License Application for a Spent Nuclear Fuel Repository in Sweden

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Abstract: The Swedish Nuclear Fuel and Waste Management Co., SKB, has applied for a license to build a final geological repository for spent nuclear fuel at the Forsmark site, situated around 70 miles north of Stockholm, Sweden. A key component in the license application is an assessment of the long-term safety of the repository. Probabilistic radionuclide transport and dose calculations are at the core of the analysis. The license application is currently (Spring 2014) under review by the Swedish Radiation Safety Authority, SSM, and a report from SSM to the Swedish Government is expected in 2015.

This paper *i*) gives an overview of the probabilistic dose calculations of the safety assessment in the license application and *ii*) presents some new results related to the probabilistic calculations obtained after the completion of the assessment.

Keywords: Nuclear Waste, Spent Nuclear Fuel, Probabilistic Safety Assessment, Sweden, SR-Site.

1. INTRODUCTION

The Swedish Nuclear Fuel and Waste Management Co., SKB, jointly owned by the nuclear reactor owners, is responsible for the management of the nuclear waste arising from the nuclear power operations in Sweden. Within SKB's program for the management of spent nuclear fuel, an interim storage facility and a transportation system are today (Spring 2014) in operation. A principal remaining task in the program is to build and operate a final repository for the spent nuclear fuel.

In April 2011, SKB applied for a license to build a final geological repository for spent nuclear fuel at the Forsmark site, situated around 70 miles north of Stockholm, Sweden. A key component in the license application is an assessment of the long-term safety of the repository [1], called the SR-Site assessment. Probabilistic radionuclide transport and dose calculations are at the core of the analysis. The license application is currently under review by the Swedish Radiation Safety Authority, SSM, and a report from SSM to the Swedish Government is expected in 2016.

Several decades of research and development has led SKB to put forward the KBS-3 method for the final stage of spent nuclear fuel management. In this method, copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500 m depth in groundwater saturated, granitic rock, see Figure 1. The purpose of the KBS-3 repository is to isolate the nuclear waste from man and the environment for very long times. The primary safety function of the repository is to completely contain the nuclear fuel within the canisters. Around 12,000 tonnes of spent nuclear fuel is forecasted to arise from the currently approved Swedish nuclear power program (where the last of the 10 operating reactors is planned to end operation in 2045), corresponding to roughly 6,000 canisters in a KBS-3 repository.

The principal acceptance criterion according to Swedish legislation is a requirement that the annual risk of individuals in the most exposed group of contracting cancer or hereditary effects from repository derived radionuclides must not exceed one in a million. The risk is obtained as a time-dependent mean value of probabilistically calculated doses, multiplied by a dose-to-risk conversion factor of 0.073/Sv, provided the Swedish regulator and in agreement with recommendations from the International Commission on Radiological Protection, ICRP. The assessment is required to cover one million years after repository closure.

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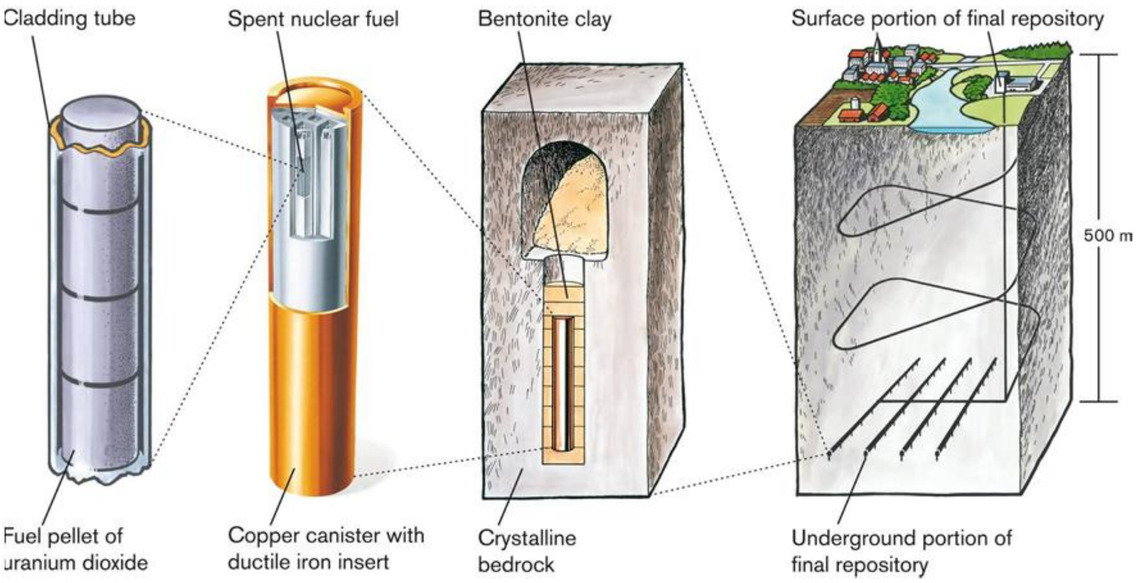
Figure 1: The KBS-3 concept for disposal of spent nuclear fuel.



This paper *i*) gives an overview of the probabilistic dose calculations of the safety assessment in the license application and *ii*) presents some new results related to the probabilistic calculations obtained after the completion of the assessment.

2. OVERVIEW OF PROBABILISTIC DOSE CALCULATIONS

2.1. Assessment Strategy

2.1.1. General

The overall approach for achieving safety in the KBS-3 repository is to *i*) locate the repository in an environment that is expected to be stable in the long time frames during which the spent nuclear fuel poses a hazard to man and the environment, and *ii*) to use materials that are compatible with that environment in the construction of protecting barriers in the repository.

