Chapter 8

Safety and Licensing

Authors and Contributors

Safety and Licensing

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Overview

The preliminary safety examination of the ESS facility indicates that the ESS is able to meet very high safety standards in case of a proper design. However, after 40 years of operation the ESS target will produce a similar quantity of radiation to a single used core of a neutron beam research reactor. Thus, the ESS safety and licensing must be given careful consideration. This summary briefly describes the preliminary safety examinations and outlines the safety assessments that must be completed prior to construction of the ESS. Fuller conclusions and facts about ESS safety and licensing are presented in chapter 8.9.2.

The status of the ESS safety subtasks

Legal Aspects

Basic requirements and regulations exist for licensing and authorisation of the ESS Facility in normal operation, for design basis accidents (DBA) and for design extension accidents (DEA) and disposal. This is true for radiotoxic and chemically toxic aspects of the inventories. However, due to the innovative character of a multi-MW spallation source the formal licensing and authorisation process is not fully established in all European countries as, until now, large amounts of radioactivity have only occurred in conjunction with fissile materials. Although no fissile materials are present in ESS it is assumed that beyond design basis accidents will probably require a nuclear emergency plan due to the large radiotoxic inventory. For DBA national regulations apply.

Shielding

Shielding guidelines for ESS have been developed as a basis for the current design. A dose rate of less than 0.5 μSv/h, consistent with European regulations, has been adopted for all areas, where non-radiation workers may be present. For the accelerator, compressor rings and beam transport lines beam losses of 1 W/m have been assumed for the majority of the structure. However, in some zones greater losses must be considered.

For 'accidental' beam losses a method of analysis has been developed. Shielding requirements for the accelerator, transport lines, targets, beam-dumps and neutron instrument facilities are estimated on this analysis. A controlled entry area around the accelerator provides further protection. It was demonstrated that the shielding proposed for the protection of workers is also sufficient for radiation protection of the public. In the light of final disposal further studies on concrete and soil as accelerator shielding material were performed. This indicated that a slight increase in the concrete thickness from 1m to 1.5m would be beneficial for the prevention of soil activation and lead to a large reduction in final disposal quantities.

Activity Transport with Ground Water

The shielding of the ground near high beam loss points, such as targets and beam dumps, is mainly required as protection against water and soil contamination and depends strongly on the conditions at the chosen site. Studies performed indicate, that the dominant nuclides are H-3 and Na-22, because of their comparatively fast movement in many kinds of soil. A study of activation of soil and ground water and activity transport was undertaken. The Jülich site with its comparatively high ground water level was used as an example. However, the model under development is also applicable to other sites. The model is based on one developed originally for pesticide transport by ground water (PEGASE). It has been extended to include separate routines for the calculation of activation processes and nuclide transport. The main output from these studies is the maximum activity of the ground water at the Jülich site boundary, as a function of shielding conditions and site-specific parameters. Adequate shielding will be specified for the protection of soil and ground water and typical values will be incorporated in the design.

Target Radioactive Inventory

A comparison of calculations for SNS, JSNS and ESS of the radioactive inventory in a mercury target revealed excellent agreement of the total activity of short-lived[1] nuclides, but some differences for long- and very long-lived nuclides. Further study of these differences by SNS and ESS with support from PSI/SINQ and JSNS resolved most of these discrepancies. Radioactive inventories in the used core of a neutron beam research reactor and in ESS targets are of comparable size. But it must be noted, that the ESS mercury is never changed, whereas research reactors use many fuel cores in their lifetime. Nuclides in the ESS targets with $t_{1/2} > 500$ y include Hg-194 ($t_{1/2}$ = 520 y) and Ho-163 ($t_{1/2}$ = 4700 y). Hg-194 with its short-lived daughter nuclide Au-194 shows very pronounced radio toxicity. Safety problems associated with the radioactive inventory can be solved.

Thermal Hydraulic Safety

Thermal hydraulic safety examinations dealt with the adaptation of a CFD code to hydrogen release from a moderator and the formation of inflammable gas combinations. Parameter calculations indicate that a suitable CFD-model is now available. For the worst-case conditions the maximum tolerable delay time for proton beam shutdown following the loss of mercury flow is estimated as 5 – 6 s, which requires a sophisticated target control system.

Decommissioning, Waste Storage and Disposal

The absence of ultra long-lived actinide elements in ESS facilitates final waste handling and disposal compared with fission systems. However, there are other long- up to ultra long-lived nuclides in ESS (e.g. Hg-194, Ho-163, Gs-150, Dy-154). As liquid radioactive mercury is not suitable for long-term storage or final waste disposal solidification is required prior to final disposal. R&D is required to find the best way of achieving this. The 30 t of mercury in the ESS targets may exceed the chemo toxicity limits on the current license of a German nuclear waste repository. This and the disposal of the total tritium activity in ESS targets, which was estimated from SNS data, require more detailed studies, but no insurmountable problems are foreseen.

Preliminary Safety Analyses

A preliminary, probability oriented safety analysis of ESS has been made, based on similar analyses for SNS. A basis for the final ESS safety assessment and for the documents required for the licensing procedure has been established. The design of ESS safety features can be, and has been, optimised from these safety analyses. The preliminary safety analysis indicates that:

- Accidents with significant consequences have extremely small occurrence frequencies.
- The ESS concept can fulfil the European licensing requirements. These requirements, in some cases, are more stringent than for SNS.
- With proper design, ESS will achieve a very high safety standard.

[1] The definition of 'short lived' respective 'long lived' nuclides follows the customs of nuclear safety research concerning corresponding halve life:

Short lived	*Medium lived*	*Long lived*	*Very long lived*	*Ultra long lived*
Days – weeks	Month – some years	10 y - some centuries	1000 – 10000 y	> 10000 y

Public Relations (PR) and Safety Concerns

A paper summarising safety related, scientific, technical, and legal and licensing, aspects was prepared and is included as section 8.9.2. PR experts will use this for the preparation of an ESS Safety Fact Sheet, which informs the public and politicians about ESS safety issues.

Outstanding Safety Assessments Prior to Construction

1. Full Probability Safety Analysis for ESS.
2. Safety analyses for external events.
3. Design studies for ESS safety systems.
4. Calculations of cooling water and air activation and on formation of chemical toxic gases in ESS buildings (Similar to ISIS as beam losses will be the same).
5. Activation studies of accelerator components (Similar to ISIS as beam losses will be the same)

Safety Assessments to be Continued

1. Extension of the national legal basis for ESS licensing (to be initiated by site proposers).
2. Experimental determinations of long-lived nuclides in SINQ proton irradiated mercury and corresponding examinations of the completeness of the nuclide vector.
3. Studies of chemistry, process technology and costing of solidification of the target mercury and on the selection of a suitable waste container.
4. Continuation of work on activation transport by ground water (within EU FP6 programs SAFERIB and EURISOL).
5. Realistic estimations of ESS accident consequences.
6. Conservation of general ESS safety related knowledge (by participation in ADS and EURISOL EU FP6 programmes).

8.1 INTRODUCTION

8.1.1 Background to ESS safety

As with any other large facility of its type, the ESS has to be designed for safe operation, commissioned safely, and operated safely. Although the ESS will be the largest spallation neutron source in the world, the important safety issues are - at least to a large extent - well known and understood. Extensive experience is available from the ISIS and SINQ spallation neutron sources already operating in the UK and Switzerland, and guidance is also available from the safety issues being addressed at the SNS spallation neutron source currently being constructed in the USA and the J-PARC facility under construction in Japan.

The radioactivity generated in the target, which is of comparable size to that of a neutron generating research reactor, and the radioactivity in other parts of ESS together with the chemical toxicity and volatility of mercury require, that considerable effort has to be spent in safety examinations and in preparation of the formal safety report to support the procedures required to obtain authorization to build and operate the ESS.

A key issue for a facility like the ESS is to meet very high safety standards. The ESS is planned to be licensable in an urban area for instance directly adjacent to a university campus. Accurate estimation of radiation levels from a detailed understanding of the facility is therefore a must. The acceptable (in terms of common regulatory regimes) radioactive releases in design basis accidents at ESS should be at least 2 orders of magnitude smaller than those, acceptable for SNS (which is largely due to SNS's location being far from inhabited areas).

In order to make sure that safety issues get proper attention when the ESS project is re-launched again the incorporation of safety and licensing activities should appear in the WBS structure at the same level as the other major tasks(see figure 9.3.3.1)
.
In order to guarantee a high ESS safety standard, the fundamental design principle of ESS requires, that during operation and in frequent abnormal events the conditions within the facility (temperatures in the target, pressures etc.) are sufficiently far from dangerous ones. Other measures undertaken to increase ESS safety are:

- Sufficient shielding for reduction of radiation from the facility
- Multiple enclosure of dangerous materials present in ESS ('defense in depth')
- Control of emissions and their reduction (filters etc.)
- Redundancy and diversity of active safety equipments
- Application of as many as possible passive and inherently safe features

Details of ESS safety design are described in the accelerator, target and ring chapters of this ESS update report and in [ESS, 2002].

Throughout the EU (as in most other countries) safety considerations are based on three different categories, distinguished mainly by their frequency of occurrence:

- Normal operation and frequent abnormal events or 'incidents' ($>10^{-2}$ y^{-1}),
- Design basis accidents, DBA (10^{-2} y^{-1}- $5{\cdot}10^{-6}$ y^{-1})[2],
- Design extension accidents, DEA, 'hypothetical' events ($<5{\cdot}10^{-6}$ y^{-1})[1].

Besides event sequences, which result from failure within the facility itself ('internal initiation'), accidents based on external events (natural hazards like earthquakes, induced accidents like gas explosions or aeroplane crashes etc.) have to be part of the safety analysis.

In estimating radiation doses the following paths must be considered:

- External irradiation:
    a) Direct radiation from the facility
    b) Submersion in any emitted nuclide cloud (cloud shine)
    c) Ground shine (from any nuclides deposited on the ground)
- Internal irradiation by incorporation:
    d) Inhalation of activity from any emitted nuclide cloud
    e) Ingestion of activity (nuclides possibly deposited on the ground or in surface water or incorporated via a food chain)

Doses b) – e) are, may be, induced by nuclide emissions in normal operation (H-3, C-11, Ar-41) or in accidents; however direct radiation may also contribute to some extent to ingestion doses via ground water and soil contamination. Only internal paths have to be taken into account for chemical toxicants (Hg). Dependant on their chemical status, incorporation via the skin has to be considered for mercury and tritium as well as their inhalation or ingestion.

For all toxicants 'early' and 'late' consequences must be distinguished. 'Early' consequences (e.g. acute radiation syndrome) and their degree are characterised by a threshold dose and by a dose-rate dependence on the threshold dose. For 'late' consequences (e.g. radiation induced cancer) no threshold dose and no dose-rate dependence exist, but a linear relationship between consequences and doses is assumed for conservatism.

Normal operation, frequent abnormal events (incidents) and *design basis accidents (DBA)* need good estimations of accident consequences. Analyses of *'hypothetical' events* are necessary for planning of emergency countermeasures and demand best estimate consequence assessments. Emergency countermeasure planning, for these very unlikely accidents, estimates areas outside the facility for potential evacuation, relocation, sheltering, foodstuff ban or iodine tablet distribution measures.

