What if criticality in the final repository?

1   Introduction

In the safety assessment SR-Site (SKB 2011), supporting SKB's licence application for a final repository at the Forsmark site, it was concluded that nuclear criticality will not occur in the final repository, either during the operational or during the post-closure phase, including situations where a canister is assumed to be water filled.

In its review of SKB's licence application, the Swedish Radiation Safety Authority, SSM, has requested SKB to give an account of the consequences of nuclear criticality in the post-closure phase as a residual scenario, in accordance with the general recommendations concerning the application of SSM FS 2008:21. This Memo provides the requested information.

2   General system development

Since nuclear criticality in the post-closure phase of the final repository was argued to be ruled out in the safety assessment SR-Site (SKB 2011, section 13.3), some assumptions in violation of the conclusions of the safety assessment are required in the definition of a residual scenario to illustrate consequences of nuclear criticality. This is in line with the procedure for defining so called "what if" scenarios in a safety assessment.

In the following, it is assumed that the spent fuel configuration in a canister is such that it will become critical if the canister is filled with liquid water, contrary to the conclusions in SR-Site.

For water to enter the canister, it is required that the canister containment fails, and the two failure modes that could not be ruled out in SR-Site are considered. These are *i*) failures caused by sulphide corrosion, at an enhanced rate due to the loss of the protecting buffer through erosion and *ii*) failures due to earthquake induced secondary shear movements in fractures intersecting the deposition hole.

The following sequence of events is considered in a deposition hole where the buffer is missing due to buffer erosion.

- The canister containment fails, in the form of an opening in the copper shell, caused by sulphide corrosion.
- Water intrudes into the canister insert and slowly fills the void volume, including that between the fuel elements. Even for a deposition hole with a high groundwater flow rate of 100 L/yr, it would take several years to fill the 1 m^3 (SKB 2010a, Table 4-4) canister void, if all water flowing into the deposition hole is assumed to enter the canister interior. Any realistic description of the size of the corrosion induced opening, the transport resistances in the canister interior and the role of corrosion of the cast iron insert including impacts on water flow of the expected evolution of hydrogen in that

corrosion process, suggests that it would take hundreds of years to fill the canister interior with water.

- At some point, when a sufficient amount of water has entered the canister, it is assumed that the fuel configuration becomes critical, i.e. that $k_{eff}= 1$. Since, in reality, no critical configurations will be allowed in the canister, it is in this what-if case assumed that this occurs for an almost completely water filled canister, i.e. for one with an almost maximal amount of moderating water.

- Fission commences and the temperature increases. As the temperature increases k_{eff} is expected to decrease due to the negative temperature coefficient of the configuration with respect to criticality, caused by the nuclear Doppler effect and the thermal expansion of the moderating water. More water is then required in order to resume critical conditions. The further development is determined by the competition between the inflowing water, leading to an increased k_{eff}, and the increasing temperature, yielding a decrease in k_{eff}. The system is expected to evolve slowly due to the negative feedback loop and the slow rate of water ingress. In principle, a certain water level in the canister should correspond to a given fuel temperature, up to the boiling temperature of water under repository conditions. A slow increase in water level and hence k_{eff} is expected until the boiling temperature is reached. For the hydrostatic pressure of 5 MPa at a depth of 500 m, the water boiling point is 264 °C (Weast and Astle 1979). The system is maintained in steady-state at this temperature and a certain, constant power is generated by the fissions. If the temperature rises slightly beyond the boiling point, some water is expelled from the canister through boiling, rendering the system subcritical. Additional water enters and the cycle is repeated. This would be a "near-steady-state" situation where the temperature is again close to the boiling point of water and where pulses of energy release rates slightly higher than those in the steady-state situation are compensated by intermediate periods of subcritical conditions.

In a case where the buffer is assumed to be in place, e.g. if failure occurs due to a shear movement in a fracture intersecting the canister position, a similar sequence of events is considered, but with an even lower water filling rate due to the low hydraulic conductivity of the buffer. As water enters the canister through the gas-tight buffer, the internal pressure will increase as the void volume decreases. Hydrogen generation due to corrosion of the cast iron insert is also expected to contribute to the pressure build-up and may in the long term lead to considerable internal overpressures (SKB 2010b, section 2.3.1). The buffer, if functioning as expected, is able to contain overpressures of up to 20 MPa. Should, however, criticality in the buffer lead to temperatures above the boiling temperature at the external hydrostatic pressure, this will cause boiling of the water outside the buffer. This would lead to drying of the buffer and loss of its gas-tightening ability. Hence, the internal pressure would be brought back to the external hydrostatic pressure and the boiling temperature of water in the canister interior would be stabilised at that corresponding to the external hydrostatic pressure also in this case.