A central element in the demonstration of safety, i.e. in the safety assessment, is, therefore, to assess how the repository environment and the engineered barriers evolve in time after closure of the repository. A reference evolution spanning approximately 120,000 years is studied as a starting point. This time frame is the period of future glacial cycles that largely determine the boundary conditions for the evolution of the repository. In the reference evolution, a repetition of the last glacial cycle is assumed. This is a starting point, based on which other scenarios are considered and analyzed.

2.1.2. Scenario Disaggregation

In principle, the product of dose consequences and likelihoods of all possible future evolutions of the repository should be weighed together and presented as a time-dependent risk. The spectrum of possible evolutions is, however, very wide and cannot be captured in a detailed sense.

The usual approach taken in safety assessments, and also in the SR-Site assessment, is to work with scenarios and variants that are designed to capture the broad features of a number of representative possible future evolutions. Together, these are intended to give a reasonable coverage of possible future exposure situations.

The scenarios are selected based on a number of specified safety functions of the repository. For example, the canisters should withstand isostatic loads in the repository and they are designed to withstand loads of 45 MPa. In order to critically analyze the potential to withstand isostatic loads, a dedicated scenario is constructed, in which all factors of importance for the isostatic load are considered. Such factors are the thickness of future glaciers above the repository determining hydrostatic pressures on the canisters and the likelihood of manufacturing flaws rendering a canister more vulnerable to isostatic loads.

Each scenario may be subdivided into several variants and each variant may, in the probabilistic consequence calculations, be represented by several calculation cases.

2.1.3. Overestimation of Risk

The formulation of scenarios, variants and calculation cases, and the subsequent weighing together of these to give a total risk aims at an over prediction of risk. SSM's regulation requires that the annual risk should be less than 10^{-6}. There are a number of uncertainties that cannot be managed quantitatively in any other rigorous manner from the point of view of demonstrating compliance than by pessimistic assumptions.

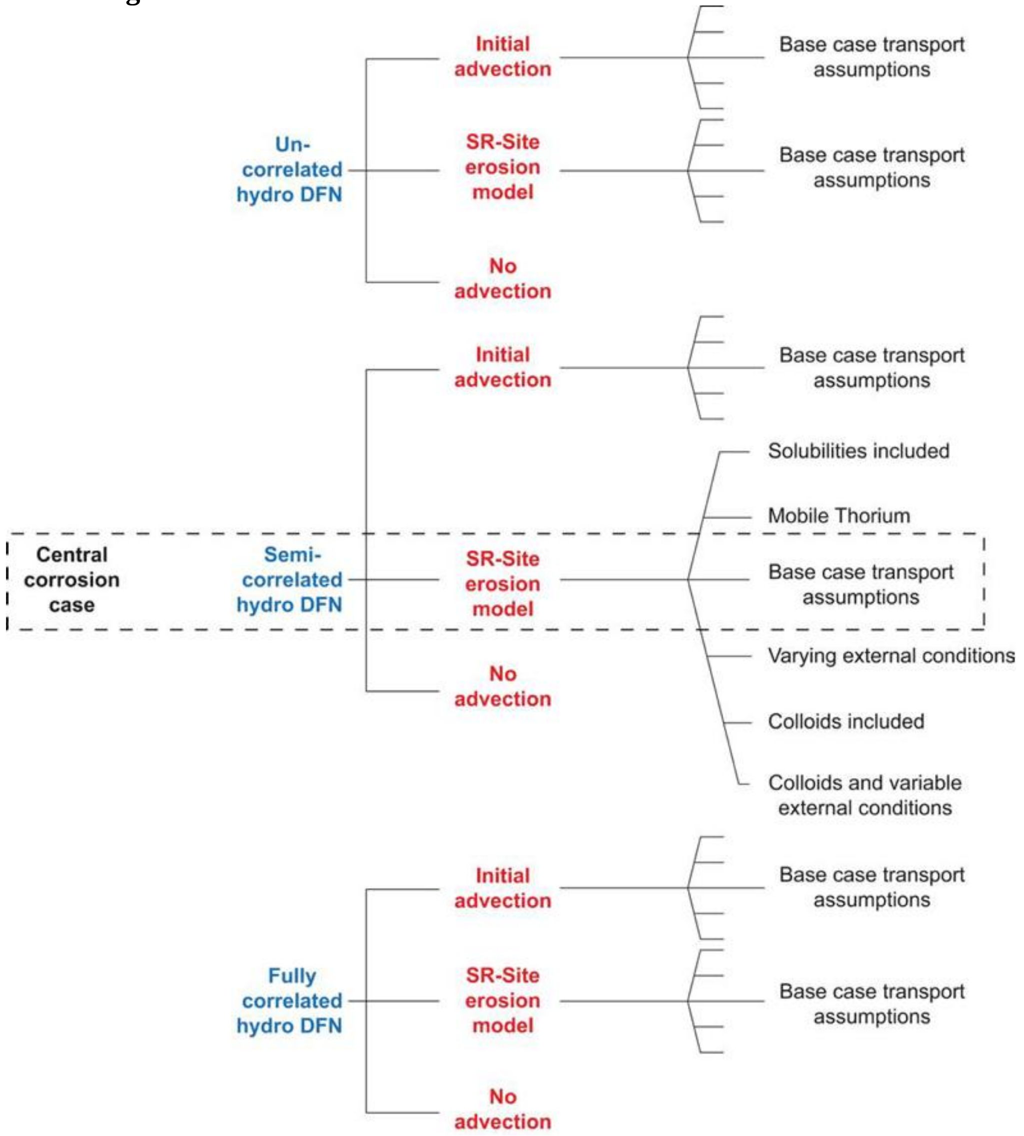
Another situation in which risk has to be overestimated concerns scenario probabilities. Regarding e.g. future climate, both repetitions of conditions reconstructed for the past 120,000 year glacial cycle and an alternative where this development is considerably perturbed by a global warming effect can be envisaged. Although the two are mutually exclusive, both must be regarded as possible. In the risk summation, the logical position is adopted that the summed consequence of a set of mutually exclusive scenarios can, at any point in time, never exceed the maximum of the individual scenario consequences. For scenarios and variants where defensible probabilities are difficult to derive, a scenario or variant giving high consequences can pessimistically be assigned unit probability and other scenarios and variants yielding lower dose impacts can be "subsumed" under the one with the more severe consequences.