When considering hazards from the facility, the facility staff and the public must be treated slightly differently. During normal operation and frequent abnormal events there are different maximum annual radiation doses for both groups and even higher allowable annual doses for staff classified as radiation workers. For DBA only dose criteria for members of the public are applicable, whereas such accidents are accepted as an occupational risk for staff.

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[2] Looking at the worldwide progress of nuclear licensing over the last 20 years it is clear that the lower frequency limit of DBA is continuing to decrease with increasing safety demands. Currently it is not exactly fixed by regulations. In the future, a lower limit of 10^{-6}/y is expected.

Regulatory requirements during commissioning for a facility like ESS are different throughout Europe and this will result in some cost differences. In some countries a license to operate is required and in others (like UK) different mechanisms are used. For convenience the term 'licensing' is used in the rest of this paper to represent the regulatory process. It will be advisable to check the licensing requirements beforehand for all countries interested in proposing a site for ESS. This should preferably be carried out with the help of a national working group from each country proposing an ESS-site. Specific licensing requirements may even result from conditions on a proposed site (e.g. high probability of earthquakes or aircraft crash onto the plant, possibilities of ground water and soil contamination etc.).

Safety studies form part of the licensing procedure, but safety features must be optimised throughout the design stage. Similarly shielding should be further assessed in parallel with the design work as an optimised shielding design is essential for cost minimisation. Decommissioning and waste disposal must also be further considered during the design stage. Experience with nuclear power plants indicates, that omission of these aspects during plant design may lead to large decommissioning and waste disposal costs.

8.1.2 Structure of the ESS safety assessment

In order to establish a basis for ESS licensing and to support optimisation of the ESS safety design, the safety assessment was structured in four main areas. These are legal aspects, technical problems, safety analyses and public relation (PR).

Table 8.1.2.1: Structure of the ESS safety assessment

1.8 SAFETY and LICENSING

- Legal framework concerning ESS in different European countries
- *Selected safety related technical items:*
  - Shielding guidelines for ESS
  - Activity transport with ground water
  - Radiotoxic inventories
  - Thermohydraulic safety work
  - Decommissioning and waste disposal issues
- ESS safety analyses
- Safety work related to PR

More detailed safety studies that will be continued are listed along with safety related work which must be undertaken prior to ESS construction.

8.2 LEGAL FRAMEWORK FOR ESS LICENSING IN DIFFERENT EUROPEAN COUNTRIES

The aim of the legal framework for licensing ESS is limitation and definition of the risk to the public, the environment and the ESS staff. The main risks are associated with the substantial amounts of radiotoxic, chemically toxic and combustible materials present in ESS. Conventional licensing requirements also have to be fulfilled. Only the radiotoxic, chemically toxic and combustible or explosive aspects will be considered here as conventional licensing is not specific to ESS and is not expected to lead to any problems. Requirements for several European countries are studied.

8.2.1 Radiological aspects

A common base exists [EURATOM, 1996] for all EU countries for the protection of workers and public from the dangers of radioactivity. This directive limits the total annual dose to any member of the public to 1 mSv (assuming the continuous presence the person at the worst position of the fence around the facility). Additionally, fulfilment of the ALARA principle (doses have to remain As Low As Reasonably Achievable) is required. Figure 8.2.1.1 shows the main protection levels.

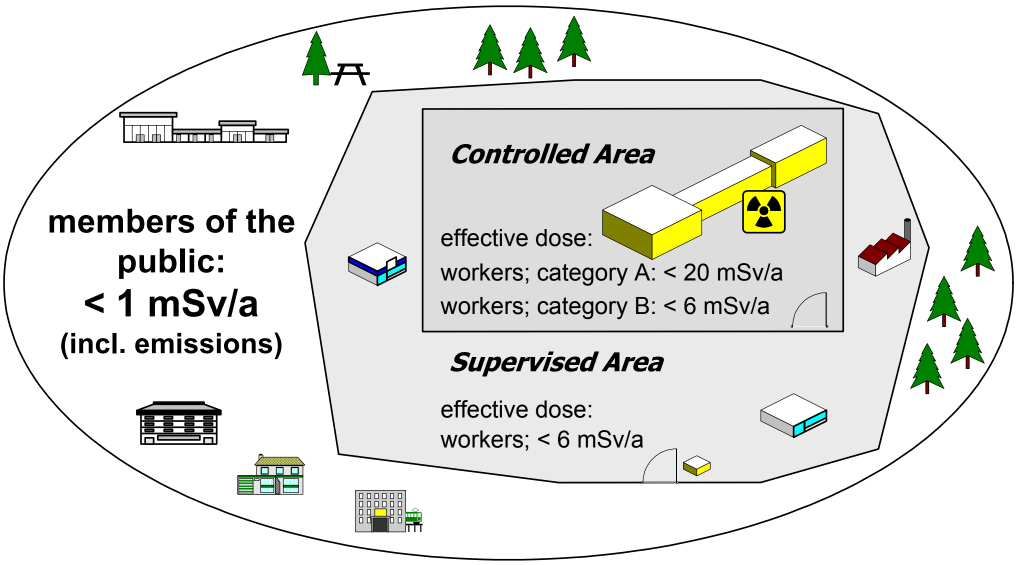


Figure 8.2.1.1: "1 mSv effective dose concept" of Council Directive 96/29/EURATOM

The formal licensing procedures differ in different EU countries. Dependant on the country-specific legislation a licensing procedure comparable to nuclear power plants may be required for a facility like ESS. In this case, the Environmental Impact Assessment Directive EU directive 85/337/EEC [EC, 1985][3] also has to be taken into account and this often leads to involvement of the public in the licensing process. Several countries (United Kingdom, Germany, Sweden) treat the ESS as a non-nuclear facility, whereas rules in other countries (France, Belgium, Italy[4]) require a nuclear licensing process for ESS. However, because a

[3] EU directive 85/337/EEC [EC, 1985] is an important part of EU environmental legislation. It requires Member States to carry out environmental impact assessments (EIA) on certain projects before they are permitted to proceed, in cases where it is believed that the projects are likely to have a significant impact on the environment.
[4]USA: In order to avoid classification as a nuclear facility already during initial operation, SNS will operate far below the nominal power within the initial operation phase [Freeman, 2003].

facility like ESS has never been licensed in Europe until now, there are still some uncertainties concerning the formal licensing process (see chapter 8.2.3). The effort required will depend on the local law which has to be applied at the site of the facility. The formal requirements for the radiological licensing process were compiled for each of the countries proposing a site (UK, Sweden, Germany and additionally France) in chapter 8.2.1of [ESS, 2002]. An updated version of this chapter is found in [Moormann, 2003]. Directive EURATOM 96/29 is mainly applicable to normal operation of nuclear related facilities. There are still substantial differences among different EU countries in the treatment of DBA during the licensing process. In contrast to that, it appears that safety analyses to be performed for planning of emergency countermeasures are similar for all countries. These are all based on guidelines of the International Committee on Radiation Protection [ICRP, 1993].

As a consequence each country that proposes a site for ESS must provide a (preliminary) safety report in accordance with the local requirements. This (preliminary) safety report will be based on the site independent safety work performed as a common ESS duty.

8.2.2 Conventional toxicity and combustible gases

Apart from the radioactivity contained in the target material, the chemical toxicity of mercury represents a hazard to both people and environment if accidentally released. Not only have large and fast releases to be considered, but also small, continuous releases and minor incidents. An undetected continuous small release of mercury affecting the staff is unlikely to occur because of the easy detection of radioactivity in the target material. It should be checked, however, whether or to what extent such an escape may remain undetected for a still un-irradiated target. Furthermore, incidental mercury intake by the staff in course of filling the target must be considered.

Due to environmental concerns numerous regulations have been created since the 1980s which have led to a marked decrease in the world-wide Hg consumption. Existing restrictions mainly refer to discharges of various bio-accumulative chemicals or emissions from coal-fired power plants, or to potential exposure at the work place. The rules for treating accidental mercury release (safety guidelines for chemical plants are applicable here) are different in different countries. A collection and comparison of these rules is found in chapter 8.2.2 of [ESS, 2002]. An updated version of this chapter which now also contains combustible gases, is documented as [Verfondern, 2003]. As well as for EU and European countries rules are described there for USA and Japan which are both developing a spallation source. This was done, because - in contrast to rules for the radiological licensing - the legal framework concerning toxic mercury did not yet part of the preliminary safety analyses for these spallation sources.

Concerning the EU the Seveso-II directive requires accident analyses for (non-radioactive) elemental mercury, if the total amount within a facility exceeds 50 t; this is not the case for ESS. Nevertheless, it is expected, that the aspect of conventional toxicity has to be considered in detail in accident analyses for the radiological licensing process. The same holds for formation of explosive mixtures [Verfondern, 2003] by failure of the moderator enclosure, which is a relevant initiating event for accidents with radioactive releases. Independently of potential radioactive releases some EU countries require studies of explosive mixtures for occupational safety reasons, if it cannot be excluded, that combustible gas mixtures may be formed in areas, where workers are present.

8.2.3 Concluding remarks on legal aspects

The general conclusion from these examinations of licensing requirements in different European countries to be drawn is, that main differences between rules of different countries belong to the radiological dose estimation part for design basis accidents (DBA), where independent national rules apply and where - for that reason - substantial differences are found, and - concerning the licensing procedure - whether the public has to be involved into the licensing process or not. For normal operation, guidelines and rules concerning radiation protection of the staff and of the public are very similar; they are all based on EURATOM-directive 96/29 [EURATOM, 1996]. Despite to differences in handling of DBA, a substantial part of the safety analyses also for accidents is probably independent of the particular site, i.e. large parts of the system analysis, of the source term estimation and of dose estimations for emergency planning. The legal basis for safety analyses to be performed for planning of emergency countermeasures is expected to be similar for all European countries. Differences are detected concerning design requirements for nuclear related facilities against a (military) airplane crash, which is necessary in some EU countries only. It is expected however, that these design requirements will be valid in future all over EU.

Due to the innovative character of a multi-MW spallation source in Europe, the formal licensing/authorisation process is not completely established in various European countries. This is caused by the fact, that up to now large amounts of radioactivity occurred only related to fissile materials, and accordingly, regulations concerning the formal licensing/authorisation process take the presence of fissile materials (which do not occur in ESS) as main criterion. This is particularly relevant concerning the necessity of an environmental impact assessment (EIA) including the involvement of the public – having the analogies of ESS and research reactors with respect to the amount of the radioactive inventory in mind, corresponding to the fact that an EIA with involvement of the public is recommended for research reactors. Beyond design basis accidents are treated on basis of the total amount of radioactivity in ESS, and therefore a nuclear emergency plan is probably required[5]. Urgent need for extension of regulations concerns official dose factors and clearance values for a large number of nuclides, which are characteristic for a mercury target. As a consequence of that and in order to avoid delays in ESS licensing/-authorisation, activities of the site proposers are advisable within the next years to examine, whether a sufficiently detailed legal basis for the formal licensing procedure exists in their respective country, and if not, to induce the creation of respective regulations. It has to be noted, that the above mentioned need for more detailed regulations concerning the formal licensing/authorisation procedure exists only for the target due to its radioactive inventory, whereas for the accelerator complete regulations are available.

