Strål
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Swedish Radiation Safety Authority

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Technical Note

2015:51

Initial State of Spent Nuclear Fuel

Main Review Phase

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SSM:s perspektiv

Bakgrund

Strålsäkerhetsmyndigheten (SSM) granskar Svensk Kärnbränslehantering AB:s (SKB) ansökningar enligt lagen (1984:3) om kärnteknisk verksamhet om uppförande, innehav och drift av ett slutförvar för använt kärnbränsle och av en inkapslingsanläggning. Som en del i granskningen ger SSM konsulter uppdrag för att inhämta information och göra expertbedömningar i avgränsade frågor. I SSM:s Technical note-serie rapporteras resultaten från dessa konsultuppdrag.

Projektets syfte

Det övergripande syftet med projektet är att ta fram synpunkter på SKB:s säkerhetsanalys SR-Site för den långsiktiga strålsäkerheten för det planerade slutförvaret i Forsmark. I denna rapport granskas och utvärderas SKB:s redovisning av bränslets initialtillstånd med fokus på beräkningar av radionuklidinventarium, resteffekt och ytdosrat.

Författarnas sammanfattning

Flera av de koder som används av SKB i beräkningarna av radionuklidinventariet, resteffekten och ytdosraten, såsom Origen-S och MCMP 5.2 är väl etablerade både inom kärnkraftsverkan och inom forskarsamhället och koderna valideras och jämföras regelbundet.

Det allmänna intrycket är att de fall som beaktats i beräkningar av ytdosraten är relevanta med konservativa antaganden. De indata till simuleringarna ges och resultaten presenteras på ett, till stor del, begripligt sätt.

Den allmänna bedömningen är dock att de granskade rapporterna behöver kompletteras med ytterligare uppgifter.

Ett urval av de uppgifter som bör genomföras i rapporteringen är att:

- Bredda omfattning i rapporteringen genom att ta mer än ett referensscenario för driften av de svenska kärnkraftsreaktorerna.
- Komplettera data och resultat i rapporteringen med en strikt hantering av osäkerheter. Felfortplantning bör utföras för relevanta resultat (resteffekt, kriticitet, dosering osv.)
- Komplettera rapporteringen med mer information om optimering av inkapslingsprocessen
- Komplettera rapporteringen med separata kapitel för analys, diskussion och slutsats. Utan dessa kapitel, tjänar rapporteringen lite syfte. SKB bör visa hur de tolkar sina resultat, vad som påverkar resultaten och vilka följder och konsekvenser resultaten har
- Komplettera rapporteringen med förklaringar på om och hur SKB planerar att experimentellt verifiera simulerade resultat, och om villkoren för sådana mätningar (vilka egenskaper kommer att mätas, vilken omfattning mätningarna kommer att vara, hur lång tid det tar för olika mätningar, nödvändiga noggrannhet och preciseringar etc.)

- Förbättra hanteringen av följande strukturella frågor i rapporten; syfte med de utförda beräkningarna, beskrivning av hur simuleringarna utfördes, användning av referenser, förklaringar till figurer och tabeller, förklaringar till uppgifter (tabeller) i bilagorna.

Projektinformation
Kontaktperson på SSM: Jinsong Liu
Direktupphandling
Diarienummer: SSM2013-5485
Aktivitetsnummer: 3030012- 4112

SSM perspective

Background

The Swedish Radiation Safety Authority (SSM) reviews the Swedish Nuclear Fuel Company's (SKB) applications under the Act on Nuclear Activities (SFS 1984:3) for the construction and operation of a repository for spent nuclear fuel and for an encapsulation facility. As part of the review, SSM commissions consultants to carry out work in order to obtain information and provide expert opinion on specific issues. The results from the consultants' tasks are reported in SSM's Technical Note series.

Objective

The general objective of the project is to provide review comments on SKB's postclosure safety analysis, SR-Site, for the proposed repository at Forsmark. This technical note reviews SKB's reporting of the initial state of spent nuclear fuel, with the emphasis on calculations of radionuclide inventory, decay power and radiation at canister surface.

Summary by the authors

Several of the codes used by SKB in calculating the radionuclide inventory, decay power and radiation at canister surface, such as Origen-S and MCMP 5.2 are well established both within the nuclear industry and within the scientific community and the codes are validated and benchmarked repeatedly.

The general impression is that the considered cases in calculations of the radiation at canister surface are relevant and concern conservative assumptions. The input data to the simulations are given and results are presented in a, largely, understandable way.

The general assessment, however, is that the reviewed reports need to be supplemented by additional information.

A selection of the information that we would like to see included in the reports are:

- Broadening the scope of the report by including more than one reference scenario for the operation of the Swedish nuclear fleet.
- Complement the data and results in the reports with a rigorous handling of uncertainties and that error propagation must be per formed for relevant results (decay heat, criticality, dose rates etc.).
- Complement the report with more information on the optimisation of the encapsulation process.
- Complement the reports with separate chapters for analysis, discus sion and conclusion. Without these chapters, the reports serve little purpose. SKB must show how they interpret their results, what influ ences the results and what implications and consequences the results have.

- Complement the report with explanations on if and how SKB plans to experimentally verify the simulated results, and on the conditions for such measurements (what properties will be measured, what the scope of the measurements will be, what are the required measure ment times, the necessary accuracies and precisions etc.).
- Improve the handling of the following structural issues in the report: the purpose of performing included calculations, descriptions of how the simulations were performed, the use of references, explanations to included figures and tables, explanations to included information (tables) in the appendices.

Project information
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Contents

1. Introduction

The Division of Applied nuclear physics at Uppsala University has been asked to review parts of the SKB application to build an encapsulation facility and a geological storage. The assessment has been performed by the five individuals who are listed as authors to this report. The three documents reviewed by us are:

- /SKB TR-10-13/ *Spent nuclear fuel for disposal in KBS-3 repository* (including the relevant references therein),
- /SKBdoc 1221579/ *Aktivitetsinnehåll i kapslar för slutförvar (Radiation activity in canisters for final disposal)*, and
- /SKBdoc 1077122/ *Strålskärmsberäkningar för kopparkapslar innehållande BWR, MOX och PWR bränsleelement (Calculation of radiation shielding for copper canisters containing BWR, MOX and PWR fuel assemblies)*.

The specific questions raised by SSM are that we should make a scientific judgment of each document, assess the credibility of the report information, especially the calculations of radio nuclide inventories, radiation doses and radiation strengths from the spent nuclear fuel. We are also asked to give an assessment on general use and validation of the software tools used to provide results in the reports. It is not part of this review to make our own calculations or simulations. Based on this information and the announced scope of the document, we have identified areas of high interest and will assess the document with respect to these areas:

- General comments on the report including problem formulation, objective, limitations, scope of the report and scientific approach,
- Calculations of radio nuclide inventory in the spent nuclear fuel,
- Calculations of decay heat,
- Calculations of dose rates and radiation protection,
- Calculations of criticality, and
- The use of software tools and their validation.

We have requested extra documentation, and successfully received the following reports:

- /SKBdoc 1193244/ *Criticality safety calculations of disposal canisters*
- /SKBdoc 1222975/ *Beräkning av fissionsgasfrigörelse för bränslet i slutförvaret (Calculation of release of fission gas from the fuel in a final disposal repository)*.

We have chosen to structure this report with the three documents listed above as separate chapters.

2. SKB TR-10-13, Spent nuclear fuel for disposal in the KBS-3 repository

/SKB TR-10-13/ is the main document for the review. The purpose of that document is to describe the types and quantities of fuel to be encapsulated, fuel properties and parameters, fuel handling procedures including inspections and sealing and to describe the expected values of parameters of importance for the assessment of the long term safety of the encapsulated nuclear fuel.

2.1. SKB's presentation

The beginning of the report contains a section on objectives and limitations of the document, together with an overview of how the different production reports describing how the KBS-3 repository is designed, produced and inspected. The focus of this specific report is however the description of the planned operation of the Swedish nuclear power plants and the resulting nuclear fuel inventory, calculations of nuclide inventories and radiation activities in copper canisters, as well as handling of the spent nuclear fuel and encapsulation of it based on fuel type and decay heat load.

2.1.1. General comments on the report

The document is a technical report describing several aspects of the fuel and its handling in connection to the future encapsulation facility. The identified objectives of this report are relevant and the stated limitations explain the reference scenario chosen for the investigations. The report has a clear structure which is easy to understand. We have however a number of general comments on the report.

Our view is that this is not a scientific report in the sense that it lacks analysis, reflections and discussions of shown results, that is has very sparse use of references and is completely void of peer-reviewed references. The main report of about 100 pages contains only 11 references, all of which are documents produced by SKB themselves. A large number of facts, in many cases without uncertainty analysis and sometimes without clarification of selection criteria and their motivations, are given. This complicates the assessment of the quality of the work for us as external reviewers and is, in our mind, a general short-coming.

Early on in the report, there is a schematic figure describing how a few reports fit together. It is however not clear from this figure or the text that /SKB TR-10-13/ is indeed the "Spent fuel report" although it is implicit from the information on page 14, section 1.4. A related problem is that there are some reports mentioned with bold text, such as *SR-Site*, *SR-Operation* and *Design premises long term safety*, without specifying proper references to them. The reader cannot be expected to know which reports these are, the way of referring to them needs to be properly done. We were also not able to understand which results in this report that are of relevance to other reports in the full SKB application (e.g. nuclear safeguards aspects), nor what other reports in the application that discuss e.g. long term safety aspects of the repository (e.g. radiotoxicity and storage time).

On a general level, we find that the technical level of detail in the report is somewhat limited. There exists a wide selection of tabulated data without explanations of what they have been or should be used for. In other areas, we have not been able to find any of the information we expected to find, nor to references where we could look it up. Examples of such cases are e.g. general aspects of the terms “safety” and “long term” which appear to be key words in this report although it is not mentioned what aspects of safety that are included in this report and which aspects that are covered by other reports (radiotoxicity, copper corrosion, climate change etc.). Another example is that the report includes results on calculations of activity levels in the canisters, but there is no information on what this information is used for nor how it is related to the radiotoxicity of the fuel or its storage time.

