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NUCLEAR ENERGY AGENCY
RADIOACTIVE WASTE MANAGEMENT COMMITTEE

International Peer Reviews

Exchanged comments and responses as preparation for the workshop between the OECD/NEA international review group and the JNC staff

This document supports NEA/RWM/PEER(99)2, titled "OECD/NEA International Peer Review of the Main Report of JNC's H12 Project to Establish the Technical Basis for HLW Disposal in Japan".

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INTRODUCTION

The Project Overview Report of the JNC[1] H12 study has been peer reviewed by a group of international experts[2] assembled by the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). This review was meant to fulfil a specific recommendation made to JNC by the Atomic Energy Commission of Japan and, in the process, help JNC to assess their own achievements. The review is documented in NEA/RWM/PEER(99)2

The conduct of the OECD/NEA international peer review was as follows:

- The Project Overview Report and other supporting documents were received in mid-May 1999.
- A meeting of the IRG took place in Paris in early June. At this meeting, JNC representatives made two presentations about the "General Background and Specific Features of JNC's Second Progress Report". In addition to providing a description of the approach used in the H12 study, these presentations also described, to the IRG, the general policy framework in Japan within which the H12 study was prepared.
- Preliminary comments and questions were provided during the months of July and August, and JNC offered written responses to all comments and questions.
- A week long workshop with JNC staff took place in Japan the 4th week of August. During this time, the IRG also visited the JNC R&D facilities at Tokai. The workshop allowed in-depth discussions of all topics identified in the preliminary comments and questions and allowed, as well, to identify and discuss new topics. The workshop was attended also by observers from Japanese institutions other than JNC. After the workshop, two members of the IRG visited the Tono mine.
- Preparation of the review document went through several iterations within the IRG, until finalization and submission to JNC mid-October 1999.

The present document collates the preliminary comments and questions by the IRG as well as the initial responses by JNC. These comments and questions were put forward, recognizing that, at this stage in development of the Japanese repository program, an opportunity exists to promote discussion in the hope of making improvements. Some comments and questions reflect as well that, although the review is restricted to technical matters, it is not always possible to separate technical considerations altogether from the societal requirements that will impinge on them as the program proceeds.

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1 Japan Nuclear Cycle Development Institute

2. Referred to as "the IRG", for International Review Group

It is to be understood that:

1. these comments, questions, and responses were produced in the June-August 1999 time frame, before the visit of the IRG in Japan;
2. only very limited editing has been applied;
3. these comments and questions represent unstructured first impressions by the individual members of the team, and initial replies from JNC; and
4. on specific issues, changes in views and perspectives may have taken place after these comments, questions, and responses were produced.

These comments, questions, and responses are made available in order to help interested readers to comprehend better the depth of the review that took place, as well as to allow them to have more detailed information on specific items. The report NEA/RWM/PEER(99)2 documents the official, final views and review of the IRG.

1 GENERAL COMMENTS

0-1 The overall document is very well done and reflects an approach to the solution of high level radioactive waste disposal in line with other national programmes and international perspectives. JNC is to be commended for the meticulous care with which the programme is described. In addition, the quality of the translation is superb.

=> This acknowledgement is gratefully noted.

0-2 The comments and questions are put forward, recognising that, at this early stage in development of the Japanese repository programme, an opportunity exists to promote discussion in the hope of making improvements, and that, in the future, the programme will be subject to the more sceptical reviews and criticisms, which must be anticipated. Some comments and questions reflect as well that, although the review is restricted to technical matters, it is not possible to separate technical considerations altogether from the societal requirements that will impinge on them as the programme proceeds.

=> We believe that this set of comments makes a number of thoughtful statements that are relevant to the development of an acceptable solution to the waste-disposal issue, and we thank the reviewer for this valuable input. Many of the points will, however, contribute more to post-H12 research, rather than H12 itself.

0-3 Storage versus disposal

In discussing fundamental principles, the issue of timing for disposal is discussed. The belief is stated that, from an institutional point of view, HLW can be stored safely for hundreds of years. The statement of this belief is common among waste management practitioners and regulators internationally. This is an important issue as regards the time frame for proceeding with disposal, and is often used in political and public circles to delay progress on disposal. Such an important issue requires more analysis to make human factors a more legitimate part of the discussion on storage versus disposal, and on the nature and timing of the transition from storage to disposal.

=> We agree with the importance of this issue, and would like to treat it in the Supplementary Report, which is planned to provide a more general background on basic issues relating to geological disposal, since this issue is not be directly incorporated in the scope of H12 (although this report is in only Japanese).

0-4 Retrievability

One reason given for the suitability of the geological disposal concept is that retrieval would not be impossible; however, no specific provision for retrieval of waste from the repository is required in the programme. Retrievability is a most important issue with both lay persons and technical experts in society. A more explicit analysis of retrieval under various conditions would be appropriate. Design modifications that could assist retrieval, as long as they do not adversely affect safety, should be considered in the future. The cost of retrieval and its practicality under realistic physical and institutional conditions will be important issues for implementation.

=> Retrievability is certainly a key outstanding issue for future research, but it was not appropriate to consider it in great detail in H12 because of its generic nature. We discussed this issue a little, however, in Chapter IV.

0-5 Independent review

It is important for success that the broad scientific community supports the repository plan. The programme reflects a substantial involvement of experts from many areas and institutions, which serves well to bring in a range of technical viewpoints. However, it may be better perceived and advantageous to the programme to highlight some independent views in the progress reports. An independent report of an academic committee not aligned with the proponent or the nuclear industry could, for example, be included as an appendix or a separate volume. The independent view is especially important as the safety case relies very heavily on expert opinion.

=> We agree, and this is an important role of the NEA review. Complementing the NEA review, there will also be a detailed assessment by independent experts. This is currently ongoing.

0-6 Role of isolation barriers

To provide flexibility in future siting, the repository concept places emphasis on the engineered barrier system (EBS) and places less reliance on the barrier functions of the geosphere. Moreover, the safety analysis for the normal evolution of the reference case assumes a very high reliability of the engineered barriers. JNC is to be commended for the analysis of the processes affecting the performance of these barriers and for the high caliber of science and engineering that support the analysis. Nevertheless, it must be expected that the emphasis on the reliability of the EBS will be a focus of criticism in the future. The inapplicability of standard engineering methods to long-term disposal and the relatively short history of engineering can be used as arguments in spite of the best scientific information. As well, drawing inferences from scientific information inevitably involves judgment and can therefore be debated among scientists. Redundancy is a strong argument in such circumstances and that argument is weakened by de-emphasizing the geosphere barrier functions. Therefore, there would be a significant advantage to developing a stronger case for the barrier functions of the geosphere, while continuing to strengthen the case for the EBS.

=> We think the idea of a robust EBS is now widely accepted. For countries like Japan and Switzerland, it is the most logical approach, especially in the absence of site data. The geosphere has a critical role, but it is principally that of providing a suitable environment for the EBS, ensuring its longevity and favouring its performance. The practicality and robustness of such EBS concepts may be further improved as we progress from 1st generation feasibility demonstration designs to 2nd generation optimised designs, and as we accumulate site data in future stages. We agree that it may be possible (and would certainly be desirable) to take more credit for the geosphere transport barrier, as site-specific data becomes available. We do not, however, wish to pre-judge the findings of site characterisation, since this may not only reduce uncertainties, but may also reveal unanticipated features.

0-7 Safety standards

Could some clarification be given regarding safety standards? Individual risk is directly related to individual dose. In Section 2.3.1, the individual risk standards mentioned (10^{-6} y^{-1} to 10^{-5} y^{-1}) would correspond to mean individual dose rates of roughly 0.01 mSv y^{-1} to 0.1 mSv y^{-1}. The lower end of the range is an order of magnitude below the lower end of the range (0.1 y^{-1} to 0.3 y^{-1}) to which it is stated that guidelines of each country generally conform. As well, standards are sometimes explicitly connected in regulatory documents with time frames of the analysis, and this is not discussed.

=> The connections between safety standards and timeframes of the analysis are covered, to some extent, in Chapter VI. We will, however, describe these issues more carefully in the final version. It will be emphasised that H12 is designed to contribute to the formation of regulations.

0-8 Collective dose

In spite of the generally held view among international practitioners in waste management that individual dose (or risk) is the appropriate indicator of safety of disposal, there exists a strong desire among many persons, including some informed scientists, to focus on collective dose (or risk). It would be beneficial to be prepared with some discussion of collective dose.

=> We recognise that collective radiation dose have often been considered in optimisation studies for present day operations, but have not usually been considered in any detail in performance assessment studies for HLW. This is even more the case for collective risks. It is, however, acknowledged that collective doses and (partial) optimisation are now of growing importance in the radwaste community (*e.g.* the new SSI regulations and criteria in the USA for limiting the release into the accessible environment, that were derived, in part, on the basis of limiting the collective health impacts of such releases). Thus, this may need to be considered when formulating Japanese regulations. We will mention these discussions on collective dose (or risk), but only in the Project Overview Report or Supplementary Report in H12.

0-9 General evaluation of SA

Given the assumptions made for the safety case, the results are generally consistent with similar analyses carried out in other countries. That is to say, with all systems operating as intended, the estimated exposure of the public is extremely low. The methods used for the treatment of important factors in the normal evolution scenario are consistent with the state of the art internationally.

=> Great pleasure for us.

0-10 Mobile long lived radionuclides

An important consideration in achieving the low dose estimates is the absence of the mobile, long-lived components I-129, C-14 and Cl-36 from the waste inventory. Some discussion of the fate of these nuclides in the fuel cycle may be warranted.

=> We agree that this point needs to be clarified in the H12 inventory section. We will add a sentence as to why there are insignificant amounts of these nuclides, compared to spent-fuel disposal, allowing them to be screened out in the performance assessment. These radionuclides, which are separated from high-level radioactive waste during reprocessing, are also discussed in the context of TRU waste management. The overall radioactive waste management programme will be discussed in the Supplementary Report.

0-11 Critical group

The estimation of dose in the safety analysis does not employ the critical group concept. An analysis employing the critical group concept would be more conservative, and some safety standards internationally are specified for the critical group. As an example of where this may affect the analysis, the reference case assumes a dilution of nuclides in 10^8 m^3 y^{-1} of river water. This may not be a conservative assumption in the case of a critical group close to the release from the geosphere. Also, is it obvious that using well water only for drinking is more conservative than use of the well for irrigation of a small garden by a relatively small critical group?

=> Although use of the critical group concept is not explicitly stated, the approach taken in H12 is, in fact, to identify a range of hypothetical (critical) exposure groups (page V-45 to V-51). Since the "critical group" would be the group that received the largest dose for any given release, calculations have been presented in H12 that allow the comparison of the impact of the different exposure pathways/geosphere-biosphere interface combinations (page V-80 to 82 and V-98).

Taking account of the JRG comment, we think this point needs to be made more explicitly and we will include a discussion along the following lines:

> The issue of (critical) exposure group definition is the subject of many national and international discussions, since there are many problems associated with trying to define future hypothetical groups of people who might be exposed to radionuclides released into the biosphere. Unfortunately, a single international definition of (critical) exposure groups cannot be achieved since there are differences in regulatory guidance/criteria, potential future biospheres, repository locations and types, and assessment approaches, e.g. related to the stage in repository development, such as proof of concept as opposed to full regulatory compliance required for final disposal.
>
> In Japan, the AEC Guidelines do not specify the approach to be used for the definition of exposure groups. In practice, one of two approaches is normally adopted for performing exposure group dose assessments. In the first approach, the locations and characteristics of the potentially exposed individuals are defined first, together with consistent assumptions about the appropriate exposure pathways. Thus exposure groups and their characteristics (such as size, age, and behaviour) are defined before the conceptual model of contaminant migration is developed. The problem is that the critical exposure media may not have been recognised at the start, and so critical exposures omitted. In the second approach, exposure pathways are defined according to assumptions about how particular radionuclides emerge from the geosphere and migrate through various biosphere media. Human characteristics are then assumed which are appropriate to human interactions with those biosphere media. This latter approach was used for H12, since it is more logical to identify the important media first, without prejudicing exposure group assumptions. Although the AEC Guidelines do not specify the approach to be used for the definition of exposure groups, they do state that

> continuation of current human lifestyles is to be assumed. From this guidance, it is assumed for H12 that there will be constant present-day environmental and societal conditions (including climate, and farming and fishing practices). Thus any hypothetical exposure groups identified will be assumed to behave according to Japanese current lifestyles associated, but with the use of local resources and consumption of locally produced and consumed produce.

We suppose the last comment refers to the last paragraph of page V-80, which mentions the comparison of different conceptual models, namely the H3 (only drinking water pathway) and H12 (a range of exposure pathways), by using "flux to dose conversion factors". This paragraph is not intended to say which model is conservative, but rather to illustrate, via flux to dose conversion factors, the sensitivity of assessment results to certain biosphere model assumptions. It is found that both conversion factors for almost of all radionuclides are within one order of magnitude in spite of different exposure pathways. We will therefore replace the following sentence in the paragraph:

" ... Figure 5.5.1-5). Thus, the inclusion of a range of exposure pathways is conservative." with

" ... Figure 5.5.1-5), although both conversion factors lie within an order of magnitude of each other for almost all radionuclides."

0-12 Durability of waste containers

The assumption is made in the normal-evolution scenario that all containers last for their design life of 1000 y. The use of this assumption in the normal evolution scenario would force defense of the proposition that all manufacturing defects and emplacement errors can be avoided. A more easily defensible conservative analysis for normal evolution would have a small fraction of the containers with initial defects.

=> Such an analysis has already been covered on page V-99. The initial-defect case is treated as a perturbation scenario where it is assumed that all overpacks fail after 100 years.

0-13 Gas generation

The argument is made, on the basis of estimated rates of gas production and diffusion as well as experimental evidence from the Swedish programme, that gas generation is not an issue in the safety case. There would seem to be some difference of opinion internationally on this point. While some countries believe gas generation is a sufficiently resolved issue to use carbon steel containers, some avoid the use of carbon steel altogether, even internal to the containers because of the perceived difficulty of analyzing the fate of the gas reliably.

=> We recognise that there are various opinions internationally, as pointed out. The debate on this issue will be explicitly described in the final version.

0-14 Siting requirements

The assumptions made in the safety analysis for the reference case put significant limitations on the characteristics of a future site. A partial list of the requirements imposed follows:

- **The outer boundary of the repository must be located at least 10 km from any major active fault.**
- **All containers must be located at least 100 m from any fault or fracture zone that could have enhanced hydraulically conductivity. This implies locating in an area where inter-zone blocks have a linear dimension of over 2 km or segmenting the repository, which could present practical difficulties.**

- **The site must be sufficiently remote from underground resources (to be consistent with the discussion on human intrusion).**
- **The rock must not have a high fracture density.**
- **The maximum hydraulic gradient over any important zone must be less than 0.1.**
- **The site must be at least 50 km from volcanic regions.**
- **Failure of isolation by any means is not a realistic possibility at the site.**

Such limitations on the qualities of an acceptable site could present difficulties in selecting a site, which must be socially and politically acceptable as well as technically acceptable. As the margin between the results of the analysis and the safety standards is very large, it is likely that the assumptions for the reference case can be made far more conservative, thereby easing the limitations on the characteristics of an acceptable site. Furthermore, a more detailed analysis of disruptive events (*e.g.*, risk from proximity to an active fault) could serve to ease the burden to prove that such events are not possible. An easing of the limitations on site characteristics would also ease the burden of proof that characterization methods can reliably establish that the required characteristics in fact prevail at a specific site.

=> In the Workshop, we will present details of how to deal with the influences of fault movement and volcanic activity in the safety assessment, mainly focusing on the time scale over 100,000 years. We will also discuss the acceptability of the concept due to the flexible siting.

The distances mentioned in the Overview Report, such as 10 km around an active fault and 50 km from a volcanic centre conservatively define potential areas to be avoided in an initial generic siting approach. The distance was chosen to demonstrate that there exist suitable areas for the repository in Japan, even using very conservative criteria. The influence of individual fault movements and volcanic activities could be limited to smaller regions than those defined by the distances mentioned above. Thus, when more data become available, much greater areas could eventually prove suitable, compared to those mentioned in the report. It should be noted that the distances are not specified in terms of 'exclusion distances', but 'separation distances'. The aim was not to exclude any area, but rather to include areas with the highest possible degree of certainty. In other words, areas further away from active faults than 10 km and further away from volcanic centres than 50 km are highly likely to be suitable and are 'included'. This is NOT to say that areas closer than these distances are excluded.

Regarding layout of the repository, emplacement panels in smaller blocks of good rock are considered in most repository designs.

0-15 Scenario analysis

The scenario analysis relies heavily on judgment that the FEPs considered are comprehensive and complete, and to eliminate FEPs from consideration in the safety analysis. Consequently, the characteristics of the team conducting the analysis have enhanced importance. It would be beneficial to document both the methodology for establishing the team, including qualification procedures, and the methods of seeking expert opinion. The most defensible analysis would involve technical experts from a very broad range of backgrounds and personal views. Finally, it should be recognized that analysis of some scenarios, even if not required from a technical perspective, may be required to satisfy persons in the general public.

=> The FEPs and scenarios have been developed through discussions within experts from different fields, for instance, at internally organised workshops. They have been also compared with international studies including FEP lists developed by NEA and BIOMOVS. Developed FEPs and its procedures were internally documented (Umeki *et al.*, 1996a and 1996b; Naito *et al.*, 1996). However, this documentation does not expand to aspects such as methodology for establishing the team, qualification procedures, or the methods of seeking expert opinion. We have also presented our scenario approach at international conferences and at bilateral meetings with other international repository programmes.

Analysis of some scenarios which are not required from the technical scope of H12 were excluded. We would concur, however, that the requirement to gain the confidence of the general public must guide consideration of future scenario analyses.

0-16 Nuclide and chemotoxic inventory

0-16.1 The nuclear fuel cycle produces - apart from HLW - long-lived low and intermediate level waste. Is this type of waste planned to be disposed of in the same geological repository? Its impact on the environment may not be negligible.

=> The discussion on co-disposal of long-lived low and intermediate level waste (LLW and ILW) with HLW is out of the scope of H12. We will mention both LLW and ILW by adding sentences to the 3rd and 4th paragraphs on page I-1 in relation to Figure.

0-16.2 The selection of nuclides considered in the safety assessment is based on a potential hazard index related to the dilution water volume. The relation between the volume limit and a dose of 1 μSv/y is difficult to understand. Why a volumetric flow rate of 10^6 m^3/y. What about solubility? To a wide extent, the selected nuclides correspond to the nuclides investigated in other studies. In the reference case of the German safety assessment on the EU project SPA [1], the two nuclides Mo-93 and Rb-87, which were not selected here, caused a significantly higher dose than e. g. Zr-93 and Tc-99.

[1] Spent fuel Performance Assessment (SPA), EUR xxx EN, to be published.

Luehrmann, U. Noseck, R. Storck: Spent Fuel performance Assessment (SPA) for a hypothetical repository in crystalline formations in Germany. Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-155, (to be published).

=> Taking account of this IRG comment, we think this point needs to be made more explicitly to avoid confusion, and we will modify the text along the following lines:

The radionuclides to be considered in the nuclide migration analysis were selected taking into account their half-lives and a screening criterion based on a maximum nuclide concentration in hypothetical well water.

The following assumptions were made in the simple screening calculations;

- The dissolution of the nuclides from the glass waste will not occur until 1,000 years after disposal when the overpack fails;
- Water containing dissolved nuclides is immediately transported into an aquifer near the ground surface;
- The contaminated aquifer water is ingested as drinking water via a well.

Under these conservative assumptions, only the glass dissolution rate and the dilution volume in the aquifer affect the nuclide concentration in the well water. The effects of radioelement solubilities, as well as nuclide retardation in the engineered and natural barriers are conservatively neglected. The concentration in well water for each nuclide, C [Bq/m^3], is calculated by the expression:

$$C = \left(\frac{A \bullet N}{T}\right) \bullet \frac{I}{Q}$$

where:

A [Bq/canister] is the maximum activity of radionuclide per canister after 1,000 years after repository closure;

N [canister] is 40,000, the total number of canisters;

T [y] is the necessary time for the vitrified glass to dissolve completely. This value is conservatively calculated to be around 6,500 [y], in which the glass dissolution rate is 10 times greater than the value used in the Reference-Case;

Q [m^3/y] is the volumetric flow rate in the aquifer, set to 10^6 m^3/y, which is determined based on the order of magnitude for extraction rates of deep wells in Japan.

The calculated concentration in well water for each nuclide are compared with the maximum permissible concentration in water. The Science and Technology Agency (STA) specifies this concentration limit for each radionuclide for the routine releases from nuclear facilities, assuming human consumption, and this limit corresponds to 1 mSv/y for each nuclide.

Our screening criterion is based on the ratio of the calculated concentration in well water to their maximum permissible concentration in water. The nuclides whose ratio is greater than 10^{-3} are included in the safety assessment. However, daughter nuclides with half-lives of less than one year are excluded from the nuclide migration calculation on the assumption that they are in equilibrium with their parent nuclides. The remaining selected nuclides included in the migration calculations for the engineered and natural barriers were verified by comparison with assessments in other countries, and are listed in Table 4.1.3-1 of Supporting Report 3 (page IV-11).

Rb-87 was screened out by this criterion.

We think that the Mo-93 comes mainly from the irradiated structural material of fuel assemblies, and its transfer to a vitrified waste through reprocessing is limited. According to the SPA project, its inventory in spent fuel is on the order of 10^9[Bq/MTU], whereas its inventory in vitrified waste is lower by two orders of magnitude .

0-16.3 Water protection regulations usually require concentration limits or constraints for chemotoxic substances in the waste package. How is this taken into account?

=> Regarding Japanese water protection regulations for chemotoxic substances in water, there exists the Water Works Law, which was established in 1958 to provide the national drinking water standard. This law contained mainly microbiological indices, limits for inorganic substances and aesthetic items. Revisions to this standard and several supplemental criteria have been added by the Ministry of Health and Welfare (MHW) in the form of tentative guidelines taking account of effects arising from halogenated organic compounds and pesticides. MHW is now discussing reconstructing the standard to take into account of the second edition of "Guildelines for drinking-water quality" published by the World Health Organization (WHO), which specifies the permissible uranium concentration from the viewpoint of its chemotoxity.

We have used this permissible uranium concentration to compare the calculated releases of (4n+2) natural decay series nuclides to a river, as shown in Figure 6.2-3 (page VI-11 of Project Overview Report).

0-17 Legal framework

The legal framework and the respective requirements (e. g. time scales, human intrusion, safeguards) may have an impact on R+D, design, site selection and safety assessment. How will this be taken into account?

=> By the term 'legal framework' we assume the reviewer means 'regulatory basis' and enabling legislation. The background information on these issues is summarised in Section 1.3. Note that Figure 1-3 indicates that decisions on the development and establishment of final safety standards in Japan are scheduled by the basic policy described in the "Long-Term Programme for Research, Development and Utilization of Nuclear Energy 1994 by the Atomic Energy Commission, Japan", to be after the completion of this H12 report. We fully expect that such regulations will be developed with full recognition of their impact on R&D, design, siting and site characterization, as also shown in Figure 1-3.