With respect to the toxicity of inactive mercury, the EU-directive (Seveso-II) guideline [EU, 1996] has not to be taken into account, because the total amount of mercury in ESS is below a limit of 50 t. The occupational safety concerning mercury and the maximum tolerable drinking water contamination are similar for most countries. Except to UK, where a slightly different commissioning process has to be fulfilled, the toxicity of inactive mercury is handled within the radiological licensing procedure, where a complete picture of the risk of the facility has to be outlined; a separate licensing is not required in other countries.

[5]Following German regulations, severe accidents with catastrophic releases may occur only, if the total amount of radioactivity exceeds the clearance limits by more than a factor of 10^{10}. Concerning the ESS targets, this limit is already reached by its Gd-148 inventory. If massive releases cannot be excluded in the beyond DBA range, a nuclear emergency planning is required

8.3 SHIELDING GUIDELINES FOR ESS

8.3.1 Introduction

Shielding of a spallation neutron source is of major safety relevance, because it has to guarantee safe working conditions within the facility and negligible radiation burden for the public and environment outside of the facility. However, shielding against fast neutrons is costly during construction, in decommissioning and final waste treatment. Accordingly, shielding should be optimised, which means, that reliable guidelines for the shielding design have to be determined. This chapter describes such general guidelines for design of the shielding of key components in ESS [Berkvens, 2003].

Concerning the ESS shielding one guideline is given by the need for a fenced ESS area with access control. An open, unfenced ESS area is not desirable, because in this case the more restrictive dose limits for the public have to be applied to the shielding design. Shielding guidelines for the ESS have to take into account the radiation released by accelerator, ring and proton beam lines due to normal (continuously occurring and unavoidable) beam losses, and the radiation induced by local losses, which consist of a short term loss of the whole beam or parts of it into a certain position ('accidental' or point losses) by failure in the equipment. The dominant type of radiation affecting shielding is by fast neutrons. The consequences of this radiation have to be considered at the shielding surface (direct radiation) near working personnel and outside the ESS fence, where the public could be affected. In addition, activation processes of surface and ground water and of soil, interacting with water, have to be taken into account.

In this chapter, shielding guidelines will be evaluated on basis of regulatory limits for workers. Then it will be demonstrated that these guidelines are also sufficient for protection of the public. In a final section, selection of shielding material and its influence on the decommissioning/final waste treatment will be discussed.

A detailed study to shielding requirements in ESS accelerator and ring is given by [Berkvens, 2002]. Details of shielding calculations performed for the ESS accelerator and the rings, using the Moyer model, are found in chapter 8.3.1 of [ESS, 2002]; these shielding calculations are based on the maximum tolerable dose rate outside the shielding, but do not yet consider waste items. Shielding calculations for the target are described in chapter 4, and in the literature cited there.

8.3.2 Legal background

EURATOM directive 96/29 [EURATOM, 1996] limits the annual individual effective dose for the public by nuclear related facilities to less than 1 mSv (see figure 8.2.1.1). The same holds for workers, but in this case an annual working time of 2000 h has to be assumed (instead of continuous presence at the fence, i.e. 8760 h/y, as for the public), which leads to an upper limit for the dose rate in areas, where (non radiation) workers may stay, of 0.5 µSv/h. Within the facility only incorporation by inhalation of emitted radionuclides has to be considered for 2000 h/y (together with groundshine by depleted nuclides). For the public the incorporation by ingestion has also to be taken into account within the total annual limit of 1 mSv; radioactive burden from activation of ground/surface water as mentioned above, have also to be taken into account, too.

For radiation workers the annual dose limits are 6 mSv (class B) and 20 mSv (class A) as seen from figure 8.2.1.1). The EURATOM directive 96/29 contains the ALARA principle, which requires doses to remain below the given limits, if that is possible by practicable cost effective measures.

A proof of compliance with the EU regulations has to be given within the licensing/authorization process for the ESS. All events, occurring with frequencies greater than 10^{-2} y^{-1} have to be assumed to occur within the lifetime of the facility. The consequences of these events have to be considered for both, workers and public. In contrast, events with frequencies of between 10^{-2} y^{-1} and $5\cdot10^{-6}$ y^{-1} belong to design basis accidents (DBA), whose consequences have to be considered for the public only. Regulations concerning DBA are different in various EU countries, but the ALARA principle is valid in all European countries.

8.3.3 Shielding requirements for accelerator and compressor ring

A comprehensive overview on shielding of high-energy accelerators is found in [Sullivan, 1992], [Thomas 1988]. In the first step, shielding guidelines based on dose limits for workers will be evaluated. Normal beam losses and local ('accidental') losses have to be treated separately and then added together.

Normal beam losses
For normal losses in high-energy proton accelerators a value of 1 W/m is often assumed, based on experience in various accelerators (Los Alamos, CERN, ISIS), showing normal losses less than or equal to 1 W/m in stable operation and on computer simulations for SNS and J-PARC accelerators. Recent examinations for the SNS accelerator [Holtkamp 2002] indicate however, that in limited transition zones between different accelerator components (i.e. normal – superconducting magnets) the beam losses are expected to be larger by in average up to a factor of about 10. Accordingly, we propose for normal losses in stable operation phases a value of 1 W/m except for these transition zones, where an extra shielding is required in order to reduce the fast neutron irradiation by an additional factor of 10. These values are not suitable for the start-up phase at the beginning of accelerator operation (see separate discussion in a separate subchapter) and for beam tuning periods. For beam tuning we expect – based on accelerator experience – a continuous loss about a factor of 10 larger than in normal operation; the duration of beam tuning periods is assumed to be, 10 % of the total operation time. Accordingly, the shielding against normal losses has to be based on an effective normal loss of 2 W/m respectively 20 W/m in transition zones.

Local ('accidental') losses
Local losses are known to occur due to several causes in accelerators, but a full power beam loss was observed only once during 15 y of ISIS accelerator operation to date. Nevertheless, there is insufficient statistical data concerning local losses, which adequately considers the different reasons for these losses. Therefore, a conservative treatment of losses is required. The first conservative assumption to be taken into account deals with the beam shutdown system, which has to stop the operation of the ion sources in case of beam losses: Experience with operating accelerators indicates that these systems are highly reliable and the assumption of beam shutdown after 5 pulses in our studies is conservative. The same holds for the assumptions on how often full power losses occur and that the mean time between failures is 24 h, which could lead to beam losses. A mean time between failures of 24 h is in line with experience, but in practice, only a very small fraction of failures leads to substantial beam loss. Further on, we assume that the localization of a beam loss is stochastically distributed along the accelerator length and that a reference worker is present for 2000 h/y near the

shielding surface, with a stochastic distribution of his location. For the radiation profile of the beam loss at the shielding surface we assume a window of 10 m × 10 m with equal dose rate, which is an overestimation by a factor of between 2 and 2.5 compared with the calculated Gaussian profile. Assuming a total length of the linac of 500 m, the probability p_1 that the reference worker (working 2000 h of 8760 h/y) will be exposed to a certain beam loss is

$$p_1 = 10/500 \cdot 2000/8760 = 0.0048$$

The probability, that the reference worker will be exposed to n full beam losses per year, if N full beam losses occur, is given by:

$$P(n,N) = \frac{N!}{n! \cdot (N-n)!} p_1^n \cdot (1-p_1)^{N-n}$$

Finally, the probability that the reference worker will be exposed to n full beam losses per year, if the average number of full beam losses of 5 pulses is N_{av}, is calculated to:

$$P(n,N_{av}) = \sum_{N=n}^{N=2N_{av}} P(N,N_{av}) \cdot P(n,N)$$

For a mean time between failures of 24 h, $P(n,N_{av})$ is presented in fig. 8.3.3.1 (9 month/y of operation): Within the probability limit of 10^{-2} y^{-1} the maximum number of beam losses the reference worker is exposed to is expected to be 5. The corresponding 25 pulses result on the surface of shielding, designed on base of normal losses as described before, is an additional annual dose of 0.11 mSv, which requires a small extra shielding of 0.1 m of soil (total shielding: 6 – 8 m, mainly soil and to a minor part concrete [ESS, 2002], calculated using the Moyer model [McCaslin, 1987]). Accordingly, even in the case of a conservative treatment as here, the local losses play a minor role compared to normal losses.

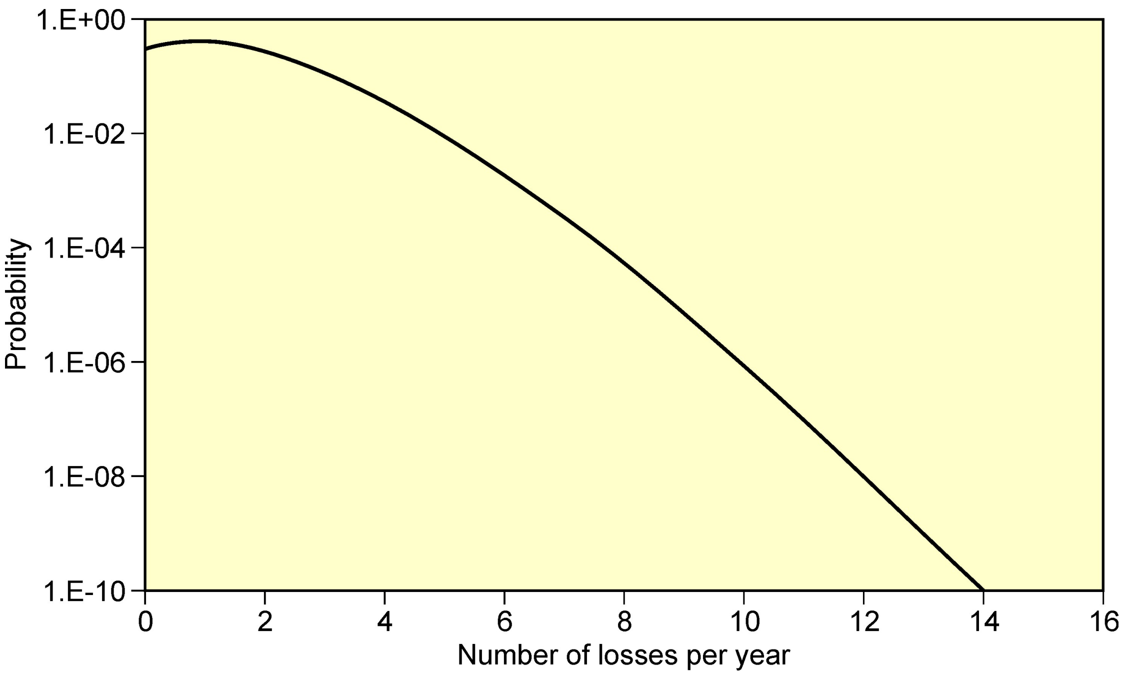
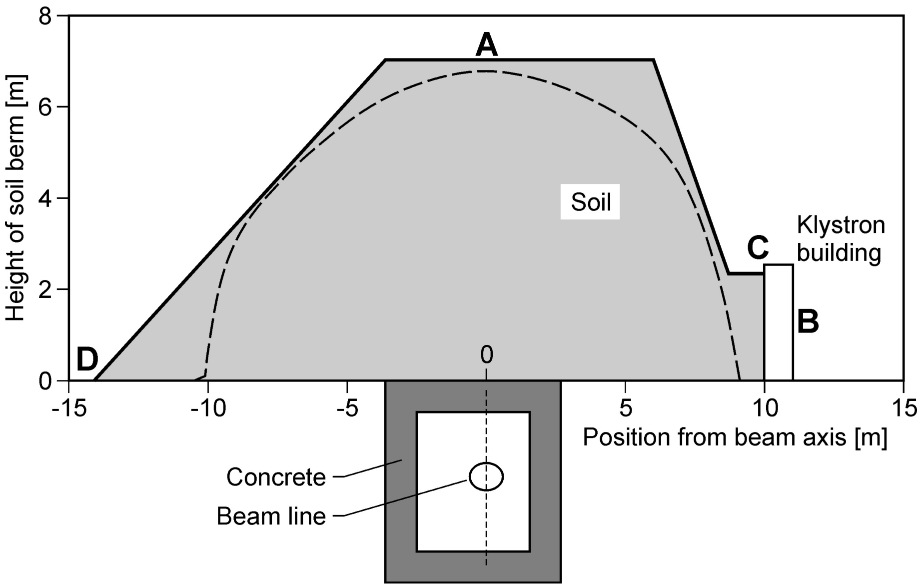