A feature that is frequent both in /SKB TR-10-13/ and in some referenced reports is that some statements are repeated several times. This may be motivated as the same text is included in multiple sections and different contexts. The problem we have discovered is that several times the reference contains the same general statement without further explaining the matter, or it may contain a reference to yet another report with the same type of general statements. This makes it very difficult to assess the reliability of the work. Specific examples related to criticality are described in subsection 2.2.1 of this assessment.

2.1.2. Chapter 2: Spent nuclear fuel to be deposited in the KBS-3 repository

This section of the report is dedicated to describing the spent nuclear fuel to be deposited. It described the Swedish nuclear power reactors, fuel quantities together with properties such as burnup and age, and fuel dimensions.

The fuel burnup is shown in three histograms (Figure 2-2), while the fuel age is shown in two scatter plots (Figure 2-3 and 2-4) where black markers show already residing fuel in Clab and red markers show the expected future fuel. The black markers are associated with no uncertainty in burnup, while the red markers are shown with assumed standard deviations in burnup of ±3MWd/kgU. From a scientific standpoint, we are not sure what is meant by an *assumed* standard deviation, nor do we understand where the numerical value of 3 MWd/kgU comes from. This should be clearly explained in this report, or in a referenced report. Furthermore, the caption mentions “batch average discharge burnup”, but does not specify how many spent nuclear fuel assemblies that are included in one such batch. In addition, the already existing fuel in Clab should be associated with some uncertainty in burnup, this is not mentioned at all.

Section 2.3.2 in /SKB TR-10-13/ explains (page 22) that the variability of the elements N, Cl, Ni and Nb “in the different kinds of BWR and PWR assemblies has been investigated by randomly selecting a number of fuel types and comparing the amounts of construction materials in these assemblies with the amounts in Svea 96 Optima 2 and Areva 17x17 respectively.” It is assuring to read that the amounts of these elements are similar for all BWR and PWR assemblies, but we expect the determined variability to be properly quoted. We also expect a more detailed description of how the random sampling from other fuels was done. If the comparison was limited to investigating random product sheets, what is the reason for not having investigated all of them?

We find that there is a limitation in the report by considering only one reference scenario for the foreseen future operation of the Swedish nuclear power plants. The reason is that any implications of changes in operation will influence not only the capacity, but also the operation of the final repository by requiring new calculations of e.g. criticality and decay heat.

2.1.3. Chapter 3: Requirements for the handling of the spent nuclear fuel

This chapter of the report aims at describing the requirements on the handling of the fuel before the encapsulation, including design premises imposed on the canister by the spent nuclear fuel. The chapter is split into one part dealing with the long term safety requirements, and one other section dealing with requirements of the operation of the KBS-3 system.

Our general comment on this part of /SKB TR-10-13/ is that we would have liked to see more details on the actual requirements related to the operation of the KBS-3 system. Specifically we expected more detailed information on the fuel selection process and on foreseen needs for measurements and inspections.

Long-term safety

Long-term safety is connected mainly to the decay heat from the spent nuclear fuel and to criticality.

Regarding decay heat, section 3.1.1 (page 25) and 4.4 (page 30) state that this parameter is one of the important parameters for long-term safety of the repository, and specifies an upper limit of maximally 1700 W per canister. It is also written that assemblies shall be selected so that the limit is not exceeded. How the suitable fuel assemblies are in practice selected, or what the result of this selection process is, is however not described.

Further, in section 6.4 (page 56), it is written that "the current selection of assemblies is made so that the total calculated decay power of the assemblies in a canister does not exceed 1,650 W". A safety margin to the limit of 1700 W has been added in order to "ensure that the actual decay power confirms to the criteria 1,700 W". In the same section it is written that the uncertainty of calculated decay power is estimated to 2 %. (See other comments on the 2 % uncertainty statement, in section 2.1.4 of this assessment.) Assuming a 2 % uncertainty (one sigma), about 30 percent of canisters filled with 1650 W will actually have decay powers lower or higher than 1650 W +/- 33 W, i.e. outside the range 1617-1683 W. There is no discussion on how to handle this situation, e.g. on how to reduce the uncertainty on estimated decay power. Also, there is no discussion on why a limit of 1650 has been set when it clearly should be either smaller, or the confidence level of the uncertainty should be larger.

The last paragraph in section 3.1.1 mentions that it may be possible to allow for higher decay power in peripheral deposition holes. It is not elaborated on why this is so but one may assume that the peripheral position implies a lower temperature in the buffer since there a less neighbours that contributes to the heat. The report would benefit by clarifying this issue.

It can also be noted that a discussion on the relation between calculations of decay power and calculations of activity content is not available.

We are aware of that criticality issues are the focus for reviews by other instances, but as it is also part of the documents we are reviewing, we have decided to include some comments here. We have found the information on criticality rather sparse in /SKB TR-10-13/, but as /SKBdoc 1193244/, *Criticality safety calculations of disposal canisters*, is referred to several times in this context, we give dedicated comments on that report here. The purpose of the document /SKBdoc 1193244/ is to show that the criticality criterion with a neutron multiplication factor never exceeding 0.95 if the canister is filled with water can be met if burnup credit is included. A lot of the criteria and methods are based on the US NRC regulatory requirements for transportation and storage of spent fuel.

- In /SKBdoc 1193244/ section 3 (page 6), there is a good introduction to the report and the used codes, but the uncertainties due to different nuclear data libraries should be evaluated and discussed. Some of the used codes and modules should be explained better, and it can be noted that no input files or other relevant material are included in the document in order to enable verification of the calculations. In the end of section 3 there is an unclear statement regarding how many neutrons that have been followed per generation, it should be clarified. There is also a claim that small changes in the results when varying input parameters are not due to the change of parameters but due to the statistical uncertainty. This is not certain, and although the effect is small there is no information here that supports the assumption.

- In /SKBdoc 1193244/ section 5.2 (pages 9-10), the Tables 2-6 show material specifications for cast iron, steel, bentonite and the bedrock (in the document it is called "continental earth crust"). The given contaminations or mixtures should be investigated and varied. In order to know the impact of impurities it would be good to compare with the results when using a pure material. Materials such as bentonite and the bedrock may vary in composition and some discussion regarding the effects of different compositions should be included.

- In /SKBdoc 1193244/ section 5.4 (page 11), Table 7 shows the main parameters used in the burnup calculations. It is essential to state how much each parameter can vary during normal operation, and if the effects of such variations have been investigated.

- In /SKBdoc 1193244/ section 6.7 (page 23), Figure 9 seems to indicated a negative feedback loop as increased temperature will reduce k-effective and thus no overheating should occur. But what assumptions have been made?

- In /SKBdoc 1193244/ section 6.10 (page 26), Table 22 shows the calculated k-effective for a partly loaded PWR disposal canister that is being filled with water. The last line of Table 22 shows the case for when there is only one fresh fuel assembly in the canister. Remarkably, k-effective is above one. As this would violate any design criteria for inserting fresh fuel into a nuclear reactor, it should be properly explained and verified.

- In /SKBdoc 1193244/ section 9.12 (page 40) we have several comments:
    - Section 9.12 is denoted "Calculation uncertainties". We would here expect some discussion on calculation uncertainties related to the use of the Scale 44-group ENDF/B-V library, and possibly also other sources of uncertainties, but the section only deals with the uncertainties in calculating k_{eff}.

- There is a statement "In these cases the calculations were done with 3003 neutron generations." It would be good with some explanation why calculations with 3003 neutron generations were considered sufficient.
- There are two statements "The constant K=1.72 is picked from /11/." and "The constant K=2.026 is picked from /11/." It would be beneficial if the role of the constant K in the estimate of the one sided tolerance limits is properly explained. We are aware of the risk of mix-up with the effective multiplication constant k_{eff} that is also used in the text, but as the given reference 11 /Owen/ has two different definitions for the constant, "K_P" and "k" depending on if the mean and standard deviation are known or not, it would be helpful if K is written properly.

* In /SKBdoc 1193244/ section 9.12 (pages 45-48) we have several comments:
    - It is not clear from the information given if the data from Table 33 was calculated in this document or in the given reference 14 /NUREG/CR-6811/. A look in the reference /NUREG/CR-6811/ reveals no information about experimental data, while references 12 which is /ORNL/TM-12667/ and 13 which is /ORNL/TM-13317/ may be correct. As both these references contain data for a number of different cases an explanation about the details would be welcome, as well as a justification of whether these data are of relevance or not for the present case.
    - Irrespective of how the data has been obtained and/or calculated, the method for determining the uncertainties should be explained, as well as why it can be justified to calculate standard deviations for data sets with as few as three measurements.
    - We are not convinced that the method where the deviation between selected experiments and simulations are used as an estimate of the uncertainty of the simulation is correct. One cannot generalise the deviation between selected experiments and calculations to the uncertainty of the calculation. See a more thorough discussion in section 2.2.1 regarding this issue, where similar problems are discussed for the main document /SKB TR-10-13/.
    - Similar problems in the three bullets above are observed for Table 36 and the given reference 15 /ORNL/TM-13315/.

Operation of the KBS-3 system

This section of the report mainly discusses the encapsulation procedure, which is discussed in detail in a later section of this report (see subsection 3.1.4).

We have no comments on subsection 3.2.3 of /SKB TR-10-13/ apart from stressing that we have not been given access to the safeguards document, which makes an assessment here impossible.

Other comments

In section 3.1.3 (page 26), it is very good that it is stated clearly that the design of the copper canisters allow deviations due to deformed fuel assemblies. We assume that it follows that all non-regular fuel will fulfil the requirements with respect to criticality, though the subsection “Check of criticality” on page 33 indicates that in the worst case it may be necessary to reconstruct individual assemblies. Although this is considered a last resort, some elaboration on the subject would be welcome.

In section 3.1.4 (page 27), the design premise for reducing the amount of nitric acid is not fully explained, and the information given in /SKB TR-09-22/ *Design premises long-term safety* does not give supporting information. On what basis are the levels of > 90% argon and less than 600 g water given? What are the uncertainties?

2.1.4. Chapter 4: The handling of the spent nuclear fuel

This part of the /SKB TR-10-13/ report is stated to describe the transport and delivery of fuel assemblies, their interim storage and the selection of assemblies and encapsulation.