0-18 Time scales

0-18.1 The radiotoxicity is said to be similar to that of a natural uranium ore body within tens of thousands of years. Has e. g. MOX fuel taken into account? A more conservative comparison of spent fuel with natural enrichments (K, Th, U) shows a time period of 10^5 years for the activity, heat production and toxicity of a repository to decrease to values in enrichments. Even after much longer time scales, however, the concentration of nuclides is much higher in a waste package [2].

> **[2] Röthemeyer H, Herrmann AG, Salewski H (1996): The influence of radioactive waste disposal on natural activity, heat production, and radiotoxicity. Kerntechnik 61: 245-250.**

=> The issue of MOX fuel and potential for direct disposal touches upon the sensitive issue of Pu-cycle in Japan. While the comment is technically very sensible and appreciated, the entire matter of direct disposal of any type of spent fuel, LWR or MOX, falls outside the proscribed scope of the H12 report, which is restricted solely to consideration of disposal of processed HLW in borosilicate glass. Of course, if in the future political decision is made regarding Pu-recycle, the potential for MOX disposition may have to be considered.

0-18.2 The report raises the question whether the geological environment and/or the constancy of the stress field can be predicted. A stability for the next 100,000 years is extrapolated from the observation of the stability of the conditions over the past several hundred thousand years. This applies to earthquake activities as well as to the location of volcanoes and active faults. It is pointed out in the text that this predictability must be supplemented by tectonic research activities. It remains open, however, to what extent results are already available. Extrapolation to the next 100,000 years based solely on the past constancy of conditions over a similar time scale may not be sufficient for the aspects of volcanism and earthquakes.

=> The basic concept to consider the (future) evolution of the geological system for a time of 100,000 years is based on the statement of the AEC Guidelines. This concept was also accepted as reasonable by a technical review team of Japanese geologists as being consistent with geological evidence and well-accepted plate tectonic model for the tectonic development of the Japanese islands (for the last several hundred thousand years) (see text 3.2.2).

Geological evidence for past volcanic activity shows that the active regions in Japan have not migrated significantly (<10 km) for the last several million years (see section 2.4.1 of Supporting Report 1). This suggests the possibility that the active volcanic regions will not show much change in future. On the other hand, the consequences of earthquakes have not been extrapolated from the past geological evidence because of the limited influence of ground motion associated with faulting on any deep repository environment. Allowance can be made for earthquakes by the engineering design of the repository system.

We feel that at a regional scale, such extrapolation of past geological evidence to constrain geological events in the next 10^5 years is reasonable and justified on the basis of available evidence and the current, well-accepted plate tectonic model of Japan. However, we agree that on a local scale, site-specific studies will be needed to strengthen and confirm that evidence of geological stability in the past can be extrapolated with confidence to the next 10^5 years or more.

0-18.3 In the H 12 report and in the „Guidelines" of 15 April 1997 information up to the time period of maximum influence on man and his environment is required. This time period may be considerably longer than 100,000 years. Does the requirement refer to the expected value output of a performance assessment only or does it include the uncertainty range of the results?

=> Yes, the AEC Guidelines (Part 1, Section 3.1 (2), page 13) require to evaluate the uncertainty range of the dose calculation results as follows:

"In the second progress report, the safety assessment will proceed on the basis of a main scenario which includes the key phenomena expected to determine the performance of the disposal system (*reference scenario*) and additional scenarios which take into consideration alternative assumptions and parameters (*alternative scenarios*)."

Based on these requirements, we have developed the *"reference scenario"* as a Reference Case of Normal Evolution Scenarios, which includes the key phenomena expected to determine the performance of the disposal system. As *the alternative scenarios*, additional calculational cases that take into consideration alternative design cases, alternative geological environment cases, data variation cases, alternative conceptual model cases, and "Perturbation Scenarios" are made to evaluate uncertainties related to data uncertainty, conceptual model uncertainty, and scenario uncertainty related to extremely long-term phenomena (see Figure 5.3.1-2).

Along these framework, maximum dose and peak arrival time are evaluated for 32 calculational cases and some alternative cases in the total system performance analysis (see Figures 5.5..3-3 and 5.5.3-4).

0-19 General Comments on Safety Assessment

0-19.1 a) The extension of the Safety Assessment chapter in report JNC TN1400 99-010 is small in comparison with the extension of the other parts of the document. It is deemed more information about the Safety Assessment is needed in the Overview.

=> Based on discussion in the Workshop for our response to specific comments (*e.g.* comments on scenario analysis and defining calculational cases) in order to improve transparency and traceability of Chapter V, we would extend this chapter to include adequate level of information.

0-19.2 b) To follow the Safety Assessment chapter it would be suitable to present a small description of the design and parameters used in the analysis like a basis design data and parameters for the assessment. Some information about the consideration of the geometry of the repository is necessary.

=> Geometries of the engineered barriers and repository layout for the safety assessment are shown in Figure 4.1.2-1 and Figure 5.2.2-3 of Supporting Report 3, respectively. A summary of repository design parameters will be also presented in the final version of the Project Overview Report.

0-19.3 c) It would be useful for the reader to have a brief description of the codes used in each analysis (for example, FINAS, ABAQUS, DACSAR for the thermo-mechanical analysis) and in the case of the THM coupling to know the name of the code and its description.

=> We appreciate this comment and will add brief descriptions of each code as well as the name of the THM coupling analysis code in the final version.

0-19.4 d) There is information about the analysis of the TM response of the near field, but there is no information about the TM behaviour of the far field for the long-term.

=> With respect to thermal behavior of the far field, thermal effect on the groundwater flow will be negligible if low permeability of the rock mass is considered. In addition, the repository temperature does not affect the ground surface temperature from our past T response analysis [1].

Thermal stress of the rock mass around the engineered barriers are roughly estimated as 1Mpa for soft rock and 11Mpa for hard rock. These values are smaller compared to the overburden pressure (11MPa:for soft rock system, 27MPa:for hard rock system).

For soft rock system; $\sigma_{temp} = E*\alpha*\Delta T$=3500 (MPa)*10.0e-6*15(°C)=0.5MPa ~ 1MPa

For hard rock system; $\sigma_{temp} = E*\alpha*\Delta T$=37000 (MPa)*10.0e-6*30(°C)=11.1MPa ~ 11MPa

Thermal load is strain-controlled; so a small amount of rock deformation can relieve the stress.

[1] Okamoto, J., Fujita, T., Hara, K., and Sasaki, N. (1991): Effect of Heat from High-Level Waste on Performance of Deep Geological Repository, Proc. 3rd Int. Conf. on Nuclear Fuel Reprocessing and Waste Management, RECOD'91, vol. II, pp.1034 – 1040.

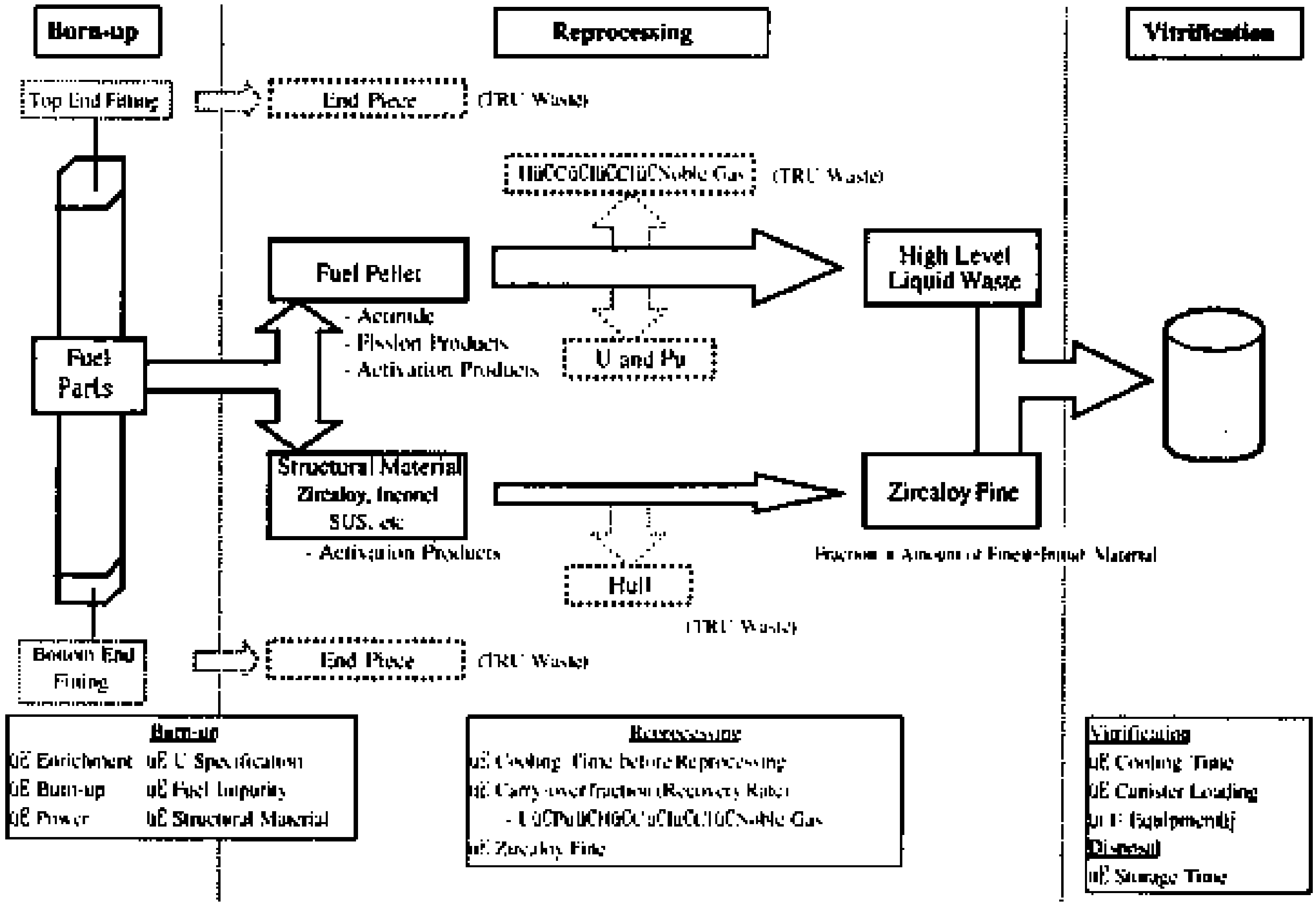
0-20 Thus, even if Mo and Rb are not present in Japan's HLW, three relevant questions regarding the H12 study are:

0-20.1 Decisions to analyse the fate of any particular nuclide, as well as to screen any particular radionuclide, should, in principle, have been documented. Has this been done? Is this documentation also available as part of the H12 study?

=> We are preparing a technical report (although in Japanese) on a procedure of identifying the inventory including its history. This report is one of technical reports for providing detail information for Supporting Report 3.

0-20.2 How well has the radioactive inventory of the waste been established?

=> The inventory in accordance with the fuel reprocessing flow chart is shown in the following Figure.



0-20.3 Similar to the the previous question: important decisions to screen this or that scenario, to screen this or that mechanism, to use this and not that code, should have been documented. Is the decision trail - and the rationale for decisions - generally documented?

=> The rationale for decisions is summarized in Supporting Report 3 and JNC Technical Report, which are publicly available. In addition, the main decision trail has been recorded internally as follows.

The H12 project has started since 1995 with establishing a project management structure, which has a Core Group and nine Working Groups.

More than 700 Technical Memoranda according to the work breakdown structure (WBS) which is corresponding to the preliminary table of contents for three H12 Supporting Reports had been prepared in order to summarize the rationale for decisions made in the Core Group and the Working Groups. Important decisions in the meetings of these Groups had been recorded with signature of the Core Group leader for approval.

The FEPs and scenarios have been developed through discussions with experts from different fields, for instance, at internally organized workshops. They have been also compared with international studies including FEP lists developed by NEA and BIOMOVS. Developed FEPs and its procedures were internally documented (Umeki *et al.*, 1996a and 1996b; Naito *et al.*, 1996). However, this documentation does not extend to aspects such as methodology for establishing the team, qualification procedures, or the methods of seeking expert opinion. We have also presented our scenario approach at international conferences and at bilateral meetings with other international repository programmes.

Since the Coordination Conference on R&D for Geological Disposal was set up in 1997 (see Page I-10), drafting of H12 Reports have been initiated. A Working Group and three Task Forces, established by the Coordination Conference, have carried out detailed technical reviews of identified key issues and the feedback from these reviews has been incorporated into revising the drafts. Important decisions in the meetings of these Groups have been recorded and finally approved by the Conference.

2 COMMENTS ON CHAPTER I

1-1 Page I-4: last paragraph before section 1.2

Hazard of the waste expressed as volume of water required to dilute the activity to drinking water limits. This index is obsolete. If hazard indexes are mentioned it might be worthwhile to refer to a more recent review [LILJENZIN J.-O., RYDBERG J., Risks from Nuclear Waste, Revised edition, SKI report 96:70, Stockholm (1996)].

=> We will include the above recent review document for the hazard index as one of the references. However, we would not necessarily agree that the index we refer to is obsolete: it depends on the illustrative purpose that is intended. Our text says only that it is one alternative given as an example.

1-2 Number of waste packages

Which number of waste packages will be disposed of in the repository? There are no indications in this chapter for which volume of waste the repository is to be designed. The number of waste packages to be disposed of may be an important factor to consider in siting the repository. Is more than one repository foreseen? On page IV-70 a scenario of emplacing 40,000 packages is discussed. What is the background to the number and why is it not discussed in Chapter 1.1.1?

=> In H12, 40,000 waste packages are assumed for design study and performance assessment. The number is based on the following discussion in the Nuclear Subcommittee of the Advisory Committee for Energy, which was established under the MITI (also see page III-9 of Supporting Report 2).

"Assuming that a disposal facility will have a capacity of equipment to receive 1,000 waste packages in a year, the relationship between the scale of the disposal facility (i.e. the total number of vitrified waste package received in the disposal facility) and the disposal cost per one package is estimated. In result, if the number of waste package exceeds 40,000 packages, the disposal cost is almost independent of the number of waste package. Therefore, it is appropriate that the scale of the disposal facility is assumed to be 40,000 waste packages as the premise for the cost estimation of the geological disposal project.

In addition, 40,000 waste packages corresponds to the total electric power that is generated at the nuclear power plants in Japan from 1966, when the first domestic and commercial reactor started operations, to around 2015."

We don't think this is really essential for Chapter I, although the fact should perhaps be mentioned that the total inventory to be emplaced in a single repository is not, as yet, defined and that the inventory assumed for H12 is hypothetically based on reactor operations until 2015. The above paragraph ("in addition, 40,000 ...") could perhaps also be included as a footnote on page IV-70. In addition, options for further repositories should probably be explicitly mentioned in Chapter I.

1-3 Comments on 1.1.1

This section is meant to be widely readable but it isn't; instead it turns out to be confusing.

1-3.1 Is it true that 50 kg of U are "consumed" per MTU ? Please specify initial enrichment. Wouldn't it be simpler to say that a certain amount, Y, of U is transformed in energy, X, in fission products, Z, and in transuranium actinides, W? Is it not worth mentioning also the production of activation products that end up in HLW?

=> We understand this suggestion. We believe it will be sufficient to add text that notes formation of actinides in addition to Pu as well as fission products and activation products, but to avoid citing specific numerical amounts. Remember, this is merely a background section within a summary volume.

1-3.2 "Spent fuel" is not normally meant to include fuel being burnt in reactors. If it is defined that way, then one should introduce the category of "spent fuel in storage".

=> We will make the following changes to clarify the text: "...,spent nuclear fuel is discharged from a reactor and is left..."

1-3.3 Have you checked for readability by the intended audience ? If the target audience are "the authorities and other interested parties in Japan", they will not need this type of introduction.

=> Based on extensive discussions with policy and technical representatives in Japan, we feel that this material is appropriate for this overview volume. Remember that a separate Supplementary Report, written only in Japanese, is being written to provide more information to the wider audience of the Japanese public.

1-4 Comments on 1.1.

1-4.1 Perhaps it is worth noticing that after about 1000 years, radioactivity is only a factor ten higher than "natural" ore.

=> We will add this clarifying remark to the text.

1-4.2 It may be worth noticing that high heat release rates would occur during the first 1,000 years, thus there would be a "thermal period" to the repository.

=> We feel that this comment regarding impacts on repository temperature is somewhat pre-mature in this section, because the concept of geological disposal has not yet been introduced. This matter is addressed on a technical basis in Section 4.3.2.

1-5 Comments on 1.2.1

This section seems week. Two references only, one of, which is very old and minor. Does the report need to mention principles? If yes, more and better has to be done. On the other hand, the mission of JNC is to fulfill the AEC mandate; it is there that principles should be found, as well as the rationale behind those principles.

A better reference to [OECD/NEA, 1977] is ICRP-77 (§2(c); Sect. 6.2.3)

=> We count three references, but will gladly also include your cited additional reference. Again, please recall that a much deeper analysis of this same topic is included in a separate Japanese only volume Supplementary Report of H12; we think that this admittedly brief treatment here is appropriate for the purpose of this overview volume.

1-6 Comments on 1.2.2

Same general comment as in 1.2.1. Besides, some of the itemized statements seem to pre-judge the results of the present work.

=> Please see our response to comment 1-5. The comment on "pre-judge" was not clear to us, and perhaps it could be amplified.

1-7 Comments on 1.3

1-7.1 The report could well do without sections 1.1 and 1.2 and start with this section.

=> The Coordination Conference on R&D for Geological Disposal (see Figure 1.4, page I-10) desired these sections should be briefly described on the top of Project Overview Report so as to remind reader's understanding of the background on HLW disposal in spite of recognising that the intended audience are experts and involved organisations.

1-7.2 Only in this section - and at the end of it - the reader finds out that geologic disposal is the "favored option for high-level waste in Japan". Can't this be said earlier? Can't a reference be given to Japanese sources, including why this is the favoured option in Japan?

=> As described in section 1.2.2 (page I-5), the AEC decided in 1976 that the focus should be placed on geological disposal "for the time being" and this was later adopted as official Japanese government policy. Since then PNC (now JNC) and other involved organisations have been developing R&D programme on this policy. The policy has been not yet changed. This section is intended to briefly summarise the history and the current status of Japanese R&D programme for HLW disposal, especially toward H12. As a result of these R&D programme so far, geological disposal can be regarded as the most feasible and practical option in Japan. This AEC (1976) reference is also cited at the beginning of section 1.3.2.

A2.3 Comments on Chapter II

2-1 Section 2.3.1: Definition of safety goals

Terminology in this section is not consistent with international use. Principles about protection of future generations is not a 'rule'. Mention also difficulty of predicting future environmental conditions. 0.1 mSv y^{-1} and 0.3 mSv y^{-1} are not dose limits but constraints.

=> We will modify the text in section 2.3.1 so that the terminology is consistent with international use.

Concerning the protection of future generations, we will modify the following text in page II-3:

"The IAEA has suggested a rule (IAEA, 1989) which states that the safety of future generations should not fall below a level which is acceptable to the current generation.", to

"The IAEA has suggested a principle (IAEA, 1989) that states that the safety of future generations should not fall below a level which is acceptable to the current generation."

For the second comment, we should modify the following text in page II-3 to II-4:

"Individual dose is generally calculated ...", to

"Due principally to the difficulty in predicting future environmental conditions, individual dose is generally calculated ...".

"Moreover, the safety standards and guidelines applied in each country generally conform to the recommendations of the ICRP (ICRP, 1985), which specify that the annual dose limit for individuals should range from 0.1 mSv y^{-1} to 0.3 mSv y^{-1}.", to

"Moreover, the safety standards and guidelines applied in each country generally specify annual dose constraints for individuals that range from 0.1 mSv y^{-1} to 0.3 mSv y^{-1}, taking account of the recommendations of the ICRP (ICRP, 1977, 1985, 1991).".

and add the following references.

- ICRP (1977): Recommendations of the ICRP, ICRP Pub.26, Annals of the ICRP 1(3), Pergamon Press, Oxford. UK.

- ICRP (1991): 1990 Recommendations of the International Commission on Radiological Protection, ICRP Pub. 60, Annals of the ICRP 21 (1-3), Pergamon Press, Oxford, UK.

2-2 Section 2.3.2: Measures to be taken: middle of page II-6

I would add a few words about the need to develop conceptual models. Validation of modelling results regarding long term performance of disposal system is impossible. Reasonable assurance is the best that can be achieved. Page II-7: Oklo is not necessarily the best natural analogue; it is quoted too much.

=> We would add few words on conceptual model development in the middle of page II-6, i.e. "Next, conceptual models are developed that represent the evolution of the system. From these, mathematical models are derived, for which relevant input data are acquired"..

Concerning the comment on "validation", we think our usage here is consistent with the reference given (IAEA: Radioactive Waste Management Glossary, 1993). We recognise that full validation of all assessment results can be defined as an impossible ideal. We may, therefore, modify the text:

"... models and data are fully verified and validated...", to

"... computational codes are fully verified and models and data are adequately validated for the purposes of the assessment ...".

If as here, the intent is to validate the confidence in the interpretation of the model results within the context of the assessment, then validation is an entirely appropriate word (see *e.g.* Kristallin-I).

The last comment on the Oklo study is true, but it is the best analogue of the fundamental feasibility of geological disposal, even if it is much less useful for "supporting modelling of radionuclide migration behaviour".

We may, therefore, modify the text:

"... the Oklo deposit is used to support modelling ...", to

"... the Oklo deposit is used to support the fundamental feasibility of geological disposal, whereas other observations of natural systems are used to support modelling ...".

2-3 Comments on 2.2.2 and 2.2.3

2-3.1 Figure 2-1 does not capture the complexity of the geologic environment in Japan. In particular, the rock could be pictured as being fractured at several scales.

=> This figure is not intended to capture such complexity; it's purpose is to illustrate the key components of a geological disposal system in Japan. Pictures of the potential fracture complexity are adequately shown in Figures 3.3-1 and 3.3-20, for example.

2-3.2 One condition for gradual dilution and long transport times is situating the waste packages sufficiently far-away from important fracture zones. The ability of finding sites with sufficient "respect distances" is an important consideration not mentioned here.

=> We will add a cross-reference to the information contained on page IV-6, as well as Figure 4.5-11 and section 5.4.1, Figure 5.4.1-1, and page V-38 subsection 2) of section 5.4.1. We will add text in section 4.2.2 that discusses the need for adequate "respect distance" between the EBS and major fractures. Finally, we will also note that as part of our calculational cases, we have evaluated the relative impact of "respect distances" on repository performance and safety: see Figure 5.5.1-8.

2-3.3 Overall, the role and functions of the individual barriers does not seem to be clearly captured in these sections. See SKB's SR-95 (and perhaps SR-97) for a typical description.

=> The safety functions of the various barriers, as summarized in Table 5.2-1, will be cross-referenced in this section.

3 COMMENTS ON CHAPTER III

3-1 Page III-1; paragraph 3

Statement that evolution of geological system should be predicted for 100,000 years is inconsistent with PA requirement to assess peak impact. It needs to be reworded. It could become a discussion point where it is explained how reliability of predictions decreases with distance in the future. This should be done throughout the report.