Figure 8.3.3.1 Probability distribution P(n,270) for the number of full beam losses seen by a single worker per year for a mean time between failure of 24 h

Start-up phase of accelerator/compressor rings

The start-up phase of an accelerator requires special attention, because losses are substantially larger than in stable operation, as indicated by experience. However, this is compensated by the fact that the effective power during start-up is far below maximum values. For the ESS accelerator it is assumed, that during start-up only one (of two) sources is operating, that the pulse repetition frequency is 1 Hz instead of 50 Hz, and that the peak current and the peak length are both 10 % of their nominal values. This yields to a total reduction of beam power by a factor of 10^4 down to 1 kW. For the compressor rings a reduction factor of 500 down to 10 kW is assumed by a repetition frequency of 1 Hz and a peak current of 10 % of the nominal value. Fig. 8.3.3.2 contains profiles of the accelerator with a minimum shielding as calculated on basis of normal and local losses in the subchapters before and of the design shielding, as required from statics examinations. The resulting dose rates in certain positions are also given in a table for a full, continuous localized beam loss. Fig. 8.3.3.3 contains the same for the ring. As the dose rates indicate, a full, localized beam loss will lead in certain positions (A and C for the accelerator, A for the rings) to a transgression of the dose rate limit for continuous radiation of 0.5 µSv/h. Based on the ESS party line it has to be assumed, that the start-up period is completed after about 6 months, followed by 3 months of normal operation in the respective year. Thus, at least the areas around positions A in fig. 8.3.3.2 and 8.3.3.3 should be assigned to radiation supervised areas[6] for the limited time duration of the start-up.

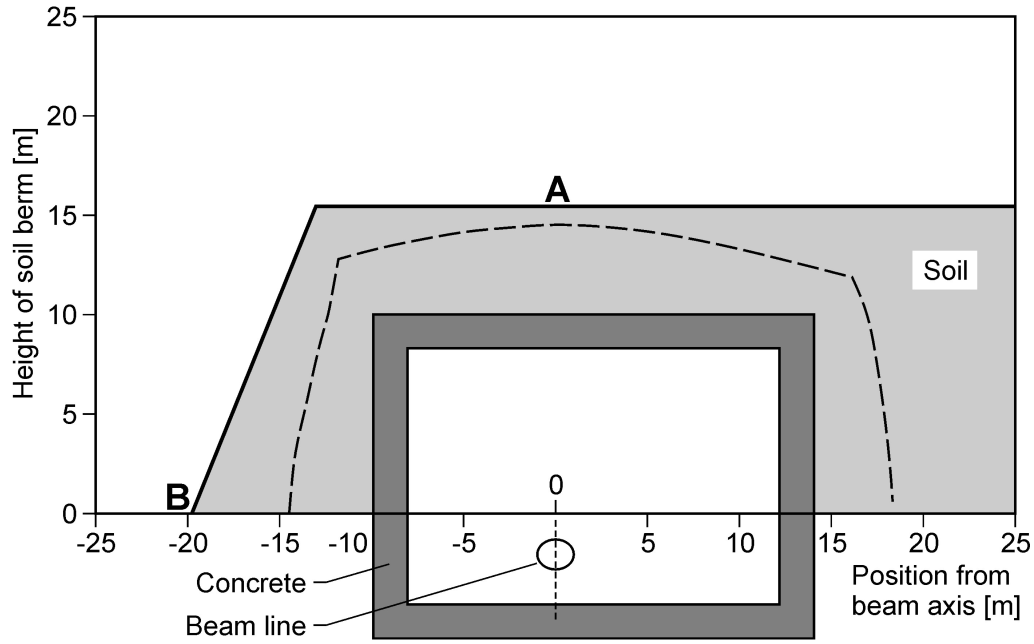
Due to the innovative character of the ESS accelerator it cannot be completely excluded, that an efficient start-up is not possible with the power reduction factor given above. Following ISIS experience only dose rates by up to a factor of 50 higher than given in fig. 8.3.3.2 will allow for an efficient, fast start-up. Nevertheless, this does not lead to serious problems, because assignment of the accelerator and a limited zone around into a (fenced) radiation controlled area only for the start-up phase will be foreseen in the general design of the facility. Additional shielding is not required, because large dose rates (< 2 mSv/h) are expected even in this case only on top of the accelerator, which is easier to control from the point of view of radiation protection.



Position	A	B	C	D
Dose rate [µSv/h]	37	0.8	2.74	0.92

Figure 8.3.3.2: Minimum soil profile (broken line) and real soil profile (solid curve) for the high-energy part of the linac; maximum dose rates are given in the table for start-up periods with 1 kW continuous losses

[6] For most European countries holds (see fig. 8.2.1.1): Radiation supervised areas have to be established for dose rates of 1 – 6 mSv/y, radiation controlled areas for > 6 mSv/y (radiation workers with 2000 h/y working time have to be considered). Closed areas have to be declared, where dose rates of > 3 mSv/h cannot be excluded.



Position	Dose rate [µSv/h]
A	63
B	0.6

Figure 8.3.3.3: Minimum soil profile (broken line) and real soil profile (solid curve) for the ring; maximum dose rates are given in the table for start-up periods with a continuous 10 kW loss

8.3.4 Shielding requirements for targets, beam dumps and instrument area

Target

The target shielding consists mainly of an inner iron layer of several meters and a smaller concrete layer. Design calculations for the target shielding are relatively complex due to the large number of penetrations; usually Monte Carlo methods have to be applied [Koprivnikar, 2001]. The target building has to be a radiation controlled area and the major requirement for the target shielding is that outside of the target building a dose rate limit of 0.5 µSv/h will be met. This restricts the dose rates inside the building but outside the shielding block to values, which allow reasonable conditions for radiation workers.

Special considerations are necessary for the proton beam line within the target building, because of space limitations and the competing requirement of accessibility to areas around the proton beam line. Due to its good shielding properties for fast neutrons, iron (clad with concrete) has to be used for proton beam shielding here. Reasonable design goals for this component are:

- Normal loss assumptions as described above for the accelerator/rings, except for the proton beam window and the collimator area. For the latter two components, losses of 3% of the beam have to be considered in shielding design.
- For local losses a full beam loss of 5 pulses must not lead to higher doses than 1 mSv in all areas surrounding the beam line, where radiation workers may be present. With this requirement, the dose (which will be monitored) a radiation worker may receive from a local loss remains more than an order of magnitude below the annual limit.

Considering the assumptions on mean time between failures, presented in 8.3.3, an approach to the annual dose limit can easily be avoided.

Additional shielding requirements in the target building would be needed for ancillary plant, handling cells, ventilation plant and water circuits, as ISIS experience shows. For example, irradiation of water by fast neutrons generates neutron and gamma emitting nuclides, which may be transported by flow outside of a sufficiently shielded zone.

Beam dumps
Beam dump shielding has to guarantee an annual dose less than 1 mSv/y outside their buildings. Due to the intermittent operation of beam dumps, a maximum dose rate of 3 µSv/h may be taken as design goal for shielding, provided, the beam dump building is situated within a radiation supervised area.

Instrumental area
Outside of the instrumental buildings and outside of the neutron beam line shielding, the maximum dose rate of 0.5 µSv/h has to be obeyed. Although an easy accessibility is desirable for the instrumental complex, where even short term scientific visitors are frequently expected, the experience at ISIS indicates clearly, that the instruments itself cannot be taken out of a radiation controlled area. However, general areas in the experimental hall and the instrumental offices should be freely accessibly in order to facilitate the experimental research.

8.3.5 Shielding for protection of the public - Sky shine and direct radiation

A proof is required, that the shielding determined for protection of workers is also sufficient for the public. Here, sky shine (long range neutron irradiation influenced by air scattering) has to be considered in addition to the direct radiation [Sullivan, 1992]. Calculation methods described in [Sullivan, 1992] are used. As a first, conservative approach all neutron sources in the ESS were taken together, assuming, that they are in a distance of 300 m from a reference person. A distance of 300 m is the smallest distance between a major ESS neutron source (target, accelerator, ring) and the site boundary. Assuming a continuous direct radiation of 0.5 µSv/h on all shielding surfaces within the site, the sky shine contribution is 4.7 nSv/h and the direct radiation is 0.35 nSv/h.

For start-up with power reductions of 10^4 for the accelerator and 500 for the rings, the ring dominates with a sky shine dose rate of 90 nSv/h, whereas direct radiation and contributions to sky shine from accelerator and other components remain in the same range as for normal operation. For the first year of operation, assuming 6 month of beam set-up and 3 month of normal operation, a total annual dose of 0.4 mSv results for the reference person, whereas in years with normal operation the annual dose remains below 0.1 mSv. These values are sufficiently below the total dose limit of 1 mSv, provided the contributions from emissions remain small [Berkvens, 2002].

If this simplified calculation is performed for start-up phases with losses in accelerator by a factor of 50 larger (s. subchapter 'start-up-phase…' in 8.3.3), the sky shine contribution of the accelerator becomes significant and the total dose for the first year doubles to 0.8 mSv, which is probably not far enough from the limit. However, a more detailed calculation, considering the real distance distribution of emissions to the reference person, reduces the total sky shine for the first year and conservative loss assumptions for the accelerator during start-up to annual doses less than 0.5 mSv, which seems acceptable.

Altogether, the shielding guidelines presented in 8.3.3 are sufficient concerning sky shine and direct radiation.