Transport and delivery of fuel assemblies

SKB presents in this part of the report information on the flow of spent nuclear fuel to the interim storage. The text in this section is very brief and since we do not regard transportation to be our main competence area, we have not comment on this part of the report.

The inspections mentioned in this part of the section are not dealt with in any detailed way, and our comment on this is that it is not possible to properly assess this section of the report. We lack information on what part of the documentation that will be surveyed, for what purpose and with which criteria the fuel assemblies will be visually inspected, what fuel properties that will be stored in the nuclear safeguards database as well as what “properties of importance for the operation of SKB’s facilities and the long-term safety” that will be stored in other databases. Regarding burnup, it is mentioned that SKB will receive calculated estimates of this parameter and that it will be accurately measured. There is however no description of whether this will be measured for all fuel assemblies, or using what methodology or equipment or what the results will be used for.

Interim storage

We have not reviewed the contents of this section.

Selection of assemblies and encapsulation

This part of the document describes the requirements for the fuel selection and encapsulation process. It is described that this will be done with respect to decay power, criticality, radiation dose at the canister surface and minimisation of the number of canisters and number of fuel movements.

Regarding the decay heat determination, section 4.4.1 (page 31) contains the preliminary selection of fuel assemblies to put in a canister is based on calculations of decay heat. It is mentioned that a well-documented and verified code will be used for the calculations. As it will be many years before the final fuel selection for each copper canister is made, it may be wise not to specify at this time what particular code that will be used. Codes, underlying models, and quality of nuclear data may improve over time. But it should be shown that such a code exists at present, how it has been verified and that it has the verified capability to calculate the decay heat for the fuel assemblies to be encapsulated, according to the reference scenario. It is further mentioned in the same section (page 33) that the preliminary selection is to be made so that the sum of calculated decay heat in the canister does not exceed 1650 W to allow for uncertainty in calculated decay heat compared to the true decay heat in the fuel assemblies. It is stated, with a reference, that the uncertainty has been estimated to about 2 %. We would like to know what confidence level that has been used for this estimate of that uncertainty. The 2 % uncertainty is somewhat contradicted by the fact that the uncertainty of the radionuclide inventory is calculated with a 12/20 percent uncertainty for fission products/actinides in /SKBdoc 1221579/ (see comments on that document).

We found no uncertainty of decay heat measurements using calorimetry presented in the report. We would also like to see how much the uncertainties in the calculated nuclide inventories influence the uncertainty in the decay heat determination. Furthermore, a sensitivity analysis on how decay heat and dose rate on the surface of the canister depends on uncertainties in the calculated nuclide inventories would provide a better understanding on the reliability of the results.

Regarding criticality, we understand from section 4.4.1 (page 33) that the criticality criterion will be ensured by calculating the (by SKB denoted) “loading curve” and comparing with individual fuel assemblies. We found no indication of any kind of verifying measurement of this. It would be good if the safety margins with respect to the calculations are explained in order to motivate that no measurements are needed. If measurements are needed, this would be a suitable place to indicate it.

Furthermore, we understand from section 4.6.2 (page 35) that the criticality calculations have been performed with the assumption of maximally of 600 g of water will be present inside the canisters. It is not mentioned here where the prerequisite comes from, or how it is verified that no fuel assembly will hold more water. The reference to the report /SKB TR-09-22/ *Design premises long term safety* gives no further information except for the repeated criterion.

We did not find any information in this subsection on how the fuel will be handled in order to comply with safeguards requirements. Regarding inspections, there is a very brief description of all stages of the fuel handling that is planned. We have not been able to assess whether this inspection is adequate or not since there is no detailed information on what tools that will be used for the purpose, or to what degree they are inspected. It is also unclear in many subsections, such as section 4.2.2, what properties of the spent nuclear fuel that are of importance in the inspection. In other places as in subsection 4.3.2 it is mentioned that “…if required it is possible to measure…” but there is no information on how this should be done or what the measurement requirements are.

On a general level regarding the encapsulation process, we have the following two comments:

- In section 4.4.1 (page 32) Figures 4-3 and 4-4 describe the loading curves for BWR and PWR canisters, but there are no uncertainties specified, for instance with error bars and explanations. The distinction between the two loading curves should also be explained explicitly in the text. As the information on this important subject is rather limited we have decided to look closer at the given reference, /SKBdoc 1193244/ see section 2.1.3. for comments on this report.

- What precautions are taken to ensure that SKB or the power plant operators do not accidentally associate a spent fuel assembly with the wrong assembly documentation? What impact could this have on the calculations of decay heat, criticality and fuel handling?

2.1.5. Chapter 5: The canisters to be deposited

This section of the report is stated to describe the number of canisters required to deposit the spent fuel based on the requirements on the handling and selection of assemblies. This work is done by simulations, supposedly accounted for in /SKBdoc 1221567/, but neither in this document nor in /SKB TR-10-13/ is there any explanation of the code being used, or a description of the used algorithm. Probably the simulation can be performed with a relatively simple sorting algorithm and be performed in a spreadsheet program like Excel, but it is remarkable that the procedure has not been specified. It should also be noted that the referred report /SKBdoc 1221567/ is incorrectly named ”*Simulering av fyllning av kapslar för slutförvaring av utbränt kärnbränsle*”, while the correct name of the report is ”*SKB - Simulering av inkapsling av använt kärnbränsle för slutförvaring i KBS-3-förvar*”. The lack of description of this process render it difficult to e.g. interpret figures such as Figure 5-1 in section 5.1 (page 37) since we have no information on how the data was obtained and whether it is applicable to all types of fuel to be encapsulated. Similarly, it is not possible to verify the conclusions drawn from Figures 5-2 and 5-3 in section 5.2, since we do not know if the uncertainties in the decay heat calculations for the selected fuel assemblies are small enough. Also, a concern of ours is what the consequences are if the real encapsulation rate differs from the simulated one by being either faster or slower.

Regarding criticality, it is mentioned that this cannot occur for the selected assemblies. This statement is based on results from /SKBdoc 1193244/, “*Criticality safety calculations of disposal canisters*” and is connected to (among other things) the materials in the disposal canisters as well as the bentonite clay and the bedrock. In this referenced report, we have not found any references regarding the steel material compositions except for various SS numbers. In addition, the reference contains only one single sentence summarizing those variations in bentonite composition and also mentioning that the bedrock gives only small or no changes in the reactivity of the disposal canister. However, there is no description of where this conclusion comes from.

The encapsulation process

The encapsulation is briefly mentioned in subsection 3.2.1 in /SKB TR-10-13/, but is further elaborated on in chapter 4 of the same report. The full fuel selection process of the fuel is, without any details, described in section 4.4.1. The simulated results of the fuel selection process at encapsulation are presented in chapter 5 and appendix C. We have chosen to collect all our comment on the encapsulation process from the different sources of information and place them here in our assessment report.

It is stated in /SKB TR-10-13/ that the assemblies were selected to give a combined heat power of 1700 W at the time of disposal. It is also claimed that for each simulated year of encapsulation, the inventory of assemblies in the Clink facility and their decay power was calculated. The results of those simulations, or references to other reports containing these results have not been found. What we have found in appendix C, are only two tables with the results of the simulated encapsulation (Table C3 and C4), followed by 14 uncommented tables that, due to lack of explanation, that serve no purpose in this document.

The decay heat conclusions in document /SKB TR-10-13/ appear to be based on results from /SKBdoc 1221579/. These results are summarised in Figure 5-1, which presents the minimum cooling time required to give less than 1700 W as a function of burnup. One can see from this figure that e.g. a BWR fuel having a burnup of 34.2 MWd/kg requires a minimum cooling time of 20 years before being encapsulated. In the referenced document /SKBdoc 1221579/, one finds several cases where the isotope composition at the time of encapsulation is claimed to be reported. This is however not correct, which can be exemplified as follows. Consider the "low burnup BWR fuel"-category which is characterised by a burnup of < 30.7 MWd/kgU. Judging from copper canister optimisation and Table 11 in /SKBdoc 1221579/, a cooling time of 20 years has been considered for these fuels (the motivation behind this choice of cooling time is not clear, but it may be related to the decay heat limitation of 1,700 W per canister). Looking at Figure 4 in /SKBdoc 1221579/, it appears that at year 2023, when the encapsulation is supposed to start, there are very few assemblies with a burnup around 30 MWd/kg and 20 years of cooling time. Rather, the average cooling time at 2023 appears to be more than 30 years for low burnup fuel (the exact number is unknown as the calculated inventory in Figures 2, 3, 4 and 5 of /SKBdoc 1221579/ is not presented in a histogram or a table but only in a scatter plot). There are even very low burnup fuel assemblies (BU ≈ 10 MWd/kg) with a cooling time of 47 years. The conclusion from this is that the decay heat at the time of encapsulation, as reported in /SKBdoc 1221579/, is highly overestimated for most fuel assemblies. Chapter 5 in TR-10-13, and specifically table C3 which presents the simulated encapsulation of BWR assemblies, further assumed that mainly low burn up fuel (BU < 38 MWd/kg) will be encapsulated between 2023 and 2037. This means that in 2037, the average cooling time of low burnup fuel will be, as we understand it, around 44 years with some as long as 61 years. The most dominating heat sources after 20 years cooling time is the decay chains of Sr-90 and Cs-137 with half-lives of around 30 years. After 44 years, their combined decay heat will be roughly half of that after 20 years, and after 61 years it is around 25%. Hence, in combination with the actinides, the decay heat at the actual time of encapsulation could in fact be less than half of the maximum at 1,700 W. At the same time, during the final years of encapsulation, most canisters will not be filled since the combination of high burnup (BU > 42 MWd/kg) and short cooling time (25 years for the final core of O3) means that the maximum decay heat of 1700 would be otherwise exceeded.

In this respect, the information in Appendix C (Table C3 and C4) seems to be somewhat contradictory to that presented in /SKBdoc 1221567/. In the reference it is stated "Den gemensamma utgångspunkten för de olika strategierna var att simulera inkapsling av bränsleelement med relativt hög resteffekt tidigt i syfte att så långt som möjligt ha kvar bränsleelement med låg resteffekt att kombinera ihop med de bränsleelement med hög resteffekt som tagits ur reaktorerna sent". (Eng: The common starting point for all strategies was to simulate the encapsulation of fuel elements with a relatively high decay heat at an early stage. The aim is to, as far as possible, save fuel elements with a low decay heat and combine these with high decay-heat fuel elements that are taken out of the reactors in late stage).