An additional case of the what-if scenario would be one with a rapid increase in k_{eff}. In the highly improbable, postulated case of criticality in a failed canister where k_{eff} is rapidly increased from 1 to 1.1, the energy would be released instantaneously. Rechard et al. (1997) have analysed the energy release that would be expected from a criticality event at Yucca Mountain. A fast assembly of critical mass is estimated to result in $\sim 10^{18}$ fissions, corresponding to an energy release of 18.7 MJ. The rapid event leads to fracturing and redistribution of the fertile material such that the system is expected to be sub-critical after the event. A rapid release of this energy is comparable to the use of 4.5 kg of TNT, which would result in a failure radius of 0.8 m in fractured tuff (Rechard et al. 1997). Thus, even at Yucca Mountain, where the waste contains higher levels of fissile material than Swedish spent nuclear fuel, such an event would cause insignificant damage to the immediate rock. This case is not further treated in this Memo.

In the following the steady-state situation envisaged above is further considered.

3 Steady-state critical conditions in a failed canister

3.1 Maximum temperature

Since liquid water is required in the canister for criticality conditions, the canister temperature, increased by the assumed fission reactions, cannot exceed the boiling temperature of water at the hydrostatic pressure, P, of around 5 MPa prevailing in the final repository for temperate climate conditions. This temperature, T_B, is around 264 °C (Weast and Astle 1979). (This temperature is similar to the operational temperature in a BWR fission rector, which is typically 285 °C.)

3.2 Maximum power

From an estimated maximum temperature at which criticality conditions can be maintained, a maximum energy generation rate from the fission process can also be estimated. If the canister is maintained at the maximum temperature, then the heat transport away from the canister at that temperature must be balanced by the power developed in the fuel. The heat transport is in this case through heat conduction in the rock, neglecting the minor contributions from the slowly circulating ground water.

It is straightforward to calculate the heat transport away from a spherical volume maintained at a constant temperature in an infinite medium, whereas the corresponding calculation for the canister's cylindrical shape is much more involved. Therefore, a case with a spherical body of radius 1.2 m maintained at a constant temperature is considered. Such a body has an area of $4 \cdot \pi \cdot 1.2^2$ m$^2 \approx 18.1$ m^2 and a volume of $4 \cdot \pi \cdot 1.2^3/3$ m$^3 \approx 7.2$ m^3, whereas the canisters' area and volume are 17.8 m^2 and 4.2 m^3, respectively.

The time dependent temperature $T(r,t)$ in an infinite medium (here the host rock) outside a spherical volume of radius R maintained at a constant temperature T_R is given by (Crank 1975, Equation 6.60)

$$\frac{T(r,t) - T_0}{T_R - T_0} = \frac{R}{r} \operatorname{erfc}\left(\frac{r-R}{\sqrt{4\kappa t}}\right) \tag{1}$$

where T_0 [°C] is the initial, assumed constant, temperature in the rock and κ [m^2/yr] is the thermal diffusivity of the rock. This expression is valid when the rock thermal conductivity and heat capacity are independent of temperature, which is not strictly true for the temperatures of interest here. Such second order effects are, however, disregarded in this stylised scenario.

From this expression for $T(r,t)$, an expression for the rate of energy transport away from canister, $q(t)$, required to maintain the surface at T_R, can be derived by considering the heat flux per unit area over a spherical interface of radius r centred at $r = 0$, $\phi(r,t)$. This is obtained by differentiating Equation (1) with respect to r and multiplying by the thermal conductivity of the rock, k:

$$\phi(r,t) = -k\frac{\partial T}{\partial r} = -kR(T_R - T_0)\left(\frac{-1}{r^2}\operatorname{erfc}\left(\frac{r-R}{\sqrt{4\kappa t}}\right) - \frac{1}{r\sqrt{\pi \kappa t}}\exp\left(-\frac{(r-R)^2}{4\kappa t}\right)\right) \tag{2}$$

Evaluating this expression for $r = R$ and multiplying by the surface area of the sphere of radius R yields the power developed in the system at steady state as

$$q(t) = k4\pi R^2 (T_R - T_0)\left(\frac{1}{R} + \frac{1}{\sqrt{\pi \kappa t}}\right) \tag{3}$$

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With the following data for the host rock at Forsmark (SKB 2011, Figure 10-14)

$T_0 = 11.2$ C
$k = 3.57$ W/(mK)
$c = 2.06$ MJ/(m^{3}K); heat capacity of the rock
$\kappa = k/c = 55$ m^2/yr

one obtains a time dependent (peak) power in the canister for a canister temperature of 264 °C according to Figure 1. As seen in the Figure, the power exceeds 20 kW only during the first few days and then decreases to a final value, q_{Final}, of around 14 kW in 10 years. The final value readily follows from evaluating Equation (3) in the situation when the time dependent term has vanished:

$$q_{Final} = k 4\pi R(T_R - T_0) \tag{4}$$

The limited cooling capacity of the host rock, together with the boiling temperature of water at the relevant pressure, thus puts a fundamental limit on the steady-state energy generation rate in an assumed critical canister in the final repository.