Although the primary aim with risk calculations is to demonstrate compliance, there is also the clear ambition of clarifying the sensitivities of the calculation results. For this aim, the calculation cases should be, in principle, as realistic as possible in capturing uncertainty. One quantitative tool for this is the use of probabilistic evaluations of calculation cases followed by sensitivity analyses of the results. For this reason, *i*) pessimistic simplifications are avoided where a sound scientific basis exists for a quantitative treatment and *ii*) pessimistically neglected features of the system are included in discussions of sensitivities.

2.2. The Structure of Scenarios and Calculation Cases

For each scenario (or scenario variant), the containment potential of the repository, i.e. the potential of the canisters to completely contain the spent fuel, is analyzed in a first step. If the system evolution is found to lead to failure of the canisters, the extent of canister failures and the failure times are quantified. This information is used as input to probabilistic radionuclide transport and dose calculations. For each scenario, a number of probabilistic calculation cases are formulated. The assessment strategy, when implemented, thus leads to a structure of scenarios (or scenario variants), each with a number of probabilistic calculation cases. Figure 2 shows an example for the scenario that yielded the highest risk contributions in the SR-Site assessment, the so called corrosion scenario, where canister failures due to corrosion are analyzed. The first two subdivisions of calculation cases arise due to two factors that are of importance for the scenario, and that can not readily be treated probabilistically: the hydrogeological model of the repository (three model variants, blue text in Figure 2) and the model used to quantify long-term loss of the protecting clay buffer around the canisters (three different models, red text). Together the two factors give rise to 3×3 calculation cases. Additional factors related to radionuclide transport and dose calculations give rise to a further multitude of calculation cases. A central corrosion case, based on reasonable assumptions regarding all the mentioned factors, is used as a starting point. The factors related to radionuclide transport and dose were studied in a first step, and this led to the conclusion that the central assumptions regarding these do not yield significantly different results than the alternative assumptions. The remaining 3×3−1 combinations of hydrogeological and buffer loss cases were therefore analyzed only for the central assumptions regarding radionuclide and dose factors.

Figure 2. Overview of calculation cases for the corrosion scenario.



2.3. Key Probabilistic Results

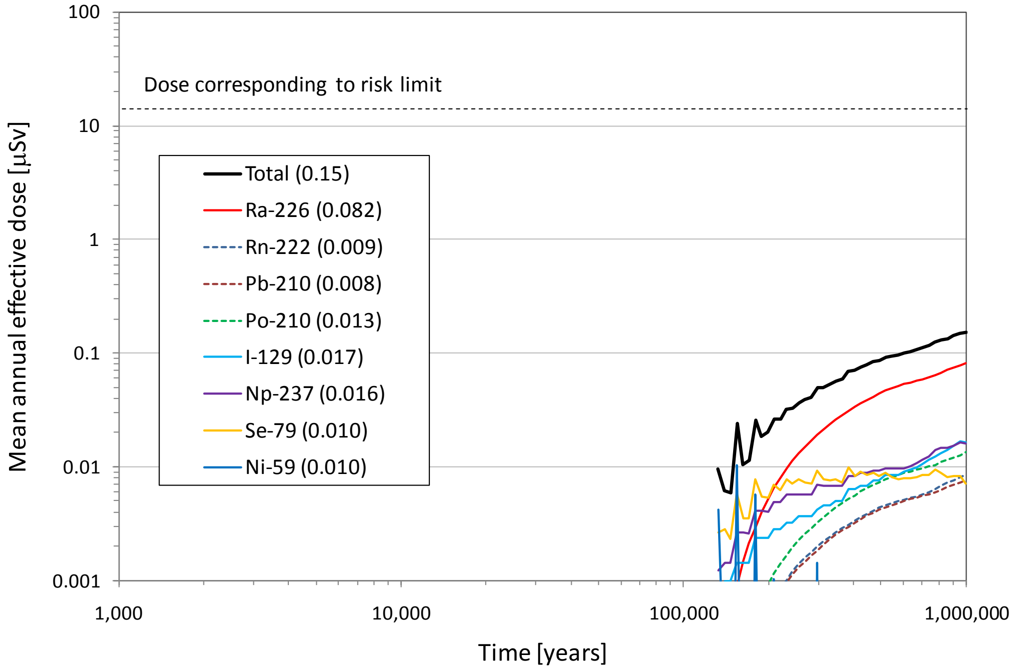
2.3.1 Input Data Compilation

A protocol for the compilation of input distributions for the probabilistic calculations was established early in the project. The protocol requires, among other things and for each input data entity, critical evaluation of data sources, discussion of correlations between input parameters, and a qualification of an input probability density function, PDF. This is briefly described in chapter 9 of the safety assessment report [1], and in more detail in a dedicated Data report [2].

2.3.2. Key results

The result of the probabilistic calculation of the central corrosion case is shown in Figure 3. The figure shows the mean annual effective dose as a function of time, obtained with 10,000 realizations and using Latin Hypercube Sampling. The increase in dose over time is essentially caused by the increasing likelihood for canister failure as corrosion proceeds over time.