In summary, the filling of spent nuclear fuel into copper canisters is not optimised with respect to decay heat, or to minimizing the number of canisters. It is hence not clear to us how, or with respect to what, the encapsulation optimisation has been performed. From the results presented it seems as the decay heat is highly overestimated for the assemblies that should be encapsulated during the first 14 years. In combination with a majority of the canisters being only "half filled" during the last 20 years of encapsulations (2050-2070) it seems as if the encapsulation scheme is highly non-optimised, and is a rather wasteful utilisation of a scarce resource, i.e., the final repository.

2.1.6. Chapter 6: Initial state – encapsulated spent nuclear fuel

This section of /SKB TR-10-13/ aims at describing the spent nuclear fuel properties in terms of their radionuclide inventory, fission gas release, decay power and radiation properties at the canister surface. We have chosen to omit fission gas release in our review, due to a lack of time.

Nuclide inventory calculations

Nuclide inventory calculations are necessary in order to e.g. estimate the decay heat of the spent nuclear fuel assemblies, to calculate the activity content for the copper canisters and to estimate the radiation dose.

We find that relevant codes have been used, but we are not satisfied with the explanations by the authors on how the codes have been used.

Connecting to the purpose of making these calculations, we have looked for and not found, a motivation in the reports on what the dose rate calculations will be used for. We have noted the maximal limit of 1Gy/h for the canisters, but apart from that the motivation for the activity calculations and radiation dose estimations is rather unclear to us. We guess that the calculations are the foundation for both estimates regarding radiation protection in the handling of the fuel, as well as for calculations of radiotoxicity and hence the required storage time for the spent nuclear fuel in the geological repository, but such information is not included in the reports we have reviewed. We would however like to make a comment that this type of information would be highly relevant in these reports.

On a more detailed level, we have noted that section 6.2.2, Table 6-1 (page 43) contains 13 different nuclides out of the in total 52 nuclides that are tabulated in Appendix C. We have not understood what the selection criteria for focusing on these 13 nuclides are: if it is a combination of activity and half-life, why is Nb-94 more important than Ni-59, or why is Cl-36 more important than Tc-99? No matter what the reason is, it should be clearly stated in the report.

In subsection 6.2.3 (page 43) it is stated: "At the time for the closure of the final repository when the encapsulation and deposition is finished, the burnup, irradiation and power history and age of the assemblies in each canister will be known and the radionuclide inventory can be calculated for each individual canister. However, at this stage it is not reasonable to calculate the inventory in individual canisters." This is a reasonable approach, but to validate this approach, it would be valuable to have indications of with what precision and accuracy the nuclide inventory in each canister can be obtained. It is also unclear to us if the inventory only will be calculated using simulations, or if it also will be guided by measurements? Further descriptions are needed.

Subsection 6.2.6 discusses uncertainties in the data. Four different important sources to uncertainties are discussed. Many sources of uncertainties are however missing, e.g. thermal scattering cross-sections, engineering quantities (densities, geometry, etc.), nu-bar, angular distribution, fission neutron spectrum, Q-value etc. The claim that fission yields should be insignificant for the inventory must be investigated further. In this context, we would like to see a comparison of results in /SKB TR-99-74/ with those of an external party e.g. /Tech. Rep. 113 696/. Furthermore, the claim that a typical uncertainty is 5% for the fission products seems to be an underestimation if compared to the values in e.g. /Tech. Rep. 113 696/. Since most information on the uncertainty propagation is contained in /SKBdoc 1198314/, which is not available to us, it is impossible to make a full judgement of all the arguments presented, but from what we have read it has not been shown that the uncertainties are taken into account in a conservative or rigorous manner.

The table in Appendix C-2 is referred to in section 6.2.2 and 6.2.5 as being the total inventory or full inventory of all radionuclides. It is not clear if the tabulated data shows the nuclides given by the codes, or if they have been selected by the authors. Additional relevant questions are if any radionuclides of importance are missing in these tables, how large fraction of the total activity do the given radionuclides give rise to, and how much is missing?

Decay power

Because the radionuclide inventory is the basis for estimating the decay power, it is discussed again in this section of the document.

As stated already in the previous chapter of /SKB TR-10-13/ (section 3.1.1) SKB acknowledges that the decay power of a fuel assembly depends on the burnup, age and mass of uranium/heavy metal. In this part of the report, where the radionuclide inventory is actually calculated, we expected to find an elaboration on how the irradiation history is relevant. In section 6.2.2 it is briefly mentioned that the irradiation history can be neglected, but there is no explanation to how this result was obtained.

It is mentioned in section 6.4 (page 56) that decay heat of each assembly can be measured in conjunction with the delivery to Clink, but there is no further explanation of how it is foreseen to be done. We would also like to know how it is ensured that the intended measurement can produce results in due time, within the uncertainties needed for selection of assemblies for encapsulation.

Radiation properties at the canister surface

The given values for the radiation dose rate in Sect. 6.6. are correctly imported from the quoted document. However, there is severe criticism of /SKBdoc 1077122/ as discussed in section 4 of the assessment document. It should be noted specifically here that the referenced document only contains simulations of the radiation dose rate for one specific case, while the main part of the work concerns *equivalent* dose rates. Furthermore it is stated that in the discussed case the content of the canister would exceed the allowed decay heat. It would be very useful to have a specific section discussing the close link between decay heat and radiation dose. However, this is not done here and no reference to another section in the report that verifies this statement is made.

2.1.7. The general use of software tools and their validation

The report /SKB TR-10-13/ mentions several codes but is lacking comprehensive information about them. In chapter 6.2.6, *Uncertainties in the calculated fuel matrix radionuclide inventories*, the use of Origen-S (see reference /Herrman and Westfall 1998/) is discussed, but without proper references to the code itself. Instead there are references to other SKB reports where the calculations are reported. Origen-S is a well validated code used both in industry and scientifically, but the report and the referenced reports lack enough information in order to validate how the code has been used, and whether the referred validations are relevant for the KBS-3 copper canisters.

In chapter 6.2.7, *Uncertainties in the calculated radionuclide inventories in construction materials and crud* , the codes IndAct and CrudAct are mentioned, once again with no proper references to the codes themselves, the only reference is to /SKBdoc 1198314/ which we do not have access to. It is therefore not possible for us to validate the code or its use.

2.2. Consultants' motivation of their assessment

The motivation of our assessment is based on the requirements expressed to us by SSM in connection to taking on this assignment. They are hence listed explicitly below.

2.2.1. Scientific judgement of the report

Fulfilment of the report objectives

We have listed four objectives and evaluated whether they are met. The stated objectives are to describe:

1) spent fuel types and quantities to be deposited in the KBS-3 repository,
2) fuel properties and parameters of importance for the assessment of the long-term safety,
3) how the fuel assemblies are handled, inspected and selected for encapsulation and deposition, and
4) the initial state of the spent fuel, i.e. the expected values of parameters of importance for the assessment of the long-term safety of the encapsulated spent nuclear fuel.

Objective 1 and objective 2 are partly fulfilled, depending on what the expected level of detail is both for the selected scenario for future nuclear power, the fuel properties and the parameters of importance. We would have liked to see either a dedicated section in the report, or a reference to another document, which explained detailed information on these properties such as actual distribution of enrichments, burnup, age and power history in the fuel inventory. We also lacked information on whether these fuel parameters were solely determined by simulations or whether verifications of these estimations would be performed, in addition one may ask what the accuracy in the fuel parameter determination is and how accurately must it be known. This could be important information when designing and applying measurement equipment for the verification of these fuels.

Objective 3 is possibly met, depending on what the expected level of detail is. Subsections are entitled according to this requirement, but the following text did not reveal any details concerning e.g. how the actual fuel selection at the encapsulation is done (what software tool is used, which fuels are considered in the selection process, how is the canister optimisation actually done etc.). We would also like to have seen an analysis in this section that explains the consequences of making the simplifying assumption that identical fuel assemblies occupy all positions in the canister. As mentioned earlier in this assessment, there is very limited information on planned inspections where explanations have a general nature – "… shall be reviewed in accordance with SKB's managing system", "If there are uncertainties […], it can be measured for verification", "Inspection […] is documented by photography" etc. Without knowing more about inspections and measurements (possible instrumentation, measurement and evaluation criteria, requirements etc.) we cannot draw any conclusion on the adequacy of the proposed methodology.

Objective 4 relates to the long-term safety of the encapsulated nuclear fuel, which is in this report interpreted as connected only to the radionuclide inventory calculations for criticality and decay heat determination. There is some very general information on criticality, but it is not possible to assess the soundness of these checks or inspections without reading also the references document. There are however other safety aspects related to the copper canister and the bentonite buffer, but such effects are not even mentioned in a reference.

By considering more than one reference scenario, one of the main limitations of this report (connected to objectives 1, 2 and 4) can be avoided. The total radionuclide inventory and hence the number of canisters, is very dependent on the operation scenario for the Swedish nuclear fleet. Since only the reference scenario is under investigation here, the total radionuclide inventory calculated in e.g. Table 6-1 can only be seen as an indication of what amount of radionuclides we expect in the inventory. This needs to be pointed out in the report more clearly. In the limitations it is only stated: *"Alternative scenarios for the operation of the nuclear power plants are not included."* There are no references to these alternatives. The reader can consequently not judge if the quoted inventories or number of canisters are realistic, to what actually will be deposited.

The handling of uncertainties in the report

After having reviewed this report, it is very unclear to us how uncertainties have been handled throughout the work. What factors are by SKB perceived as critical for safety aspects, how were the uncertainties in these parameters estimated, how large are the estimates and what is the interpretation by SKB from the results? This reasoning can be applied to in principle all areas covered in this report: fuel parameters, decay heat calculations, dose rates, assumptions concerning identical fuels residing in the copper canisters etc. For this reason, we have chosen to explicitly discuss the handling of uncertainties in a dedicated subsection of our motivation for the assessment.

