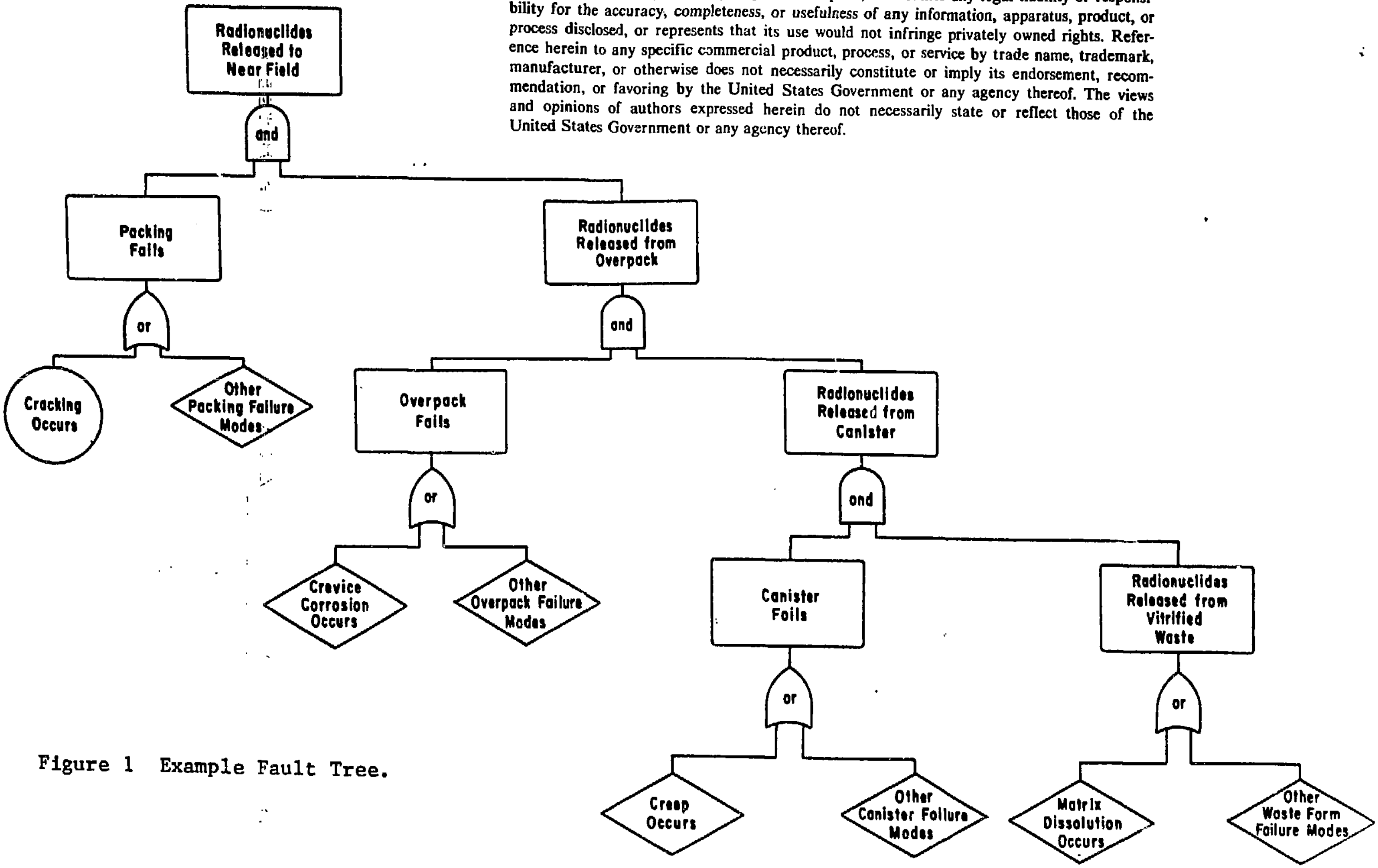

Figure 1 Example Fault Tree.