=> There is apparently some confusion regarding the 100,000-year figure that must be clarified within H12. The AEC Guidelines identify 100,000 years as an *initial target* for the confirmation of geological stability for a potential disposal site with respect to natural events (volcanism, earthquakes, uplift and erosion, and, climate and sea-level change). Based on well-accepted plate tectonic models by Japanese geoscientists and supplemented by collected and interpreted data reported in H12, we conclude that it will be possible to identify geologically stable sites in which significant perturbation by natural events will be unlikely for the next 100,000 years.

There is no intent by either the AEC or JNC, however, to use this figure of 100,000 years in a performance assessment. Consistent with international safety assessments of HLW, H12 safety calculations extend far into the future to capture peak dose release rates. Because H12 considers perturbation scenarios occurring over a range of initiation times or rates, some calculations extend as far as 10^7 years into the future.

We believe it will be possible to identify and confirm candidate disposal sites that will geologically stable for the next 100,000 years. Hence, most of the H12 perturbation scenarios are initiated after that time. We recognize, however, that there may be concern regarding potential unexpected perturbation events earlier than 100,000 years. Therefore, we have also included certain perturbation scenario calculations that are initiated before 100,000 years. But in all cases, we intend to extend the calculation time to encompass the time of peak dose release rate.

3-2 Section 3.1.2: Long-term stability of the geological environment

I suggest to modify list; it should have 5 points. A point iv. should be added as follows: iv. erosion/denudation.

=> The list is based on the items given in the AEC Guidelines. The proposed point iv) is, we believe, already covered by point iii).

3-3 Section 3.2.3: Fault movement: page III-11

Apparent belief that faulting risk can be handled by staying away from active faults may be overly optimistic. In a tectonically active region like Japan the risk of faulting can be reduced by careful siting but not eliminated. I believe this section will need to be revised to a certain extent and the changes reflected in the safety assessment.

=> Japan is located in one of the most tectonically active areas in the world, and fault movement is more significant than in most other countries. In Japan, the study of major fault movement ("active faults") has therefore progressed and a lot of knowledge has been collected. Accordingly, evidence for fault movement in Japan during the past several hundred thousand years suggests that future fault movement in the area of interest can be estimated (reasonably and reliably) by extrapolating information on existing faults. These findings also suggest that the possibility of generating new major faults without any relationship to existing faults is slight. If this message is misleading in the text, it will be revised.

This treatment is consistent with that in other countries: eg Switzerland, where tectonic movement is also significant.

The possible influences of new fault generation is under consideration as an optional scenario for safety assessment. In the Workshop, we would like to present examples of the influence-analysis of fault movement on safety assessment, mainly focusing on a period of 100,000 years.

3-4 Oxidizing conditions; page III-26

In the performance assessment for a granitic repository in Sweden, the consequences of intrusion of oxidizing waters after a glaciation will have to be analyzed even if it is not a main scenario. The volcanic waters in Japan have very low pH (down to pH 2) and probably also contain aggressive solution species. Has the consequences of temporary ("accidental") intrusion of such waters into the repository been analyzed? If not, what is the reason?

=> Concerning the first comment, we think it is reasonable only for locations in which continental ice-sheets are expected.

As shown in the Overview Report, page III-25, the influence of volcanic activity for groundwater chemistry can be expected up to around 20 km at maximum from the volcanic centre. The intrusion of the volcanic water, therefore, can be excluded by appropriate siting (excluding volcanic areas).

The accidental intrusion scenario of volcanic water in repository system is under consideration as an optional scenario for the safety assessment. In the Workshop, we are going to present the possible influence by volcanic activity focusing on the time after 100,000 years.

3-5 Fault movement

It is assumed that future fault movements take place in existing faults. Even if that is the case in an overwhelming number of situations, it cannot be 100% true since the active faults must also have been created once. Has the consequences of rock movement through the repository been considered as an accident scenario? If not, what is the reason?

=> See responses to comments 3-3 and 5-5.

3-6 Section 3.2.5. Uplift/subsidence and denudation/sedimentation

Not enough consideration is given to localized erosion processes.

=> As a localised erosion process, deepening by rivers is particularly important in Japan. The process of river bed denudation progresses linearly in the long-term, while repeated short-term filling and deepening processes develop within the glacial-interglacial cycle. The geological (topographical) evidence suggests that, in Japan, the denudation rate of a river bed is typically several tens of meters during a 100,000 years (see III-32). We think this level of information is enough for the present generic study.

3-7 Section 3.2.6. Climatic and sea-level changes

I believe that too much importance is given to future climate. Firstly, impact on repository of climatic change can be expected to be fairly insignificant Secondly, predictions of future climate in consideration of possible major modifications caused by human actions are quite questionable.

=> We agree. Discussions are, however, required by the AEC Guidelines. We therefore need to take into account future climatic change, even though it is relatively unimportant, and difficult to predict, compared to other natural phenomena. In this chapter, the evidence for climate change that has been observed (*e.g.* changes in temperature, rainfall and sea-level) is described. The influence of these changes on a repository system is discussed and analysed as an example in chapter V (safety assessment, see V-99). We may, however, consider adding a brief discussion of the uncertainties related to the effects of present and future human activities on the climate.

3-8 Faults

The question arises if active fault zones can also develop at larger distances than 10 km from fault zones that are active today during the time periods of concern. The answer to this question may require a special safety assessment of possible effects of future tectonic processes at the active faults.

=> See response to comment 3-3.

3-9 Volcanoes

Should the safety distance around a known volcano be based on a radial distance or on a distance from volcanic belts?

=> The influencing distance of volcanoes estimated from the geological evidence written in the report is the distance from the center (or central area) of a volcano. The distance mentioned i.e. 50km is the most conservative (or unrealistic) length. Realistically, any actual repository site will be chosen to be at a sufficient distance from volcanic influences after the characterization of influences of each specific volcano or volcanic region.

The safety distance will *operationally* be defined as a distance from a volcanic front of a volcanic region (page III-18), although *effectively* this distance is derived from analysis of thermal perturbations measured in a radial distance from volcanic centers as shown, for example, in Figure 3.2-12. The actual safety distance is somewhere between 10 and 50 km, the latter number being a conservative upper bound.

3-10 Groundwater

3-10.1 Data referring to the movement and chemistry of groundwater are mainly based on the Tono and Kamaishi area. Are they representative for a possible disposal site?

It is generally assumed that the permeability decreases with depth. This is not confirmed in Supporting Report 1, chapter 3.3.1 (see fig. 3.3-6). It may have an impact on the assumed decrease of the groundwater movement with depth. Fractures and fluid components have been observed in several 1000 m in Germany [3]

> **[3] Kessels W (1996): Geohydraulische Erkenntnisse aus der kontinentalen Tiefbohrung (KTB) Oberpfalz. DGG Mittlg 4: 77-81.**

=> Most data used in H12 SA analyses are derived from a number of data obtained from literature including Tono and Kamaishi data. Although Tono and Kamaishi are not necessarily representative for a possible disposal site, some data that are not found in literature (*e.g.*, transmissivity distribution) are guided by data obtained at these sites.

We also recognized that permeability obtained below sea level at Tono is smaller than those obtained above sea level at Kamaishi. The permeability of host rock at the depth of future repository is expected to be smaller than those obtained at shallow depth. We would like to know further information about the permeability decreases with depth observed in Germany.

Finally, general summarized data such as Figure 3.3-6 should not be over-interpreted, as claiming that all potential sites will display decreasing permeability with depth. Such measurements of permeability vs. depth will be of prime importance for collection in any future site-specific characterization programme.

3-10.2 The groundwater chemistry depends e. g. on the sea-level changes and the boundaries and the mixing of salt and freshwater. Do reliable long term predictions need further investigations?

=> At this stage of generic assessment for H12, it is probably meaningless to evaluate any long-term extrapolation when site specific information on geological environment conditions is lacking. Therefore, at this stage we have introduced simple but bounding assumptions for perturbation scenarios to illustrate possible impacts of external perturbation factors on the expected system performance under the normal evolution scenario.; *e.g.* for the climate/sea-level change scenario, the 10^5-year cycle climate change which will cause the change of groundwater type from the saline-type water (SRHP) to the fresh-type water (FRHP) in the repository is assumed. Verifying the credibility of such an assumption would form an important part of any future site-specific characterisation study.

3-11 Minor comments

3-11.1 Page III-43: There is an inconsistency between parameter (intrinsic permeability) and units (m^2/s) for fractures measured at Kamaishi.

=> Parameter is not "intrinsic permeability" but "transmissivity". We will correct the text.

3-11.2 Page III-60: Table 3.3-2: There is a negative value for the linear thermal expansion coefficient of crystalline rock (-4.53).

=> This data is based on the data from basaltic lava. Negative values of thermal expansion coefficient are presumably caused by contraction of the test sample due to the dehydration of water in hydrous minerals by heating.

4 COMMENTS ON CHAPTER IV

4-1 Emplacement configuration: page IV-4

Configurations (i) and (ii) have been to "provide a good contrast with one another". To me they seem to be nearly identical. The only difference is a 90 degree turn, while (iii) and (iv) are different from (i) and (ii) by having one package per deposition hole rather than may. Please explain the "contrast" between (i) and (ii).

In configuration (ii), how is it foreseen that the weight of the column of overpacks is supported during emplacement?

=> "(ii)" actually should be "(iv)" in the first and second paragraphs on p.IV-4 in the Project Overview Report (typos; see Supporting Report 2, where the description is correct).

In configuration (ii), the weight of the column of overpacks could be supported, because the lower construction tunnel would be backfilled before emplacement commences.

The "contrast" relates to engineering, rather than long-term performance. This should be clarified in the text.

Finally, general summarized data such as Figure 3.3-6 should not be over-interpreted, as claiming that all potential sites will display decreasing permeability with depth. Such measurements of permeability vs. depth will be of prime importance for collection in any future site-specific characterization programme.

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In configuration (ii), the weight of the column of overpacks could be supported, because the lower construction tunnel would be backfilled before emplacement commences.

The "contrast" relates to engineering, rather than long-term performance. This should be clarified in the text.

4-2 Oxidizing conditions of groundwater: page IV-16

The granite groundwater is not reducing until at a depth of 500 - 1000 m. This is considerably deeper than in Swedish granite. Is the reason for this difference known?

=> According to data from the Tono area on current properties of granitic groundwater at about 200m to 300m below ground surface, the Eh is around 0 mV, which is reducing with respect to the Eh at the ground surface. The groundwater at depths greater than 500 m is more strongly reducing, at about ~300 mV (see Project Overview Report p.III-54). Though the reason for the difference compared to Swedish granite is not known, an indication is provided by tritium, which has been found in the Tono groundwater at a depth of about 100 m. This indicates that surface water can permeate relatively quickly along faults and fracture bands. The permeating water becomes more strongly reducing with increasing depth, as it reacts with the rock. There is, however, a significant possibility that strongly reducing groundwater will also be found in granite at shallower depths in the future. The contents of the first paragraph on page IV-16 in the Project Overview Report is intended to mean that "it is highly likely that groundwater is strongly reducing at depths of about 500 m, even in granite".

The description will be modified in the final version to avoid confusion.

4-3 Buffer: page IV-19

There are no design requirements on the ability of the buffer to mechanically buffer minor shear movements in the rock. Are such movements totally excluded as possible events or are they considered as upset conditions for which no safety margins are required?

The buffer must have good thermal conductivity to avoid overheating of the glass wasteform with crystallization as result. What is the maximum allowed temperature for the glass?

On page IV-23, the maximum overpack surface temperature is set to 150°C. Is this constraint the design basis and not the wasteform temperature?

=> Active faults, with shear deformation of the rock mass, are assumed to be avoided by site selection. The ability of the bentonite to mechanically buffer minor shear movements in the rock is implicitly considered as a design requirement for stress buffering. In order to estimate the influence of shear movements in the surrounding rock on the EBS, test equipment is being prepared, referring to the results of a study by SKB.

The maximum allowable temperature for the vitrified waste is set at about 500 °C, as indicated in the footnote on page IV-67 of the Project Overview Report. The thermal conductivity of the buffer is a more important factor in ensuring that the temperature constraint on the bentonite is met, than it is for the temperature constraint on the glass.

The temperature of 150 °C is set conservatively in order to evaluate the material strength of carbon steel, which is used for calculating the pressure resistance thickness of the overpack. It is not a constraint on the EBS design, and this will be clarified in the text. As shown on page IV-69 of the Project Overview Report, the design ensures that the maximum temperature of the overpack does not exceed 100 °C (since it ensures that the maximum temperature of buffer material is below 100 °C).

4-4 Overpack thickness

There is a slight logical problem in the way the corrosion depths are estimated. To do this, the surface area of the container must be known and this will require knowledge

of the container's actual wall thickness. Have you considered as a different approach to determining the overpack thickness to:

1. **determine the thickness necessary for mechanical reasons.**
2. **determine the thickness necessary for radiation shielding.**
3. **determine the thickness necessary for corrosion protection.**

=> The thickness of the overpack is estimated taking account of corrosion resistance, pressure resistance and its radiation shielding function. These aspects are interrelated. The corrosion allowance of the overpack is, for example, used as boundary condition for the calculation of pressure resistance, because the consolidation reaction force due to corrosion expansion is applied to the buffer. In the procedure indicated in the reviewer's comment, if the mechanical interaction between the overpack and the buffer is considered, the thickness necessary for pressure resistance is determined based on an assumed corrosion allowance, and the thickness necessary for radiation shielding is assessed against pressure resistance thickness plus corrosion resistance. The thickness of the overpack is, therefore, determined in an iterative way.

4-5 Corrosion due to oxygen: page IV-24

Equation 4.3-2 is incorrect. It should read: $P = Xm + 7.5Xm^{0.5}$ (see Supporting Report 2).

Please explain the reasons for the choice of this equation. Equations of the type:

$$P = kt^n$$

have been tried by others, including your ref. Honda (1995).

=> Equation 4.3-2 will be corrected according to the reviewer's comment.

In the evaluation of corrosion, it has been conservatively assumed that all the oxygen trapped in the buffer and backfill materials in the disposal tunnels contributes to the corrosion of the overpack after emplacement. Average corrosion depth can be calculated by assuming chemical reaction with all the trapped oxygen. To obtain from this the maximum depth of corrosion front, the equation, $P = Xm + 7.5Xm^{0.5}$ has been used. Honda *et al.* (1995) applied an equation of the type $P = kn^t$ to describe empirically experimental results obtained for time dependency of maximum corrosion depth. In H12, however, corrosion by reaction with trapped oxygen is assumed to be a rapid, so that we conservatively neglect the time-dependency of this process.

4-6 Bacterial corrosion: page IV-24

The current understanding is that the bacterial processes are not totally dependent on the amount of organic matter. The presence of hydrogen and methane may also contribute. Would this change the estimated corrosion attack due to bacterial processes?

=> There is the experimental evidence that the activity of the sulfate reducing bacteria (SRB) is low in the compacted bentonite (see p.IV-16 of Supporting Report 2). In H12, the effect of the SRB on corrosion is assessed based on the general understanding that the SRB are heterotrophic bacteria, and the transport of sulfate ions does not affect the estimated corrosion depth. However, even if it is pessimistically assumed that:

- all the sulfates ions in the bentonite are reduced to sulfides,
- the iron is corroded to ferric sulfide (FeS),
- the concentration of sulfide in the groundwater is very high (3.0×10^{-2} M) and contributes to corrosion,

- the transport of sulfide determines the corrosion rate,

then, even in the severest case of disposal pit vertical emplacement in a soft rock system, the corrosion depth is estimated to be only 5 mm in 1000 years (compared to 2 mm given in H12).

In H12, the corrosion rate due to water reduction is set to two times the measured value, to take account of the uncertainties in the environment (see p.IV-21 of Supporting Report 2). Based on the measured value, however, the corrosion depth obtained after 1000 years is approximately 5 mm (compared to 10 mm in H12).

Thus, even if a highly pessimistic estimate of bacterial corrosion is made, the conservatism in the corrosion due to water reduction suggests that there is no need to change the estimated value of the corrosion depth. This discussion will be incorporated into the final version.

4-7 Containers resistance: page IV-27

The pressure resistance calculations are based on the provisions for second-class containers documented in "Notification No. 501, Technical Standard for Construction of Nuclear Power Plant Components".

Do these provisions take into account the load situation in the repository? A brief account for the design basis cases in Notification No. 501 would be appreciated.

=> No, Notification No. 501 is not directly applied. Rather, we refer to ASME Code Section III, as described at pages IV-27 of Project Overview Report and IV-41 of Supporting Report 2.

4-8 Radiation shielding: page IV-28

4-8.1 What is the dose rate at the waste-form surface at the time of disposal of the waste?

=> Calculated absorbed doses on the overpack wall surface, at the time of repository closure, are shown in Figure 4.1.1-19 (p IV-43) of Supporting Report 2.

4-8.2 The reference Marsh *et al.* 1989 seems to be incorrect. I can find no contribution from Marsh in Corrosion vol. 45 (1989). Please advice.

=> Concerning Marsh *et al.*'s paper (hereinafter, the Marsh paper) entitled "Corrosion of Carbon Steel Nuclear Waste Containers in Marine Sediment" (Marsh, G. P., Harker, A. H. and Taylor, K. J. (1989), Corrosion Vol.45, No.7, pp579-589.), the following four points are referred to in the text from page IV-28 to IV-29 of Project Overview Report.

- Equation 4.3-3 is derived from two equations presented in the paper. This derivation process is described on page IV-44 of Supporting Report 2.

- The λ (the radiation attenuation factor) value of 6.25 comes from the description in p.581 of the Marsh paper.

- The G value of 2.13 (converted for bivalent oxidizing chemical species) comes from the description in p.581 of the Marsh paper.

- In the first paragraph on page IV-29 of the Project Overview Report, it is stated that "Localised corrosion of iron due to radiolysis dose not develop if the cathodic current density is lower than the passive current density of carbon steel." This is based on the following fact presented in the paper.

> "Assuming for the moment that a protective layer will form on the containers, then the minimum requirement for the bulk of the metal surface to act as cathode to the localised corrosion sites is that the flux

of oxidizing species to the metal surface should support a cathodic current that at least balances the anodic dissolution current flowing through the protective layer. The mass transport of oxidizing species to the container surface will be under diffusion control, and therefore the above balance can be expressed as:

I_{cath} = - n F D ?c/?x

Where I_{cath} is the maximum cathodic current density: ?c/?x, the concentration gradient of oxidising species at the container/sediment interface; and D, the valence and diffusion coefficient of the oxidising species; and F, Faraday's constant."

4-8.3 The dose rate will decrease with time and the highest dose rates will occur at the time when there is oxygen from other sources present in the repository. The radiolysis will not be important for the corrosion processes (i.e. pitting) until the oxygen level in the repository has dropped to levels when passivity cannot be maintained any longer. When does this happen? Also, adding the corrosion allowance to the radiation protection seems overly conservative. The dose rate drops to nearly zero during 1000 years. That is after the corrosion allowance has been used, there is no longer a need for radiation protection. Please comment.

=> As pointed out in page IV-23 of Supporting Report 2, the transport of oxygen in compacted and water-saturated bentonite is extremely limited. Our experimental results (Honda *et al.*, 1995) suggest that the flux of oxygen is insufficient to drive localised corrosion when the thick compacted bentonite (*e.g.* over 100 mm) is saturated with groundwater. Because compacted bentonite acts as a strong barrier for mass transport, the oxygen flux equivalent to the passive current density therefore cannot be maintained under such conditions, and localised corrosion will thus not be propagated.

Table 4.4-4 (page IV-89) of Project Overview Report shows that estimated time for the buffer to resaturate would be in the range of 5 years to 50 years after repository closure except for an extreme case. Around this time scale dose rate arising from gamma and neutron radiation is still so high that oxidants produced by radiolysis at the overpack surface could replace the oxygen flux from other sources in terms of driving localised corrosion. The calculated highest dose rate (at the time of repository closure) is therefore used taking account of uncertainty on both the resaturation time and the radiolysis process, and, from this, the required radiation shielding thickness for the overpack is determined to be 150 mm (page IV-29 of Project Overview Report).

To determine the thickness of overpack, three factors are taken into account, namely, corrosion resistance, pressure resistance and influence of radiation. As described in page IV-29 of Project Overview Report, the overpack thickness is specified by summing the corrosion allowance (40 mm) and the required radiation shielding thickness (150 mm). The resulting thickness can satisfy the required pressure resistance thickness (30 - 110 mm). A uniform thickness of 190 mm will be adopted for both the end and cylindrical shell sections for both hard rock and soft rock (Table 4.3-9, page IV-29 of Project Overview Report).

Adding the corrosion allowance (40 mm) to the required radiation shielding thickness (150 mm) might be conservative as pointed out in the comment of the reviewer, but the pressure resistance thickness (30-110 mm) and corrosion allowance are independently required at least up to 1000 years after closure. Taking account of uncertainties in the other influences of radiolysis, we have concluded that the difference of thickness (40-120 mm) between pressure resistance and influence of radiation could be accounted for in the safety margin for corrosion allowance, as the dose rate will decrease with time.

4-9 Page IV-34

A better reference for Hallberg *et al.* is: Applied Geochemistry Vol. 3 (1988) 273-280.

=> We will refer to it. Thanks.

4-10 Page IV-34

All sulfates in the bentonite have been converted to sulfide. How does this take place? Is microbial activity expected in the bentonite? If so, how has this been taken into account for the steel corrosion?

=> In the calculation of porewater chemistry in the buffer material, it is assumed that an oxidation/reduction equilibrium in the sulfate/sulfide system is achieved, even though there is the experimental evidence that the activity of the SRB is low in the compacted bentonite. In the case of the seawater system, because sulphate in the groundwater is reduced to sulfide, it is assumed that sulphate in the bentonite would be also reduced to sulfide. This assumption is conservative even if the mechanism is undefined. Stress corrosion cracking (hydrogen induction cracking) of the carbon steel is examined based on this assumption. With respect to the effects of the SRB on the corrosion depth, please refer to the answer to comment 4-6.

4-11 Mechanical properties: page IV-38: point d)

"...to alleviate the effect of mechanical changes that may occur around the engineered barrier system". Which mechanical changes are design bases?

=> The corrosion expansion of the overpack and creep deformation of the rock mass (especially in the soft rock system) are considered as mechanical perturbations to the engineered barrier system, in both the overpack and buffer design. For example, the curve e. of stress-buffering function for the soft rock system in figure 4.3-19 (page IV-47) of the Project Overview Report has been calculated taking into account these mechanical changes.

4-12 Disposal configuration: page IV-53

In figure 4.3-23 it looks like the pit disposal version has only one package per pit. (See also figure 4.3-41) This is different from what was discussed on p. IV-4, where the pit disposal concept had more than one package per pit. What is the reason for this difference?