When uncertainties in a simulated or measured quantity are quoted, good practice is to quote the best estimate with an uncertainty and specify at which confidence level the uncertainty is quoted (typically one sigma). Alternatively, confidence or tolerance intervals can be used. Neither of these approaches have been adopted in the report. If uncertainties are at all quotes, the reader must guess what confidence level the results refers to (e.g. on page 26 "The effective multiplication factor (k_{eff}) must not exceed 0.95 including uncertainties.") This makes it impossible to judge the probability that at least one canister fails the specified requirements. Furthermore, most tables and figures are completely missing error bars. In addition, in documents where uncertainties are referred to, there is no mentioning of whether these uncertainties are systematic or random. In summary, the concept of accuracy vs. precision is not treated at all. This may have a large effect on the final safety of the repository.

In the report in a few different places (e.g. page 33, last sentence in section "*Preliminary selection*") the deviation between selected experiments and selected simulations are used as an estimate of the uncertainty of the simulation. One cannot generalise the deviation between selected experiments and calculations to the uncertainty of the calculation. The experiment can only be used to constrain the input parameters of the simulation, since most simulations depend on multiple input parameters. Consequently, the deviation between the experiment and the simulation cannot be used to quantify the uncertainty of the simulation. In addition, the experiment itself does contain uncertainties. In the case where experiments are performed on all assemblies, the uncertainty of the experiment can be used, however then the simulations would not be needed. From the report it is not clear if experiments on all assemblies are planned.

A lot of the work on uncertainties calculations performed is presented in /SKBdoc 1198314/. Since we have not had access to this document we cannot assess if the uncertainty calculation is correct. Either /SKBdoc 1198314/ needs to be made available for review, or more information from /SKBdoc 1198314/ needs to be included in /SKB TR-10-13/ if the results should be possible to review.

The use of references

Regarding the use of references, we have identified that SKB uses both open documents and non-released documents, and that almost all sources of information are produced by themselves. The use of references to reports which are difficult to access makes this review difficult. Despite the (limited) number of references listed, it is not obvious to us how the quality assurance of previously (un)published documentation has been performed; worth noticing is that in some place we even found that the referenced document contained different values than those imported by SKB for this report. In addition, we have found several examples of referencing to documents with the wrong names, and to documents which contain no more information than the original one does. In several occasions the referenced document in turn contained no further information, apart from further references that the reader had to go and look for.

Examples of bad handling of references in the report:

- We have found several examples of where /SKB TR-09-22/ has been used as a reference for criticality calculations presented in /SKB TR-10-13/, e.g. in section 3.1.2 (page 25). It is claimed that information is imported from section 3.1.4 in /SKB TR-09-22/, but in fact this reference contains no useful information except the same statement accompanied by the reference of yet another document: /SKB TR-06-09. In this second reference, we find in section 7.4 (page 190) the same information which was initially reported in /SKB TR-10-13/.

  Expecting to find more information in a second reference also mentioned in this context /SKB TR-11-01/ *SR-Site* we also read this report and found a number of similar statements as in the previous reports. Finally in section 13.3 (page 652), we found a satisfying explanation. In this case, the relevant reference with details was /SKBdoc 1193244/. As this report is referred to in section 4.4.1 (page 32) in /SKB TR-10-13/, it would have been more efficient to refer to/SKBdoc 1193244/ also here on page 25, instead of sending the reader on a long odyssey through a number of reports that refer to each other in an intricate manner.

- Related to references is the list of references at the end of /SKB TR-10-13/: The second and tenth report in the list are actually the same report, /SKB TR-09-22/, but they are listed in such a way that they appear to be two different reports.

- The unpublished document /SKBdoc 1221567/ is listed with the title "*Simulering av fyllning av kapslar för slutförvaring av utbränt kärnbränsle*" but its proper name is "*SKB - Simulering av inkapsling av använt kärnbränsle för slutförvaring i KBS-3-förvar*". The unpublished document /SKBdoc 1198314/ is sometimes entitled "*Källstyrkor för bränsleelement...*" and sometimes "*Källtermer för bränsleelement...*" As we have not been able to locate this document we do not know the correct title of it.

Assessment of the credibility of the report

It is possible that SKB has done a very thorough work in the construction process of the documentation that we have reviewed, but we have not been given this impression due to the inadequate descriptions of calculations, results and their interpretation. We have not been able to find a section where the results' implications are explained or discussed, and there are very few comments about the contents of the purpose of the tabulated data in the appendices. This makes it difficult to know how SKB interprets the presented results, what they perceive as valuable information and what the implications on the encapsulation process or the repository are. There is also inadequate information on whether or not actual measurements will take place (for what purpose, using what measurement technology, under what conditions etc.). In addition to this, the handling of uncertainties and references has not been presented in a correct way.

The fact that we have not had access to all referenced documentation, made it even difficult to assess its credibility. It is for instance difficult to assess if the report contains enough relevant information about e.g. the nuclide inventory (and hence the burnup) for accurate nuclear safeguards conclusions (whatever they may be) to be drawn, as we have not been able to read the safeguards document that is referenced to.

In order to correctly asses the credibility of the report, we would like to see it supplemented with more information in the areas just mentioned.

2.2.2. Calculations of radionuclide inventories, radiation dose and radiation strengths

We have chosen to specifically address this issue as it was one of the main issues raised by SSM for this review. We have many comments and concerns about this, and they are mentioned explicitly in the reviews of the documents /SKBdoc 1221579/ and /SKBdoc 1077122/.

Our concerns regard mainly implications of the dose rates around the canister for the geological repository, selection of cross-section libraries, handling of uncertainties, presentation of results and the lack of analysis and discussion. There is also a lack of supporting information such as input files, definition of tallies and post-treatment of output data.

2.2.3. Assessment on general use and validation of the software tools

A number of computational codes have been used for the work in /SKB TR-10-13/ and its references. Several of the codes such as Origen-S and MCNP 5.2 are well established both within the nuclear industry and within the scientific community and the codes are validated and benchmarked repeatedly. Sections 6.2.6 and 6.2.7 in SKB TR-10-13 are examples where the codes and the associated uncertainties are explained quite well.

The comments in this section will focus on where there are reasons to question how these codes have been used, and there will be comments on calculations made with less established codes. Some general comments:

- The used codes are mentioned only briefly, in a few cases only by name. For reports with multiple safety aspects it is necessary to clearly explain the use of the codes, why the particular code was selected, capabilities and limitations of the codes, and associated uncertainties of the input and output from the codes.

- In many cases it would be useful to have input files for the particular codes attached in appendices. In order to assess the credibility and be able to back-track some of the work such files are absolutely necessary.

- Codes that are dependent on input data, such as tabulated cross-sections from evaluations (e.g. ENDF-B/V...) need to be validated for the specific application at hand. There are several evaluated nuclear data files, sometimes they deviate significantly from each other. In several places of /SKB TR-10-13/ a dedicated sensitivity analysis is lacking, following the choice of cross-section evaluation. There are also cases where one may question the definition of material compositions. A sensitivity analysis where comparisons are made with the scenario of no contaminants is useful in order to estimate how accurate the material composition is in the calculation.

- For Monte Carlo codes, such as MCNP, it is necessary to give detailed information about how the simulations have been performed. The use of tallies should be specified so that it is clear what physical properties that have been evaluated. Any post-simulation treatment of output data need to be explained. The number of initial events and the resulting statistical uncertainties in the output should be clearly stated. Other sources of uncertainties, such as the selection of a particular nuclear data library, should be clearly stated. In the document /SKBdoc 1077122/ all this information is lacking.

- In the document /SKBdoc 1077122/ one may question if the authors have understood what they are simulating. It is not clear if the output from the tallies have been normalised to any common unit or if they are given per tally (i.e. there is a big difference between dose rate per unit area and dose rate per tally). The division of tallies on the top and bottom of the copper canister has been done in quadrants, but the results are only given as average values as function of distance from the centre. Any information about an increase in dose rate due to uneven loading of fuel assemblies is therefore lost. Furthermore, one may question the relevance of the information given in the figures where the dose rate is plotted versus the distance from the origin, because it is not clear what the normalisation of the dose rate is.

2.3. The Consultants' assessment

Our assessment is that the reports need to be supplemented by additional information. We cannot, with the report in its present state, say that we are convinced that the information presented by SKB on the spent nuclear fuel for disposal in the SKB-3 repository is sufficient and adequate information. We recommend the following supplements to the report:

General recommendations:

- In order to, to a larger extent, meet the objectives of the report, we recommend that SKB goes beyond the reference scenario to estimate what the consequences for the encapsulation process and the geological repository may be.

- We recommend that the report is supplemented with information on identified risks in the handling of spent nuclear fuel, as perceived by SKB. An example could be given as to how SKB prevents the risk that documentation errors occur, that a fuel assembly other than the intended one is picked in the canister selection process etc.

Specific recommendations:

- **Uncertainties**
  It has not been shown in /SKB TR-10-13/ that handling of uncertainties has been done in a rigorous and satisfactory manner. We recommend that SKB complements the calculations in this report by proper estimates of uncertainties and that error propagation is performed in order to obtain uncertainties in quantities such as e.g. decay heat, criticality, dose rates etc. We also recommend that SKB takes on a probabilistic approach in their evaluations, which could answer questions related to the probability of a canister to pass different thresholds (e.g. what is the probability that k_{eff} is above 0.95, 0.96, 0.98, respectively 1.00 in at least one canister?).

- **Additional calculations**
  If SKB decides to explore additional scenarios for the operation of the Swedish nuclear power plants, additional calculations need to be performed. This concerns especially criticality calculations and the determination of the radionuclide inventories with implications for the decay heat and dose rate.

- **Information on measurements**
  We have in the report found multiple references to simulation results and general statements that it is possible to perform measurements. We have two recommendations here.

  We recommend that SKB explains if and how they plan to experimentally verify the simulated results. Normally, experiments are used to validate simulations. This is done by performing experiments in a number of cases where simulations have been performed. If experiments and simulations agree within the uncertainties of the simulations and the experiments, this strengthens the belief in the simulation results. In this process, all uncertainties in the simulations input parameters need to be propagated through the simulations. Alternatively, a subset of the experiments can be used to constrain the input data, and another subset to validate the simulations. The latter alternative will likely reduce the uncertainty in the simulation results, but requires some more knowledge in uncertainty quantification.