Quantitative Reliability Analysis

Quantitative reliability analysis provides a numerical value for the reliability of the waste package system. The analysis is performed in two general steps: (1) development of a general model for overall system performance; and (2) operating on the model with a random sampling technique. Step 1 is the more complex one. Indeed, because uncertainty in the basic physical data and models results in an uncertainty in the reliability estimate, initial sources of uncertainty should be identified and quantified, if not eliminated, at this stage. Uncertainty in the physical data should be quantified by providing probability distributions, or uncertainty ranges at least. Uncertainty in predictive models should be established through statistical evaluation of the scatter of the reference data, and through generally accepted engineering judgement if the basic data are deemed inadequate to characterize the models. These uncertainties can be represented by a factor (uncertainty factor) which multiplies the results of the equation and which could have its own probability distribution. Thus each basic model is statistically characterized by the probability distributions of its parameters, including its uncertainty factor. In step 2, one treats the input data of the overall system performance model as random variables with given distribution functions, samples among these with an appropriate technique based on a random number generator approach, and determines performance for each sample case. The process is repeated several times in order to simulate any combination of data considered possible for the design. References [4] and [5] identify Latin Hypercube Sampling [7] as a general random sampling technique available for preparing random inputs to a generic, deterministic model. One important feature of this technique is that it allows the control of correlation between the input data.

The approach outlined above is known as Monte Carlo Simulation in the literature. Monte Carlo simulation of a complex system is practical only if the models used for the analysis are simple enough that each calculation takes limited computer time and memory. This would not dispense the analyst of having detailed process models, as these are needed to provide guidance in the usage of the simpler ones.

SIMPLIFIED WASTE PACKAGE RELIABILITY CALCULATION

A simplified calculation of the reliability of the waste package is summarized below. The criteria used to judge success are: (1) containment is 'substantially complete' if the waste package is not releasing radionuclides at an annual rate greater than 1 part in 10^8 during the first 1,000 years after emplacement; (2) controlled release is complete if, based on the inventory at 1,000 years, the waste package is not releasing radionuclides at an annual rate exceeding 1 part in 10^5 between 1,000 and 10,000 years after emplacement. The first criterion is arbitrary and not endorsed by the NRC. It is used here for illustration purposes in order to give the packing material partial credit for radionuclide containment. In fact, due to the nature of the dispersion equation for radionuclides in a saturated porous medium, all dispersion models would predict an instantaneous failure of the package as soon as the canister fails if the zero release rule were interpreted literally. The second criterion is conservative with respect to 10 CFR 60 in that the latter poses the requirement on the engineered barriers (i.e., the waste package and the underground facility).

The waste package design considered in this analysis is one described in the Site Characterization Report for the Basalt Waste Isolation Project.[8] It entails a borosilicate glass waste form, a carbon-steel canister, and a basalt-bentonite packing in horizontal emplacement holes. Our simplifed analysis of the system uses a limited scope FMEA, i.e., without a judgement of the probability of other failure modes. The only failure modes considered are: (a) pitting corrosion of the canister followed by (b) leaching of the glass, and (c) transport of radioisotopes through the packing material. It is further assumed that the packing material is saturated with water and that the chemical composition of the water is not modified by the effect of the ionizing radiation. Mathematically, this results in a performance model consisting of the following component models: (a) a temperature-time model, (b) a canister corrosion model, and (c) a combined, radioisotope leaching-and-migration model. Details for the formulation of these models are provided in Ref. [4]. The pitting corrosion model has been obtained by analyzing the statistical scatter in experimental data for uniform corrosion, adjusting the resulting correlation through a pitting factor incorporating a pitting correction and the uncertainty of the model with respect to the data. The pitting model accounts for the effects of time and temperature, as well as of the chlorine and oxygen concentrations in the water in contact with the canister. Values for chlorine and oxygen concentrations are supplied as separate input data with appropriate uncertainty ranges and distributions (Table 1). The leaching model depends on time and temperature. Transport through the packing materials depends on time, temperature, porosity of the medium, water velocity, dispersivity, diffusivity, and a radionuclide-specific distribution coefficient. In particular, the analysis has focused on the behavior of Technetium and Plutonium alone, since these elements display a very large difference in their degree of retardation in soils.

The above models have been implemented in the code WASTE and then interfaced with the Latin Hypercube Sampling routine (LHC). Typically, LHC prepares input to deterministic codes such as WASTE from statistically characterized inputs supplied by the user. Values which have been used with WASTE through LHC are reported in Table 1. In this table, every variable is characterized by its lower and upper quantile values and by its probability distribution function (p.d.f.) type, except for quantities taken as known with negligible uncertainty for which the upper and lower quantiles coincide, no distribution is given and are treated in the program as point values.