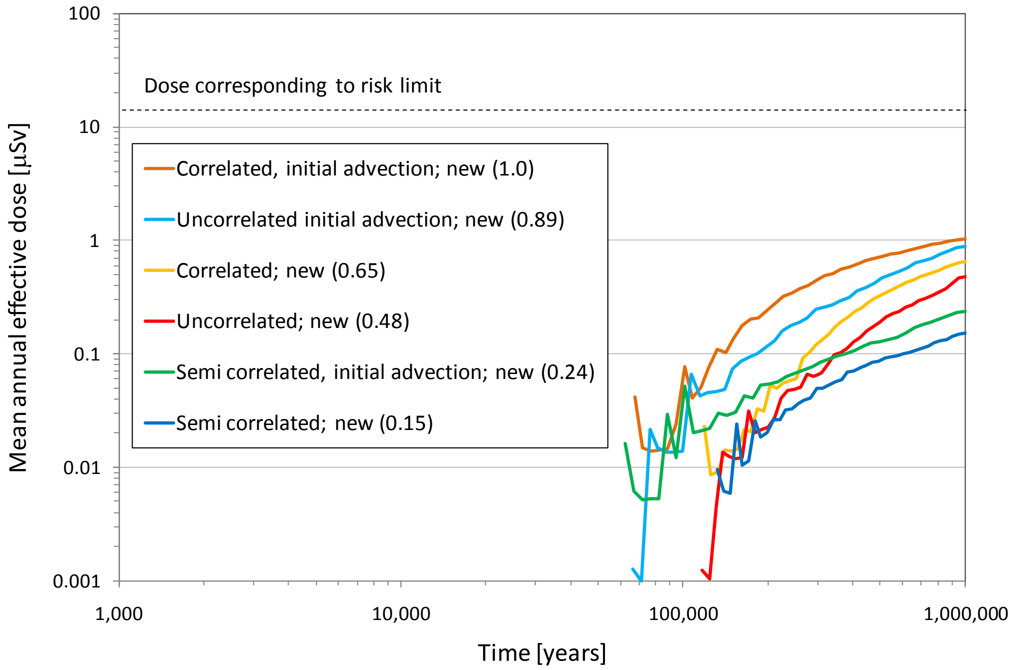
Figure 3. Mean annual effective dose equivalent release for a probabilistic calculation of the central corrosion case. The average number of failed canisters is 0.12. The peak mean annual effective dose over one million years in µSv is given in brackets in the legends.



The results of all 3×3 main corrosion cases are shown in Figure 4. Since the three "no advection" cases, representing the cases where the protective buffer remains intact, yield no canister failures and thus zero releases, the nine cases generate only six release curves.

The corrosion scenario yields the highest risk contributions in the safety assessment. The cases shown in Figure 4 are argued to, together, cover uncertainties in the quantification of the corrosion scenario and to each be an upper bound on the cases they represent. Since the calculated mean doses are all below the dose corresponding to the regulatory risk limit over the entire one million year assessment period, these results are key elements in arguing that the proposed repository fulfils the regulatory risk limit at the Forsmark site.

Figure 4. Summary of far-field mean annual effective dose for probabilistic calculations of six variants of the corrosion scenario. The peak mean annual effective doses, all occurring at the end of the assessment period, are given in parentheses in μSv.



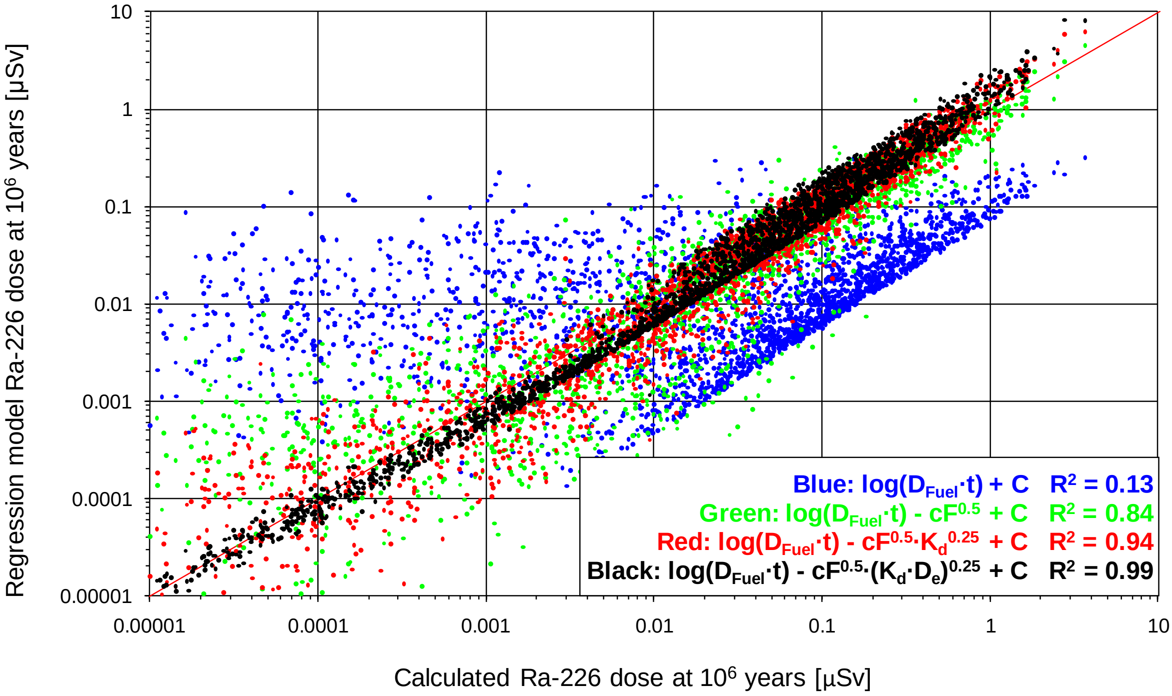
2.5. Sensitivity Analyses

Regarding sensitivities, it is of interest to determine *i*) the input parameters that correlate most strongly with the dose over the entire dose range and *ii*) the input parameter values that are related to high and low doses. Ra-226 dominates the dose in most of the realizations in the dose dominating corrosion scenario and it is thus of particular interest to clarify sensitivities of the Ra-226 dose to input parameters. The first purpose is achieved with two methods: *i*) determination of standardized rank regression coefficients and *ii*) determination of variance based sensitivity indices. The former is presented in the safety assessment report [1], whereas the latter has been applied after the completion of the report [3].