We also recommend that the report is complemented by information on all types of measurements that will be performed. This regards e.g. what spent nuclear fuel properties SKB plans to measure, what the measurement criteria are, how and using what equipment the measurements will be performed. We therefore recommend that this type of information is added to the report.

- **Analysis and discussion**
  We have not found any analysis of the modelled calculations or of any other results. We recommend that the report is supplemented with information on the purpose of simulations, evaluation of the results and a discussion on the implications for the encapsulation process and the geological repository of these results.

- **Reference handling**
  We have noted several shortcomings in the handling of references. We recommend that SKB complements their list of references with a larger set of references and recommend that these are not produced by SKB themselves, but rather are peer-reviewed documents. This would support the content of the report. We further recommend that SKB themselves review their use of references such that they are equipped with the (consistently) correct title and that they refer to the relevant document.

- **Optimisation of encapsulation**
  In our judgment, the encapsulation procedure is not optimised for efficient utilisation of the final repository. In the early stages of the described optimisation process, it appears that the strategy is overly conservative. The result is that a large fraction of canisters in the later stages of the encapsulation must be only partly filled. This is identified as a wasteful utilisation of a scarce resource, i.e. the repository.

3. SKBdoc 1221579, Aktivitetsinnehåll i kapslar för slutförvar

This document contains simulations of the encapsulation process, in order to meet the requirements on decay heat. 1650 W was assumed to be the target value in the simulations of the radionuclide inventory simulations reported on in here.

3.1. SKB's presentation

The document has three main chapters except for the introduction. These chapters are "Methdology" (Metodik), "Fuel data" (Bränsledata) and "Calculation of nuclide inventory in canisters" (Beräkning av nuklidinventarium i kapslar). There are no chapters for analysis, discussion or conclusions.

3.1.1. Chapter 2: Methodology (Metodik)

This chapter, covering just half a page of information, states with what software the simulations were performed and comments briefly on what factors such as e.g. the nuclear fuel configuration and geometry, that influence the modelling.

Regarding the selected software that has been used for the simulations, a reference is given in Section 2 (page 3) that presumably shows that Scale 5.1 has been validated. This is in one sense correct, but SKB will encapsulate fuel assemblies (e.g. MOX) that were not used in those validations. For this reason, it is important to show that the selected software is applicable to all fuel types that will be encapsulated in Sweden. Furthermore, the quoted three references for the validations are all performed by the same company which also produced the code. No external peer-reviewed validation is referenced. It should also be noted that the validations performed for BWR fuel assemblies are associated with larger uncertainties than for PWR assemblies, due to uncertainties in void distributions during irradiation in the reactor.

Regarding the cross-section library used, 44GROUPNDF5, it is based on the ENDF/B-V evaluation, but there is no explanation on what is included in this particular cross-section library. A more detailed explanation regarding the selection would be welcome, together with a discussion of the advantages and disadvantages of this particular library and whether or not it is possible to select other cross-section libraries for use in Origen.

There is also a statement in the same section (section 2, page 3) that the geometry of the fuel assemblies and the neutron spectrum affects the neutron cross-sections used in the burn-out calculations. To be specific, it is the grouped macroscopic cross-sections that are affected, not the microscopic cross-sections.

3.1.2. Chapter 3: Fuel data (Bränsledata)

This section of /SKBdoc 1221579/ presents information on the fuel types selected for the simulations. The chapter consists almost entirely of poorly commented tables, without explanations to the reader about why this information is relevant for the interpretation of the results.

In general we would appreciate a more comprehensive explanation on how the data are used and why it may be justified to use these fuels as references cases. Table 4 is not mentioned at all in the text and no sources for the given values are quoted.

Specifically, we note that:

- In tables 3, 5 and 6 we would like to see a sensitivity analysis being performed on the implications of the impurities. This is relevant in the discussion of understanding the effect of the impurities.

- Table 7 (page 8) displays data for some physical parameters in the burnup calculations. It would be useful if the table also displayed spans for the parameters during a typical fuel cycle, with a comment on whether such variations have been analysed or not.

3.1.3. Chapter 4: Calculation of nuclide inventory in canisters (Beräkning av nuklidinventarium i kapslar)

This section of /SKBdoc 1221579/ presents information on the relationship between burnup, cooling time and decay heat. The encapsulation process with a variation of BWR and PWR fuels is presented, together with a selection of estimated canister activities and masses of certain radionuclides. The decay heat is finally estimated for certain categories of spent nuclear fuels. The report also contains a section on uncertainties relevant for the calculations.

In this chapter, or associated with this chapter, we lack an analysis and a discussion on the results. In its current form, it is not at all obvious what the importance of the results is or what their implication on the encapsulation process is. Numbers and results have no meaning unless they are interpreted, and that interpretation is unfortunately not part of the current scope.

We also lack a description of the key isotopes and for which other reports (or areas) the results will be useful.

Specific comments on this chapter are:

- Uncertainties are treated in a separate section, but not mentioned in relation to the respective simulations that are described. Uncertainties should be associated with each calculated quantity.

- Figure 1 describing the relation between burnup and cooling time, with an inserted line marking the 1700 W limitation for a canister consisting of identical fuel elements, has no error bars.

- Figures 2-4 (page 11-12) describing the cooling time and burnup for PWR and BWR fuels in 2023 and 2070 contains error bars only for the red markers which denote fuel that is not yet residing in Clab. We have three comments on this: 1) why are there no uncertainties shown in burnup for the already existing fuels, 2) where does the ±3 MWd/kgU for the future fuel assemblies come from, and 3) why is there no line showing the combination of BWR fuels that gives 1700 W in a filled canister in Figure 4-5?

- Tables 13-21 in sections 4.3. and 4.4. contain a varying number of displayed radionuclides. What is the reason for this and what selections have been made?

Worth noting is also that none of the tables or the related text explain the uncertainties in the calculated values.

- Tables 17 and 18 list calculated activities (Bq per canister) after up to 1.000.000 years. Why is this relevant for the objectives of the report? If quoted at all, and given the time scale considered here, it seems more relevant to give these values as radiotoxicity.

- In the uncertainty section 4.6 (page 26) it is written that the uncertainty on the content of actinides are judged to be about 20 % and about 12 % for fission products. These uncertainties are generally larger than the uncertainties reported in Table 24 on page 28. One may assume that this discrepancy can be explained by the method used to calculate the average using data from, i.e. Tables 22 and 23 on pages 26-27, but is not stated explicitly. It should be noted that using such an average calculation does not give a representative view of the uncertainties in calculated quantities such as decay heat or dose rates. In this case it is the uncertainties of the individual isotopes relevant for the calculation that are important. In this context it is quite unfortunate that Table 24 that has been reproduced in /SKB TR-10-13/ (see text below for a longer discussion).

- The decay power from a filled canister is given Tables 20 and 21 for BWRs and PWRs, respectively. It can be seen in the tables that the major part of the contribution to the decay power comes from a small subset of nuclei. It is however unclear how much of the decay power comes from the nuclei <u>not</u> listed in Tables 20 and 21. Further, there are no estimates of the uncertainties in the decay power contributions. The uncertainties of individual nuclides have been estimated in Table 22 based on measurements and calculations, but the uncertainties have not been propagated to the entries in Tables 20 and 21 to find the resulting uncertainty in the calculated decay power.

- With regards to the uncertainty in decay power, it is mentioned in section 4.6 (page 26) preliminary calorimetric measurement results on decay heat indicate that the difference between calculated and measured decay heat is generally small. We are not sure how to interpret this and would like to see in detail what the differences actually are in order to be able to assess what “generally small” means and how large the differences are when they are not small. This information must be available for all fuel assemblies to be encapsulated, and the results need to be definitive and not only preliminary. Furthermore, the decay power is one of the most critical parameters to the encapsulation process. For an individual canister it is not relevant if calculations are on average correct, based on a large number of assembles. What is interesting to know in this context is the (average and) maximum spread in the agreement between calorimetric measurements of decay power and calculations. We would for this reason like to see a validation of the results after the completion of the project.

- A simple calculation for the BWR MOX canister in Table 20 has been performed by us in this review, by propagating the uncertainties from the rightmost column of Table 22. In this trial simulation, we obtained a spread in decay heat of about ±220 W, with a standard deviation of about ±70 W. Is this value representative for the uncertainty in decay power? If so, a margin of only 50 W, which has been used the encapsulation optimisation /SKBdoc 1221567/ is too low.

- A different number of nuclides are listed in Tables 22 and 23 (pages 26-27) in the description of the difference between measured and calculated values. We also do not know how the standard deviation has been determined, and are very hesitant that it is at all meaningful to express this quantity for a set of only three measurements.

- Table 24 (page 28) shows a summary of the results. How have the average ratios been determined? A discussion around the results and the consequences of large discrepancies for certain nuclides would be welcome.

3.2. Consultants' motivation of their assessment

SKB has included a paragraph on objectives with this report, which states the purpose of the present radionuclide calculations. However, this does not seem sufficient as there is no information on why this is relevant, or in which context.

We have not been able to understand from the report, how selections of radionuclides have been done by SKB, or for what purpose. We have however noted that some tables contain a different number of radionuclides than other tables, without any explanation.

The report contains a separate section on uncertainties, which is well motivated in this context, but we lack uncertainty estimates in a majority of the tables and for many results. Very limited error propagation from uncertainties in one table to the values in the next seems to have taken place. We did however, as noted in section 3.1.3 of this assessment, find one example where error propagation had taken place, but it does not appear to be done correctly and, as far as we can understand from the material presented by SKB, led to the conclusion that the end result appears to be better than it actually is.

In addition, several tables in the report (e.g. tables 12 and 16) are completely uncommented in the text and we are not sure why they are even included.

3.3. The Consultants' assessment

Our assessment of this report is that it needs to be complemented. We recommend that supplemental information in the following areas:

- **Decay heat determination**
  From our motivation in the subsection above, we recommend that the report is complemented with additional calculations and information on the decay heat determination. We also request more information on the calorimetric measurements already performed at Clab, as well as those that are currently being pursued.

- **Uncertainties**
  We recommend that all tables and results should be accompanied by uncertainty estimates on some explicitly stated level of confidence. Proper error propagation should also take place from input data, throughout the radionuclide inventory determinations and to fuel parameters with relevance to the safety of the encapsulation process or the geological repository.