8.3.6 Shielding materials in relation to decommissioning/waste disposal

There are huge amounts of shielding material required in a facility such as the ESS (>10^5 m^3), part of that will be substantially activated, which may lead to additional effort to be spent in final waste treatment and disposal [EC, 1999]. Thus shielding materials should be considered with respect to their behaviour during final waste treatment and disposal. With regard to volumes, mainly concrete and soil should be examined, although the comparatively smaller volumes of iron cannot be completely neglected. Because iron shielding is essential in the target area, the main concern has to be the selection of iron qualities with impurity levels, which do not lead to a strong long-term activation. The same criterion is also relevant for concrete and soil, but in addition, another aspect has to be discussed: To some extent soil can be replaced by concrete (and vice versa); this is relevant due to the fact, that concrete (less leachable, less erosion) is much easier to handle as radioactive waste than radioactive soil. Consequently, soil to be treated as radioactive waste should be avoided as much as possible.

There are substantial differences in various European countries concerning regulation threshold values, which require waste to be handled as radioactive. In France all contaminated material having a radiation level higher than the natural environment has to be treated as radioactive waste. Other countries define thresholds. In the UK a free release is possible for all nuclide activities of less than 0.4 Bq/g. In Sweden, a material may not to be treated as radioactive waste, if the total dose resulting from this waste is less than 10 µSv/y for a person of the public. The latter criterion is also the basis of the German regulations [StrlSchV, 2001], which define individual thresholds for each nuclide. For selected nuclides (having the highest activities in the particular soil) typical activity values in soil are presented in table 8.3.6.1. These are calculated for the ESS accelerator in the soil adjacent to the concrete shielding. A concrete thickness of 0.6 m and soil as found at CERN/Geneva were taken into account[7], the activity is given for a time after shut down of 10 y after 30 y of full operation [Berkvens, 2002a]. German thresholds are also given in table 8.3.6.1. If various nuclides are present, the thresholds for individual nuclides decline, as described in [StrlSchV, 2001]. It becomes obvious from table 8.3.6.1, that the thresholds for non-radioactive waste are exceeded for several nuclides (Co-60, Eu-152). H-3 is however not a problem. Due to the large amount of waste, disposal at landfills as normal waste is probably not allowed. Additional activation calculations indicated that a concrete thickness of 1.5 m is sufficient for the respective soil to avoid a classification as radioactive waste under the German regulations. This increase of concrete thickness imposes a cost increase of about 15 M€. Because of the differences in regulations and site conditions in various countries (for France, a substantial larger increase of concrete thickness is required in order to avoid soil rated as radioactive waste), a final decision about the concrete thickness depends on site selection. Nevertheless, an increased concrete thickness seems desirable.

[7] Soil at CERN was selected for these estimations, because at that time a complete activation study was available only for this kind of this origin.

Table 8.3.6.1: Specific activities (Bq/g) in soil adjacent to the inner concrete shielding of the ESS accelerator (10 y after ESS shut-down, thickness of the inner concrete shielding: 0.6 m) and corresponding German thresholds (Bq/g) for waste classification

Nuclide	Maximum Specific Activity [Bq/g] in ESS soil	German Limit of Specific Activity [Bq/g] for		
		Free handling	Normal waste*	Radioactive waste
H-3	1.5	< 60	< 10^3*	> 60** (> 10^3*)
Na-22	0.08	< 0.1	< 4*	> 0.1** (> 4*)
Co-60	0.2	< 0.09	< 4*	> 0.09** (> 4*)
Eu-152	2	< 0.2	< 8*	> 0.2** (> 8*)

*(*Applicable only to a waste amount < 1000 t/a. For > 1000 t/a treatment as normal waste is not permitted, i.e. transgression of free handling limits leads to radioactive waste **Amount >1000 t/a)*

8.3.7 Conclusions

Even for innovative large-scale spallation sources such as the ESS there is a sufficient basis for shielding guidelines. Besides the detailed design work, particularly for the target, future work is required on site-specific activation of soil/groundwater and transport of contaminated groundwater. In addition, selection of shielding materials for the inner shielding of the accelerator and ring in the light of final waste disposal remains an open point, which is partly due to different regulations in various European countries. Activation of inner components of accelerator and ring and suitable material selection for reduction of this activation, which is a related question but not discussed in this chapter, remains to be examined in detail.

8.4 ACTIVITY TRANSPORT WITH GROUNDWATER

8.4.1 Introduction

The operation of accelerators and related targets can lead to an increase of radionuclide concentrations in the groundwater due to activation of soil and groundwater mainly by fast neutrons. This increased radionuclide concentrations are related to specific circumstances, e.g. energy of the beam, insufficient shielding, washing out of radionuclides to groundwater, depth of water table etc.

The question of sufficient shielding is directly linked to the question of the costs of this shielding. An activation of soil and groundwater can be acceptable if regulatory limits are met or, in general, if the public and the environment are protected against any harmful radiation. In detail this means, that the annual dose limit of 1 mSv/y is met for a person of the public living 365 days per year next to the ESS "fence", exclusively using drinking water from local sources (e.g. groundwater)[8]. As outlined before, other radioactive burden (by food, inhalation, direct radiation) has to be considered in the annual dose of 1 mSv/y.

Ignoring these questions during the planning of a new accelerator or the upgrading of an existing facility can cause a major delay during planning or even lead to the licensing procedure being halted due to critical discussions with the licensing authorities or the expression of environmental concerns by the public with regard to the project. These expressing of concerns can be part of an authorization procedure. Accordingly, it is advisable to examine the contamination of ground water already in an early stage of ESS safety examinations. For that, a computer model is under development, which allows for determination of the activity concentration at the ESS fence. The code will be applied at first to the proposed ESS site of Jülich, but it will be formulated in a flexible manner: Exchanging site specific input data, the computer model will be applicable to all proposed ESS sites. A more detailed description of this issue is found in [Heuel-Fabianek, 2003]. It should be noted, that this work is still in an initial stage but will be continued within EU FP6.

8.4.2 Approach

Radionuclides in groundwater on accelerator (and target) sites are usually activation products due to fast neutrons, which have a comparatively large penetration depth within most materials. For modelling of radionuclide transport in the groundwater it is necessary to determine the affected area and the production rate of the relevant radionuclide. Factors influencing the kind of radionuclides produced and their production rate are:

- ✓ the energy of the beam,
- ✓ the composition of the irradiated material (concrete, soil, groundwater etc.),
- ✓ the shielding of the beam.

The nuclide production cross section and the chemical composition of soil / groundwater for the Jülich site are taken from [Probst, 1992]. Using space dependent neutron spectra based on Monte Carlo calculations performed for spallation source facilities [Koprivnikar, 2002] the desired production rate of radionuclides is obtained for Jülich site conditions [Schaal, 2003]. The calculated radionuclide production rate due to activation is one key input parameter of the computer model. In combination with other parameters (e.g. hydraulic conductivity,

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[8] *In Germany, a dose limit of 0.3 mSv/y exists for incorporation by drinking water*

groundwater level, partition coefficient of the radionuclide between mobile and stationary phase depending on soil / sediment / pH) and an assumption of a (worst case) scenario the computer model will calculate concentrations of radionuclides in the groundwater downstream of the site. Washout of activated soil by precipitation and transfer of the dissolved nuclides into the ground water has to be considered, too. Table 8.4.2.1 contains the most relevant nuclides concerning activity transport in ground water on the Jülich site; the examinations are still limited to mass numbers up to 59 and have to be extended to larger masses: As seen for CERN-soil (see 8.3.6), nuclides with larger masses may be formed, too. Very short-lived nuclides are not considered in table 8.4.2.1.

Table 8.4.2.1: Nuclides relevant for estimations on ground water contamination on Jülich site

Nuclide	Half-life	Production cross section in ground water (thermal neutrons)	Production cross section in ground water (fast neutrons)	Production cross section in Jülich soil (thermal neutrons)	Production cross section in Jülich soil (fast neutrons)	Retention on Jülich soil*
H-3	12.3 y	Negligible	Large	Negligible	Large	Small
Be-7	53 d	Negligible	Large	Negligible	Large	Large
Na-22	2.6 y	Negligible	Small	Negligible	Large	Small
P-32	14.3 d	Small	Small	Small	Medium	Medium
Cl-36	310000 y	Medium	Negligible	Medium	Negligible	Small
Fe-55	2.6 y	Very small	Negligible	Very small	Medium	Large

**retention means the conservatively estimated factor of the flow rate reduction of the particular nuclide, compared with the ground water flow rate: Small = < 10, Medium = 10 – 100, Large = > 100*

According to the Radiation Protection Ordinance (see sections 8.2 and 8.4.1) the habits of the population and the burden pathways are clearly defined. Combination of the calculated radionuclide concentrations and this definition of pathways and habits results in a potential radioactive intake by members of the public. If this potential radioactive intake needs to be calculated more precise the parameters and scenarios have to be checked and the assumption and bandwidths of worst case scenarios may also be narrowed in order to become more realistic without however leaving the conservative approach. This iterative process has to be carried out during the further process of planning the ESS facility.

8.4.3 Computer model

The prediction of radionuclide transport in groundwater is calculated with the TRACE/PAR-TRACE code, a model which was previously adapted to site-specific conditions as part of an ongoing research project related to the migration of pesticides (PEGASE) by FZJ/ICG-IV. TRACE [Vereecken, 1994] calculates the 3-dimensional unsaturated/saturated water flow in porous media. The code has already been applied to several ground water flow problems [Herbst, 2003]. For reactive solute transport the PARTRACE code is used. The transport is calculated with a 3-dimensional particle tracking approach based on finite elements, which allows linear and non-linear sorption, microbial degradation and radioactive decay to be taken into account. This code is also a scalable model system and has already been applied to several transport problems [Englert, 2003]. A file containing flow velocities and water contents links the water flow and the solute transport model.

8.4.4 The ESS Case Study

Exemplary of the Jülich site [Heuel-Fabianek, 2003a], a potential location for the planned European Spallation Source, the application of the model will be pointed out: The site is located in a flat area with an elevation ranging from 89 to 94 m above sea level. The whole area dips slightly to the SE. The geological situation of the site characterized by tectonic subsidence with strain-induced active NW-SE fracture zones and smaller translational faults is well known. The site is bordered by two faults and situated on a small fault block ("*Jülicher Zwischenscholle*"), which was formed by trench structures during tectonic settlements. The current tectonic settlement of the area is about 0.1 – 1 mm/y. The whole area is filled with tertiary and quaternary sediments that consist of inter-bedded sand and gravel layers with silt and clay layers, partly containing brown coal (lignite) seams. The clay layers are aquicludes that separate the water-bearing sand and gravel layers in several aquifers. Due to the thickness and properties of the first aquifer and aquiclude it is necessary to consider the site down to a depth of the first 30 m to investigate the hydrogeological conditions relevant for the questions of radionuclide migration by activation. The upper aquifer consists of sand and gravel layers from the main river terraces of the Rhine and Maas rivers. It stays in direct contact to surface water and is not under pressure. At a depth of 15 to 35 m the aquifer is limited to the Reuver clay layer with a very low water permeability of $k_f = 10^{-10}$ m/s. Therefore it is the most effective approach to consider only the upper aquifer related to the potential activation of the groundwater.