The second purpose is also achieved with two methods: *i*) the calculation of conditional mean values and *ii*) the application of so called cobweb plots. Again, the former is presented in the safety assessment report [1], whereas the latter has been applied after the completion of the report [3].

Finally, the results of the above methods are further explained by the use of a tailored regression model [1] that demonstrates how the variability in the output can be explained with analytic expressions derived from the conceptual understanding of the transport processes involved in the dose calculations. Based on the understanding and mathematical formulation of the transport processes involved in the radionuclide transport calculations, a number of tailored regression models that include successively more input variables were constructed for the peak dose of Ra-226, the radionuclide that dominates most of the numerical model realization. The highest order model yields an R^2-value of 0.99 when regressed on the results of the numerical transport models, see Figure 5 that shows both the regression model expressions as an insert, and the regression results.

Figure 5. Four tailored regression models, based on successively more variables, for the Ra-226 peak dose. For example, the black dots show the good agreement of the results of a five-parameter tailored regression model with those of the numerical transport model.



All the applied methods point to two input parameters to which the results are most sensitive: The rate at which the fuel matrix is converted, D_{Fuel}, with the associated release of radionuclides, and the transport resistance in the geosphere, F, which is closely related to the flow conditions along the transport path from the failing canister to the biosphere. This is further discussed in section 13.5.11 of the safety assessment [1] and in Reference [3].

3. ADDITIONAL RESULTS

What follows are new results in two areas, obtained after the completion of the safety assessment SR-Site.

3.1. Determination of the Variability of the Probabilistic Results

The probabilistic dose assessments were set up to obtain a statistically correct mean value, whereas the determination of output variability was simplified regarding some aspects of the problem. Specifically, in the probabilistic calculations, it was postulated that one canister fails, the mean dose of this case was calculated and subsequently multiplied by the mean number of failed canister for a particular calculation case.

To analyze the impact of this simplification, a more elaborated set-up of the calculation cases is applied and the variability is determined from these. Each case in Figure 2 is based on one of three hydrogeological model variants. The central corrosion case is e.g. based on the so called semi-correlated hydrogeological variant. In this variant, as in the other two, the network of rock fractures is described statistically. In a particular realization of a model variant, a network of fractures is generated in accordance with the statistical model related to the hydrogeological variant. This, in turn, will determine the flow condition at each of the 6,000 canister positions in the repository for the realization in question. Furthermore, the groundwater concentration of the main canister corroding agent, sulfide, is sampled for each canister position from a distribution determined from the present and future hydrogeochemical conditions at the site. These two stochastic factors, the groundwater flow and the sulfide concentration, will basically determine which (if any) of the 6,000 canisters that will fail in the corrosion scenario. The number of failed canisters, their failure times, their positions in the repository, and the flow related transport properties for radionuclides at these positions will thus vary between realizations of the hydrogeological model variant.

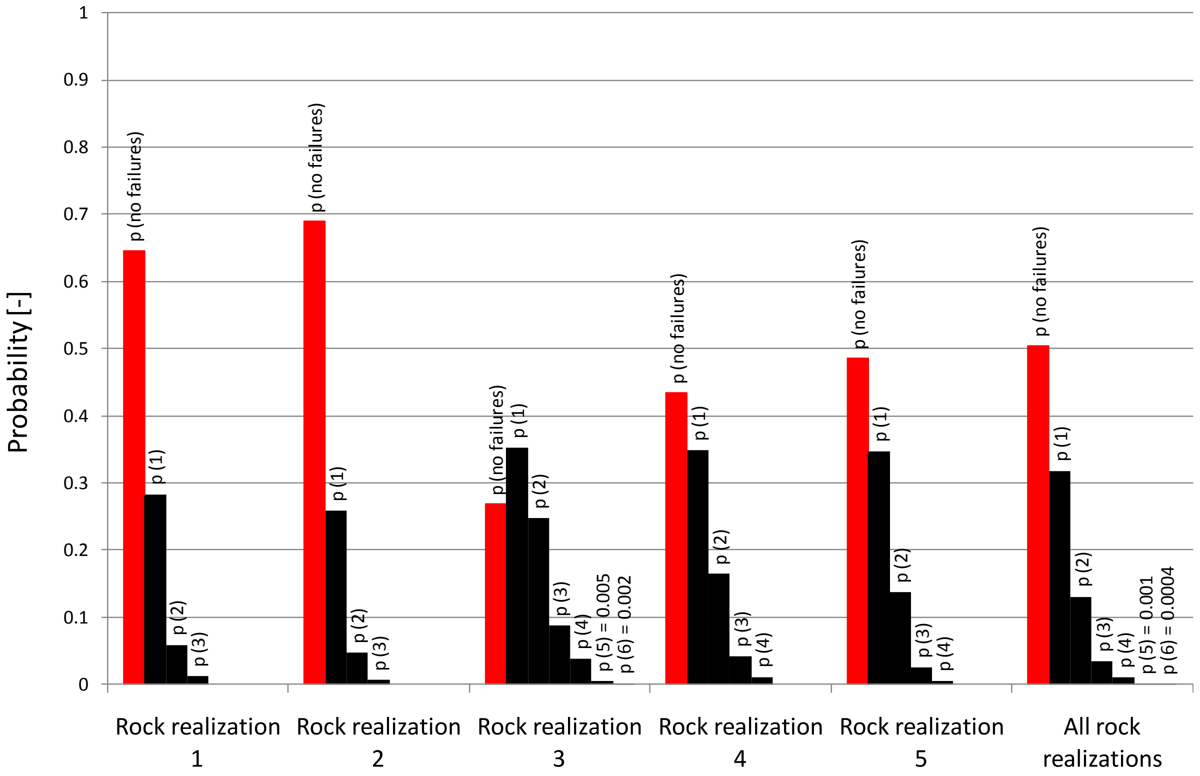
The statistically correct way of determining the variability of dose results this variation gives rise to is to determine the number of failed canisters, failure times and transport properties in each fracture network realizations, and then run a number of radionuclide transport and dose realizations for each of these. The former will essentially cover aleatory uncertainty related to the natural variability of the host rock conditions whereas the latter is essentially related to epistemic uncertainty where the fuel matrix conversion rate is a main contributor.