- **Comment tables and figures**
  Tables and figures need to be accompanied with text in order to make sense to the reader, in an uncommented state they serve no purpose at all, and the risk is instead that they give the impression that the authors did now know what was

important so they put everything in to give a rigorous impression.
An alternative approach could however be to replace the tables by a few references to information where the data can be found.

- **The use of references**
  The use of references is very limited and contains no peer-reviewed material that could strengthen the credibility of the document. Also tables like the ones in the appendix could be avoided by a proper reference to where the values are taken from.

4. SKBdoc 1077122, Strålskärmsberäkningar för kopparkapslar innehållande BWR, MOX och PWR bränsleelement

This document contains dose rate calculations for a number of geometries and canister options. Dose rates have been calculated at the canister surface and for different positions in the deposition hole with and without bentonite backfill. The calculations have been performed with the software MCNP 5.2. Simulations have been performed both for canisters with a decay heat of 1700 W, and for canisters with a decay power above 1700 W.

The document presents results for simulation of dose rates (absorbed dose per time unit) and equivalent dose rates (equivalent dose per time unit) at several locations outside the copper canister for several cases of spent fuel inside the canister. Gamma radiation and neutrons are considered. The results are presented in 42 figures and several tables.

The general impression is that the considered cases are relevant and concern conservative assumptions; however in some cases exceeding the 1700 W constraint on the decay power. The input data to the simulations are given and results are presented in a, largely, understandable way. The code used for the simulations, MCNP 5.2, is widely used for this type of calculations and can be considered validated.

However, we find that several pieces of information are missing and that some results seem to be contradictory to either other information inside the present report or other SKB documents.

4.1. SKB's presentation

This document has six chapters besides the introduction: Methodology (Metodik), Source strengths (´Källstyrkor), Calculated copper canisters (Beräknade kopparkapslar), Calculated geometries outside the canister (Beräknade geometrier utanför kapseln), Results (Resultat) and Conslusion (Slutsatser). We will comment on each in order.

In section 1 (page 5) it is explained that the canister alternatives selected for the calculations have relatively high burnup and short cooling times. For this reason, there are several cases where the calculated power is higher than the allowed 1700 W. The reason for why these cases are considered and calculated should be explained.

4.1.1. Chapter 1: Introduction (Introduktion)

This is a very brief introduction stating the software used for the calculations, with some information on burnups and cooling times considered in the work.

Our comment here is that it is not explained (page 5) why several cases of canisters with a decay heat above the allowed value of 1700 W are considered.

4.1.2. Chapter 2: Methodology (Metodik)

This is an extremely short section, consisting of only four sentences. The description is very shorthanded, but this would in fact be a good place to give a comprehensive explanation about details related to the code MCNP 5.2. Among the information of relevance we would expect are explanations about what kind of input that has been included (at least a sample input file should be included in the appendix), a general explanation about what a tally is and what kinds that have been used, how many initial events have been simulated, which nuclear data libraries have been used, material specifications, and information about any pre- or post-processing of the input and output data. This type of information is common in scientific reports in order to ensure that another independent scientist is able to replicate the work with the same results in the end.

4.1.3. Chapter 3: Source strengths (Källstyrkor)

This section displays a set of five spent nuclear fuel assemblies which have been selected for the simulations.

Section 3 (page 7) starts with a list of the selected alternatives, with respect to fuel type, burnup and decay time. We have not found, in this chapter, a motivation to why the set of five fuel assemblies were selected although we consider this to be an important piece of information. Furthermore, the reason for choosing such an odd value of burnup and cooling time as 42.4 MWd/kgU and 34.1 years should be explained (page 7).

On page 8 it is explained that the source terms for Co-60 have been homogenised over each component. Are there any effects on the calculated dose rate by doing this? Will this effect hide any extreme values in certain directions? Have the assumptions been verified?

4.1.4. Chapter 4: Calculated copper canisters (Beräknade kopparkapslar)

This section consists of a geometry description for the BWR and PWR fuel geometry.

Our comment to section 4.1 (page 11 and 13), where we have found footnotes about homogenisations, is the same as for the previous chapter in the report: Are there any effects on the calculated dose rate by doing this? Will this effect hide any extreme values? Have the assumptions been checked?

4.1.5. Chapter 5: Calculated geometries outside the canister (Beräknade geometrier utanför kapseln)

This section of the report explains the geometries outside the canister where the dose rates are calculated.

We have a number of comments here:

- Results are obtained from MCNP by using tallies. We have not found sufficient descriptions of the use of tallies in this report, it appears that the user must be acquainted with them to start with. SKB should however provide more information about the specific tallies used because there are multiple ways of how to do this in MCNP (see last bullet in this list).

- In section 5.1 (page 14): The table of levels for tallies seems to have a misprint, level 3 should have lower lever 193.4 cm, not 103.4 cm.

- Section 5.2 (page 15) explains the simulated geometry and the position of the tallies at 2 m distance from the copper canister. The table of levels for tallies seems to have some misprints: level 3 should have a lower lever at 193.4 cm, not at 103.4 cm. The upper level should be at 708.5 cm instead of at 658.5 cm, if the distances are supposed to be the same both in the top and the bottom.

- To judge from the plots in Figure 7 and thereafter, the angular intervals in the figures seem to agree well with angular tallies at a 2 meter distance from the canister (as shown in Figure 4). For Figure 3 which displays the copper surface, no angular tallies are shown at all.

- Section 5.3 (pages 16-17) explains the simulated geometry and the position of the tallies for the hole with closed steel lid. However, it is difficult to see the colors for the smaller details in Figures 5 and 6, and in addition the purpose of the steel lid is not explained. It is also not clear from Figure 6 if the lid is there or not. Its tallies are not explained, and information about the tallies used for air 1 m above the steel lid is also missing.

- In section 5.4 (page 17): Tallies are explained by dimensions and positions, but there is no explanation about what tallies actually are, which kind of tallies that are being used (dose calculations is mentioned but this has to be further explained), what property that is scored in them, and whether or not the output from the tallies has been handled or transformed in some way.

4.1.6. Chapter 6: Results (Resultat)

This part of the report collects the results and presents them in 42 figures and one table.

Our comments here are:

- In section 6 (page 18-45), Figure 7-48: None of the figures display error bars for statistical uncertainties. It is also questionable what the line between the data points mean. Straight lines or no lines at all give a more fair assessment. This becomes quite clear in Figure 9 where the curved lines indicate a maximum point at a position where there has been no simulation performed.

  Two examples are given here explaining how the current display of results is a problem, but the issue is further discussed in section 4.2. of this assessment:

  - Page 20 shows the average dose rate at the bottom (Figure 9) and top (Figure 10) of a PWR canister. The distances where the data points are given can only be explained if each data point is given as the mean value of the radius based on the area for each segment. If this is the strategy then it should be thoroughly explained, and any difference between different segments, or the lack thereof (depending on how the division into segments relates to the geometry of the fuel bundles within the copper canister), should also be discussed. A thorough explanation of whether or not the dose rates have been normalised with respect to the area of each segment should also be included.

  - Comparing Figure 10 with Figure 33 (PWR), and Figure 14 with Figure 35 (BWR), we would expect some similar trends as function of increasing radius; we expect the steel lid to give similar effect in both cases. The plots show a deviating behaviour that needs to be explained. No such discussion is made. Similar comparisons for Figures 41- 48 raise similar questions that need to be answered.

- Section 6 (page 20), Figures 9-10: The distances where the data points are given can only be explained if each data point is given as the mean value of the radius based on the area for each segment. If this is the strategy then it should be explained, and any difference between different segments, or the lack thereof (depending on how the division relates to the geometry of the fuel bundles within the copper canister), should also be discussed.

- Section 6 (page 38): Comparing Figure 10 (PWR) and 14 (BWR) with the corresponding figures for the closed steel lid (figures 33 and 35) show some deviating trends that need to be explained. No such discussion is made. Similar comparisons for Figures 41- 48 raise similar questions that need to be answered.

- There is no interpretation, analysis or discussion provided together with the results. It is commonly known in the academic community that all figures and tables must be commented on in the text so that the reader understands their purpose as well as their implications. The lack of this information is a serious shortcoming of this report.

4.1.7. Chapter 7: Conclusions (Slutsatser)

The conclusions drawn by SKB are, among other things, that the surface dose rates vary but are large for PWR canisters and that the implication of using MOX fuel assemblies is small.

Our specific comments are:

- We have not understood the meaning of the sentence (page 48) "*Förutom de två 1700 W kapslarna har valda fall varit konservativa och resteffekten har överstigit dimensioneringsgränsen 1700 W per kapsel med 60-70%*." (Eng: Besides the two canisters with 1700 W, the assumptions has been conservative and the decay heat exceeded the limit of 1700 W with 60-70%.) There is no further explanation, but exceeding the decay heat limit by 60-70% seems very severe to us. Perhaps the sentence has an unfortunate formulation, otherwise more thorough explanations are needed.

- What are the limiting factors regarding the dose rate? This is not clear to us after having read the report. We also have no understanding on whether or not the reported results are acceptable or problematic to SKB, since the analysis and discussion of the results is lacking. Is e.g. the high surface dose rate of the PWR canisters a problem and if so, will it affect the design of the copper canisters?

- SKB reports on equivalent doses (in the unit of mSv) and equivalent dose rates (in the unit of mSv/h). How is this relevant, as equivalent doses are intended for the estimation of damage in human tissue with weighting factors for different types of radiation and specific organs? It would be more relevant to express the calculated (energy) dose in terms of Gy.

- As the report contains no information on uncertainties on the calculated dose rates etc, no scientifically based conclusion can be drawn. Therefore, in our opinion, the conclusions made in section 7 are not valid from a scientific viewpoint.

4.1.8. Appendices A and B

The appendix contains several pages of tabulated values of energy spectra which are difficult to digest. We believe that figures of the neutron and photon spectra would be helpful in order to better understand the input source terms.

4.2. Consultants' motivation of their assessment

The use of complex codes such as MCNP comes with a responsibility to show the reader that the user knows how to handle the code properly. There are a number of issues with this report where relevant questions are raised because it is unclear to us if the user has the proper knowledge for handling the code, to understand the physics output, to analyse the results and to make the results understandable to a wider audience, not the least for those who in the end will decide if the KBS-3 method will fulfill the requirements.