Furthermore, a power of 14 kW developed by the 2 tonnes of fuel in the canister corresponds to a specific power of 7 kW/tonne. This is more than three orders of magnitude lower than the circa 30,000 kW/tonne developed in a power reactor where typically 100 tonnes of fuel may generate 3GW of thermal power output.
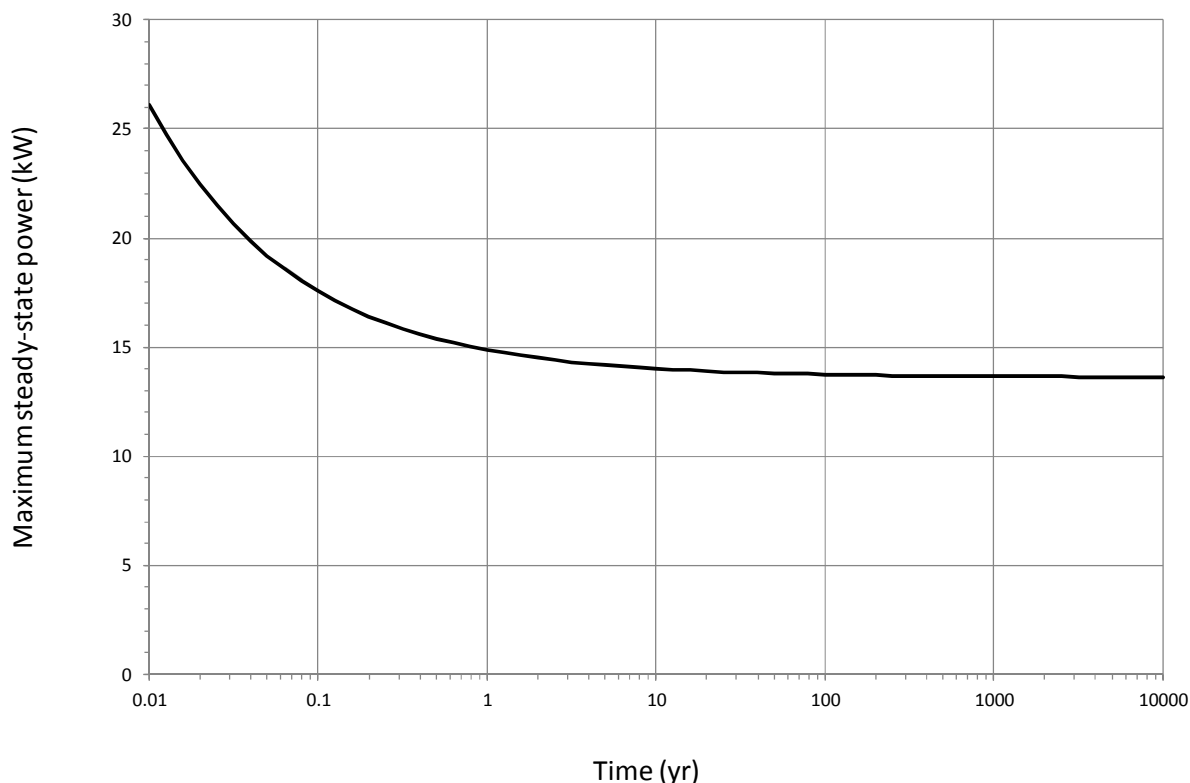

Figure 1. Steady state power for a canister maintained at 264 °C. The background temperature of the rock is 11.2 °C in this calculation.

3.3 Impact on adjacent deposition positions

The temperature increase at adjacent deposition positions can be estimated with Equation (1). The canister spacing at the Forsmark site is smallest, 6 m, in the main deposition area. Figure 2 shows the calculated temperatures in the three positions nearest to the canister that is assumed to be critical at 264 °C, i.e. in positions 6 m, 12 m and 18 m away from the critical canister. The calculation is done under the assumption that the thermal pulse from the fuel residual power at deposition has declined to insignificance. As seen in the figure, the centre of the nearest deposition position experiences a temperature of around 60 °C after around 100 years if the temperature of the critical canister is 264 °C. In the buffer parts closest to the critical canister in the adjacent deposition position are at a distance of 5.2 m, and here the temperature is around 70 °C. This is considerably lower than the thermal design requirement for the buffer, that states that the buffer temperature must not exceed 100 °C for the buffer to maintain its safety functions in the final repository (SKB 2011, section 8.3.2). For canisters farther away, the temperature increases are considerably lower. It is also noted that it would take a temperature of 400 °C (at the 1.2 m radius sphere surface) to reach 100 °C in the nearest buffer parts of an adjacent deposition position. Since this would require internal canister pressures of more than 25 MPa, the margin to a temperature exceeding the design temperature of the buffer is considerable.