In summary, the following procedure was followed to evaluate one of the calculation cases:

1. Generate a realization of the network of water conducting fractures in accordance with the statistical model of the hydrogeological model variant. This yields a flow rate and transport related rock properties for each of the 6,000 canister positions.
2. Sample a groundwater sulfide concentration for each of the 6,000 positions. Determine, based on the flow rates and the sulfide concentrations, which (if any) of the 6,000 canisters will fail due to corrosion during the one million year assessment period; and the associated failure times. Repeat this step 1000 times to capture the variability caused by the distribution of sulfide concentrations, yielding 1000 sets of canister failure times and associated transport conditions.
3. For each of the 1000 sets generated in step 2, determine the dose consequences by running 1000 realizations of the radionuclide transport and dose model, sampling the additional parameters required by the model.
4. Repeat steps 1 through 3 for a number of realizations of the network of fractures. There are typically between 5 and 10 such realizations.

The result in terms of the number of failed canisters of this considerably more elaborate calculation scheme is shown in Figure 6 for the 5 realizations of the fully correlated hydrogeological model variant (see Figure 2). This variant was chosen since it gave rise to the highest doses in the safety assessment. Figure 6 shows the distribution of the number of failed canisters for each of the rock realizations when the sulfide concentration for the 6,000 positions is sampled 1,000 times. Also the results when combining all realizations is shown. As seen, there is a considerable probability that no canister will fail, and 6 canisters fail in the sulfide realization yielding the highest failure count. The mean number of failed canisters is 0.733, which is in good agreement with the theoretically correct 0.729 obtained with the simplified calculation scheme.

Figure 6. Distribution of the number of failed canisters for each of the five realizations of the fully correlated hydrogeological model.



The distribution of the dose at 10^6 years for each rock realization is shown in Figure 7. These distributions were obtained by running each of the 1,000 canister failure realizations (for a particular rock realization) 1,000 times, sampling the parameters related to radionuclide release and transport. Note that the zero dose outcomes are not shown in the distributions. Hence the areas under the distributions are not unity, but reduced in accordance with the probability of a zero dose outcome.

Figure 7 also shows the distribution obtained with the simplified scheme aiming at calculating a correct mean value. As seen, the mean value calculated with that scheme, 1.01 µSv, is in good agreement with that calculated from all the realizations of the more elaborate scheme, 1.02 µSv, confirming that the simplified scheme yields a correct mean value. As expected, the variability of the results obtained with the elaborate scheme is higher, but the difference is not dramatic. Note that the area under the PDF curve of the simplified scheme is 1, since this distribution, by the nature of the calculation scheme, has no zero doses. Note also that the regulatory risk limit relates to the mean value and that the mean values of all the distributions are well below the risk limit of 14 µSv.

Figure 8 shows the distribution of peak dose taken over the one million year assessment period. The results are similar to those in Figure 7 showing the results at 10^6 year, which is expected since the peak dose in general occurs at the end of the 10^6 years assessment period.

Figure 9, finally shows the distribution of peak release from the near field over the one million year assessment period, converted to dose using the same biosphere dose conversion factors as for the results in Figures 7 and 8. It is noted that the near field doses are higher than the far field doses by a factor of about 8. The retention in the geosphere thus reduces the doses by less than an order of magnitude. Also the standard deviation is higher by about a factor of 8, meaning that the variability is not increased substantially by the transport of radionuclides through the geosphere. This is to a large extent explained by the fact that there is a two order of magnitude variability in the release rate of radionuclides from the spent fuel, controlling the variability of near field releases and playing a dominant role for the variability also after radionuclide transport through the geosphere.

Figure 7. Probability density function of annual effective dose at 10^6 years for each of the five realizations, R1-R5, of the fully correlated hydrogeological model.