Below are listed a number of critical remarks that motivate the final assessment:

- One major problem with the report is the lack of a discussion of the results. It is important to analyse the results and judge their meaning. As an example it can be mentioned that the maximum equivalent dose rate was calculated to be about 250 mSv/h on the surface of the copper canister. It is not discussed here (nor in the main report /SKB TR-10-13/) whether this is acceptable or not and what measures could be taken to further reduce this value. It is, furthermore, not clear why equivalent dose rates are calculated at all since the limits to the surrounding, as discussed in the main report, concern doses to material and is considered to be 1 Gy/h.

- No analysis or discussion of the uncertainties present in the simulation is given. All the graphs only mark the calculation results without any error bars. In several figures containing just three data points (e.g. Figures 9, 10, 13, 14, etc.) the data points seem to be interpolated by an arbitrary internal Excel function. No motivation and justification for this is given. The very same figures also cover only part of the area of interest and do neither contain origin (radius of 0 cm) nor the outer edge of the canisters. It is unclear why values at about 12, 28 and 45 cm are given. These radii seem not to be connected to the defined tallies in Figures 3 and 4.

- It is not stated which radiation weighting factors that have been used. These values have been recently changed for neutrons. The new recommended values are given in ICRP 103 (2007). The new weighting factors for neutrons range between 2.5 and about 20.

- The presented results seem to be partly inconsistent. Figures 7 and 27 consider the same case but give values as dose rate in mGy/h (Figure 27) and equivalent dose rate in mSv/h (Figure 7). Since the radiation weighting factor for gamma radiation (“fotoner”) is 1, the difference in the absolute value of the peak values (180 mGy/h and 250 mSv/h, respectively) must be due to neutrons. However, according to Figure 8, the maximum equivalent dose rate from neutrons for this case amounts to only 8.5 mSv/h. It should therefore be investigated if any mistakes have been made in the simulations.

- On a similar issue, one can compare Figures 8 and 28. The difference in scale is a factor of about 60. This scaling should only be due to the radiation weighting factor for neutrons. However, as mentioned above, the maximum value for this weighting factor is about 20. It therefore seems that either of the figures (or possibly both) is wrong.

- The report /SKB TR-10-13/ refers to report /SKB TR-99-74/ by R. Håkansson. The latter report gives values on the total dose rate on the surface which are at odds with the table given in Section 6.5 of /SKBdoc 1077122/. One may, e.g., compare the values “Kapselyta Mantel” (Eng: canister surface area) given in the first, fourth and fifth column with the corresponding values in /SKB TR-99-74/ in A1-53, A1-50 and A1-52, respectively. The latter values are about a factor of three higher than those given in the present report (e.g. 780 mSv/h in the 60 MWd/kgU case with 30 years cooling time in A1-53 of R-99-74, compared to 260 mSv/h in the present report).

- The comment of page 50 concerning revision 2 is somewhat disturbing as it may be interpreted as some references being omitted because they were considered as “icke godkända” (Eng: not approved). How can a reference be removed without implications for the contents of the text? The removal of the references seems to be perceived as an improvement of the report without further needs to scrutinise. In this context it should be noted that the present report contains only two references.

- Appendices A and B contain lengthy tables which are difficult to assess. It would be good to add plots of the presented numbers to simplify a review of the results.

4.3. The Consultants’ assessment

With respect to the long list in the previous subsection of this assessment, we recommend major supplements to this report to address the specific issues mentioned above. We recommend additional information in the following areas:

- **Introduction**
  The introduction should clearly state what the purpose of calculating the dose rates are and in what context the results will be used. It should also be clear from the start for what reason the energy dose (expressed in Gy) and the equivalent dose (expressed in Sv) are relevant.

- **Methodology**
  We recommend that SKB complements this report with details on how the simulations in MCNP were performed, including input information and the use of tallies.

- **Comments on figures and tables**
  This report contains a large number of figures, but there is no explanations to them in the text. If they are not worth commenting, they should be removed. If they serve a purpose, this should be stated.

- **Analysis and discussion**
  The report includes no discussion of the results or their implications. Without such information it is not possible to assess the work done, because we are simply not aware of how they are interpreted by the authors or what the conclusion of this is.

- **Uncertainties**
  No uncertainties are included in this report. This needs to be inserted before any conclusion can be drawn based on the report.

5. Consultants' motivation of their assessment

We have reviewed three documents that are part of SKB's application for a geological repository. For each of the three reports, we have motivated our assessments and delivered our overall assessment. This chapter will for this reason contain only a summary of our views, which can be found in either section 2.2, 3.2 or 4.2 in greater detail.

We have found two types of limitations in all three reports: such that are of a structural or pedagogical character and such that are of a factual matter character:

- On a *structural level*, we had difficulties navigating through the reports partly because SKB uses an unsatisfactory reference system. This concerns both lack of externally and scientifically reviewed work, referencing to unpublished and thereby inaccessible work and several examples of referencing to documents with the wrong names or to documents which contain no more information than the original one does.

  It is possible that these structural problems make it more difficult to access the factual matters in the report. For this reason, it has sometimes been difficult to assess whether certain information is actually included in the reports (and in that case where it can be found) or if it has been omitted.

- On a *factual matter level*, we have identified several severe limitations in the reports. The most substantial ones are:
  - The lack of analysis and discussion of presented results, and the inclusion of several uncommented tables in the appendices.
  - The lack of uncertainty handling and error propagation in the data. The insufficient handling of uncertainties in the reports can be applied to in principle all areas covered: fuel parameters, decay heat calculations, dose rates, assumptions concerning identical fuels residing in the copper canisters etc.
  - The lack of supporting data for presented calculations such as e.g. input files, definition of tallies and post-treatment of output data so that the reader can understand how the software tools have been used.
  - The lack of information on future measurements (purpose, extent, precision, accuracy etc.) for validation of modelled results on e.g. decay heat, as well as for verifying the encapsulation process itself and the nuclear fuel to be encapsulated.

These limitations in the reviewed work make the interpretation and evaluation of the reports' contents very difficult. In addition it is often not possible to neither understand the purpose nor the implications of it. This in turn means that we have had difficulties assessing the credibility of the reports. It is possible that the underlying job is of good quality, but its current presentation does not allow to professionally judge this.

6. The Consultants' overall assessment

We have reviewed three documents that are part of SKB's application for a geological repository. For each of the three reports, we have motivated our assessments and delivered our overall assessment. This chapter will for this reason contain only a summary of our views, which can be found in either section 2.3, 3.3 or 4.3 in greater detail.

Our general assessment, based on the motivation in the previous chapter, is that the reports need to be supplemented by additional information.

A selection of the information that we would like to see included in the reports are:

- Broadening the scope of the report by including more than one reference scenario for the operation of the Swedish nuclear fleet.

- Complement the data and results in the reports with a rigorous handling of uncertainties and that error propagation must be performed for relevant results (decay heat, criticality, dose rates etc.).

- Complement the report with more information on the optimisation of the encapsulation process.

- Complement the reports with separate chapters for analysis, discussion and conclusion. Without these chapters, the reports serve little purpose. SKB must show how they interpret their results, what influences the results and what implications and consequences the results have.

- Complement the report with explanations on if and how SKB plans to experimentally verify the simulated results, and on the conditions for such measurements (what properties will be measured, what the scope of the measurements will be, what are the required measurement times, the necessary accuracies and precisions etc.).

- Improve the handling of the following structural issues in the report: the purpose of performing included calculations, descriptions of how the simulations were performed, the use of references, explanations to included figures and tables, explanations to included information (tables) in the appendices.

7. References

Published documents:

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SKB TR-09-22. **Design premises long-term safety, SKB 2009.** Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses. Svensk Kärnbränslehantering AB (2009)

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ORNL/TM-13315. O. W. Hermann and M. D. DeHart, **Validation of the Scale (SAS2H) Isotopic Predictions for BWR Spent Fuel** (1998)

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NUREG/CR-6811. I.C. Gauld, ed., **Strategies for application of Isotopic Uncertainties in Burnup Credit** (2003.)

Unpublished documents:

SKBdoc 1193244. Agrenius L, **Criticality safety calculations of disposal canisters,** Svensk Kärnbränslehantering AB (2009)

SKBdoc 1221567. Agrenius L, **SKB - Simulering av fyllning av kapslar för slutförvaring av utbränt kärnbränsle**. Svensk Kärnbränslehantering AB (2010)

Documents we have referenced, but no access, to:

SKBdoc 1198314, **Källstyrkor för bränsleelement under driftskede för Clink, slutförvarsanläggning och slutförvar**. Svensk Kärnbränslehantering AB (2009)

Coverage of SKB reports

Table A:1

Reviewed report	Reviewed sections	Comments
SKB TR-10-13	*Spent nuclear fuel for disposal in KBS-3 repository*	
SKBdoc 1221579	*Aktivitetsinnehåll i kapslar för slutförvar*	
SKBdoc 1077122	*Strålskärmsberäkningar för kopparkapslar innehållande BWR, MOX och PWR bränsleelement*	

2015:51

The Swedish Radiation Safety Authority has a comprehensive responsibility to ensure that society is safe from the effects of radiation. The Authority works to achieve radiation safety in a number of areas: nuclear power, medical care as well as commercial products and services. The Authority also works to achieve protection from natural radiation and to increase the level of radiation safety internationally.

The Swedish Radiation Safety Authority works proactively and preventively to protect people and the environment from the harmful effects of radiation, now and in the future. The Authority issues regulations and supervises compliance, while also supporting research, providing training and information, and issuing advice. Often, activities involving radiation require licences issued by the Authority. The Swedish Radiation Safety Authority maintains emergency preparedness around the clock with the aim of limiting the aftermath of radiation accidents and the unintentional spreading of radioactive substances. The Authority participates in international co-operation in order to promote radiation safety and finances projects aiming to raise the level of radiation safety in certain Eastern European countries.

The Authority reports to the Ministry of the Environment and has around 300 employees with competencies in the fields of engineering, natural and behavioural sciences, law, economics and communications. We have received quality, environmental and working environment certification.

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