These results indicate that nuclear criticality in a deposited canister does not have a detrimental effect even on the most closely positioned canisters. It is also noted that the increased rock temperatures may affect the rock nearest to the critical canister negatively as regards mechanical properties and effects. However, the rock mechanical properties are important in particular to ensure intact containment of the canister and in the criticality case the canister is failed by definition.
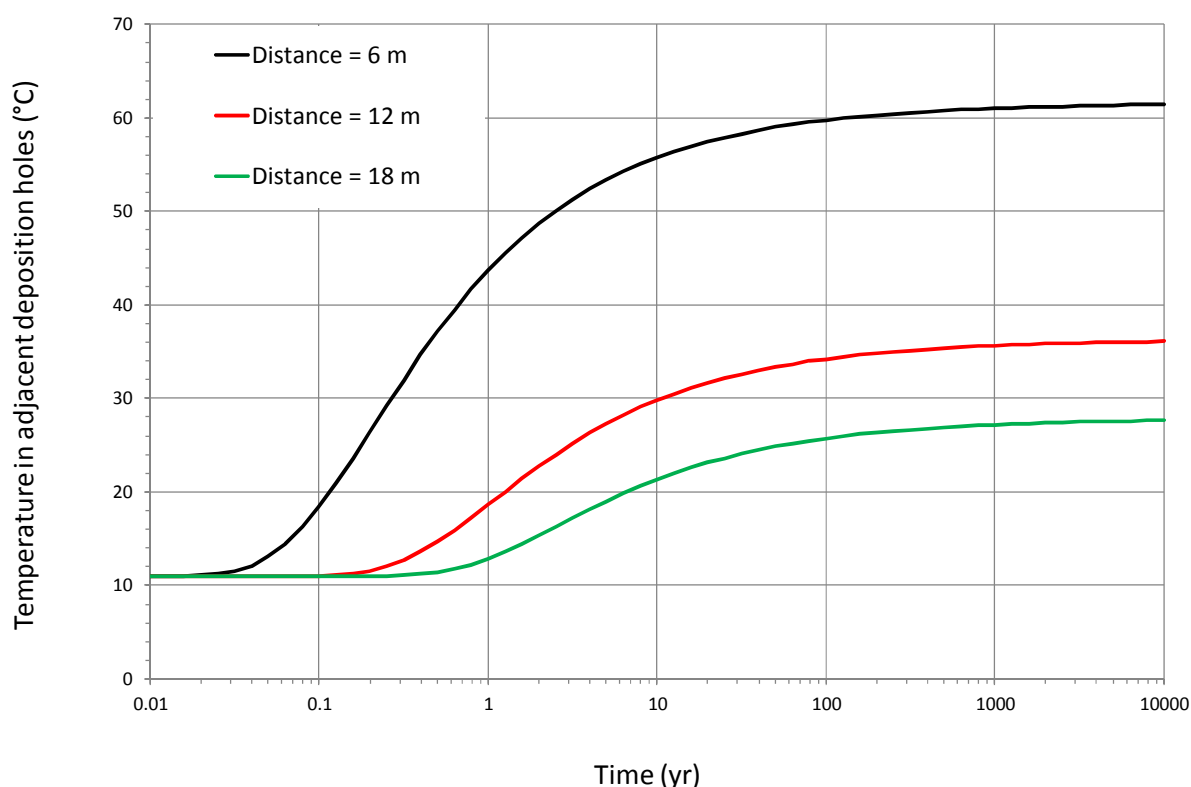

Figure 2. The temperature development in the three nearest deposition positions to a critical canister maintained at 264 °C. The three positions are located 6, 12 and 18 m from the critical canister, respectively. The background temperature of the rock is 11.2 °C in this calculation.

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3.4 Longevity of critical conditions

Since the occurrence of criticality in a canister is hypothetical, a discussion of the longevity of such conditions must also be based on unrealistic hypotheses. The following simple reasoning is offered.