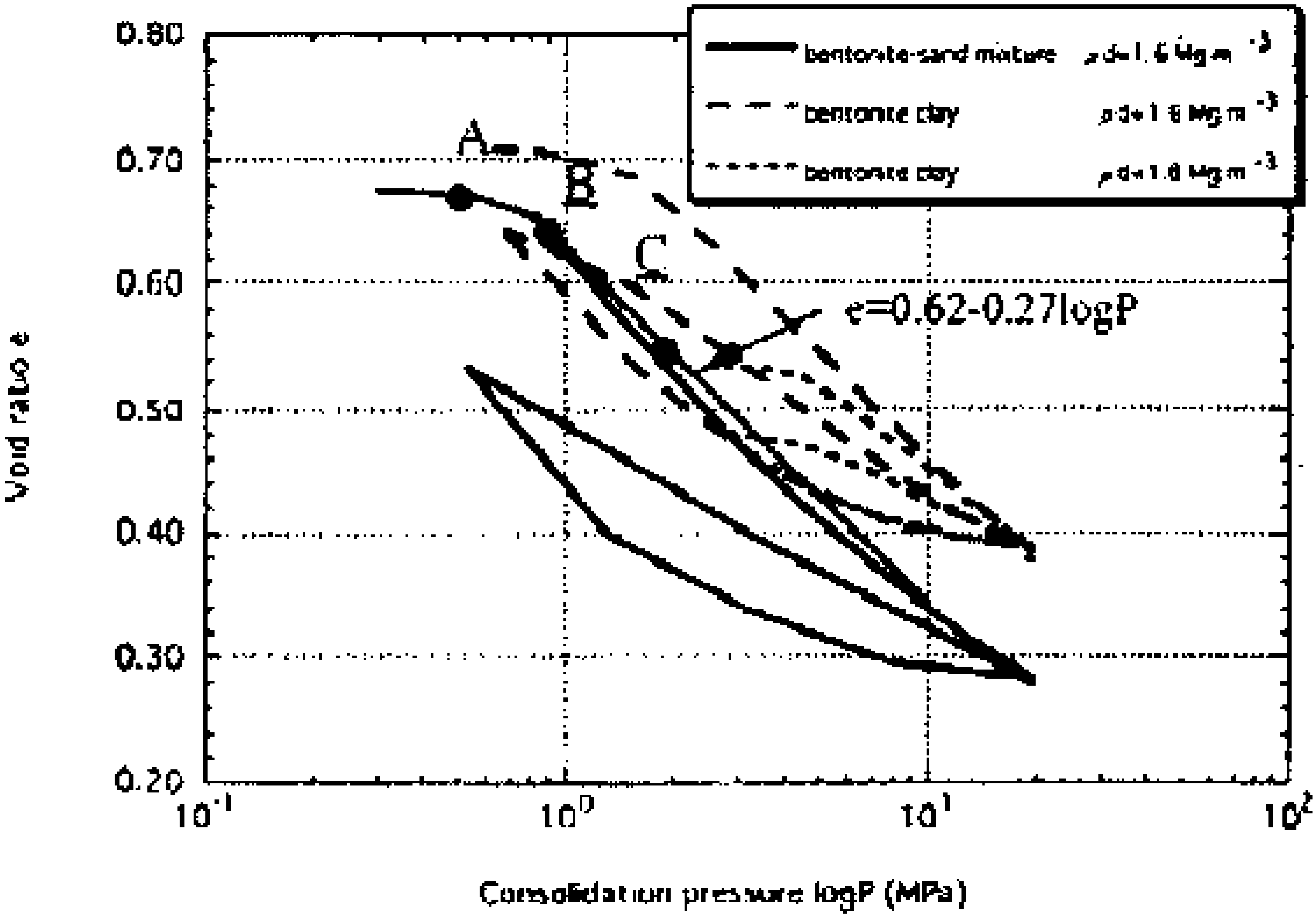
=> The confusion was given by the typographical error in the first and second paragraphs on page IV-4 when read in connection with figure 4.2-2, equal to the comment on the emplacement configuration at page IV-4 (see the answer to comment 4-1.

4-13 Swelling pressure of buffer

It is not clearly described how the swelling pressure of the buffer contributes to the whole external pressure applied to the overpack (table- 4.3-7). The consolidation process force of buffer material due to creep deformation and rock mass and corrosion product expansion pressure (0.86 MPa) seems to be low. The bases for this value should be explained

=> Swelling pressure is included in the consolidation process force. As described on page IV-26 of the Project Overview Report, the consolidation process force of the buffer is determined from the relationship between void ratio of buffer material (e) and effective stress (P) (as shown in the following e-log P curve) obtained from consolidation tests (see Figure 4.1.2-37 at page IV-82 of Supporting Report 2). The swelling pressure of the buffer (70 wt% bentonite and 30 wt% quartz sand with a dry density of 1.6 Mg m^{-3}) is about 0.5MPa shown at point A on the e-log P curve. The void ratio (e) will decreases as rock mass creep deformation and corrosion products expansion proceeds. For the hard rock system, the void ratio (e) of the buffer at 1000 years after emplacement is calculated to be 0.64 due solely to overpack corrosion expansion (high strength of hard rock system prevents contribution by creep). The void ratio 0.64

corresponds to the consolidation process force of the buffer, 0.86 MPa, that is obtained by using a linear-approximated fit; e=0.62-0.27logP for the e-log P curve (point B on the line). In a similar manner, the consolidation process force of the buffer for the soft rock system is calculated to be 1.87 MPa at a void ratio of 0.55 (point C on the line), considering both overpack corrosion expansion and the effect of rock mass creep.



(Figure 4.1.2-37)

Note: void ratio (e) = Vv/Vs

where:

Vv: Volume of void (pore) in buffer material

Vs: Volume of solid in buffer material

4-14 Convergence limitation criterion

In order to calculate the characteristics of tunnel liners (width and stiffness) a convergence limitation criterion should be included as design requirement.

=> We think that tunnel stability can be checked by local safety factor (from the viewpoint of rock stress) and shear strain of rock (from the viewpoint of rock deformation). And tunnel convergence will be checked in the construction process in terms of "observational procedure (construction)" as shown in table 4.5-5 (page IV-112 of Project Overview Report).

4-15 Sorption capacity of buffer material

4-15.1 Justifications should be done on assumptions in the third paragraph of page IV-39 (Properties relating to movement of substances) "If the quartz sand mixing ratio is 30 wt %, the distribution coefficients of various radionuclides that are high in 100 wt % bentonite are not appreciably affected by the added quartz sand".

=> We agree. The distribution coefficients must decrease theoretically in batch sorption system, when quartz sand which is inert for sorption is added. In H12 safety assessment, distribution coefficients for buffer material are derived from measured apparent diffusion coefficients (Da), which represent non-steady state diffusion including retardation by sorption, and effective diffusion coefficients (De), which represent steady state diffusion. It is confirmed for some elements that the influence of 30% quartz sand mixture in not significant with respect to Da and De values obtained by diffusion experiments.

It is also noted that changes in the retardation by the buffer has no impact on the peak dose release rate of long-lived solubility limited radio nuclides and a relatively minor impact on peak dose of long-lived non-solubility limited nuclides.

We will modify the text in this paragraph as follows,

"The results of diffusion experiments for the compacted bentonite mixed with quartz sand show that the 30% mixture of quartz sand does not significantly reduce the retardation performance of the buffer (*i.e.*, Idemitsu, 1994: Sato, 1999)."

4-15.2 A lost of sorption capacity due to the addition of quartz sand in the buffer, would be difficult to detect if the distribution coefficient is high and it have a large variability, but theoretically, distribution coefficients would be lower than for pure bentonite .

=> See comment 4-15.1.

4-16 Backfilling and sealing

A consideration about the need to remove the concrete lining, at least in the seal zones, should be done in order to prevent that degraded concrete become a migration pathway.

=> The degraded concrete effect is considered in the safety analysis as alternative geological environment cases (table 5.4.2-1 on page V-55 in Project Overview Report). Based on these preliminary results, we believe it may be unnecessary to remove the concrete lining.

4-17 Thermal conductivity: page IV-14

The thermal conductivity value for soft rock in table 4.3-2 (2.2 W/mK) seems to be a little bit high. Figures used in other performance assessments for clay material are about 1.5 W/mK.

=> The thermal conductivity value for the silica sand rich buffer material is the same order of the thermal conductivity value for soft rock system, when the buffer material has higher density as shown in following table. The actual thermal conductivity will, therefore, be a function of clay and clastic components composing the "soft rock", as will factors such as density. This is illustrated by the data above collected for clay and quartz sand, which shows the value of 2.2 W/mK to be within this range of thermal conductivities.

	Dry Density [g/cm^3]	Quartz Sand Mixing Ratio [%]	Degree of Saturation [%]	Thermal Conductivity [W/mK]
Rock (Soft Rock System)	1.9 (Saturated Density =2.2 g/cm^3)	-	100	2.2
Buffer	1.6	0	93.1	1.438
		30	100	2.144
		50	96.3	2.192
	1.8	0	100	1.434
		30	94.4	1.972
		50	97.2	2.403
		80	95.2	3.153

4-18 Corrosion due to oxygen: page IV-24

In the H3 report, the amount of oxygen trapped in the buffer material was estimated considering the oxygen in the pores of the bentonite and the oxygen absorbed on the surface of the bentonite (estimated as 4 times greater than the amount contained in the pores). Why in this exercise the amount absorbed has not been considered? (if it would had been taken into account, the maximum corrosion depth would be 31 mm)

What is the correct transcription for the equation 4.3-1?

In the H3 and in the Project Overview Report 99 appears as $P= Xm+7{,}9\cdot Xm^{0.5}$. Using this expression, the maximum corrosion depth should be P=12.39 mm

In the supporting report 2 (Repository Design and Engineering Technology) this formulation is $P= Xm+7{,}5\cdot Xm^{0.5}$, that seems to be the correct expression.

In any case, it seems that P is very sensitive to the value of Xm, and this introduces a large uncertainty.

=> In the analysis, the adsorbed oxygen on the bentonite is taken into account. The amount of adsorbed oxygen is assumed to be 0.02wt% of the bentonite weight in both the H3 and Project Overview Report. In the H3 case, the buffer consists of bentonite and only horizontal tunnel emplacement method is considered. In this case, the amount of absorbed oxygen corresponds to 4 times of the amount of oxygen in pore space. In the Project Overview Report, however, the buffer consists of a mixture of bentonite and silica sand. In addition, the content of the bentonite in the backfilling material for pit disposal is only 15%. The ratio of adsorbed oxygen to the oxygen in the pore space of the buffer is reduced in Project Overview Report because the contents of bentonite both in the buffer and the backfill is reduced in comparison with that in H3 case. Furthermore the dimensions and geometry of the EBS are different between these cases.

The surface area of an overpack in the Project Overview Report is changed from that in H3. Therefore the equation for the relation between averaged corrosion depth and maximum corrosion depth is changed from

$P=Xm+7.9Xm^{0.5}$ to $P=Xm+7.5Xm^{0.5}$. We would view the relationship not as introducing "uncertainty", but rather as providing a conservative bounding value for potential early failure by pit formation.

4-19 Increase of corrosion rate: pages IV-24 and 25

The average corrosion rate was set to 5 μm/y based on long-term test results and doubling it (corrosion rate 10 μm/y) to take into account heterogeneity in the overpack. Later, in the *summary of the assessment of the lifetime of the carbon steel overpacks* it is said again that the average corrosion rate is doubled to account for heterogeneity. It looks like corrosion rate is doubled twice for the same reason.

=> We agree that this description is inadequate in the Project Overview Report. The average corrosion rate of 0.005mm/y (as measured in lab tests) is doubled for covering the uncertainty of environmental conditions. For example, if we assume the groundwater of very high sulfide concentration (about 0.03mol/l) and also employ the same analysis procedure as that applied in the copper case, 5mm/1000y will be corroded by the sulfide arising from natural groundwater concentration and sulfide produced by the potential reduction of sulfate in bentonite.

The average corrosion rate of 0.01mm/y is doubled again in order to take localized corrosion into account. The factor of 2 for localization is obtained from the acceleration tests (Taniguchi *et al.*, 1999). We will add clarifying text into the report to resolve this confusion.

4-20 Comments on pages IV-27 and -28

4-20.1 Figure 4.3.6 should present a detailed zone for the consolidation reaction forces between 0 and 5 MPa.

=> We agree. Detailed figure between 0 and 5 MPa will be added in figure 4.3.6.

4-20.2 It seems that there is an error in table 4.3.7. The values for the external pressure for hard rock should be 0.91 MPa instead of 0.86 Mpa, according to the text.

=> 0.86 MPa is the correct value, and the text will be corrected.

4-21 Calculation results for the pressure resistance of the overpack: page IV-28

Table 4.3-8 gives values for the required pressure resistance thickness. However, in the text below the values given for the thickness are different (it seems that the values are rounded up, but in the case of the hard rock this rounding is very rude (10%).

=> We will revise the text to state: "The required overpack thickness (rounded to the nearest 10 mm) for pressure..."

4-22 Radiation shielding properties: page IV-28

4-22.1 The equation (4.3-3) should be:

$$I_{cath} = -n \cdot F \cdot \frac{p \cdot G' \cdot E}{100 \cdot Av} \cdot \frac{p}{\lambda}$$

G′: G value (number of molecules of a given species produced per 100 eV of absorbed energy) converted for bivalent oxidising chemical species (molecules/eV)

Av: Avogadro's number (= $6.023 \cdot 10^{23}$ molecules/mol).

=> It is true, and we will modify the above equation including notations and their unit as pointed out.

4-22.2 It should be justified why the required overpack thickness is specified to be 150 mm instead of 130 mm obtained according to figure 4.3-7

=> As described in the second paragraph of IV-29, we have not so much available information, especially on the validity of the passive current density of carbon steel. We will add the following text: "Considering these uncertainties, about 20 mm is added as a safety margin for the overpack thickness in case the passive current density decreases by about one order of magnitude."

4-23 Specification of thickness: page IV-29

4-23.1 It seems that there is an error in table 4.3-9. The values of column b) for "soft rock" should be 80 mm for the end section and 30 mm in the shell section, according to the text.

4-23.2 The paragraph above should say: "because the required thickness taking the influence of radiolysis into consideration, is greater than the pressure resistance thickness."

=> Table 4.3-9 and the paragraph will be corrected in agreement with the above comments.

4-24 Overpack corrosion

4-24.1 On p. IV-24-25 the corrosion of the overpack due to water reduction is discussed. It is said that average measure rate is 0.005 mm/y. Because of heterogeneity in the overpack, this is doubled to 0.01 mm/y. Then further down that page it is said that the average corrosion rate was set conservatively to 0.01 mm/y and doubled tom 0.02 mm/y to account for heterogeneity in the overpack (!). This seems overly conservative to account for heterogeneity twice. Please explain.

=> See previous response to comment 4-7. The text will be revised to clarify our approach.

4-24.2 On page IV-80, when the Overpack corrosion expansion analysis is performed a corrosion rate of 0.04 mm/y is used. Why are different corrosion rates used?

=> Our overpack design objective is 1000 years of containment. Based on initial evaluation of corrosion processes, including corrosion by trapped oxygen, microbial corrosion and general anaerobic corrosion, we estimated a corrosion allowance of 40 mm to achieve the 1000-year containment objective. We, therefore, adopted a value of 0.04 mm y^{-1} as a conservative initial bounding value in our investigations on potential impact arising from corrosion product expansion.

4-24.3 As I see it, already 0.005 mm/y is a very conservative estimate of the corrosion rate, at least in bentonite equilibrated fresh water. In SKB:s SR-97 safety analysis, we use 0.0001 mm/y as most probable value and 0.001 mm/y as conservative value. These data are based on experiments performed at AEA Harwell/Culham.

=> In H12 report. averaged corrosion rate of carbon steel obtained by the immersion experiments in compacted bentonite placed at anaerobic environment have been used for the life assessment of carbon steel overpack. We are now compiling data from laboratory and long-term analogue data to derive and confirm a long-term corrosion rate. We would welcome receiving a more detailed reference to the Harwell study.

4-25 Bacterial corrosion: pages IV-34 and IV-24

4-25.1 In page IV-24 the value given for the bacterial corrosion of carbon steel overpack is the one that had been calculated for the H3 report (that is, 2 mm). However, in page IV-34 (*copper carbon-steel overpacks*) a calculation is done to evaluate the corrosion by sulphide that diffuses through bentonite from groundwater. Why it hasn't been used the same method to calculate the bacterial corrosion in both cases. If the same calculation method would be used, the loss of thickness of carbon steel overpack for corrosion due to sulphide would be 4,6 mm instead of 2 mm

4-25.2 In the document: "Repository Design and Engineering Technology" the equation 4.1.1-27 (page IV-36) for the length of buffer corresponding to one should multiply calculating the steady state flux overpack.

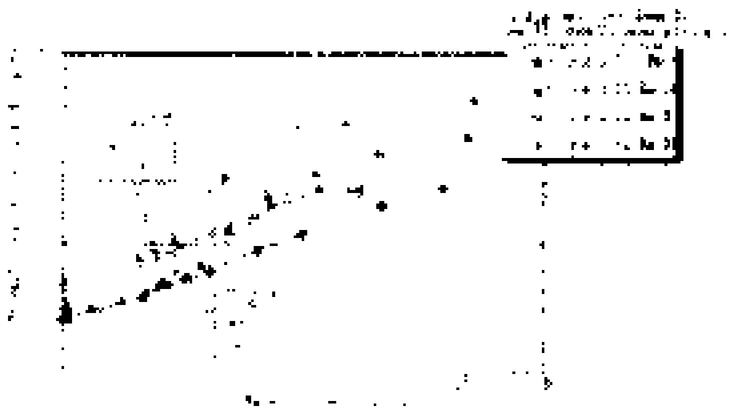
=> We will discuss this point at the Workshop.

4-26 Thermal conductivity: page IV-42

In table 4.3-17 (that should be renamed as table 4.3-11) the thermal conductivity of the buffer material in the case of 100 % bentonite (0.96 W/mK) is higher than in the case of the bentonite/quartz sand mixture (0.78 W/mK). A reason to add quartz sand to bentonite seems to be the increase of the heat transfer capacity.

=> Table 4.3-17 is typo, and it will be renamed as table 4.3-11.

As described in Section 4.3.3 2) (1) b) of the Project Overview Report (page IV-36), the thermal conductivity increases as the density increases. In table 4.3-11, the dry density of 100% bentonite (1.8 Mg m^{-3}) is larger than that of bentonite/quartz sand mixture (1.6 Mg m^{-3}), so that the thermal conductivity of the buffer material in the case of 100 % bentonite (0.96 W m^{-1} °K^{-1}) is higher than in the case of the bentonite/quartz sand mixture (0.78 W m^{-1} °K^{-1}) due to the difference of the dry density. The following figure shows our test results in relation to above comment.



4-27 Overpack corrosion expansion analysis: page IV-80

It should be clarified which is the reason for using a corrosion rate of 0.04 mm/y. In fact, this value doesn't correspond to a physical corrosion rate because it comes from:

0.02 mm/y · 1000 y + 12 mm + 2 mm ≡ 40 mm in 1000 years

water corrosion (rate) + oxygen corrosion + bacterial corrosion

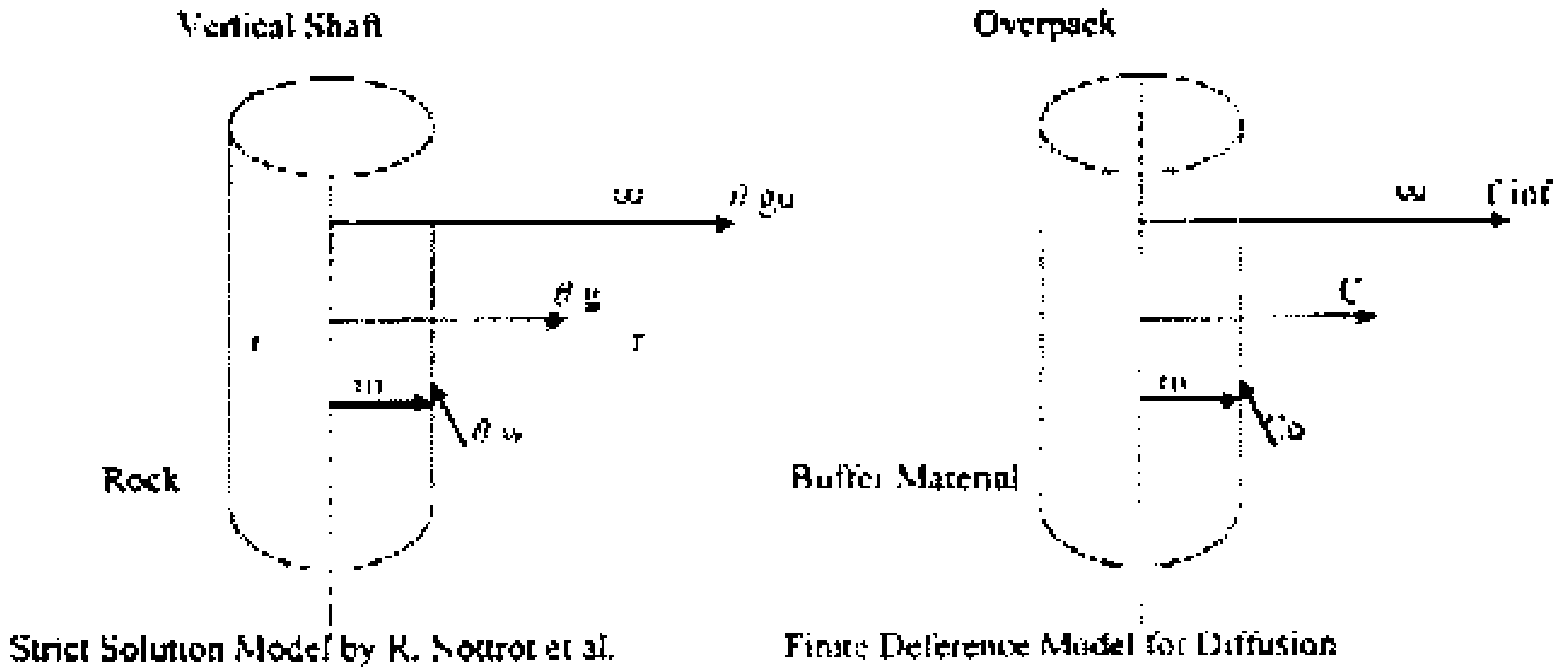
=> See previous response to comment 4-24.2. "..... when the overpack corrosion expansion ...".

4-28 Evaluation of diffusion of dissolved hydrogen: page IV-90

4-28.1 In the Repository Design and Engineering Technology (page IV-399) it is said that the equation of linear diffusion of hydrogen is solved numerically by a finite difference method but the name of the code used for the calculations is not given.

=> This simulation code was originally made using a finite difference equation. The code is not named.

This resulted needed to be checked. Therefore, we referred to a bibliography which was presented by Roel Nottrot *et al.* [1)] who solved the heat transfer process for the ventilating air and the rock - wall around the vertical shaft in an underground mine. These conceptual models are showed in the following figure. The maximum difference in values between the two methods was less than 0.3%. This conceptual model also satisfied this diffusion problem. Therefore, we applied this simulation code for the diffusion problem. We will add the above information to the final version of H12.



Conceptual model of Strict Solution and Finite Difference

where:
θw : Temperature at shaft wall
θg : Temperature at voluntary point (at r)
θgu : Temperature at far distance
Co : Concentration of dissolved hydrogen at overpack wall
C : Concentration of dissolved hydrogen at voluntary point (at r)
Cinf : Concentration of dissolved hydrogen at far distance

1) Roel Nottrot und Constant Sadee: Abkuhlung homogenen isotropen Gesteins um eine zylindrische Strecke durch Wetter von konstanter Temperatur :Gluckauf-Forschungshefte,27 Jahrgang H.4.August,PP.193-200(1966)

4-28.2 Corrosion rates of 1 μm/y, 2 μm/y and 10 μm/y are used for the calculation of gas diffusion, but how is the gas generation rate evaluated?