Table 1

	Lower 0.001 Quantile	Upper 0.001 Quantile	Distribution Function
ROCK PROPERTIES			
Geothermal Temperture (C)	54.0000	60.0000	Uniform
Thermal Conductivity (W/M/K)	1.2500	2.5000	Uniform
Density (KG/CU.M)	2410.0000	2800.0000	Uniform
Specific Heat (J/KG/K)	820.0000	1160.0000	Uniform
EMPLACEMENT GEOMETRY			
Pack Density (1/M/M)	0.00748	0.00748	
WASTE PACKAGE PARAMETERS			
Waste Age (Years)	0.0000	0.0000	
Initial Power (KW)	2.1000	2.1000	
Rock Shell Thermal Conductivity (W/M/K)	1.2500	2.5000	Uniform
Outer Diameter of Packing (M)	0.6860	0.6860	
Thermal Conductivity of Packing (W/M/K)	0.4000	1.4000	Uniform
Outer Diameter of Overpack (M)	0.3250	0.3250	
Thermal Conductivity of Buffer(W/M/K)	10.0000	10.0000	
Outer Diameter of Canister (M)	0.3250	0.3250	
Canister Thickness (M)	0.0530	0.0530	
Length of Canister (M)	4.1000	4.1000	
CORROSION INPUT DATA			
Pitting Factor	1.0000	6.0000	Uniform
Exponent of Time	0.3639	0.5736	Normal
Uniform Corrosion Coefficient (MM/YR)	0.0015	676.0000	Lognormal
Chlorine (PPM)	1.0000	100.0000	Uniform
Oxygen (PPM)	0.0100	3.0000	Uniform
LEACHING INPUT DATA			
Exponent of Time	0.1000	0.7500	Uniform
Leach Rate Factor =	(10**(X-Y/T.FAIL))*(10**Z)	(GM/((CM**2)*(DAY**EN))	
Leach Rate Factor X	3.1800	3.1800	
Leach Rate Factor Y	-2424.2200	-2424.2200	
Leach Rate Factor Z	-0.4000	0.4000	Uniform
Density of Glass (GM/CM**3)	3.0000	3.0000	
Radius of Glass (CM)	30.5000	30.5000	
Crack Factor of Glass	2.0000	40.0000	Uniform
TRANSPORT INPUT DATA			
Hydraulic Conductivity (CM/YR)	0.0001	0.3000	Uniform
Hydraulic Gradient	0.0050	0.0300	Uniform
Density (GM/CM**3)	2.1000	2.7000	Uniform
Porosity	0.0010	0.0010	
Diffusivity (CM**2/YR)	3.1500	315.0000	Uniform
Dispersivity (CM)	0.0000	1525.0000	Uniform
PLUTONIUM			
Distribution Factor (CM**3/GM)	45.0000	5200.0000	Lognormal
TECHNETIUM			
Distribution Factor (CM**3/GM)	0.0000	0.0000	

Results

Using the input values shown in Table 1, the program WASTE was run for 476 cases. There were nine cases showing failure of the canister from corrosion in less than 1,000 years. All cases showed failure of containment for Technetium and one of the cases showed failure of containment for Plutonium. Failure to meet the controlled release criterion occurred in 10 cases. From the results of 476 cases, the probability of failing the containment criterion is 2% (reliability level of 98%). The probability of failing the controlled release criterion is also 2%. This does not mean that it is expected that 2% of the canisters in a repository constructed according to this design will fail, but means that there is a 2% chance that all the canisters will fail since the causes of the uncertainty are common to all canisters. Inspection of the time to failure data shows that the failures tend to occur early, if they occur at all. This is due to the combined effect of the early high temperatures and of the decreasing rate of corrosion with time. The presence of the packing material appears to be beneficial for Plutonium but shows no significant benefit for Technetium. The dominant uncertainty in the time to failure is introduced by the uncertainty of the overall corrosion coefficient.

CONCLUSIONS

Probabilistic reliability analysis may prove a viable methodology to demonstrate with reasonable assurances that the performance of a waste package complies with the regulatory criteria set forth in 10 CFR 60. On the part of the potential applicant for a repository license it requires a well planned research effort for modeling basic waste package processes at different level of complexities and for generating probability distribution functions for the input parameters.

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