The groundwater flows to the NE at a velocity of ~ 1m/day. The ground water level varies between 1 – 7 m distance to surface. Usually the aquifer is covered by a loamy soil, which has developed on the loess. Based on an existing and already implemented groundwater model at the site ("PEGASE model"), the task-specific conditions have to be specified. In the case of the planned ESS accelerator the mesh had to be modified: Due to the hydrogeological and input conditions the grid of the model covers 1.6 km^2 with an extension of 800 m in the NE direction (X) and 2000 m in the NW direction (Y). The meshes of the grid are 100 m wide in the X- and Y direction, and in the Z direction (depth) a distance of 0.5 m is selected. Considering this orientation of the accelerator, the mesh is narrowed in the potential input area. For the modelling, the orientation of the accelerator is assumed to be parallel to the grid (and to the groundwater flow) in order to calculate the "worst case". Thus the content of radionuclides in the groundwater by activation will continuously increase while passing the accelerator.

Furthermore, for the calculation of the "worst case" it is assumed that the accelerator is built directly on top of the water-saturated zone of the uppermost aquifer. Therefore a reduction of the input of radionuclides into the groundwater by sorption processes in the soil (loess loam) in the unsaturated zone can be ignored. With these assumptions, the highest rates and fastest transport of radionuclides are taken (conservatively) into consideration.

First calculations indicate that H-3 and Na-22 are the most relevant nuclides concerning ground water contamination. More detailed results are expected for the end of 2004 (SAFERIB, EU FP6).

8.5 RADIOTOXIC INVENTORY IN THE ESS TARGET

8.5.1 Radioactive inventories

A good knowledge of the radioactive inventory within of the ESS targets and of its radiotoxicity is of major relevance for safety analyses. Detailed data about the inventories of about 1000 nuclides are available for the SNS target [SNS, 2000]; for certain relevant nuclides in JSNS independent inventory calculations were performed [Maekawa, 2002]. ESS inventory calculations are found in [ESS, 2002].

Comparing the before mentioned total inventory calculations and the calculations for selected relevant nuclides, the following results [Moormann, 2002]:

- There is an excellent agreement for the total activity of short-lived nuclides: The total inventories of all calculations are effectively identical up to a decay time of about 1 y.
- There are no large discrepancies for the inventories of individual short-lived nuclides, too.
- Concerning long-lived nuclides, pronounced discrepancies up to several orders of magnitude between SNS and ESS calculations were detected for the total activities (> 1 y decay time) and individual nuclides: These discrepancies are mainly due to Ho-163 ($t_{½}$ = 4750 y) and Hg-194 ($t_{½}$ = 520 y), whose inventories were calculated to much higher values at SNS than at ESS[9].

Because Hg-194 is a very radiotoxic nuclide[10] [11] (see table 8.5.2.1 below) and Ho-163 (despite of its probably low radiotoxicity) is of particular relevance for long term waste studies, these discrepancies were studied in collaboration[12] with SNS and JSNS, and additional calculations were performed [Filges, 2003]. The following conclusion can be drawn from these examinations:

- The former inventory calculations of ESS underrated heavily the inventories for both, Hg-194, Ho-163 and of other long and very long lived nuclides. Accordingly, some prior estimations of radiotoxicities and safety relevance of ESS are too optimistic.
- The SNS data [SNS, 2000] for Hg-194 are probably slightly conservative by a factor of < 2, but the Ho-163 concentrations were overestimated by about a factor of 80 (realistic value for a 5 MW target: 1900 GBq).

Taking that into account, the total radioactive inventory in one single ESS target after 40 y operation with 5000 h/y is presented in fig. 8.5.1.1, compared with the inventory of a fission research reactor core of 20 MW thermal power. The data for ESS are based on SNS-calculations (except for Hg-194 and Ho-163, where the before mentioned new calculations are considered); this was done in order to remain conservative: The new ESS [Filges, 2003]

[9] Concerning tritium in the target, calculations for ESS lead to substantially smaller inventories than SNS calculations; because of the minor radiological relevance of tritium in general (see 8.5.2), and because the tritium amount in the cooling water is more important than the target inventory, because it is easier released and its status (HTO=tritiated water) is completely the most radiotoxic one, this discrepancy was not examined in detail up to now. However, because of the relevance of tritium for final waste disposal (see 8.7.2) additional examinations are required, which also include possibility of formation of HgH_2.

[10] The γ-energy of Au-194, the short-lived daughter nuclide of Hg-194, is substantially larger than that of Cs-137, a nuclide comparable concerning burden pathways to Hg-194 and well known as highly relevant for fission systems. Furthermore, the halve life of Hg-194 is more than a factor of 15 larger than that of Cs-137

[11] For copper beam dumps in ESS, mercury free copper should be used in order to avoid problems with final disposal of these copper components

[12] Support by P.Ferguson, D.Freeman (SNS) and F.Maekawa (JSNS) is greatly acknowledged

calculations are in altogether sufficient agreement to SNS-calculations, but lead to slightly smaller total activities than SNS particularly for long halve lives (factor 2-3), which may be explained by the use of a simplified model in ESS calculations, not sufficiently considering nuclide production by chain reactions. This neglection of chain reaction may result in underestimation of production rates particularly for low mass nuclides and may explain the differences found between ESS and SNS calculations for Gd-148 (factor 5).

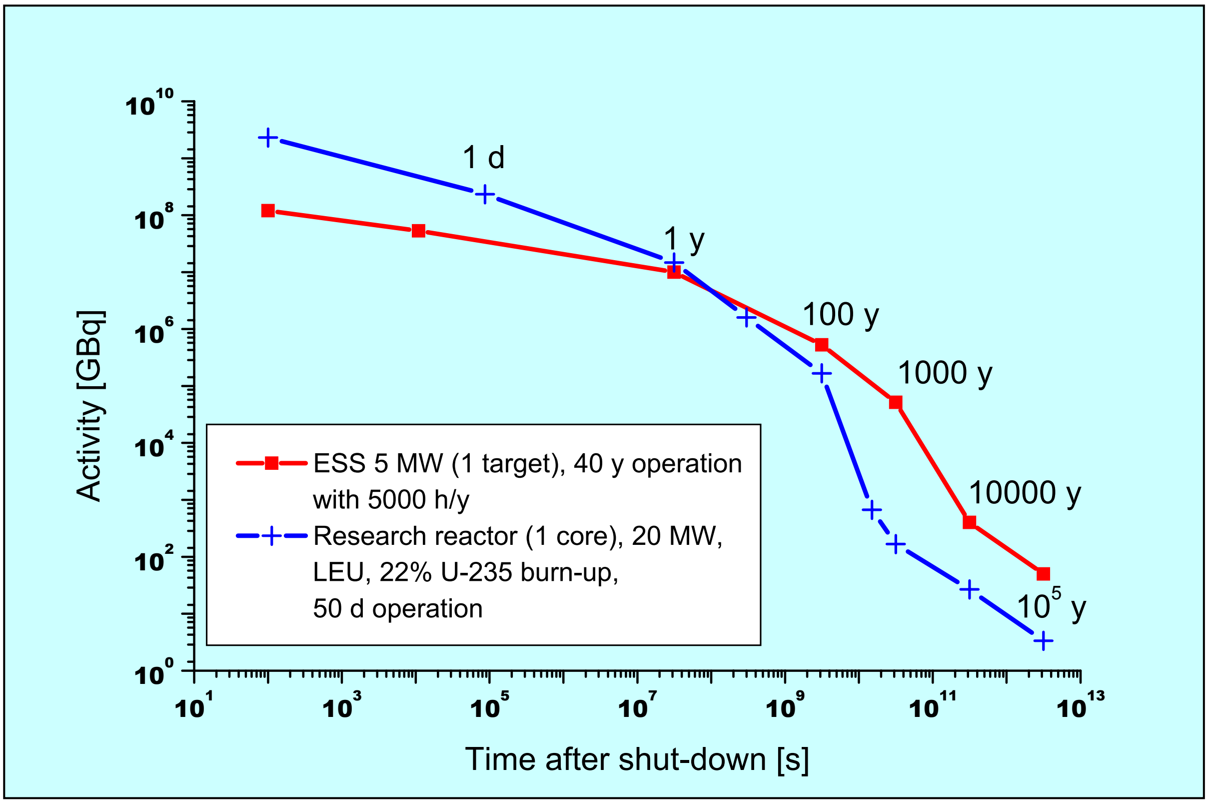

Figure 8.5.1.1: Comparison of activities in an ESS target and in a research reactor core (relevant for safety/accident analyses)

This figure reveals that the inventories are of comparable size. It has to be noted that ESS shows a comparatively large long and very long-lived lived activity, which is partly due to the fact that the ESS target is never changed, whereas research reactors run short fuel cycles[13]. The slow decay of research reactor fuel at t > 1000 y is mainly due to actinides in LEU (low enriched uranium) fuel; actinides are formed to a much smaller extent in HEU (high enriched uranium) fuel, used in certain research reactors. There is a limited amount (3 GBq) of ultra long lived α-emitting, but non fissile nuclides (Gd-150, Dy-154, $t_{1/2} > 10^6$ y) in ESS, too (see table 8.7.1.1).

As a general rule it may be taken that the equilibrium activity in a mercury target is in the same order of magnitude as in a fission system with equivalent power.

[13] The inventory of a single reactor core is nevertheless the correct basis for comparison, as long as safety considerations for accidents are the main focus. However, if final waste handling is the main point of concern, the long-lived activities of a research reactor summed up for the whole reactor life have to be compared with the inventory of both ESS-targets.

Fig. 8.5.1.2 compares the total activities to be disposed (or reprocessed in case of a research reactor) after 40 y of operation for ESS and a research reactor, depending on time after unloading of target/fuel. Here, due to decay during presence in the facility, ESS has substantial smaller activities up to 100 y after unloading. However, the total amount of long-lived up to ultra long-lived nuclides to be disposed is similar for both, ESS and a 20 MW research reactor.