=> The gas generation rate is calculated from:

$$Rg = \frac{A \cdot Rc \cdot \rho_i}{M} \cdot \frac{4}{3} \cdot 22.4[mol^{-1}]/1000[m^{-3}]$$

where:
Rg ; gas generation rate [$m^3\ y^{-1}$]
A : surface area of overpack [m^2]
Rc ; corrosion rate [$m\ y^{-1}$]
ρi ; density of iron [= 7.8E6 $g\ m^{-3}$]
M ; molecular of iron [= 55.9 $g\ mol^{-1}$]

The corrosion reaction is as follows:

$$3Fe + 4H_2O \rightarrow Fe_3O_4 + 4H_2$$ (Honda *et al.*, 1997)

We will add the above information to the final version of H12.

4-28.3 It's no clear why at the depth of 500 m the production and accumulation rate of H_2 is higher than at the depth of 1000 m, and the evolution volume and accumulation pressures tend to become smaller.

=> We are now checking the calculation.

4-28.4 In any case, the gas production rate would be the same at 500 m and 1000 m, because this value is calculated as a function of the corrosion rate. ("At the depth of 500 meters, meanwhile, the ~~production and~~ accumulation ratio of hydrogen would be.....")

=> "Production" means accumulation volume of hydrogen at overpack-buffer interface. This sentence will be revised as follows:

"At the depth of 500 meters, meanwhile, the accumulation volume and ratio of hydrogen would be a little larger than at the depth of 1,000 meters, but the elution volume and accumulation pressure would tend to become smaller."

4-28.5 It should appear as a conclusion that hydrogen will probably accumulate (for the values of gas generation and diffusion coefficient used as base values)

=> Gas diffusion analysis was firstly carried out for the purpose of confirming whether hydrogen gas is dissipated by only diffusion without accumulation in the buffer.

The parameter studies for corrosion rate and diffusion coefficient described in Section 4.3.4.1 of Supporting Report 2 show that, for the case of the corrosion rate $10 \mu m\ y^{-1}$, the hydrogen is estimated to accumulate at the overpack-buffer interface after 10,000 years (see figures 4.3.4-6 and 4.3.4-10 in Supporting Report 2).

Following gas diffusion analysis, gas transport analysis was conducted by using TOUGH2 code for the same condition in which gas would accumulate due to limited diffusive flux of the dissolved hydrogen. The results of gas transport analysis are described in Section 4.4.4 2) of Project Overview Report (pp.IV-90~91) and Section 4.4.4.2 of Supporting Report 2 (pp.IV-406~418).

We will modify the text along this context.

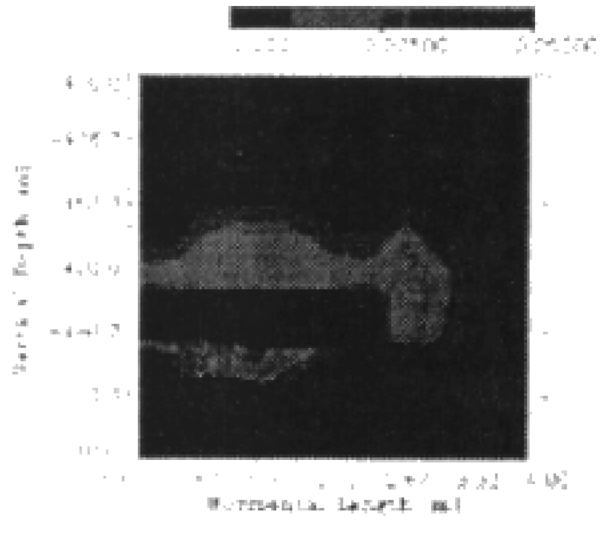
4-29 Evaluation of gas migration behaviour: page IV-91

4-29.1 Which is the gas generation rate used in the TOUGH calculations?

=> See previous response to comment 4-28.2 on gas generation rate equation.

4-29.2 a) case of pit disposal (depth: 500 m) for the soft rock data set
In the Repository Design and Engineering Technology (page IV-414) Figure 4.3.4-21 why the saturation of gas is lower for point 2617 than for point 2618?

=> For the case of analysis for soft rock system (disposal pit vertical emplacement method), the backfill material is used in the disposal tunnel over the disposal pit. Water permeability of this backfill material is very low (6E-19 m^2). Also, the water intrinsic permeability of rock is very large (1E-16 m^2). On the other hand, gas permeability of both is the same value (9.5E-16 m^2). In these conditions, gas will be initially generated around the waste package and will diffuse into the buffer and accumulate in the buffer. Afterwards, owing to the low intrinsic permeability of backfill, gas will accumulate in the surrounding rock without penetrating to the upper rock (shown in the following figure). For the case of analysis for hard rock system (disposal tunnel horizontal emplacement method), this phenomenon does not occur because the disposal tunnel are filled with only buffer materials.



4-29.3 It is said "*the saturation degree of gas would be about 99.6-99.8% in all locations....*". This values seems to be wrong, because, in the document Repository Design and Engineering Technology (pages IV-413 to 414) the gas saturation level rise up to 2 to 4% (water saturation level 96 to 98%).

=> These values are typos. The right value is "96-98%".

4-29.4 Further on, it's said "*Examining the cumulative discharge volume of gas migrating from the disposal system into rock mass, it is found that a gas production rate of $2.3x10^{-3}$ m^3 y^{-1} would almost fully migrated into the rock mass*". The value of gas generation rate seems to be wrong because in the document Repository Design and Engineering Technology (page IV-413 and Figure 4.3.4-22) the value for gas generation is $2.3x10^{-1}$ m^3 y^{-1}.

=> This value of gas generation was typo. The right value is "2.3E-1 m^3 y^{-1}".

4-29.5 b) <u>case of tunnel disposal (depth: 1000m) for the hard rock dataset</u>

The latter two comments for the case of pit disposal can be applied to tunnel disposal

=> We acknowledge this connection and will make the appropriate changes.

4-30 Thermo-hydro-mechanical analysis

4-30.1 (Pages IV-88 and IV-89):THM analysis: In order to be conservative in a thermal analysis, the data for the thermal conductivity of the buffer material should be those corresponding to the lowest water content state. This value does not correspond with the initial water content of the buffer (see Figure 4.3.3-18 (a) of the "Supporting Report 2: Repository Design and Engineering Technology").

In Figure 4.3.3-18 (a) the water content of the inner part of the buffer material is lower than the initial water content during the first 25 years. Therefore, the initial temperatures of the inner part of the bentonite in a THM analysis should be higher than in the case of an uncoupled thermal analysis with the initial water content state of the buffer but in Figure 4.3.3-20 (a) is the opposite.

Then, it should be more conservative to use the THM coupled analysis than the uncoupled thermal analysis with the initial water content of the buffer. In the case of using a thermal analysis the water content should be the lowest value of the THM analysis, but no the initial water content.

=> We carried out the parametric study to compare the maximum temperature in the buffer between the thermal analysis and coupled T-H analysis. This result is described in the JNC technical report " Coupled thermal, hydraulic and mechanical analysis in the near field for geological disposal of high-level radioactive waste". The parameters were water content, thermal water diffusivity and pore pressure in the rock mass. The water contents for analysis were 7%, it is natural water content, and 17%, it is optimum water content of in-situ compaction. The thermal water diffusivity of buffer (D_{Td}) was determined by back analysis of laboratory test and was 7.0×10^{-8} cm^2/s/°C. In the parametric study, we set four cases, case 1 is $D_{Td}\times0.5$, case 2 is $D_{Td}\times1.0$, case 3 is $D_{Td}\times2.0$ and case 4 is $D_{Td}\times3.0$. As a pore pressure in the rock mass, we used 0.0 MPa. Of course, the pore pressure in the rock mass is higher than 0.0 MPa. However, we considered that 0.0 MPa is a conservative value. The reason is that water content becomes low because water inflow rate into the buffer from the rock mass is low when the pore pressure in the rock mass is low. Lastly, we considered the case that rock mass was no flow as the special case, that is, water did not inflow into the buffer from the rock mass. In this case (Case 5), the thermal water diffusivity (D_{Td}) was 7.0×10^{-8} cm^2/s/°C. Analyses were performed using a simple model, as shown in Figure 1. Because the analysis model was simple (quasi-one dimensional model), we decreased the heat output of waste as the maximum temperature in the buffer approximately reached 100°C. In the case of an initial water content 7%, the heat output was 2/5 of the original heat output, and in the case of an initial water content 17%, the heat output was 1/2 of the original heat output.

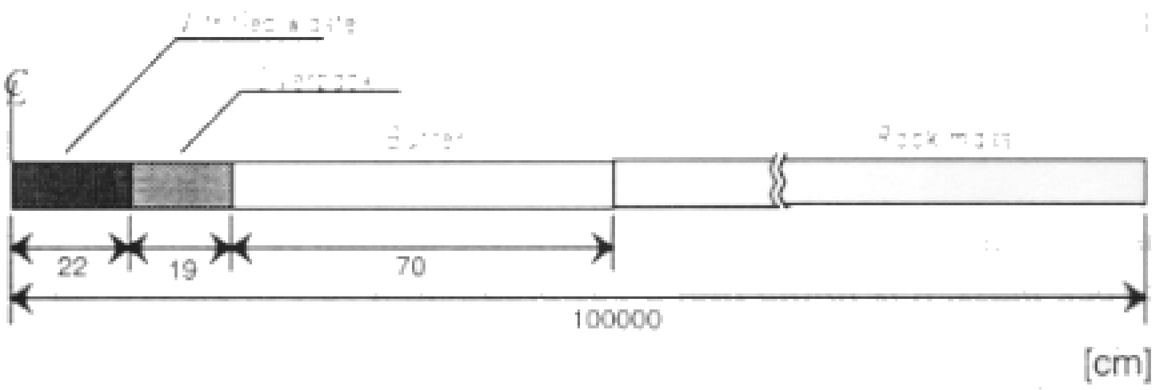


Figure 1 T-H analysis model.

Figure 2 shows the time history of water content in the buffer at the inner part of the buffer in the case of an initial water content 7% and Figure 3 shows the time history of temperature at the same point. In all cases of coupled analysis, water contents decrease from the initial values for a long time (3-20 years). However, the maximum temperatures of each coupled case are lower than that of the uncoupled case. Figure 4 shows the distribution of water content in the buffer for Case 2 (the initial water content is 7%).

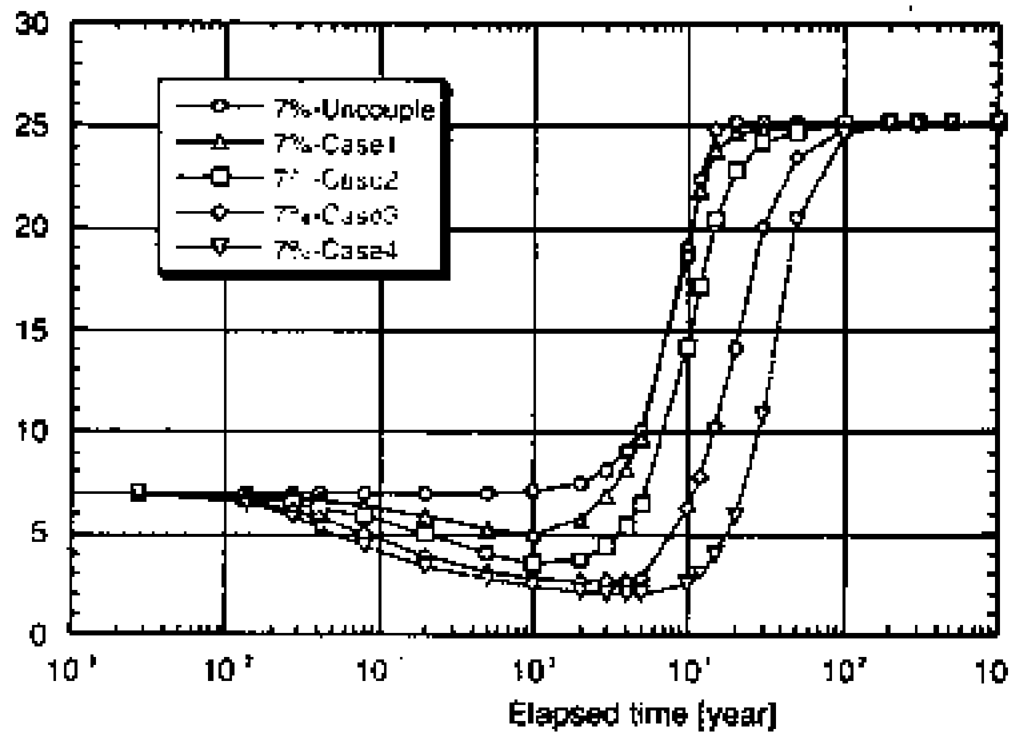


Figure 2 Time history of water content in the buffer (Inner part of the buffer).

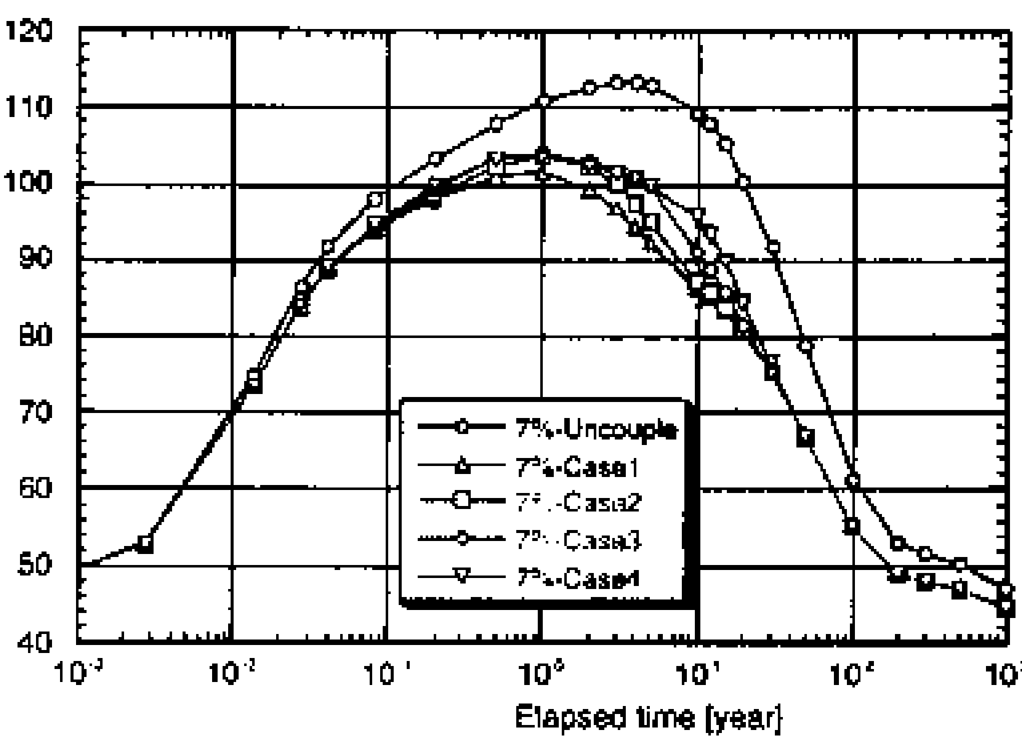


Figure 3 Time history of temperature in the buffer (Inner part of the buffer).

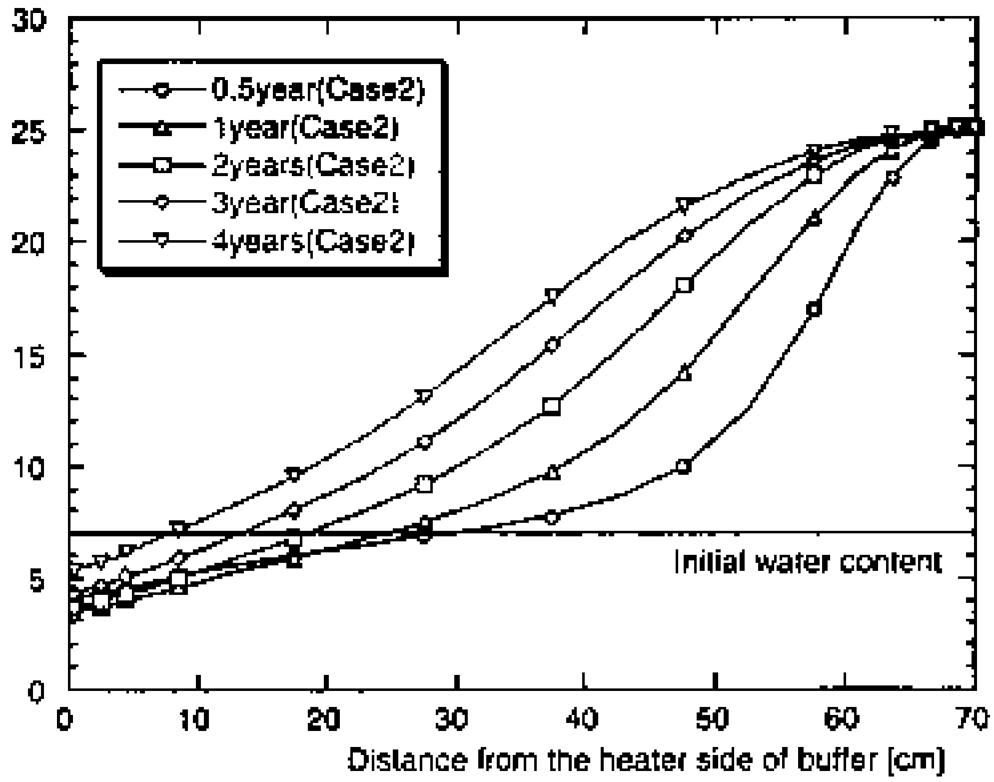


Figure 4 Re-distribution of water content in the buffer (Case2).

At the inner part, the water content clearly decreases due to the thermal effect. However, at the outer part of the buffer, the water content increases due to the water movement from the inner part and water inflow from the rock mass. As results, the total water content in the buffer increases. We consider that this is a reason that the maximum temperature by the coupled analysis is lower than that by the uncoupled analysis, because the average value of the thermal conductivity among the overall buffer area increases with increasing the water contents by the groundwater infiltration from the rock mass.

All results are shown in Table 1. In most cases, the maximum temperature by the coupled analysis is lower than the uncoupled case. When water does not inflow into the buffer from the rock mass (Case 5) or both an initial water content and thermal water diffusivity are high, the maximum temperature predicted by the coupled analysis is higher than uncoupled analysis. We consider that Case 5 to be an unlikely bounded case. Therefore, we conclude that the use of the thermal property with an initial water content 7% is conservative in the thermal analysis. As the reviewer pointed out, the use of the thermal property with the lowest water content is more conservative. However, we consider that the use of the thermal property with the initial water content 7% is sufficiently conservative.

Table 1 T-H analyses results.

	Initial water content 7wt%				Initial water content 17wt%			
	Minimum Water content [%]	Time [year]	Maximum temperature [°C]	Time [year]	Minimum water content [%]	Time [year]	Maximum temperature [°C]	Time [year]
Uncouple	7.0	-	113.2	4	17.0	-	102.4	5
Case 1	5.0	1	101.4	1	14.4	1	98.3	3
Case 2	3.5	1	103.4	1	11.7	2	99.6	3
Case 3	2.6	3	104.0	1	6.6	4	103.5	4
Case 4	2.0	4	103.4	1	3.4	10	106.9	5
Case 5	2.8	10	113.4	3	8.8	10	105.0	5

4-31 Resaturation time: page IV-89

The analysis concludes that the re-saturation time is 50 years, but in page IV-78 the value considered is 100 years. Why is there this difference?

=> THM modeling shows a range of re-saturation times, ranging between 5 and >1000 years depending on boundary conditions, especially intrinsic permeability of the rock mass. The 50 year value for case 01-1 in Table 4.4-4 on page IV-89 of the Project Overview Report should not be overinterpreted as a reference value. We adopted a re-saturation time of 100 years based on the THM calculation as a reasonable value within the expected range, and we used this value for overpack corrosion expansion analysis. In fact, the difference of 50 year and 100 year is expected to cause little effect on the corrosion expansion analysis results.

4-32 Retrievability

4-32.1 Retrievability could be a design requirement. This requirement would be decisive to choose the "disposal pit vertical emplacement method", which have better retrievability features than the "disposal tunnel horizontal emplacement method". Otherwise the advantages of the disposal pit vertical emplacement method are arguable.

=> Because retrievability is not a formal system or safety requirement in Japan, we have not formally analyzed the differences between design concepts based on ease of retrievability. It is our understanding that retrievability for both design options are at least feasible.

4-32.2 Retrievability is only mentioned as a "long-term countermeasure" (page IV-113). In this paragraph retrieval of waste before full closure is assumed to be possible following a series of steps in reverse order of those taken for construction and emplacement. Safety concerns related with the disposal method before closure should be considered.

=> If and when retrievability formally becomes a formal safety requirement in Japan, analysis of the relationship among design, safety and retrievability will be needed. At this generic stage of analysis and in the absence of formal requirements, however, no analyses have yet been conducted to include in this H12 report. See also response to the comment "One reason given for the suitability of the geological disposal concept is that retrieval would..." (see comment 0-4)

4-33 Radiation protection areas

4-33.1 One of the most important features of a disposal facility for HLW for the operational phase is the radiation designation of the areas that comprises the underground facilities (in order to develop the radiation protection strategy of the whole system) and the separation between them.

=> The basic idea of radiation protection is that the radiation control zone during operation is separated from other activity zones such as construction and closure zone. Section 5.2.1.3 Radiation control zone (pp.V-45~49 of Supporting Report 2) and Section 4.2.3.5 Position of the main and connecting tunnels (pp.IV-276~277 in Supporting Report 2) show the example of radiation protection strategy.

4-33.2 Of particular significance is the separation of the conventional mining works (construction activities) from the areas where disposal activities are carried out (operation areas) This seems difficult given the limited number of access foreseen . On the other hand , the optimisation of the radiological protection of the facilities would be one of the main reasons for dividing the disposal area in panels, but this is not clearly mentioned in subdivision 2 of 4.4.2 (page IV-5).

=> As described in subdivision 2 of Section 4.2.3.4 (page IV-275~276) in Support Report 2, the number of access tunnels is discussed taking into account the path way for each work activity during construction, operation and closure phase, and the transportability and ventilation route.

The following article will be added in the subdivision 2 of 4.2.2 (page IV-5 of Project Overview Report) and Section 2.2.2.2 (page II-8~9 of Supporting Report 2).

"The operation area can be separated as a radiation control area from the construction and closure area."

4-33.3 The statement in paragraph 2) on page IV-103, "A radiation controlled zone for the time of waste package and the buffer emplacement is not required", should be clarified further.

=> In buffer design, radiation shielding is considered as shown in Section 4.3.3 2) (2) c) Buffer thickness required for radiation shielding function (page IV-48 of Project Overview Report) for vertical emplacement method.

This sentence will be revised as follows:

"In disposal tunnels with engineered barriers for the vertical pit emplacement method, radiation is buffered or shielded by the design-based thickness of the engineered barriers, described in Section 4.3.3 2) (2) c), below the radiation level that it does not affect the human body."