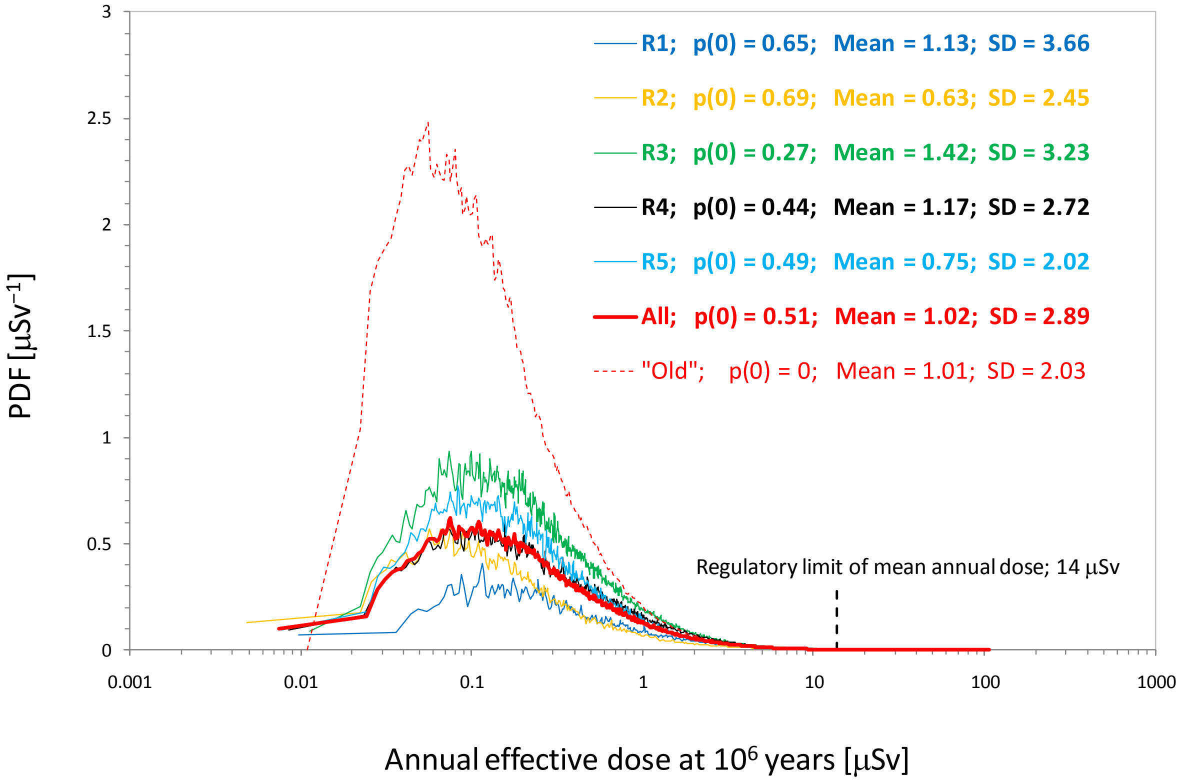


Figure 8. Probability density function of peak annual effective dose over the 10^6 year assessment period for each of the five realizations of the fully correlated hydrogeological model.

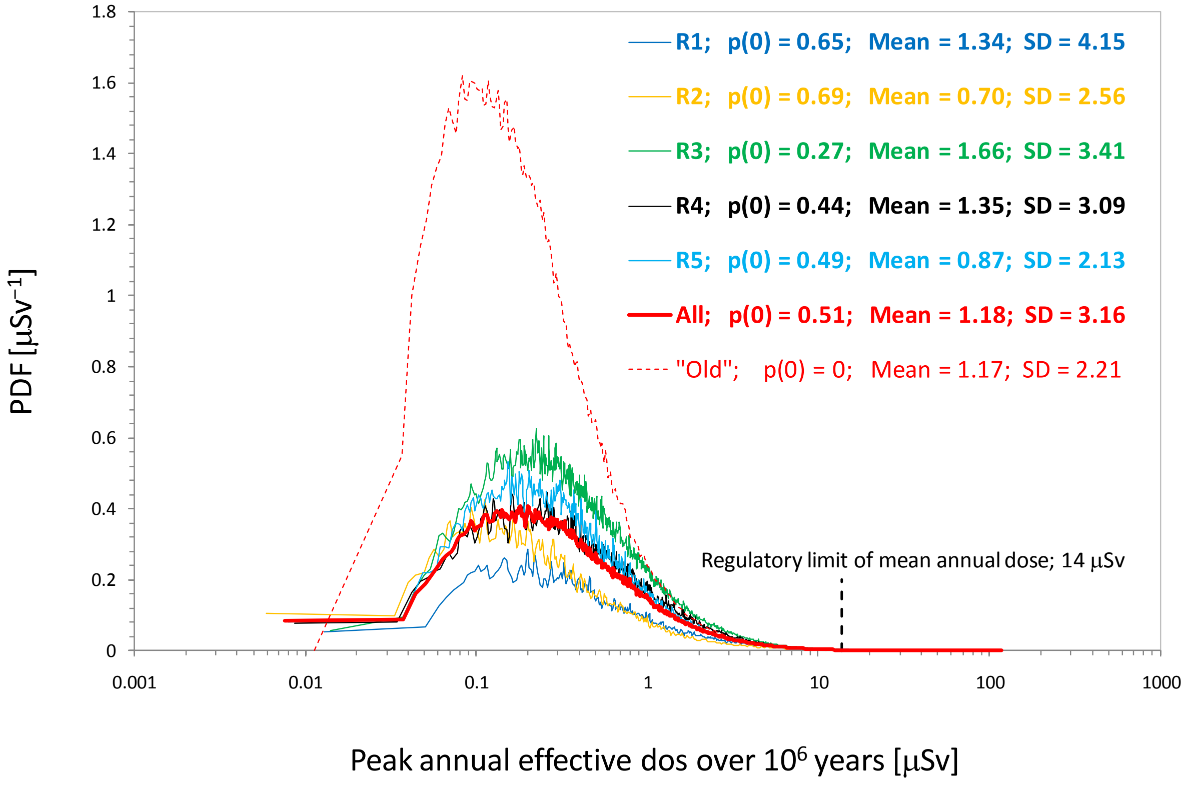
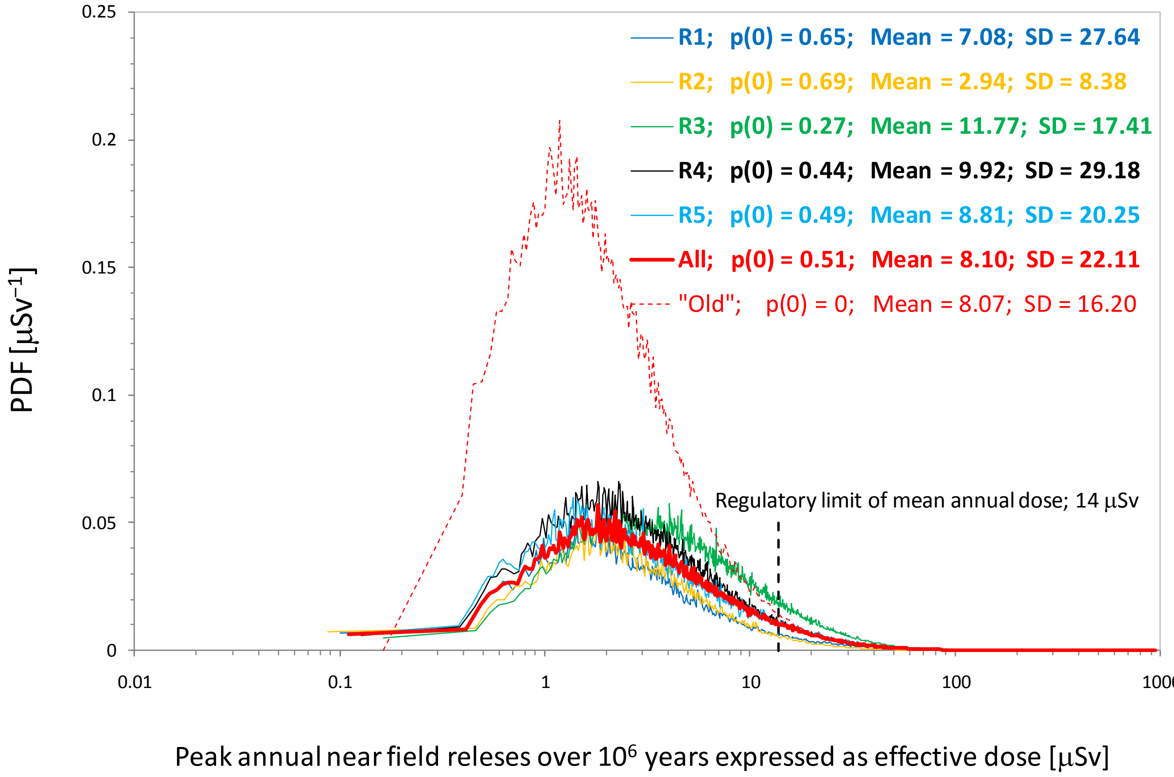