The maximum steady-state power generated in the fuel is of the order of 10 kW according to section 3.2. The fertile material remaining in a deposited canister (mainly the inventories of U-233, U-235 and Pu-239), together with the amount of Pu-239 that would be generated by neutron capture of U-238 in the fission events, put a fundamental limit on the total energy released in the fission events. If it is assumed that the canister to become critical is deposited with 2 tonnes of un-irradiated fuel, the total energy generated could at most be comparable to that generated in a power reactor by the same fuel. Assuming that the critical conditions in the deposited canister ultimately results in a high burn-up of 50 MWd/kgU, this corresponds to a duration of the critical conditions of
$2000 \text{ kg} \cdot 50 \text{ MWdkg}^{-1} / 10 \text{ kW} = 10^7 \text{ days} \approx 3 \cdot 10^4$ years. While this is a completely unrealistic scenario, it still suggests that the fertile material left in a deposited canister could in principle be sufficient to sustain a power generation of ~10 kW for thousands of years.

4 Estimate of radiological consequences

This section aims at giving a rough illustration of the consequences in terms of radiological dose for a case where the fuel in a damaged canister is critical in a steady state situation as described above.

The case considered is the central case of the so called erosion/corrosion scenario in the safety assessment SR-Site (SKB 2011, section 13.5). The failed canisters in that case have lost their protetcting clay buffer and experience high groundwater flow rates, such that all radionuclides released from the fuel are assumed to be immediately released to the geosphere, in the absence of the clay buffer. A key limiting factor for this case is therefore the rate at which the spent fuel matrix is dissolved. A discussion of fuel dissolution under steady state criticality conditions is therefore given in section 4.1. Based on the conclusion regarding the fuel dissolution rate, a recalculation of the central corrosion case with data representing criticality conditions is reported in section 4.2.

4.1 Fuel dissolution under criticality conditions

The following is an attempt to estimate a dissolution rate for the spent fuel inside a copper canister in the hypothetical case when the neutrons released by the fissions become more than the ones consumed and the system becomes critical. Since canister failures occur after hundreds of thousands of years in the corrosion scenario modelled here, it is assumed in all cases that the cladding is no longer intact so that water is in contact with the fuel matrix. A criticality event would cause changes in temperature, chemical composition and radioactivity of the fuel. When considering dissolution rate of the fuel, the change in composition is important mainly due to the effect it has on the radiation field, and the radiolysis of water.

According to section 3.2, the thermal power of a container which becomes critical is estimated to be of the order of 10 kW, based on the boiling point of water at the repository depth and the thermal conductivity of the system. Fissions will create new short-lived isotopes (mainly beta- and gamma-emitters), as well as alpha emitters. An estimation of the radiation field can be made based on the power developed, which is more than a thousand times lower than the power of a typical BWR reactor per unit mass fuel (section 3.2). The amount of oxidants produced inside the container by the **low linear energy transfer (LET) radiations** (beta and gamma) should be much smaller than under operational conditions in a BWR reactor. Hydrogen is routinely added in LWR reactors at the concentration of a few ppm, in order to neutralise the oxidant production by low LET radiation. Under strictly anoxic conditions, even small amounts of hydrogen neutralise completely the oxidant production by low LET radiations (Pastina and LaVerne 2001). In a study of the influence of hydrogen addition in the reactor water, Pastina et al. (1999) estimate that 0.23 ppm H_2 (corresponding to $1.1 \cdot 10^{-4}$ M) is sufficient to inhibit formation of oxidants at 200 °C in a 70 kW reactor.

The formation of new **alpha emitters** as a result of the neutron capture is expected to increase the alpha dose rate of the fuel. One may assume that given the much lower neutron flux in a critical canister as compared to a nuclear reactor, the generation of higher actinides as e.g. Cm-244 will be very low. Based on the ~5000 times lower power as compared to an LWR reactor, it is reasonable to assume that the content of alpha emitters during the whole critical period will be lower than in freshly discharged LWR fuel, i.e. the fuel in the critical canister will have an alpha dose rate of less than 0.12 Gy/s, or ca 200 MBq/g (which corresponds to the alpha dose rate of 100 y old LWR fuel of 38 MWd/kg U burn-up) (Table 1 in Nielsen and Jonsson, 2006, SKBdoc1221579). In a similar case for a critical canister of 14 kW power highly enriched in ^{235}U fuel, Rechard et al. (1997) estimate the amount of ^{237}Np build-up (from ^{235}U in highly enriched fuel) to be about 0.5% of the amount which builds up during reactor operation. Assuming an elevation of the alpha dose rate up to this level, it is relevant to consider the effects of hydrogen. Hydrogen will be expected to form as the iron of the canister insert corrodes in the anoxic environment of the repository. It is estimated that the water inside a failed canister will contain circa $1 \cdot 10^{-4}$ mol/L hydrogen (SKBdoc 1385070), which is sufficient to suppress the estimated radiolytic oxidation from the newly formed alpha-emitters (Trummer and Jonsson 2010).