4-34 Excavation damaged zone (EDZ)

The characteristics of the excavation damaged zone (EDZ), should be given in relation with the rock geomechanical characteristics and the selected construction method (*i.e.* TBM or boring and smooth blasting method), preferably in paragraph 4.5.2 Construction Phase.

=> The characteristics of the excavation disturbed zone (EDZ) are reported in relation with the excavation methods in Section 3.3.5 3) of Project Overview Report (pp.III-65~69) and Section 3.5.3 of Supporting Report 1 (pp.III-93~104). Hydraulic characteristics of the EDZ are also discussed in sections 4.1.3.8 and 5.1.1.3 of Supporting Report 3. Here in Section 4.5.2, it is only described that construction method is selected in qualitative manner so that the EDZ becomes smaller. We will cross-reference to the appropriate sections.

4-35 Transportation and emplacement of waste package

There is no consideration about the alignment requirements of the waste transport vehicle for the emplacements operations.

For the horizontal emplacement option, a justification for the achievement of the alignment required to place the waste package with a rail-less vehicle should be done.

=> As shown in Section 7.2.3 Specific goals of the R&D programme (page VII-3 of Project Overview Report), these above points are considered to be the future study item. We recognise that demonstration of alignment should be carried out in the future stage and will add this statement in the text.

4-36 Repository layout

Hard rock alternative. Ventilation has shown to be a problem in Sweden in some potential sites and potential repository layouts. The main reason is high radon levels that have to be vented out. The ventilation will be a bigger problem in the Japanese concept, with higher ambient temperatures in combination with a larger repository (40,000 packages). Are radon problems foreseen in the Japanese hard rock?

=> We recognise that the ventilation of the underground facility is an important issue. Radon problems will be discussed in the future stage of formation of regulations. We will address it in Supplementary Report. It will be appreciated a specific reference to any studies of this issue so it can be cited in H12.

4-37 Disposal pits

Levelling of the bottoms of the disposal pits will be necessary. If the bottom is not horizontal (within some tolerances), emplacement of the bentonite blocks around the overpack will not be possible. Has this been considered and are there criteria for acceptable leveling of the pit bottoms? Are there tolerances for the acceptable dimensions of the disposal pits?

=> We also think that leveling of the bottoms of the disposal pits or tolerances for the acceptable dimensions of the disposal pits are important. In this report, 40 mm is assumed as a tolerance to emplace bentonite block between rock surface and bentonite block. Constructing Errors in construction of the pit is not included in the assumed tolerances of 40 mm. This issue will be further considered in the detailed design after H12.

4-38 Minor Comments to Project Overview Report

4-38.1 Page IV-42: Table 4.3-17 should be table 4.3-11

=> That's right. It is typo, and we will correct it as pointed out.

4-38.2 Page IV-116: header's date. May 19999

=> Thanks. We will correct the header's date.

4-39 Minor Comments to Supporting Report 2

4-39.1 Title: Technology instead of Technoplogy

=> Thanks. We will correct the title.

4-39.2 Page IV-385: In Table 4.3.3-1 the number of the equation for the thermal conductivity expression for buffer and backfill material is 4.3.3-19 instead of 4.3.3-18. The same applies for specific heat, 4.3.3-20 instead of 4.3.3.19.

=> We will correct them as pointed out.

5. COMMENTS ON CHAPTER V

5-1 Section 5.1.1

Features of the safety assessment and general methodology: page V-2: first paragraph: The statement that codes are then validated against experimental results is not acceptable as written. Validation is only possible in a very limited way and only for short term processes. Most codes used in safety assessments cannot be validated.

=> See also response to comment 2-2.

We may replace the statement with:

"The models and data sets are tested be a variety of means, including comparison with observations of natural systems and with the results of field and laboratory experiments. This process is referred to as validation, which aims to ensure that, under relevant conditions, the models and data sets can adequately predict, or bound, the future performance of a repository".

5-2 Section 5.1.2

Objectives and scope of the safety assessment in H12: page V-3: third paragraph: In agreement with international developments a statement should be added to the effect that impacts on the environment may also need to be assessed.

=> AEC Guidelines implicitly require the protection of the environment, by requiring the protection of human beings. We will discuss this interpretation of the AEC Guidelines, as well as referring to international safety principles such as IAEA safety standards for the protection of the environment. There is certainly a growing international trend (*e.g.* SSI in Sweden) to require the impact on flora and fauna to be assessed. This is not within the remit of H12, but it should, perhaps, be considered in the site-specific R&D programme beyond H12.

5-3 Half-life of Se-79

Please observe that the "old" half-life for Se-79 has been found to be incorrect (p. V-35). Instead of 6.50E+4, the correct value is 1.13E+6. (see *e.g.* Li Chungsheng *et al.* J. Radioanal. Nucl. Chem. Vol. 220 (1997) 69-71.)

=> We recognised the half-life of Se-79 may possibly be changed to approximately 1.1×10^6 years, as shown in the footnote to Table 4.1.3-2 of Supporting Report 3). Although the value for the current safety-assessment calculations is taken from the DECAY library of the ORIGEN2.1 code, we will present calculational results using the proposed new half life for Se-79 at the Review Workshop.

5-4 Table 5.4.1-1: third column, second bullet

I don't like the hazard index expressed as volume of diluting water.

=> We will refer to a recent review on hazard indices in the final report (see the response to comment 1-1).

5-5 Section 5.4.1: the Reference Case: page V-38: second paragraph

The assumption that 100 m separation between repository and active fault may be sufficient to eliminate faulting risk may be overly optimistic. It is obvious that a significant uncertainty is attached to the assumption.

=> See also response to comment 3-3 on Section 3.2.3. The bulleted list on page V-38 is currently misleading, and will be revised along the following lines:

... it is assumed that:

- the repository is sited at a location where there is no influence from active faults, as defined in Section 3.2.3, and at least 100 m from major (inactive) faults that comprise large, highly permeable crushed zones;
- no new faults are generated that may adversely affect the stability and performance of the repository; and
- nuclides carried from the repository by groundwater eventually reach a large (inactive) fault, that is assumed to be located .....

Calculation results for scenarios in which new faults are generated will be presented in the Review Workshop.

5-6 Section 5.4.3: analysis of perturbation scenarios: page V-69

Faulting scenario needs to be analysed.

=> See response to comment 5-5.

5-7 Section 5.6.1: findings from safety analysis: page V-117: first paragraph

Validity of the models was confirmed.....I don't believe it! See previous comments.

=> See response to comment 2-2 about section 2.3.2.

We will replace:

"Validity of the models was confirmed through ...", by

"The adequacy of the models for their intended purpose was confirmed through a variety of validation measures, including ...".

5-8 Safety assessment

5-8.1 The representation of the near field is short and sometimes not clearly comprehensible. This applies e. g. to the conversion of the cylinder lid surfaces into an enlarged coating surface area. Furthermore, the term „initial volume" is not clearly defined. What is the value used for the water volume available for mobilisation? How is the time dependence of the glass volume taken into account?

=> The detailed representation of the near field is noted in Supporting Report 3. For instance, the conversion of the cylinder lid surface into an enlarged coating surface area is carried out conservatively to take account of radionuclide release from the top and bottom ends of the vitrified waste in one dimensional model (see section 4.1.2.2 of Supporting Report 3).

The initial glass volume of 0.15 [m^3] (see Table 3.2-1 in Supporting Report 2) is referred in the analysis.

We neglected the decrease of the glass volume with time due to dissolution conservatively as noted in Section 4.1.2.2 of Supporting Report 3.

In final version, more comprehensible overview of the near-field representation will be also included in Chapter V of the Project Overview Report.

The hypothetical water-filled volume around the vitrified waste and hypothetical volume between the buffer and host rock are introduced to define a mathematical boundary condition that connect glass dissolution vs diffusion in the buffer and diffusion in the buffer vs release to host rock. In the calculation, 10^{-3} [m^3] is used as a conservative volume and we confirmed that the sensitivity of these volume on calculated release rate is negligible. This explanation will be noted in the final version.

5-8.2 The solubility values of most elements are in the same order of magnitude as those from Kristallin-I. Exceptions are Th and Zr with differences of more than two orders of magnitude. These differences are, however, explained comprehensibly.

=> Thank you for your thorough analysis. To briefly summarize, the solubility values in H12 assessment were determined by the procedure described in 4.1.3.5 of Supporting Report 3. The detailed review of the thermodynamic database and the solubility setting in H12 from view point of cross-comparison with the other safety assessment reports are ongoing, and have not been completed. However, the reason for the outstanding discrepancy for Zr and Th is clear as follows:

Zirconium

In the development of thermodynamic data base for H12 assessment (JNC-TDB), we found that thermodynamic data for Zr are not mature and current experimental data show higher solubility than the calculated one. Therefore, the solubility for Zr used in H12 assessment was determined based on solubility experiments in bentonite co-existing system (Shibutani and Yui, 1998). The maximum value of the scattered experimental data was used as a conservative value, and 1×10^{-6} M was set.

Thorium

In H12, the solubility for Th was determined based on the solubility calculation with JNC-TDB. In this calculation, hydoroxo-carbonate complexes was considered and the solubility was calculated as 4.3×10^{-6} M. The dominant species was $Th(OH)_3CO_3^-$ and second dominant species was $Th(OH)_4(aq)$ with the concentration of 6.3×10^{-10} M. The dominant species of Th solubilities reported by different organizations, *e.g.* Kristallin-I (Pearson *et al.*, 1992) and SR97 (Burno *et al.*, 1997), were $Th(OH)_4(aq)$ and the difference of the dominant species caused the discrepancy.

The data for hydroxo-carbonate complexes have been reported recently (Östhols(1994); Felmy *et al.* (1997)) and their reliability was confirmed by the peer review in the development of the JNC-TDB. Therefore, we added the data for hydroxo-carbonate complex into the JNC-TDB and used them in the solubility calculation.

5-8.3 Most initial parameters used in far-field modelling are of the same order of magnitude as those in other studies. Compared to the reference cases of the SPA [1] and Kristallin-I study [4], the data for the matrix diffusion depth of 0.1 m with a matrix porosity of 2 % and a pore diffusion coefficient of $1.5 \bullet 10^{-10}$ m /s are relatively high.

=> As discussed in Section 4.2.2.2.3 of Supporting Report 3, the matrix diffusion depth derived from natural analogue studies (see Table 4.2.2-1) is the minimum estimated value (Miller *et al.*, 1994) and is considered to be a conservative value in terms of evaluating nuclide migration. On the other hand, it has been confirmed by laboratory tests that matrix diffusion occur even for unaltered granite. Therefore, for the Reference Case it is assumed that matrix diffusion occurs also in unaltered zones, but is limited to a depth

0.1 m from the fracture surface. For a Data Variation Case, the minimum estimated value based on natural analogue studies is taken into account (see Section 5.2.1.5 of Supporting Report 3). Porosity and pore diffusion coefficient are obtained based on literature review (see Sections 4.2.2.2.3, 5.2.1.6 and 5.2.1.7 of Supporting Report 3).

5-8.4 A random sampling comparison of the dose conversion factors with the project SPA [1] shows that the values of H-12 are one to two orders of magnitude lower.

=> We have checked the dose conversion factors (in H12; we refer to flux to dose conversion factors) in Tables 22 and 23 of SPA project (page 46), and made comparison with the corresponding GBI (geosphere/biosphere interface) cases of H12, specifically (1) reference case (river water GBI) and (2) deep well GBI case, as shown in Fig.1 and Fig.2. In both figures, the SPA values are normalised by the conversion factors of H12 so as to facilitate a relative comparison.

Concerning the well case, since the unit of conversion factors are different between H12 ((Sv/y)/(Bq/y)) and SPA ((Sv/y)/(Bq/m^3)), we have corrected the factors of H12 by using the extraction rate of the deep well (2x10^6 m^3/y), which is derived from the log-mean value of the distribution of waterworks capacity, where water is extracted mainly from the deep well (Japan Water Works Association, 1996) (page V-59, V-62 in Supporting Report 3)

From both figures, it seems that the dose conversion factors of H12 are slightly *higher* than the corresponding values in SPA, except for some of fission products (notably Pd-107 and Sn-126). Looking at Figure 1, it can be seen that the factors for virtually all radionuclides lie within an order of magnitude of the H12 values (ie within the range 1E+1 and 1E-1) with most lying in the range 1E+0 to 1E-1. For Figure 2, the same pattern occurs with slightly more lying above the 1E+0 line and a few falling below the 1E-1 line (those falling below the 1E-1 line are primarily the VTT WELL-97 doses which only consider the drinking water dose - as noted in the SPA project, these doses are consistently lower than other doses due to the importance of pathways other than drinking water).

According to the Table 20 of SPA project (page 44), there are no significant differences in the annual flow rate between H12 and SPA (see below Table 1), although well water extraction rate is not explicitly described in SPA, except VTT case (a fraction of 5x10^{-6} of the total release annually into the biosphere is assumed). In addition, both projects make some similar assumptions; use of biospheric compartments, consideration of variety of exposure pathways, etc.

Table 1 Annual flow rate used in river type biospheres

	H12	IPSN	NRG	SCK/CEN
Annual flow rate (m^3/y)	**10^8**	**10^8**	**7.75x10^7**	**~10^8**

However, we recognise that the comparison of flux to dose or concentration to dose conversion factors from different projects must be undertaken with EXTREME caution. In particular, it is vital to be aware of any differences in the following areas between projects:

- the assessment context
- the conceptual models
- the mathematical models
- the data values used.

Indeed some of these issues are touch upon in the SPA report. In order to compare dose factors from different projects, it is necessary to compare and contrast each of these aspects of the project in a systematic way. Only then can a truly meaningful comparison of dose factors from different projects be undertaken.

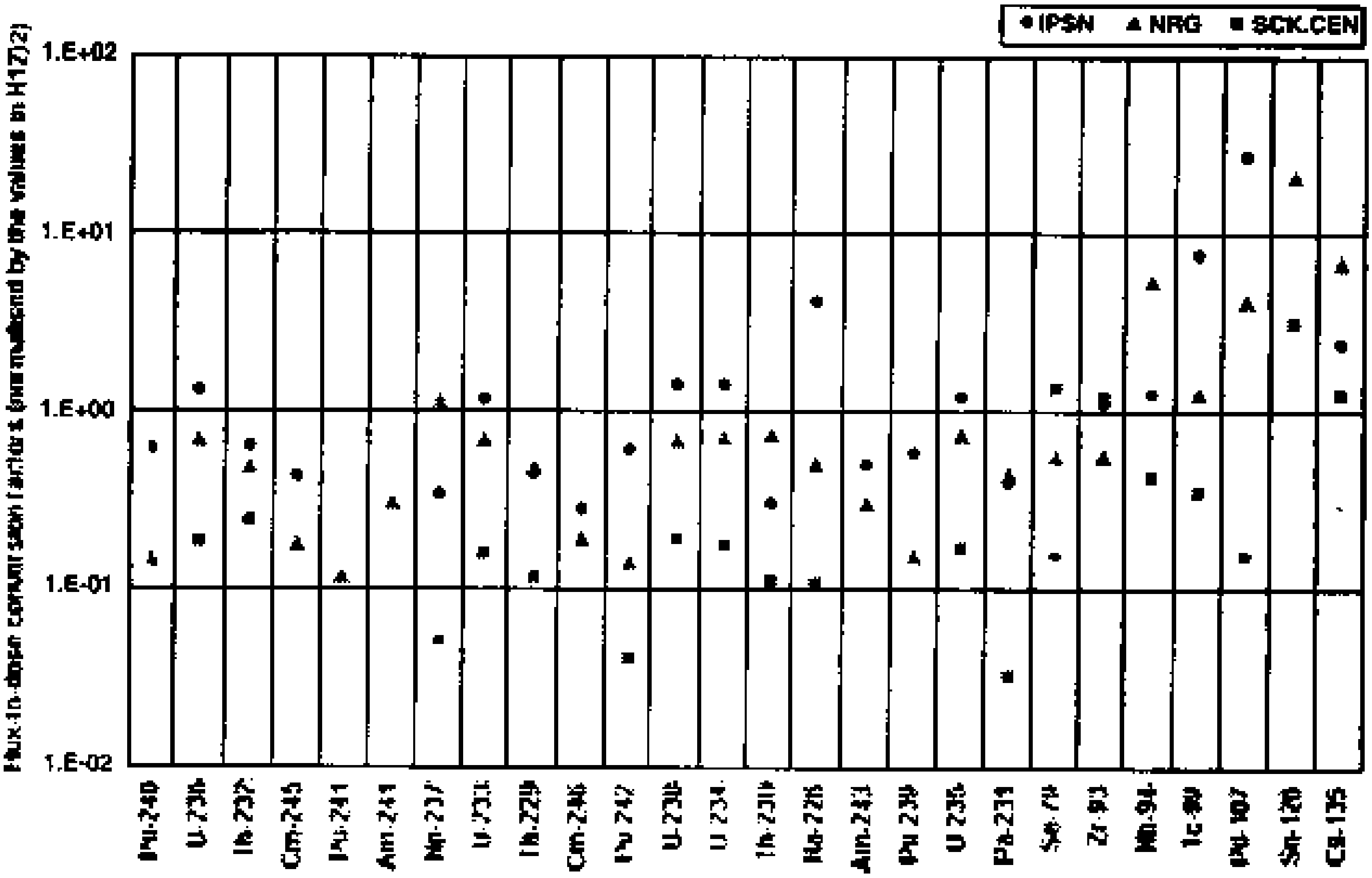


Figure 1. Biosphere conversion factors normalised by the value of H12 (for river type biosphere)

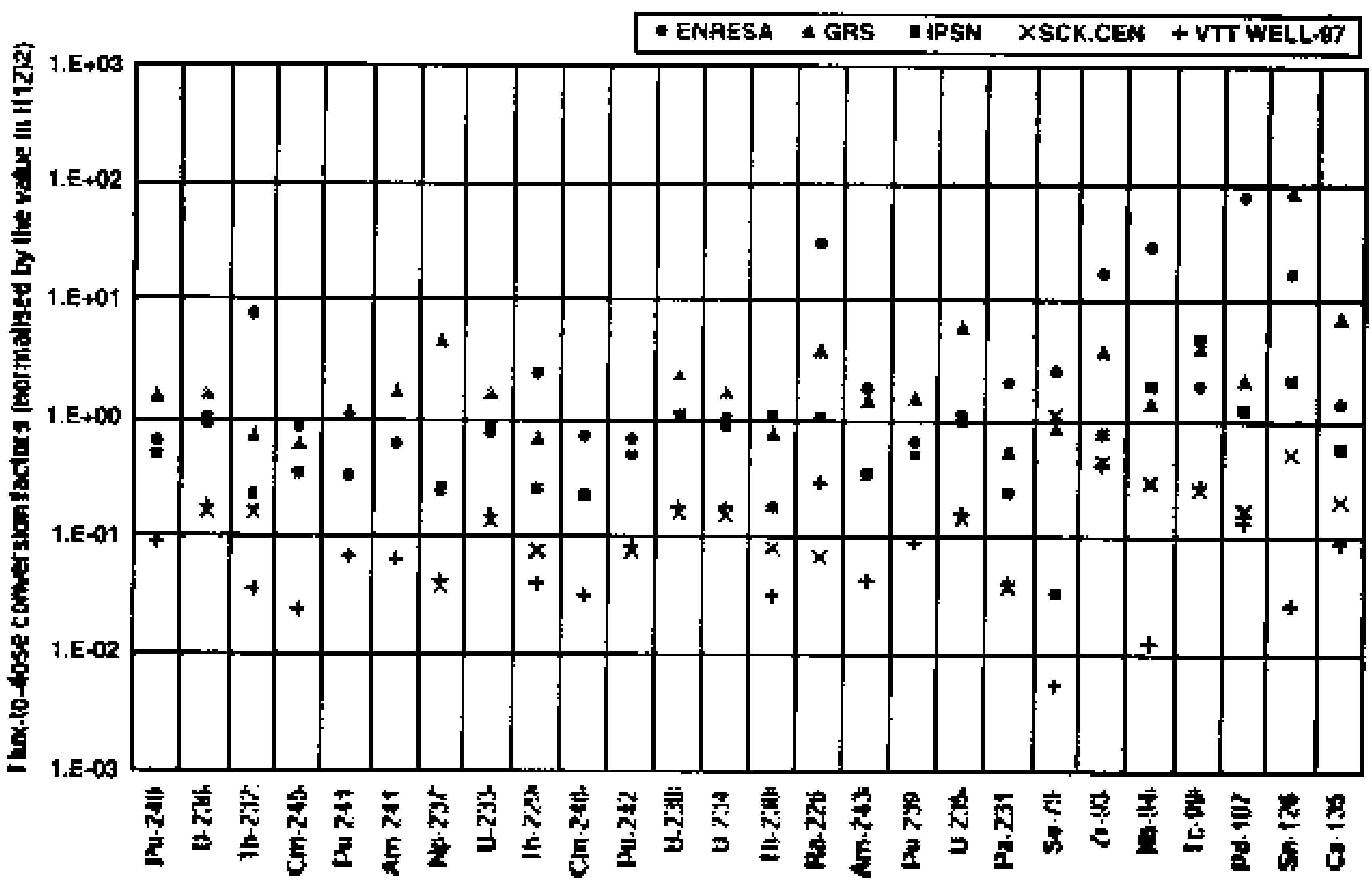


Figure 2. Biosphere conversion factors normalised by the value of H12 (for well type biosphere)

5-8.5 The effect of oxidising conditions in the near-/far-field are only considered in the case of the uplift/subrosion scenario and for the near field of 4 containers in a human-intrusion scenario (see groundwater).

=> There is strong and abundant evidence for well-buffered, low redox potentials in deep geological formations in Japan. As long as the repository remains at depth, it is credible that the assumption of reducing conditions be made (See also response to comment 3-4).

5-8.6 The maximum annual radiation exposure is determined by Cs-135 in the reference case and is about 5 • 10^{-8} Sv/a. The annual radiation exposures of the reference cases in SPA [1] and Kristallin-I [4] caused by Cs-135 are approximately a factor of 160 and 40 higher. This is caused by a strong retaining effect during the transport through the far field in the host rock in the case of H-12. A reason is the higher matrix diffusion depth, together with a higher value for the effective diffusion coefficient in the matrix and an interstitial velocity that is lower on average. In the case of a site-related study it would have to be examined if corresponding values are realistic.

[1] Spent fuel Performance Assessment (SPA), EUR xxx EN, to be published.

L. Luehrmann, U. Noseck, R. Storck: Spent Fuel performance Assessment (SPA) for a hypothetical repository in crystalline formations in Germany. Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-155, (to be published).

[4] NAGRA: Kristallin-I Safety Assessment Report. Technical Report NTB 93-22, Wettingen, July 1994.

=> The values for transport parameters are defined and defended in Supporting Report 3. We agree that these values should be confirmed as appropriate in a real site, as you commented. Note also, that the sensitivity of repository safety to potentially different values for these parameters are considered in H12 (see Table 5.4.2-8).

5-9 Further comments

5-9.1 The chapter of Scenario Analysis seems to be more focussed towards defining a large amount of calculation cases, in order to obtain different alternatives of the disposal system, than to the proper definition of pure scenarios. We think that this can give origin to the confusion of the reader. Maybe, It could be more adequate to define the large amount of the different calculation cases (up to 92 cases for sensitivity analysis and 32 for the total system performance analysis) just in the subsequent chapters dedicated to the consequence analysis.