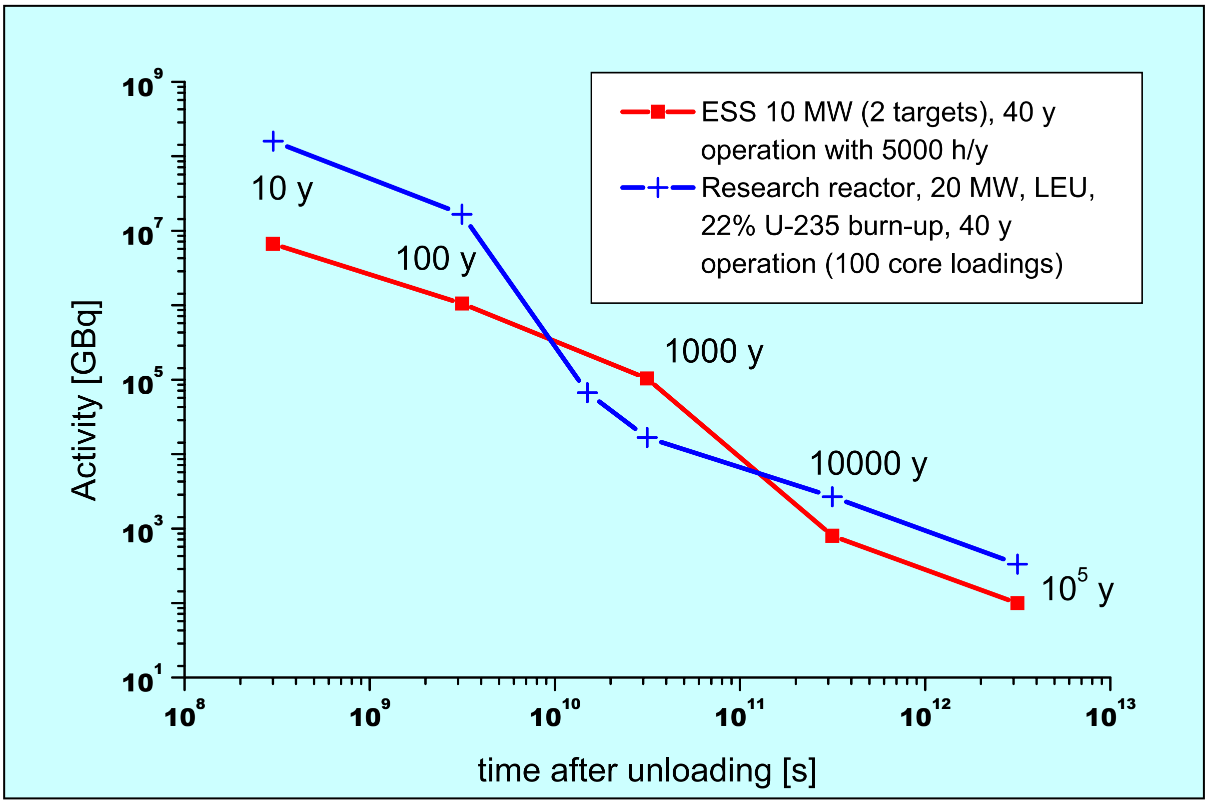

Fig. 8.5.1.2: Comparison of total activities of ESS and of a research reactor (relevant for waste disposal)

The before mentioned studies are theoretical only up to now. For validation it is advisable to determine experimentally the concentrations of long-lived nuclides in suitably irradiated mercury. More details to these intended experiments are given in subchapter 8.10.

Concerning the chemical status of iodine nuclides in mercury thermochemical examinations were performed, indicating, that iodine is mainly present as HgI_2 [Moormann, 2003]. This reduces the effective equilibrium iodine vapour pressure by several orders of magnitude compared to molecular iodine (which was assumed to occur in PSAR/SNS [SNS, 2000]), because the sublimation temperature of HgI_2 is about the same as that of atomic mercury, whereas molecular iodine already evaporates at 456 K. For several accident categories this leads even to a reduction of the iodine source terms [Moormann, 2003]. Figure 8.5.1.3 contains typical results of these thermochemical calculations.

However, concerning source term, it has to be taken into account that the density of HgI_2 is substantially smaller than that of Hg, which means, that HgI_2 swims up to the mercury surface and may in accidents evaporate faster due to kinetic reasons.

If iron (from target hull) is considered too, a further reduction of the equilibrium iodine vapour pressure by formation of FeI_2 has to be expected. More detailed even experimental examinations are advisable to this item, at least, if ingestion has to be taken into account, which is the predominant pathway for iodine.

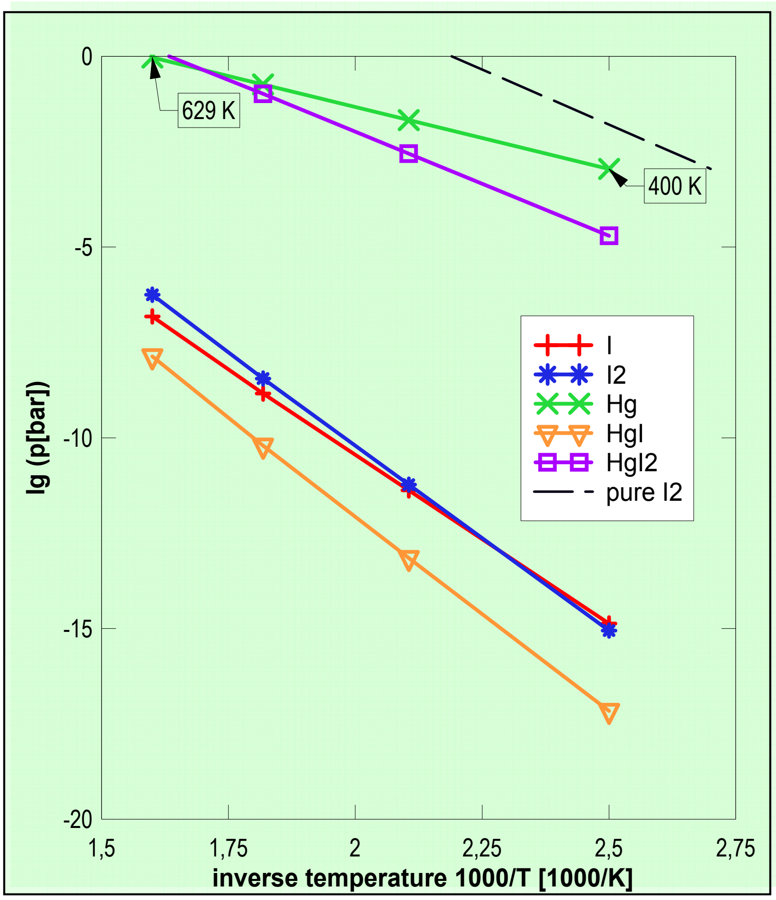

Figure 8.5.1.3: Equilibrium vapour pressures in an Hg/I system containing a mole fraction of 10^{-5} of iodine, compared with the vapour pressure of pure iodine

8.5.2 Radiotoxicities

Most relevant nuclides present in the ESS target were identified; criteria were:

- ✓ formation yield
- ✓ lifetimes
- ✓ dose factors
- ✓ volatility

Concerning volatility, the classification of PSAR/SNS [SNS, 2000] was used; nuclides were divided in PSAR/SNS into 3 volatility categories:

- High volatile nuclides (tritium, iodine, noble gases)
- Mercury isotopes (having an average volatility)
- Low volatile nuclides (most metals other than Hg)

These most relevant nuclides in the ESS target are given in table 8.5.2.1 together with their estimated inventories, their half-lives, their radiation type and their boiling points. The radiotoxicity is presented in terms of the dose, which results from an emitted activity of 1 GBq (assuming dispersion, incorporation etc. as in [StrlSchV, 2001]). Dominating nuclides for the different pathways are given in full-tone. For tritium, the inventory in the target is presented as scaled up from SNS-calculations [SNS, 2000]. The inventories are mainly best estimate values, multiplied by a factor of 1.6 in order to obtain conservative figures.

Table 8.5.2.1: Overview on radiological most relevant nuclides in one ESS-target (5 MW, 40 y operation with 5000 h/y)

Nuclide	Boiling point [K]	ESS target inventory [GBq]	Half life [d]	Radiation type	Dose/emission [1)] [Sv/GBq] Ground shine (γ)	Cloud	Inhalation	Ingestion
H-3 (HTO)	373	7.9e5 (total)	4500	*weak* β	0	0	2.3e-9 [3)]	4.6e-8 [2)]
H-3 (HT)	14						1.5e-11 [3,4)]	3.3e-9 [4)]
I-124	387	5000	4.2	β,γ	2.8e-6	1.0e-8	**3.0e-5**	**3.4e-3**
I-125	“	21700	60	γ	1.8e-6	1.0e-10	**1.8e-5**	**2.3e-3**
I-126	“	990	13	β,γ	3.9e-6	4.2e-9	**6.3e-5**	**7.6e-3**
Hg-193	629	3.1e6	0.16	γ	3.6e-8	1.7e-9	6.5e-9	2.5e-10
Hg-194	“	1.4e5	1.9e5	γ	**6.1e-3**	1.3e-13	1.3e-6	5.0e-7
Hg-195	“	5.1e6	0.42	γ	7.5e-8	1.7e-9	8.1e-9	6.3e-10
Hg-197	“	3.4e7	2.67	γ	1.1e-7	5.2e-10	2.0e-8	3.0e-9
Hg-203	“	2.4e7	47	β,γ	7.5e-6	2.2e-9	1.8e-7	8.1e-9
Gd-148	3546	5.5e4	2.72e4	α	0	0	**2.2e-3**	4.0e-7
Hf-172	4875	1.2e6	683	γ	**2.8e-4**	7.7e-10	**1.3e-5**	1.9e-9
Au-195	3081	6.6e6	186	γ	3.1e-6	6.2e-10	2.2e-8	5.9e-10

[1)] German directives for design basis accidents, infant (70 a continuous ingestion and ground shine), effective doses except for iodine (thyroid), minimum distance: 250 m, emission height: 25 m, [2)] preliminary ingestion model of German rules, [3)] including HTO resorption by skin (preliminary model), [4)] HT-oxidation to HTO in soil considered
Volatility classes: green = high volatile, blue = mercury (intermediate volatility), yellow = low volatile

The dose/emission factors of table 8.5.2.1 should only cautiously be used in realistic accident analyses: Whereas the assumptions for cloudshine and inhalation are more or less realistic, the assumptions for groundshine and ingestion are highly conservative and even unrealistic at least if applied to severe accidents with substantial source terms.

In order to obtain a figure of the radiotoxic potential of these inventories a computer code (CHI-ESS) was developed [Moormann, 2003a], which determines radioactive doses by inhalation, ingestion, submersion and groundshine for ESS-design basis accidents in the required conservative manner.[14] Using CHI-ESS dose calculations [Moormann, 2003b] were performed for these nuclides[15], oriented on German rules for design basis accidents (DBA), described more detailed in chapter 8.2.1 of [ESS, 2002] and (updated) [Moormann, 2003c]. Dose limits in DBA following German rules are 50 mSv (effective dose) and 150 mSv (thyroid dose).

[14] It has to be noted, that the upgrade of existing consequence models, leading to realistic consequence and risk figures, remains an open problem (s. subchapter 8.10)

[15] A short term emission was assumed together with worst weather conditions; an emission height of 25 m was considered (which is the approximate height of the target building; a larger effective emission height may occur in case of fires will decrease maximum doses). A distance from the emission point (target) to the fence of 250 m was taken into account; except for low volatile nuclides, where emission as aerosol is assumed, gaseous emissions are taken into account, which leads to higher doses for certain pathways. For Hg it is assumed, that 10% were converted into a soluble state. Calculations are performed for a person living unsheltered near to the fence for a 70 years period [StrlSchV, 2001]. Table 8.5.2.1 contains the estimated doses per emission for the 4 pathways, described in chapter 8.1.1; doses per emission are valid for infants and belong to effective doses except for iodine incorporation, where (because of the substantial accumulation of iodine in the thyroid) thyroid doses are calculated; effective doses are about a factor of 30 smaller than thyroid doses for iodine incorporation.

Concerning external irradiation it becomes obvious from these calculations, that a very pronounced contribution will result from long-lived Hg-194, mainly due to its short-lived daughter nuclide Au-194, and, to a smaller extent, to Hf-172. Cloud shine is obviously a less relevant pathway. With respect to incorporation pathways, particular high doses by inhalation are expected from Gd-148 (because of its α-emission), but iodine nuclides and Hf-172 are also relevant; substantial high ingestion doses may be induced by iodine nuclides; however, it has to be noted in this context, that a pronounced consideration of ingestion as done in German rules is not required in all European countries (which rely more on food ban in case of accidents).