Another factor which has to be considered in the case of the critical canister is the radiolytic effect of **fission fragments** (Olander 1995). Very little is known about the radiolysis of water by these heavy particles and their effect on fuel dissolution is therefore uncertain. However, the escape depth of the fission fragments is estimated to 6-8 μm (IAEA 2000), which is lower than for alpha particles, and the amount of recoiling fission fragments in a critical canister is expected to be much lower than for fuel with damaged cladding in an LWR reactor. The oxidant production in water by these very high LET particles as measured by Fricke´s dosimeter seems to be not larger than e.g. alpha particles (LaVerne 2000, LaVerne and Schuler 1996). The conclusion is therefore that the effect of these particles is probably minor, but in order to account for the lack of any direct experimental data, the fission fragment radiolysis is assumed to increase the dissolution rate by about an order of magnitude as compared to dissolution in their absence.

The **higher temperature** is expected to increase the oxidative dissolution rate of the fuel, as shown in fuel dissolution studies in presence of air. On the other hand, fuel tests in the presence of $8 \cdot 10^{-4}$ M dissolved H_2 and 70 °C show a much faster reduction of the pre-oxidized U(VI) and other redox sensitive nuclides than in experiments carried out at room temperature. Also, these experiments show no measurable oxidative dissolution. Above 100 °C hydrogen is expected to be activated thermally (Ekeroth et al. 2004, Parks and Pohl 1988) and together with Fe(II) should contribute to the neutralisation of oxidants produced by alpha radiation. As indicated by the study of Sunder et al. (1990), at 100 °C with $1.6 \cdot 10^{-4}$ M dissolved H_2, a reduction of the UO_2 surface in presence of alpha radiation is observed, while a clear oxidation occurs under Ar. Further, **the solubility** of UO_2 is practically the same in the interval 100-300 °C (Parks and Pohl 1982) and independent of temperature, while the non-oxidative dissolution rate of UO_2 does not affect fuel dissolution for repository relevant flow rates (SKBdoc 1372969).

In conclusion, in spite of the criticality, the conditions inside of the container are expected to be predominantly reducing, due to the presence of large amounts of metallic iron and its products of water corrosion. It seems reasonable to assume that the alpha dose rate does not become higher than 0.12 Gy/s due to build-up of alpha emitters. However, to account for uncertainties, including the effects of radiolysis by fission fragments, a constant fractional dissolution rate in the range of 10^{-7} to 10^{-5} per year is suggested. Thus, for fuel dissolution in a canister which becomes critical, a fuel dissolution rate with a triangular probability distribution in $\log_{10}$-space with lower limit, best estimate and upper limit of 10^{-7}, 10^{-6} and 10^{-5}, respectively, is adopted in the hypothetical criticality case. It is also noted that a fully elaborated discussion of the fuel dissolution rate including its uncertainties is not possible given the hypothetical nature of the criticality conditions in the canister.

4.2   Radionuclide transport calculations

In order to roughly illustrate the consequences of criticality conditions in the final repository, the central corrosion case in the SR-Site safety assessment, described in section 13.5.4 in the SR-Site main report (SKB 2011), has been recalculated with the log-triangular distribution of fuel dissolution according to the previous paragraph. This means that the fuel dissolution rate is assumed to be ten times higher than in the central corrosion case, and that this higher rate is assumed to prevail throughout the calculation period irrespective of how long the criticality conditions are assumed to last. The radionuclide inventory is not altered. This is motivated by the fact that the inventory used in the central corrosion case corresponds to a mean burn-up of around 40 MWd/kgU, implying that no substantial increase in the radionuclide inventory would occur even if all the remaining fertile material in the canister were to be consumed.

The calculation is probabilistic, with all other input data described in section 13.5.4 in SKB (2011). Simplified, analytical transport models are used in the calculation; these have been demonstrated to yield results similar to those of the more detailed numerical models used in SKB's compliance calculations, see section 13.5.10 of SKB (2011).