=> As mentioned in Chapter III of Supporting Report 3, in order to avoid a risk of introducing inconsistency and of losing traceability at the interface between scenarios and calculation cases, the scope of the scenario analysis is expanded to include the development of a number of calculation cases corresponding alternative geological disposal systems and uncertainties in scenarios, models and data.

Scenarios for safety assessment calculation are briefly described in Section 5.3.2 of the Project Overview Report, which were obtained based on system understanding described in Section 3.2 of Supporting Report 3. According to the scenarios, each calculation case was derived.

5-9.2 Features of the disposal system which are not generally accounted for in performance assessment of an unperturbed repository system, such as undetected geological structures or undetected past human intrusions, are not considered in the FEP list of Table 5.3.1-1.

=> Perturbation scenarios are developed to illustrate possible impacts of external perturbation factors (natural phenomena, future human activities and initial deficiencies of repository components) on the expected system performance under the normal evolution scenario.

The likelihood of these situation can be made low by selection of an appropriate site and design. Therefore, an illustrative assessment relevant to these scenarios was made solely on the assumptions listed in Section 5.3.3.

If there are undetected geological structures or undetected past human intrusions, these features may cause disturbance of hydrology around the repository. These features are included implicitly in a FEP, "hydrological properties of host rock", and possible impact of these features on flow around the repository could be examined implicitly by the consequence calculations regarding the uncertainties of hydrological properties of host rock (see Section 5.4.2 of the Project Overview Report and Section 5.2.1 of Supporting Report 3).

5-9.3 For some screened FEPs of Table 3.1.1, that may give origin to significant consequences (thermal expansion of host rock, gas generation effect, gas driven mediated transport), fulfilment with the criteria (1) to (4) of page V-14, is not evident and should be proved.

=> We recognize and agree that decisions on FEP screening will eventually require a formal methodology and documentation of the objective evidence and subjective judgements used in decision making, as well as documentation on the technical training and biases of the "experts". Given the generic site basis for H12, however, at this stage we have necessarily made decisions on a more subjective (expert judgment) basis with less than adequate data in many cases. We will add text in Section 2 on page V-14 to note that

"The entire scenario methodology is an iterative process. Decisions and judgements regarding the inclusion or exclusion of FEPs are always open to re-interpretation as new information is obtained. Because of the generic site information and preliminary design concepts in this H12 report, none of the following FEPs analysis should be construed as a final decision at this time."

5-9.4 Possible mechanical and thermo-mechanical effects on the behaviour of the disposal system are not considered. Processes, as thermal stresses, that can change the permeability of the fractures, or thermally induced groundwater movements, should in some way be assessed.

=> It is not strictly correct to assert that such affects are "not considered". Under slightly different terminology, such effects have been considered (for example, "thermal expansion of host rock" and "deformation of host rock"). We concur that such effects have the potential to be important under certain conditions, especially for waste packages that might display fabrication failures, with the potential for release of radionuclides during the early thermal period. Within the context of the AEC guidelines emphasizing groundwater scenarios and based on limited available information, however, some of these impacts have been judged to be insignificant.

5-9.5 On the text can be read "Isolation failure and significant impact due to natural phenomena and human activity could be avoided by appropriate site selection and repository design" (page V-17). This is very questionable, since for example, "Meteorite Impact", up to a certain extent, can be regarded as a site/repository independent scenario. The same happens with "Direct Human Intrusion" scenario, having into account that future human subsurface activities can not be scientifically predicted.

=> We will clarify that some isolation failure scenarios can be avoided by appropriate site selection and repository design. Other isolation failure scenarios, notably meteorite impact, can be excluded based on its extremely low probability of occurrence as well as its catastrophic, non-radiological impacts (see Section 3.2.4.5 of Supporting Report 3).

The possibility of human intrusion into repositories is expected to be reduced by the measures such as avoiding places where underground resources occur by site selection and choosing sufficient depth by repository design. Nevertheless, the possibility of such an event can not be scientifically precluded. Therefore, the possible consequences of human intrusion scenarios were carefully analyzed based on probability of drilling and consequence of a drill worker assuming that a borehole will hit a waste package and core the vitrified waste. The result shows the obtained risk is lower than foreign guidelines (see Appendix of the Project Overview Report and Supporting Report 3), therefore this FEP was screened out.

These evaluations will be clarified in the final version of H12.

5-9.6 It seems to be more adequate to treat biospheric subsystem FEPs as a part of the whole Scenario Analysis (See Supporting Report 3, where the biospheric FEP list and interaction matrix are treated in a separated chapter "Models and Dataset for the Reference Case").

=> For the H12 biosphere assessment, the nuclide migration processes and exposure pathways were incorporated into the compartment model. The interaction matrix is useful to show the relation between FEPs and the main biosphere components which compose the compartment. Therefore, it is convenient to include biospheric subsystem FEPs in the biosphere modelling section.

5-9.7 In the perturbed scenarios "Deficiencies of repository components", just the "overpack defect" scenario has been considered. However, a scenario as broadly considered in performance assessment, as is the "Repository Seal Failure" has not been defined.

=> Quality control will make it possible to detect, avoid or repair any initial defects in the repository. Therefore, an illustrative assessment relevant to the initial defects will be made solely on the assumption that sealing of the overpack is incomplete (see 5.4.3 2) of the Project Overview Report and Sections 3.3 3), 5.4.2 and Table 3.2.5-3 of Supporting Report 3).

Although a scenario as is the "Repository Seal Failure" has not been defined explicitly, the impact of this assumption could be examined by the results of consequence calculations assuming high flow rate or no retardation in host rock, which have been discussed in the Project Overview Report and in Supporting Report 3.

5-9.8 Perturbation scenarios of the type "Future Human Activities" as deep wells and boreholes have been considered in the evaluation, but they have been modelled using very simple and not proved to be conservative assumptions (see pages V-73 and V-74).

=> At this generic-site stage that is the basis for H12, the generation of a detailed human intrusion model was judged to possibly arouse concern without recourse to our ability to justify specific data or model assumptions. Therefore the human intrusion case, as an unexpected isolation failure scenario, is treated in a simple, illustrative manner. The likelihood of the direct human intrusion into the disposal site can be made low by selection of an appropriate site and design. Therefore, an illustrative assessment relevant to the human intrusion was made solely on the assumption of a well drilling.

As mentioned in Section 3.2.4.3 of Supporting Report 3, in accordance with the viewpoint described in NAS (1995), perturbation scenarios of the type "Future Human Activities" focus on the danger caused by damaging the planned design functions of a geological disposal system due to one borehole drilled from the surface through a region in the vicinity of emplaced waste packages. We agree with the statement in NAS that "One can always conceive of worse cases, such as multiple boreholes with each penetrating a canister, but this single-borehole scenario seems to us to hold the promise of providing considerable insight into repository performance with the minimum complication" (see page 111 in NAS (1995)). This meets the purpose of perturbation scenarios to illustrate possible impacts of external perturbation factors on the expected system performance under the normal evolution scenario.

NAS(1995): Technical Bases for Yucca Mountain Standards.

5-9.9 Scenario Analysis should define clearly the period of time considered in the evaluation and it should correspond to the time cut-off used for the calculations. However, there is an inconsistency between the chapter describing geological environment, where it is stated that JNC should consider a future time interval of 100,000 years after repository closure (see page III-1), and the radiological assessment chapter, which considers far longer time-scales, up to 100,000,000 years.

=> There is a misconception that the 10^5 years value represents an assessment-time cut-off. The value of 10^5 years is a figure proposed in the AEC Guidelines as target with respect to the ability to confirm the geological stability of a potential waste disposal site over such a period. The H12 SA calculation was conducted without specifying any cut-off time according to the AEC Guidelines which requires to present information in such a way that the risks (or hazards) at the time period of maximum influence on man and his environment.

Regarding the inconsistency between the chapter describing geological environment, please see the response to comment 3-1.

5-10 Additional comments

5-10.1 Figures 5.3.3-1 of the "Project Overview Report" and 3.4-1 of Supporting Report 3 – "Safety Assessment" are the same. In Figure 5.3.3-1, instead of "AD. Alternative design" for the Natural Barrier it should be "CM.Alternative model cases".

=> Thanks. The figure will be corrected.

5-10.2 PageV-38: The temperature in the rock at 1000 m after 1000 years is not 45 °C like the text indicates. 45 °C is the initial temperature in the rock and the emplacement system recovers this temperature after 10000 years, not 1000 years.

=> As described in Section 2.2.2 in Supporting Report 3, temperature at disposal site of 1,000m depth in crystalline rock with a ground temperature of 45°C is estimated to reach about 55°C at 1,000 years after disposal. In the case of sedimentary rock of 500m depth with an initial ground temperature of 30°C, temperature at disposal site is estimated to reach about 40 to 50°C.

The influence of heat release from the repository will be limited in the near-field rock. At 10,000 years after disposal, the rock temperature recovers the initial temperature. In addition, most of radionuclides will be retained and retarded in the EBS for a very long time as shown in Figure 6.4-1 in Supporting Report 3. Therefore, for simplicity in the analysis the ambient temperature of the host rock is considered after 1,000 years.

We will modify the following text in page V-38: "The repository is located at a depth of 1,000 meters and the temperature of host rock around the repository is 45 °C at 1,000 years after disposal." to "The repository is located at a depth of 1,000 meters and the initial temperature of host rock around the repository is 45 °C".

5-10.3 Glass leaching: For glass dissolution rate, 0.001 g/(m²d) was used based on long term glass leaching experiments. For the overpack corrosion, "safety factors" were used resulting in a much larger corrosion rate than the long-term rate measured in the laboratory. Are the measured steel corrosion rates considered as less reliable than the glass leaching rates?

=> It is more a question of different design functions requiring different degrees of conservatism than a question of different reliability. Three aspects must be considered. First, the overpack has a containment function, whereas the glass has a controlled release function. Localised differences in dissolution/corrosion rates (pits), therefore, have the potential for a much more significant impact on the overpack function (early failure) than on the glass function. Second, the release of most radioelements is insensitive to glass dissolution rate because of solubility limits. For the few radioelements that may be constrained by glass dissolution, their peak release rate from the EBS is rather insensitive to dissolution rate (see Figure 5.5.2-1) because of attenuation and dispersion of by sorption. Third, there are a great number of processes affecting iron overpack corrosion (*e.g.*, corrosion by initially trapped oxygen, long-term anaerobic corrosion, pitting, mass-transfer of reacted species, and microbial effects), whereas the long-term glass dissolution is controlled by formation of a gel layer/alteration products and mass transfer of reacted species. The larger number of potential process introduces a somewhat higher degree of uncertainty for iron corrosion rate.

5-10.4 Solubilities: The solubilities used in safety analysis seem to differ grossly between different organizations. Partly, it can be ascribed to differences in groundwater compositions, but there are discrepancies that are more difficult to rationalize. In your reference case, the solubility of Zr is given as 10^{-6} M, while using SKB's database it comes out as $2.5 \cdot 10^{-9}$ M virtually independent of pH and water composition for fresh water. An even greater discrepancy is found for Th. These deviations may be of little consequence for the safety of the repository, but will need some discussion. Could you please comment this situation?

=> See response to comment 5-8.2.

5-10.5 The overpack failure times are postulated to be 1000 years after disposal. Variations are made upwards from this number (10000, 100000 years etc.). The overpack is designed to last for at least 1000 years based on conservative assumptions. Why is no analysis based on a best estimate of overpack life time?

=> The prime factor in the relationship between corrosion rate and overpack thickness is the design objective of 1000 years for complete containment. This value is selected to prevent release of radionuclides during the period of highly elevated thermal conditions and high temperature gradients. If future measurements on corrosion rates for overpack material confirm a lower corrosion rate, the thickness of the overpack would be accordingly decreased to met the design objective of 1000 years, rather than claiming a longer containment time for the original overpack thickness. Therefore it would be somewhat misleading to cite calculations of a more extended containment time based on a lower, "best estimate", for corrosion rate. Furthermore, such calculations are somewhat irrelevant because containment times up to 100,000 years will have little impact on dose rates of key dose-contributing radionuclides that have half-lives much longer than this.

5-10.6 The total dose is dominated by Se-79 for the first several thousand years. The half-life of Se-79 used in the analysis is 65,000 years. New information shows that the actual half-life is 1,140,000 years. What consequences will this have for the safety analysis?

=> We are aware of the newer data from Chinese researchers. Unfortunately, this information reached us after considerable calculational cases were already completed. Furthermore, additional adjustments would have been required in the biosphere model. Given the severe time constraints for completing the H12 report, we found it impossible to accommodate this change. However, the information of such new data is noted in Table 4.1.3-2 of Supporting Report 3.

Furthermore, we have conducted scoping calculations regarding the potential impact of this change in half life. With respect to impact on release rates through the EBS and geosphere, the change in peak release rate for Se-79 is negligible because the peak release occurs at a time only slightly greater than the old half-life. Furthermore, the non-sorbing character of Se means that essentially none of it is trapped on buffer or rock, so that combined retardation/"rapid" radioactive decay is *not* significantly contributing to limiting the peak release rate of this nuclide.

5-11 Table 5.4.1.5: Kd values: page V-37

The buffer is set as a mixture of 70 wt% bentonite and 30 wt% quartz sand. Kd values in the buffer should be averaged with the same relationship.

Kd values used are higher or are near the higher range given in other Performance Assessments (Kristallin in ENRESA-98 for Clay).

	AGP Clay*	NAGRA**	JNC***
Ac	0.1 – 3		1
Am	1 – 3	1 – 5	10
Cm	1 – 3		10
Cs	0.2 – 0.5	0.02 – 0.2	0.01
Nb	0 – 0.2	0.02 - 2.5	1
Np	0.1 – 3	0.1 – 1	1
Pa	0.1 – 3	0.1 – 1	1
Pb	0 – 0.5		0.1
Pd	0 – 0.01	0.001 – 0.005	0.1
Pu	1 – 3	1 – 5	10
Ra	0.1 – 0.5	0.02 – 0.2	0.01
Se	0 – 0.003	0.001 – 0.005	0
Sm	0.2 – 1		1
Sn	0.01 – 3	0.001 – 0.05	1
Tc	0.01 – 0.1	0.01 – 0.25	0.1
Th	0.1 – 3	0.1 – 1	1
U	0.01 – 1	0.1 – 1	1
Zr	0.2 – 2	5 - 25	10

* **Evaluación del comportamiento y de la seguridad de un almacenamiento profundo en arcilla. Publicación Técnica ENRESA 03/99 (tabla 6.15).**

** **"Kristallin-I Safety Assessment Report". Technical Report 93-22. NAGRA (1994).**

*** **H12 Project to Establish Technical Basis for HLW Disposal in Japan "Project Overview Report". Table 5.4.1.5.**

Kd values in bentonite for Am, Cm, Pd and Pu are higher than the ones used in the ENRESA-98 (AGP Clay) and in Kristallin exercise. The values for Nb and Zr are also higher than the ones used in ENRESA-98 (AGP Clay) and are included in the range given in Kristallin.

Kd for Ra is lower that the one used in ENRESA-98 (AGP Clay) and in Kristallin. The value for Cs is lower than the one used in ENRESA-98 (AGP Clay) and is included in the range given in Kristallin

=> The reference values for Kd are based on direct laboratory measurement of sorption on intact buffer samples normalised by measurement of effective diffusion coefficient on the same intact buffer samples. We believe that our values are the most representative values that have yet been collected, especially when compared to the "batch" Kd values collected on disaggregated bentonite samples that are the data sources for the other assessments cited. Furthermore, an independent review team evaluated these Kd values. Supporting Report 3 contains further discussion of this approach, which is summarized as follows.

In H12, the distribution coefficients (Kd) for buffer material are derived from measured apparent diffusion coefficients (Da), which represent non-steady state diffusion including retardation by sorption, and effective diffusion coefficients (De), which represent steady state diffusion, by the following equation.

$$Kd = \frac{1}{\rho}\left(\frac{De}{Da} - \varepsilon\right)$$

Because we believe that this methodology can reflect the nuclide retardation in the buffer material better than using batch Kd (see Supporting Report 3, 3.2.1.5 and 4.1.3.7).

When silica sand, which is generally inert with respect to sorption, is added, the distribution coefficients must decrease theoretically. When Kds are derived from batch sorption experiment, the weighted average Kd values considering the buffer constituents should be discussed. However, in the H12 methodology, the effect of silica sand mixture is directly factored into our direct evaluations of Da and De. From the measured Da values, with a variation of silica sand content, results show that the effect of 30% sand mixture is not significant (*e.g.* Idemitsu *et al.* 1994, Sato, 1999). As for De, a small effect is recognized only for anionic species for the 30 wt% mixture (Fujiwara *et al.* 1998).

The Kd values pointed out as high in comparison with the other reports are also derived from measured Da and De values (Cm is determined by chemical analog to Am). On the other hand the Kds are basically derived form batch experiments in the other reports. This is probably the reason for the discrepancy. Note also that our buffer is composed of Japanese bentonite that may introduce some differences in Kd values when compared to different bentonite-based buffers.

In our Kd derivations, Da values measured for the specific element with relevant redox condition are used. However De for tritiated water (HTO) was used for the most elements due to lack of data.

For the cationic species such as Cs and Ra, the derived Kds from Da and De tend to be significantly smaller than batch experiments in fresh conditions. This may be because the porewater in compacted bentonite in the diffusion experiments is rather saline due to soluble impurities accompanied with bentonite.

References

Fujiwara, H., Yasutomi, I., Kato, H. and Ueta, S. (1998): Effective diffusivities of iodine, chlorine and carbon in bentonite buffer material. 1998 Annual Meeting of the Atomic Energy Society of Japan, L25 (in Japanese).

Idemitsu, K., Furuya, H., Tachi, Y. and Inagaki, Y. (1993) : Diffusion of Uranium in Compacted Bentonite in the Presence of Carbon Steel., Mat. Res. Soc. Symp. Proc. Vol.333, pp939-946.

5-12 Hydraulic transmissivity

It´s difficult to understand why the "host rock" transmisivity, (and therefore the fracture aperture), is the only parameter defined by a distribution function, when there are many parameters affected by variability and/or uncertainty. The rest of the parameters are fixed in the reference scenario, and they are only varied when doing the sensitivity analysis. This applies to the fault parameters too (even its transmisivity is fixed).

=> Radionuclide transport in a fractured rock is evaluated using a fracture network model. In the three-dimensional fracture network model, fracture parameters including orientation, size as well as transmissivity (aperture) are defined by distributions. The result of radionuclide transport in a three-dimensional fracture network system is approximated by multi-one-dimensional model described in Section 4.2 of Supporting Report 3. In the Project Overview Report, only the parameters for the multi-one-dimensional model are described. This may cause misunderstandings like the one responsible for comment 5-12.

5-13 Natural barrier parameters

The description on the conceptual model and how the natural barrier parameters are obtained (pag. V-38) we think is rather poor. It´s necessary to review "Safety Assessment" report to understand the methodology and the modelling strategy. In general, the information provided in report JNC TN1400 99-010 falls short of the minimum needed. There is room for traceability and transparency improvements.

=> The description of the natural barrier in the Project Overview Report can be improved and we will revise it to make more understandable. We would welcome whatever specific suggestions you could make.

5-14 Sensitivity analysis

There is a lack of transparency in the list of the calculation cases and the presentation of sensitivity analysis results (Tab. 5.4.2-1 & 5.4.2-8). In the same way, the set of scenarios considered in the safety assessment is confusingly presented in the summary.

=> This is an important portion of the Project Overview Report, so we are anxious to discuss, clarify and resolve your concerns. Perhaps the table formats are too comprehensive and wordy; we would welcome your specific suggestions on how we might improve transparency and reduce confusion of this material.

5-15 Alternative media

The treatment of alternative media is over simplified, as the conceptual model used for the geosphere in the reference scenario is not modified. Only the hydraulic gradient, some rock-type dependent parameters and the reference water are modified.

=> The conceptual model used for the geosphere in the Reference Case is not modified in the alternative geological environment cases, because fracture properties in various rock types except granitic rock have not yet been clarified and the connectivity of fractures in granitic rock is considered relatively high. Thus, adopting fracture properties in granitic rock for another rock types is conservative (see Section 3.2.2.1 (1) of Supporting Report 3).

5-16 Geosphere

In all the cases for the normal evolution scenario, there is no change in the geological barrier along calculation time. It's like a fixed and simple photography of the geological medium. The justification for this approach should be given.

=> As shown in Section 3.2.2.9 and 3.2.2.2 of Supporting Report 3, thermo-mechanical, hydrogeological and chemical effects on host rock around repository after emplacement are considered as insignificant. Conservative and simplified treatment of geological barrier conceptual models is introduced in H12, which is relevant for the generic SA stage. Therefore, time-dependent changes in mechanical, hydrogeological and chemical properties of the rocks are not considered in the Reference Case.

It should be noted that other previous repository safety assessments have also adopted a time-invariant reference case as a baseline (*e.g.*, Kristallin-I Reference Scenario; SPA).

5-17 Geological changes

When assessing the consequences of changes in the geological barrier induced by natural phenomena:

if the calculations were done for 1,000,000 years (instead of 100,000 years) and the erosion and uplift rates were 1 mm/year, the repository would be exposed to surface, as it's 1,000 m deep. Which would be the consequences of this?

In the "Climate and sea level changes" scenario, changes in the dilution factors in the biosphere are not considered (only changes in the chemistry of water are). Which is the justification for this?

=> Exposure of the repository to surface was taken into account in the uplift/erosion scenario (see Section 5.5.2 2) (1)). In the climate and sea-level change scenario, cyclic change of groundwater chemistry in the repository from SRHP to FRHP was assumed. However, changes in the dilution factors in the biosphere were not considered. Even if sea-sediment is considered as the geosphere–biosphere interface (GBI) for SRHP in this scenario, maximum dose increases at most one order of magnitude. which is still much lower than the regulatory guidelines in foreign countries.

5-18 Deep well scenario

The "Deep well scenario" is not quite well described. It doesn't seem to be changes nor in the groundwater flow directions neither in the hydraulic gradients induced by the water extraction. Nothing is specified about the extraction location and characteristics, only the water uses and dilution are modified.

=> Taking account of the IRG comment, we think this point needs to be made more explicitly and we will include a discussion along the following lines:

Groundwater aquifers that can be used for well water (including deep wells) in Japan are generally limited to sediments post-Pliocene and recent volcanic rock (The Geographical Survey Institute, 1977). Lowlands, such as alluvial fans, flood plains and deltas consist of Holocene deposits, including sand, pebbles, silt and clay. Generally, groundwater is found in abundance in the shallow parts of the lowlands. Tablelands and benches mainly consist of Pleistocene deposits, and produce a considerable amount of groundwater from depth. In general, thick Pleistocene formation exists beneath the Holocene in the plains. More than 90% of groundwater which can be utilised in Japan is considered to exist in the relatively shallow Holocene and the Pleistocene formation (Kurata, 1962), with little being produced from deeper Pliocene deposits. Thus, unless there is significant future uplift of an initially deep repository, we believe a scenario involving penetration of a well to repository depths is highly unlikely. Nonetheless, we feel it is prudent to at least examine the potential consequence of a "deep well scenario".