Figure 9. Probability density function of peak annual near field releases over the 10^6 year assessment period converted to effective dose for each of the five realizations of the fully correlated hydrogeological model.



3.2. Impact on Calculated Risk of Lifting Key Pessimistic Assumptions

The safety assessment in support of SKB's license applications is based on several pessimistic assumptions. In the following, a brief discussion of the potential impact of replacing some of these with more realistic assumptions is given.

First, it is noted that three of the cases in Figure 2 yield zero releases; the three variants with "no advection". These represent the situation where no advective conditions arise in the clay buffer protecting the canister. This is a possible outcome of the evolution of the buffer sub-system. The other two variants, advective conditions arising according to the SR-Site erosion model and the bounding case of initially advective conditions in the buffer, represent more pessimistic interpretations of the knowledge regarding the buffer erosion process and of those aspects of the system evolution related to the conditions required for erosion to occur, mainly low groundwater salinity and high flow rates.

Furthermore, for all cases calculated for the corrosion scenario it is assumed that a certain sulphide concentration, sampled from a specified distribution, prevails throughout the assessment period. It is only the highest sulphide concentrations in combination with the highest groundwater flow rates that yield canister failures. If, instead, the mean value of the sulphide concentrations is assumed throughout the assessment period for all canisters, reflecting a varying sulphide concentration over time, no canisters would fail. Periods with high concentrations would be followed by periods with low concentrations such that the mean value would not be sufficient to cause failures even in the deposition position with the highest flow rates.

Additionally, there is experimental evidence [4] that the oxidative fuel dissolution rate would in fact be near zero for the conditions present in the fuel tens of thousands of years into the future when the first canister failures occur. This would have as a consequence that the releases would be limited to only the fraction of radionuclides that is not embedded in the fuel matrix. This would lead to a reduction in calculated doses by several orders of magnitude for most of the assessment time, whereas doses caused by more rapidly released radionuclides, occurring during limited times after canister failures, would be unaffected.

Finally, in the safety assessment, it is assumed that the fuel matrix is converted according to a specified rate distribution. Uranium released in this conversion process is assumed to get reduced back to UO_2(s) on iron and its corrosion products in the canister and there lead to the production of U-238 daughter nuclides such as Ra-226. All such daughter nuclides are assumed to be readily accessible for release, whereas a more reasonable assumption would be that the released uranium would re-crystallize and that the subsequently generated daughter nuclides would be contained in the UO_2 grains rather than being released. An assumed particle size of only one micrometer would lead to a reduction in the release rate of e.g. Ra-226 and its daughter products Rn-222, Pb-210 and Po-210, see Figure 3, by about a factor of 1000. The reduction in calculated total dose from all nuclides would be around one order of magnitude.

4. CONCLUSION

Key elements in the probabilistic radionuclide transport and dose calculations in support of SKB's license application for a spent nuclear fuel repository in Sweden have been summarized. The results demonstrate that a safe repository of the proposed KBS-3 type can be built at the Forsmark site in Sweden.

This paper has also confirmed that the simplified probabilistic calculation scheme adopted in the SR-Site assessment yields correct mean values of the calculated dose, the entity of interest regarding compliance with Swedish regulations. The variability is, as expected, somewhat reduced compared to that obtained with the more elaborate scheme used here, yielding a correct variability.

The paper also points to several factors treated pessimistically in the probabilistic assessment and gives a rough estimate of the potential reduction in dose that could result if more realistic assumptions were assumed. All these factors are assessed to yield reductions of at least an order of magnitude.

5. REFERENCES

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