Figure 3 shows the results of the central corrosion case for non-critical conditions in the canister, i.e. the case described in section 13.5.4 of SKB (2011). The result of the corresponding case calculated with numerical models is given in Figure 13-18 of SKB (2011) and a comparison shows that the two models yield quite similar results, justifying the use of the analytical models for this case.
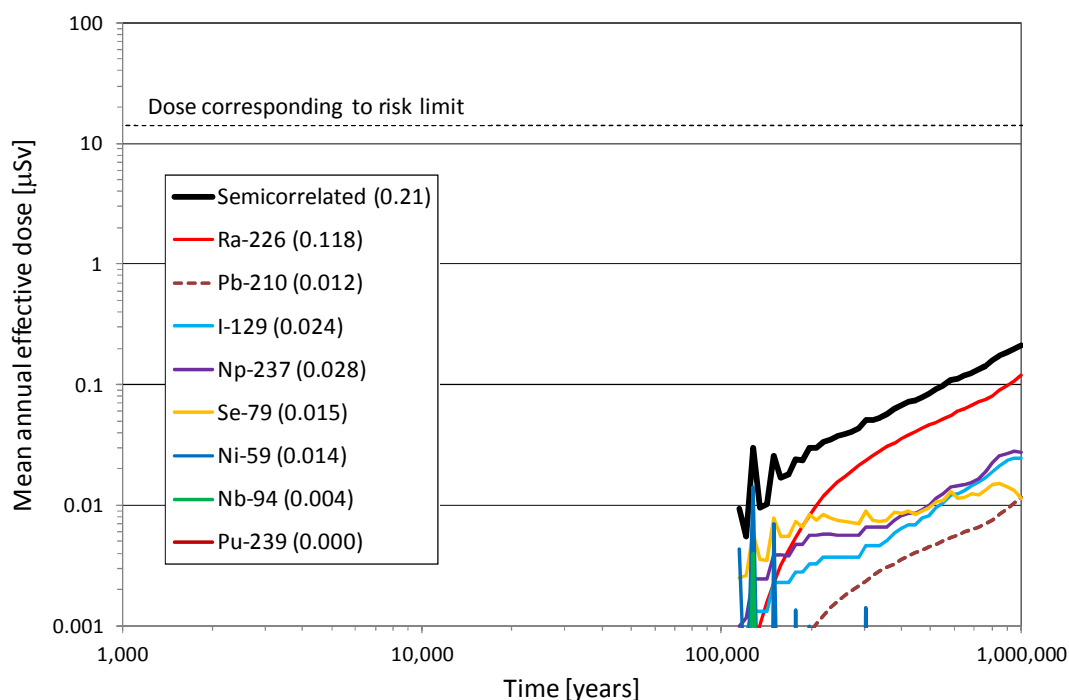

Figure 3. *Far-field mean annual effective dose for the central corrosion case in the safety assessment SR-Site, calculated with a simplified, analytical model. The figures in the legends are peak mean annual effective doses over one million years, given in μSv. The result of same case calculated with numerical models is shown Figure 13-18 of SKB (2011). The calculated peak doses differ by less than a factor of 1.2 between the models.*

Figure 4 shows the results of the central corrosion case, but with the assumption of critical conditions in the canister, reflected by the shift of the distribution of fuel dissolution rate by a factor of 10 towards higher values. As seen when comparing Figure 3 to Figure 4, the increase in dose is about a factor of 6. The main reason that the dependence of dose on dissolution rate is somewhat weaker than linear, is that for dissolution rates in the interval 10^{-6} to 10^{-5} yr^{-1}, the fuel matrix will, for many

canister failure times in the central corrosion case, be fully dissolved within the calculation period. The impact of the increased dissolution rate is thus not effective throughout the calculation period, resulting in a less than linear dependence on dissolution rate.
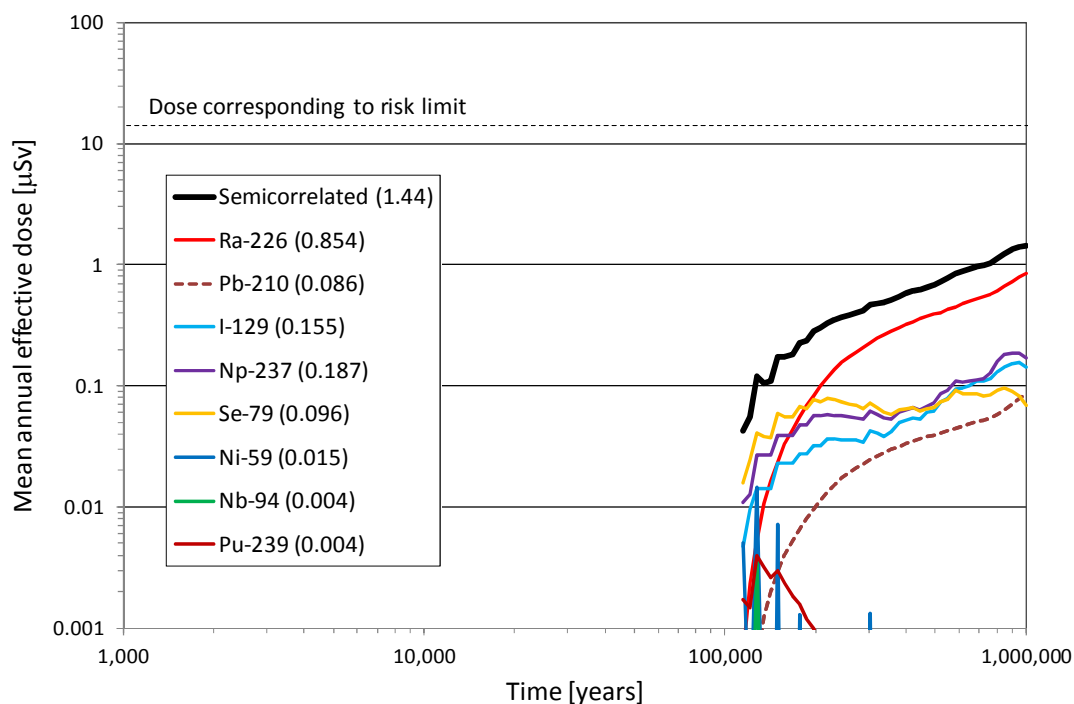