To evaluate a "deep well scenario", it is assumed that radionuclide release from the geosphere to the biosphere are taken to the surface environment from a deep well sunk into a contaminated deep aquifer (right Figure). The flux from the aquifer into the deep well is assumed to be the same as the flux from the major fault into the aquifer with no addition occurring in the aquifer. Implicit in this assumption are two further assumptions: (1) the deep well intercepts the radionuclides discharged into the aquifer; and (2) there is no sorption in the aquifer. The contaminated well water is assumed to be used mainly for drinking and irrigation. The extraction rate of the deep well is set as 2×10^6 m^3/y, which is derived from the log-mean value of the distribution of waterworks capacity, where water is extracted mainly from deep wells (Japan Water Works Association, 1996) (page V-59, V-62 in Supporting Report 3).

In the alternative case when a new well is drilled downstream of the repository after disposal (Perturbation scenario, page V-99 of Project Overview Report), there would be some impact on groundwater regime in an aquifer, especially on groundwater flow rate and direction, as pointed out in the comment of reviewers. However, it could be considered that these changes would be neglected due to the above assumptions.

Agricultural Structure Improvement Bureau of the Ministry of Agriculture, Forestry and Fisheries (1977): Investigation on Actual Status of Agricultural Water. (in Japanese)

Geographical Survey Institute (1977): Groundwater. National Atlas of Japan, p.25. (in Japanese)

Japan Water Works Association (1996): Statistics of waterworks in FY1994. (in Japanese)

Kurata, N. (1962): Groundwater in Japan. Chemistry of Underground, Series I, Jitsugyo Koho-sha, p.117. (in Japanese)

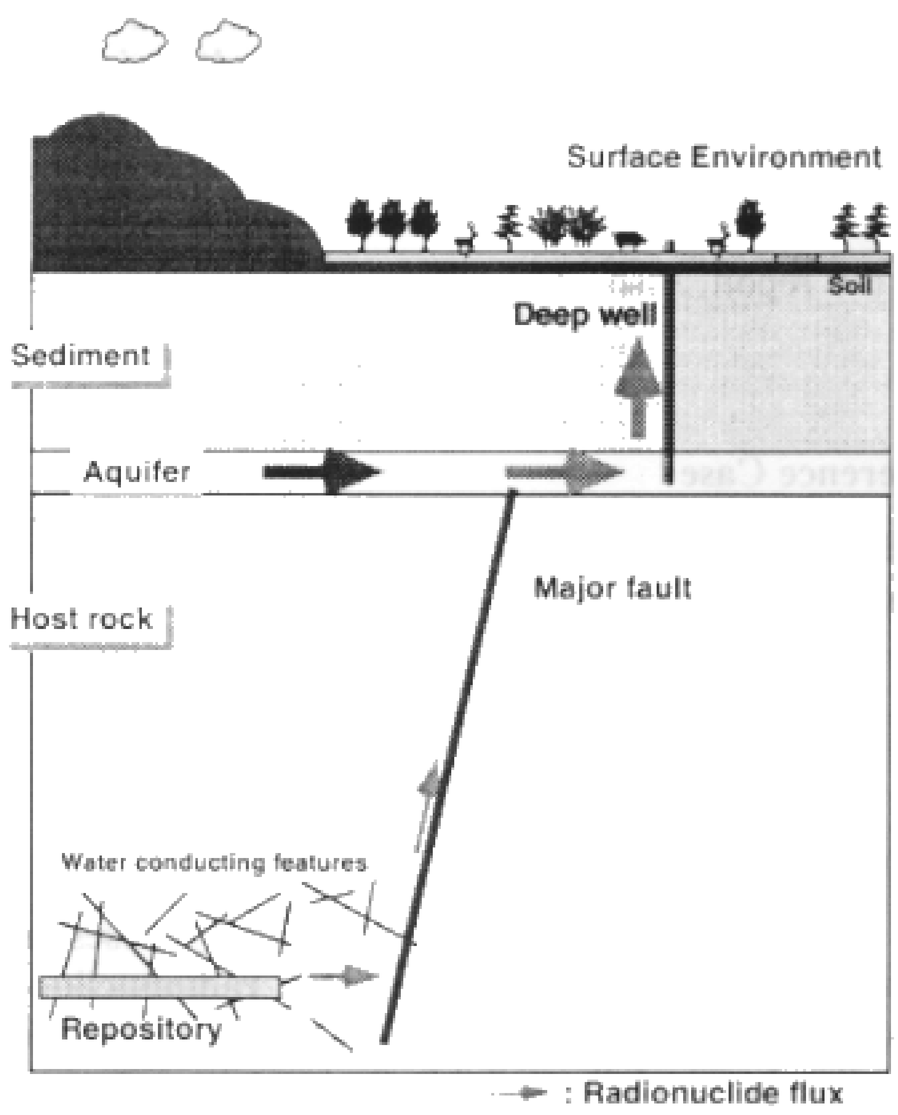


Figure Schematic view of deep well scenario

5-19 Isolation barriers

One of the conclusions in point 5.6.2 is that: "...even where no credit is taken for the geological transport barrier, the system is sufficiently robust that the calculated dose maxima remain low, provided that the host rock continues to provide a suitable environment for the EBS. The EBS thus plays a particularly important role given the inevitable variability, and associated uncertainties, of the geological environment".

Above conclusion is apparently based on Figure 5.5.1-8 "Change of maximum dose with increasing migration distance in the host rock (Reference Case, 40,000 waste package)" where doses with "migration distance in the host rock=0" remain acceptable. Comparison of Th229 peak doses in Figures 5.5.1-6/7/8 shows that, apparently, in calculations for Figure 5.5.1-8 the 800m long fault has been bypassed.

Figure 5.5.1-8 caption should state clearly that "no credit is given to the fault", and justification for above mentioned statement should be provided in the text.

=> As described in Section 5.5.3 1) (3), in order to illustrate the efficiency of the EBS performance, a totally hypothetical calculation case that assumes direct radionuclide release from EBS to the biosphere, i.e., retention of radionuclides in the natural barriers is ignored, is also considered in the total system performance analysis. The result of this "robust case" is included in Figure 5.5.3-3. We will explain the result of this robust case to support the conclusion in Section 5.6.2 in the final report.

The text related to Figure 5.5.1-8 will clearly state that no credit for attenuation by the fault is included in these analysis.

5-20 Impact of waste package number on dose

In Page V-83 of the "Project Overview Report" it would be useful to explain that doses caused by the 40000 waste packages (Figures 5.5.1-6/7) are calculated multiplying the release rates per waste package (Figures 5.5.1-2/3) by the total number of waste packages (40000) and the steady state flux-to-dose conversion factors (Figure 5.5.1-4).

=> We will add such text to this report. Also adding the term "conservatively calculated".

5-21 Geochemistry of Reference Case

In Figure 5.5.3-3 of the "Project Overview Report", it would be useful to identify the case "fresh reducing high pH+0.01 hydraulic gradient+silicic crystalline rock" as the "REFERENCE CASE".

=> The Reference Case will be identified in the figure.

5-22 Impact of geosphere on results of SA

5-22.1 In Section 5.5.3 of the "Project Overview Report" doses are calculated for several alternative geological environment cases. It's assumed that radionuclide transport is controlled by advection along the fractures, and the characteristics of the fractures are the same in all the cases. Only the properties of the "rock matrix" available for matrix diffusion change from one case to another, as shown in Table 5.5.3-2.

Since transport modelling is the same and transport parameters are quite similar for the 6 rock classes, it can be seen from Figure 5.5.3-3 that peak doses (for the same groundwater and hydraulic gradient) are practically the same for the 6 rock classes.

=> See response to comment 5-15.

5-22.2 In the "soft rocks" (Neogene sandstone and Neogene mudstone/tuff) radionuclide transport probably will be controlled by diffusion through the rock, not advection along fractures. Fracturation probably will be different for each hard rock class. As a consequence, the simplified modelling used in H12 to study alternative geological environments is not realistic and can lead to wrong conclusions about the influence of the geological formation characteristics on the repository performance.

=> It is possible that sedimentary rock systems will have a different balance of matrix diffusion and fracture transport than crystalline rock.

According to our observations conducted at various sites, in some Neogene sedimentary rock fracture dominated flow. However, there are few investigations on the fracture properties in Neogene sedimentary rock and thus these properties are not well known. Therefore, the fracture properties in granitic rock are adopted to simulate Neogene sedimentary rock where fractures dominate.

On the other hand, in some Neogene sedimentary rock fracture flow is not dominant. Thus, radionuclide transport in porous media was also evaluated using equivalent continuum models. Since it is found that performance assessment evaluated by fracture network models is more conservative than that evaluated by continuum models, we choose to represent radionuclide transports in all rock types by fracture network models for safety case evaluation (see Section 5.2.3.2 of Supporting Report 3).

Finally, as you note, it might be possible that diffusion rather than adovective flow could be the dominant transport mechanism of radionuclides through unfractured low-permeability "soft-rocks". Clearly this would lead to strong reduction in peak dose rates for the repository. But in the absence of site-specific information on fracture characteristics and permeability, we have decided not to illustrate this more favourable case.

5-23 Tc-99

In Figure 5.5.1-7 of the "Project Overview Report" the label for Tc-99 should be included, because doses due to Tc-99 are shown in the graphic.

=> We will label Tc-99 in Figure 5.5.1-7.

5-24 Cs-135 doses

In Page V-90 (conclusions about the EBS) it's said that Cs-135 doses are sensitive to the glass dissolution rate and relatively insensitive to the flow rate in the EDZ. From results in Table 5.5.2-1 it can be seen that effect on peak doses due to Cs-135 of changing one order of magnitude the glass dissolution rate is smaller than the effect of changing one order of magnitude the flow rate in the

EDZ. As a consequence, Cs-135 doses are more sensitive to the flow rate in the EDZ than to the glass dissolution rate, and above conclusions should be written in a different way.

=> See response to comment 5-25.

5-25 Radionuclide concentrations

In page V-86 it is said that "if the nuclide concentrations outside of buffer were set to zero (equivalent to a high flow rate in the EDZ), the maximum release rates of Se-79 and Np-237 from EBS were approximately two orders of magnitude greater than those in the Reference Case". Concentration equal to zero outside the buffer is not one of the variants considered in H12, and this assertion seems arbitrary (although probably correct).

Conclusion 3 in page V-90 seems to be based on calculations done assuming concentration equal to zero outside the buffer, which should be clearly stated.

Perhaps the best solution would be to include a new "alternative model case" for the EBS, with concentration equal to zero outside the buffer.

=> Release rate of Cs-135 is not linearly sensitive to glass dissolution rate and the flow rate in the EDZ, when both parameters are increased one order of magnitude compared with the values for the Reference Case. The following interpretation is described in 6.2.1 1) of Supporting Report 3:

In the former case, this response may be because the sharp peak in the release rate from the glass that appears as the glass lifetime (and hence the Cs-135 leaching time) becomes small is significantly modified by transient diffusion retarded by sorption. In the latter case, release rate of Cs-135 is limited by the glass dissolution rate.

Based on the results of the data variation cases, release rates of many nuclides from the EBS vary approximately in proportion to the flow rate in the EDZ. The release rate assuming concentration equal to zero outside the buffer, that is a mathematical treatment as a hypothetically extreme boundary condition, was presented as an information to show the upper limit of release rate due to variation of the flow rate in the EDZ.

As pointed out in the comment, conclusion 3 in Page V-90 should be derived from explicitly defined calculation cases. Therefore, we will note in V-86 that the release rate assuming this hypothetically extreme boundary condition is presented in order to discuss the sensitivity of this parameter variation to release rates from the EBS.

According to these comments, the conclusion in Project Overview Report will be written in the following way.

- The releases of nuclides that are limited by solubility are not sensitive to the rate of glass dissolution, but are sensitive to changes in solubility resulting from variations in pore water composition. The release rates of these nuclides from the EBS are sensitive to the flow rate in the EDZ.
- The release rate of Cs-135 is sensitive to both the glass dissolution rate and the flow rate in the EDZ when 10-fold decrease from the Reference-Case data, but not linearly sensitive to 10-fold increase from the Reference-Case data.

5-26 Diffusion coefficient: page V-91

Conclusion: "The correlation that is assumed between the effective ....a greater diffusion coefficient" should be revised.

In my opinion, what really happens is that:

- **transport is practically independent of the value of De (in the studied range), because diffusion into/from the rock matrix is faster than advection in the fracture. As a consequence, it´s correct to assume that the concentration in the fracture water and the rock matrix porewater are the same,**

- **since the minimum value of the Kd in granite is quite high (0.01 m3/kg for Selenium), the radionuclide traveltimes in the host rock and the fracture are independent of the rock matrix porosity,**

- **the radionuclide traveltimes in the host rock and the fracture are proportional to the rock dry density. Since in the data variation cases this parameter is changed only 1%, the effect on doses is negligible.**

Conclusion can be written as follows: "Changing the rock matrix porosity has no effect on doses. Results are insensitive to small changes in the rock dry density. Four orders of magnitude.........maximum dose equivalent".

=> Although the correlation among porosity, dry density and De are possible, there are still quite large uncertainties within this correlation. Therefore, we would like to revise the conclusion based on the comment as follows.

"Results are insensitive to small changes in the rock matrix porosity and the rock dry density. Four orders of magnitude difference in the effective diffusion dose not change the maximum dose equivalent significantly, either in cases with or without the fault, whereas one order of magnitude difference in the groundwater velocity within the fault results in around one order of magnitude difference in the maximum dose equivalent".

5-27 Minor comments

5-27.1 Page V-6: Figure 5.1.3.2: Geological environmental

=> The figure will be corrected.

5-27.2 Page V-34: Table 5.4.1-1: distribution coefficient – effective **diffusion Circumference**

=> The table will be corrected.

5-27.3 Page V-44: Table 5.4.1-8: length as a host rock

=> The table will be corrected.

6. COMMENTS ON CHAPTER VI

6-1 Section 6.2.3: Requirements for safety assessment: page VI-8: last paragraph

Statements about 'tremendous changes in the next glacial epoch' and 'this glacial epoch will have a relatively small impact on the geological environment of Japan' seem to convey contradictory messages. Last part of paragraph on next page contains another contradictory message about time frame of the assessment.

=> We will modify the following sentence in page VI-8:

"Man's lifestyle and environmental conditions will undergo tremendous changes in the next glacial epoch that is expected to arrive around ten thousand years from now. There will thus be inherently greater uncertainties involved in the results of a safety assessment for this period and thereafter. As noted in chapter III, this glacial epoch will have a relatively small impact on the geological environment of Japan.", to

"Man's lifestyle and environment will undergo tremendous changes in the next glacial epoch that is expected to arrive around ten thousand years from now. There will thus be inherently greater uncertainties involved in the results of a safety assessment for this period and thereafter. However, as noted in chapter III, changes that occur during this glacial epoch will mostly affect the near-surface environment, and will have a relatively small impact especially on the deep geological environment of Japan."

We will also consider adding at the end of p. VI-9, paragraph 2, a sentence along the following lines:

"Predictions related to the deep geological environment are, however, possible over considerably longer timescales, due to the lesser impact of these climatic influences".

6-2 First point of first list on page VI-8

Comment about use of reference biosphere is confusing.

=> We will modify the following sentence in page VI-9:

"Use of reference biosphere as a tool for measuring the performance of a geological disposal system in terms of a radiation dose yardstick (IAEA, 1996)", to

"Use of reference biosphere to avoid speculative discussion on the future evolution of the biosphere by providing a simple and robust approach to representing transfer through the biosphere to humans (IAEA, 1999)",

and replace IAEA (1996) for the following reference.

> IAEA (1999): Critical Groups and Biospheres in the Context of Radioactive Waste Disposal, Fourth Report of the Working Group on Principles and Criteria for Radioactive Waste Disposal, IAEA-TECDOC-1077.

6-3 Page VI-12: last paragraph

The use of complementary safety indicators has no impact on repository safety; it can only improve the presentation of the safety case.

=> True. We will modify the following sentence in page VI-8:

"Application of complementary safety indicators can increase the reliability of long-term safety discussed in Chapter V.", to

"Application of complementary safety indicators can increase confidence in long-term safety and improve the transparency of its presentation."

It is also noted that the presentation of the safety case is itself a key issue, in order that it is accessible to a wide audience. Thus, indicators that are not directly related to safety should also be considered.

6-4 Site selection

Requirements on a site are defined with the help of physical characteristics of the geological system. To what extent can these characteristics be explored and/or proved with sufficient certainty for a conceptual model? This can vary for different rock types. Can e. g., the fracture system determining the permeability of a jointed solid rock be described with sufficient certainty? Deficiencies in exploration may lead to conservative assumptions within the performance assessment. The aspect of explorability/provability should, therefore, be taken into account in the selection of a site.

=> We agree that the explorability/provability of geosphere characteristics are important in the selection of a site. But, we only noted in page II-5 (Project Overview Report) 1) the characteristics that a favourable geological environment should have.

Since explorability/provability is site specific and is unknown at present, we are considering the interactive relationship among site characterization, repository design and performance assessment in a real field [Ijiri *et al.*, 1999 at 4th GEOTRAP].

6-5 General comments on Chapter 6

6-5.1 This is the key chapter of the Overview Report, for it is meant to address the two primary objectives of the H-12 study:

- **Outline the technical basis for the reliability of geological disposal in Japan.**
- **Provide input into the siting and regulatory procedures.**

The chapter is not convincing. In particular it gives little solid material on which a decision to go forward could be articulated. The treatment of topics is almost scholastic. What are the real difficulties of gaining technical confidence not only by the specialists but also by the local public; what are the difficulties of managing a long-term project; how to keep focused on safety culture; how to ensure flexibility; which are the current challenges internationally at technical and regulatory level; and why is there confidence that these challenges can be met. What is the general thinking in Japan? What merit those opinions have?

=> We concur with your point that Chapter VI does not deal with all of the issues that will emerge after Japan makes a decision to develop a deep geological repository. This omission from Chapter VI, however, is intentional. The H12 Coordination Group, composed of representatives from Japanese universities, government, private industries and utilities, examined the intended scope of this H12 report. It was recommended that the H12 report must establish the technical feasibility and reliability for supporting a decision to proceed with geological disposal. It was further recommended that the H12 report should focus on R&D information needs associated with the development of site selection criteria, creation of safety standards, and the technical basis for management of an eventual disposal site.

The key non-technical issues you cite, such as dealing with the local public, non-safety aspects of site selection (*e.g.*, land ownership), establishing a safety culture, etc., are indeed potentially important and need to be considered within the context of nuclear waste management in Japan. It was the judgement of the H12 Coordination Group, however, that neither the H12 report nor JNC as a lead organisation were appropriate for this evaluation of non-technical issues.

We would, however, appreciate further dialogue with the IRG regarding improvement in the treatment of the technical aims and accomplishments of the H12 report, and especially in Chapter VI. We can also consider your suggestion regarding a separate high-level report that might examine other, non-technical factors that may affect site selection and confidence building between the technical and public communities.

6-5.2 It would seem that, in order to meet the primary objectives, JNC could usefully put together a high-level report where the necessary information - JNC's, national and international - is abstracted, commented upon, and recommendations made. JNC could pattern this report, *e.g.*, on Ch. 9 of AECL-10711 of 1994, or on the stand-aloneAECL-10721 of 1994.

=> See response to comment 6-5.1.

6-6 Siting

This section is rather "light" and states the obvious. Indeed, what we read is not necessarily the result of this study. Also, this section reflects a technical-only point of view, whereas site selection will be influenced by consideration other than long-term safety, *e.g.*, all considerations that are typically included, in Europe and North America, in an environmental assessment report. Can one do without commenting on those aspects?

=> According the requirements of the AEC Guidelines, this section mainly focuses on the technical basis of siting rather than the social aspect of siting.

Detailed requirements for siting will be discussed and prepared separately with the framework of safety regulation.

6-7 Comments to 6.1.2

6-7.1 Do you plan to identify - for screening - major crush zones ? How to you suggest topography should be factored in for selecting sites ? How about criteria on conservation and groundwater areas, land use plans, land ownership, etc.?

=> The answer to the first question is: Yes, crushed zones and other zones of wealness will be identified and avoided as part of site screening. Second, Yes, topography is considered as part of determining hydraulic gradients and rates of erosion. Third, the various criteria you discussed will certainly be used for screening but are not part of the technical basis.

6-7.2 A more generic questions is: which are the criteria to move forward that you considered initially, and which ones you did not retain and why ? Should this information not be communicated as well ?

=> The criteria mentioned in chapter 6 are the basic and primitive ones. Categorization of the criteria is not within the scope of H12.

6-8 Comments to 6.1.3

6-8.1 This section is also rather "light" and states what seems to be obvious, i.e., what we read is not necessarily the result of this study. At present, this section reads as asking for carte blanche for

doing research. The issues here are: is the technology mature to go ahead to site selection and characterization? What is needed "de minimis" to make an acceptable environmental safety case?

=> Detailed descriptions of siting criteria and/or siting approach are not within the scope of H12. Specific tools for site characterisation are discussed as part of the Supporting Reports. Additionally, characterisation methods as broadly described by many programmes. Therefore, we feel that there is an adequate technical basis for site selection and characterisation. Adequacy of the safety case is dependent on the regulatory framework.

6-8.2 Whatever is suggested in 6.1.2 and 6.1.3 should be put in perspective with the results of the present study: where are the uncertainties; how to deal with them; why can we have confidence that these uncertainties can be dealt with; what could be a step wise implementation process of geologic disposal; etc.

=> Descriptions required in this section are the basic technical requirements for siting and the present state of the methodology R&D for site characterisation.

6-9 Comments to 6.2.1

6-9.1 We learn that the layout and design of the repository _is_ flexible,"as explained in Ch. IV". But, in fact, that is not so obvious, and the issue of flexibility deserves a whole section to itself in formulating recommendations to the implementor, the regulator, and the policy maker.

=> See response to comment 6-5.1.

6-9.2 Similar comments to section 6.1.3 apply here. For instance, why is there confidence that the buffer material with desired properties can be obtained?

=> See response to comment 6-5.1.

6-10 Comments to 6.2.2

This section seems rather light. It doesn't really summarise lessons learnt. Should it be part of the report? Time needs to be taken to explain the difficulties of decision making in a step-wise manner where not only technical institutions will be involved.

=> See response to comment 6-5.1.

6-11 Comments to 6.2.3

6-11.1 In 1), is a case being made, or to be made, to decouple uncertainties in natural and engineered barriers from uncertainties in the biosphere?

=> Due to the uncertainties in the biosphere, the "reference biosphere" approach is used for the biosphere modeling. This tends to decouple the assessment of the biosphere from that of the near-field and geosphere, although the nature of the geosphere-biosphere interface is always emphasized, as is the need for consistent treatment of all components of the disposal system.

6-11.2 In.3): (i) the NEA has more recent references than the 1991 Collective Opinion: the IPAG-1 and - 2 studies, and the Confidence Report of March 1999. (ii) a safety assessment is just more than technical analyses and calculations: it is also a collection of technical judgements made that are to some extent subjective and whose bases have to be stated. Confidence aspects of producing a safety assessment should be addressed. A safety assessment also provides a basis for dialog amongst stakeholders. This aspect and its implications should also be mentioned.

⇨ See response to comment 6-5.1.