Figure 4. Illustration of the effect of steady state critical conditions in the canister for the probabilistic central corrosion case. The case is the same as that in Figure 3, except that the distribution of the fuel dissolution rate is shifted by a factor of 10 towards higher values.

While fission is ongoing, also the production and release of radionuclides with shorter half-lives than those in the presented calculation need to be considered, in particular Cs-137 and Sr-90, both with half-lives close to 30 years. With a fission power of 14 kW and an average energy release per fission event of 215 MeV (Mulligan 1980), the fission rate is around $4 \cdot 10^{14}$ s^{-1} or $1.3 \cdot 10^{22}$ yr^{-1}. The fission yield of these nuclides is approximately 0.06 per event (Mulligan 1980), meaning that the production is approximately $6 \cdot 10^{11}$ Bq/yr of each. This means that the inventory of these nuclides in the fuel matrix will grow from the onset of critical conditions, until the production rate is balanced by the decay rate of the nuclide in question. A steady state inventory of approximately $2.5 \cdot 10^{13}$ Bq is established after about one half-life. With an average fuel dissolution rate of 10^{-6}/yr (see section 4.1), this corresponds to a release rate of $2.5 \cdot 10^7$ Bq/yr. Taking into account the decay in the geosphere transport and the dose conversion factors for Cs-137 and Sr-90, this yields annual effective doses of around 0.1 and 1 µSv/yr, respectively. These doses are restricted to the period of tentatively a few thousand years when fission is ongoing, due to the short half-lives of the nuclides involved. They can therefore be compared to doses arising from the initial pulse release of the instantaneously accessible fraction of nuclides in the case without criticality. Such doses are given in Table 13-5 of SKB (2011), where it is seen that e.g. I-129 gives rise to a dose of approximately 4 µSv/yr. It is also noted that the temperature in the fuel pellets will be only marginally above the canister temperature of 264 °C in the criticality case, as opposed to the corresponding temperature during reactor operation that is well above 1000 °C. This means that the diffusion of elements like Cs and Sr is much slower, why the generation of an IRF fraction has not been considered here.

5   Conclusions

An illustration of the consequences of nuclear criticality in a canister during the post-closure phase of a KBS-3 final repository at the Forsmark site has been given in this Memo.

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Cases of rapid increase of k_{eff} beyond 1.0 are only briefly discussed. For such cases, the discussion suggests that the energy release is much smaller than in a steady-state case, that the rapid event renders the system sub-critical and that the damage to the near field rock is limited.

For a steady-state case, the following is concluded:

- The temperature in the critical canister is limited by the boiling point of water under repository conditions. This temperature is around 264 °C for a hydrostatic pressure of 5 MPa.
- The power developed in a critical canister is limited by the finite capacity of the rock to carry away the generated heat by thermal conduction. This means that a canister maintained at 264 °C is calculated to develop a power of around 14 kW in the host rock in Forsmark.
- The resulting increase in temperature in the host rock is not sufficient to cause any damage to the bentonite buffer in adjacent deposition positions.
- The fuel dissolution rate in a steady-state criticality situation is judged to possibly be an order of magnitude higher than for sub-critical conditions.
- Using this higher fuel dissolution rate, the radiological consequences for a steady-state criticality situation in a failed canister are calculated to be about a factor of six higher than for the corresponding sub-critical situation.

Finally, it is recalled that this is an unrealistic "what if" case, building on an assumption of criticality, whereas in the safety assessment SR-Site, it was concluded that criticality in the post-closure phase of the final repository can be ruled out.

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