Comparing the potential doses with the before mentioned dose limits and with the inventories for the respective volatility classes, it is found that dose limits are exceeded, if emissions of high volatile nuclides (iodine) reach about 0.25 % of the total inventory; for mercury this limit in the range of 0.025 % and for low volatile nuclides at about 0.03 %, too. These limits are decreased, if - as usually the case - coincident release of different volatility classes has to be considered: Application of CHI-ESS to release vectors typical for relevant 'loss of target confinement' accidents pointed out again the radiological dominance of Hg-194 due to its groundshine; this nuclide is responsible for > 80 % of the critical dose[16]. Taking into account the ALARA principle and the required conservatisms, and assuming that a pronounced release of volatile nuclides precedes a mercury release, the maximum tolerable release of mercury in a design basis accident was calculated to some 10^{-5} of its inventory (ground release).

With respect to low volatile nuclides it has to be mentioned, that a non negligible release pathway may be mercury spilling due to leaks or droplet formation by explosions and transport of mercury droplets (containing low volatile nuclides) by gas flow. The knowledge for estimation of corresponding source terms due to droplets is not sufficient. Some research work to this problem based on CFD-modelling was started within the ESS safety task, but had to be stopped, before reasonable results were obtained.

The official national databases for dose calculations in a licensing process do not yet cover Ho-163. Because it is an X-ray emitter, a pronounced radiotoxicity is not expected for this nuclide; in [UKAEA, 1999] it is quoted low radiotoxic.

A comparison of the radiotoxicity of mercury with its chemical toxicity was performed for the inhalation pathway and details are found in [Moormann, 2003b], [Moormann, 2002]: Main result is that the radiotoxicity dominates within the licensing framework, but that the chemical toxicity cannot be completely neglected. In case of realistic consequence estimations the role of the chemotoxicity of mercury increases.

Compared with other potential target materials (LBE = lead bismuth eutecticum, W) the chemotoxicity of a mercury target is substantially larger. However, even for radiotoxicity a mercury target does not show pronounced advantages compared to a LBE target, as an examination considering the most toxic nuclides Gd-148, Hg-194 and (for LBE) Po-210 indicates [Moormann, 2003d]. The total relevant radiotoxic inventory of a tungsten target is expected to be substantially smaller than that of a mercury target (see chapter 8.7.1): Because Hg-194 is not produced in tungsten, its accumulated radiotoxicity is smaller than in mercury by at least an order of magnitude. Due to the fact, that tungsten targets have to be exchanged several times during lifetime of a spallation source in contrast to liquid targets, the radiotoxic

[16] Accordingly, it makes no real sense to clean the mercury by distillation during the operation time of ESS, because Hg-194 cannot be removed.

inventory present in the facility (i.e. the amount, relevant for safety considerations) is even far less for a solid target. A disadvantage of solid targets is the increased H-3-production rate (due to cooling water activation), which has to be taken into account by respective design measures, limiting the release in normal operation; such measures are well established in nuclear technology. However, because of the small specific radiotoxicity of H-3 (see table 8.5.2.1), its relevance for DBA even in solid targets is limited.

Experimental determinations of long-lived nuclides in SINQ proton irradiated mercury are in preparation; it is intended to continue these investigations within European Framework Program FP6. The completeness of the nuclide vector formed in spallation systems is obviously a problem, which should be carefully addressed in future ESS studies. This problem is discussed in detail in [Freeman, 2003]. It is however expected, that in course of the before mentioned examinations and due to experience with SNS and JSNS the knowledge with respect to mercury targets will be improved. Nevertheless, additional adjacent theoretical studies and calculations of the nuclide vector are still required.

Another inventory to be released in certain accidents is associated with the cooling water of the target/moderator system, which contains relevant amounts of H-3 as HTO (about 10 - 50 % of the total tritium inventory in the target, as given in table 8.5.2.1) and some Be-7; potential doses connected to this inventory remain comparatively small, but the release probability in accidents respectively during normal operation is greater than that of the target inventory, and, accordingly, the activation of cooling water is of safety relevance, too.

8.5.3 Potential for radioactive releases

The following facts have to be considered in estimation of the risk by radioactive releases:

1. The radioactive inventory in the ESS targets (2.5×10^8 GBq at the end of life of the ESS) and its radiotoxic potential is similar to that of typical neutron generating research reactors.
2. There are no fissile or fertile materials and no criticality or chain reactions in ESS; the spallation stops, if the proton beam is shut down. However, a highly reliable beam shutdown system is required for ESS.
3. The power density of the decay heat in a mercury target is so small, that even in complete loss of forced cooling a boiling of mercury by decay heat is excluded.
4. There is no chemical reaction between mercury and the main cooling material water or between mercury and air; such reactions might enhance the release rates of the radioactive inventory.
5. The volatility of the radiotoxic inventory is comparatively high. This is due to the low boiling point of mercury and due to the fact, that liquid targets do not have a release barrier by the crystal lattice as solid state targets have; the absence of this barrier may facilitate the release of volatile spallation products, if other barriers (target hull etc.) have failed in an accident.

8.6 THERMAL HYDRAULIC SAFETY EXAMINATIONS

Safety studies for ESS require the detailed examination of accidents with gas explosions due to leaks within the moderator system (see chapter 8.8). Accordingly, computer models have to be adapted to this problem. The CFD (computerized fluid dynamics)-code CFX was applied to this problem; the results of a first parameter study are presented below. It has to be noted, that due to the still rough design of the moderator, these calculations may be only taken as an indication for the general applicability of the CFD-model, but not yet as part of an accident analysis.

The detailed objective of this parameter study is the examination of the mixing behavior of different gases and the potential formation of flammable gas mixtures, respectively, in a closed room simulating the ESS target hall under abnormal conditions. In a simple approach and due to lack of more detailed data, the room dimensions are set to be 3m x 4m x 5m with no internal structures or components present. The room is filled with helium at 0.1 MPa pressure. At time zero, the breach of a liquid hydrogen containing pipeline of the moderator system is assumed releasing instantaneously vaporized gaseous hydrogen at a temperature of 20 K, represented by a hydrogen source in a central position on the right side wall of the room. At the same time, another central opening in the room ceiling is assumed to have developed allowing air of ambient temperature (288 K) to flow into the room. In order to avoid a pressure buildup inside the room, an outlet is defined on the left sidewall. The ingress flow rate of hydrogen is assumed to be 0.05 kg/s over 30 s; the total mass of 1.5 kg represents the entire hydrogen inventory in the moderator system. For the air, a continuous inlet mass flow of 0.15 kg/s is assumed translating into an air exchange rate of 7.5 per hour. The plot in Fig. 8.6.1 shows the situation during the initial phase after 1.1 s into the accident scenario considered.

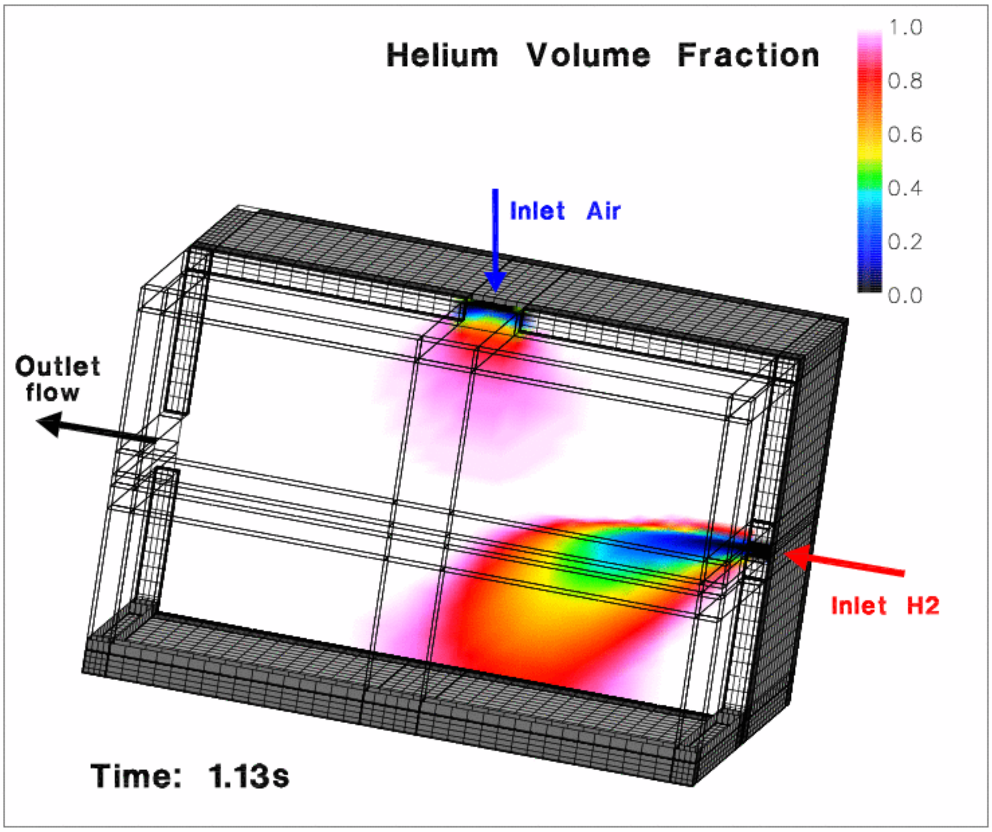

Figure 8.6.1: Helium concentration in target room approx. 1 s after initiation of the accident

The colors are a measure for the helium concentration in the room with the white color standing for 100 vol-% of helium. It is zero right at the two inlet locations. The colored plumes visualize the hydrogen and air flow into the helium. Both flows are directed downwards to the ground, since both air at 288 K and hydrogen at 20 K are heavier gases than the helium is at 288 K. During the release phase, hydrogen is mostly concentrated in the right half of the room with values in the range between 30 and 50 vol-%; the concentration in the left part of the room is 10 vol-% or less. After the end of the release phase, hydrogen concentrations are gradually decreasing reaching an almost homogeneous distribution around 20 to 25 vol-% after 50 s. In terms of flammability sensitive gas mixtures will develop near the boundaries where the two inlet flows are interfering; these areas, however, have not been evaluated quantitatively.

Other thermal hydraulic safety examinations dealt with the temperature development until failure of the proton beam window in case of loss of mercury flow: Stagnation of mercury will lead to a heat-up rate of the mercury adjacent to the proton beam window of up to 900 K/s, which means, that within < 0.5 s mercury boiling starts. Due to bubble formation, the heat removal from the proton beam window is reduced and heat-up to failure conditions occurs without significant delay time. However, a sudden complete stagnation of mercury has not to be expected even in case of a complete failure of the mercury pumps, as flow examinations and experimental work at SNS indicate: Due to inertia effects at least 5 s of (decreasing) flow seems to be realistic, which means, that the tolerable time span, until a correct proton beam shut-down has to be obtained, increases to $\geq$ 5 - 6 s.

8.7 DECOMMISSIONING AND FINAL WASTE DISPOSAL

8.7.1 Introduction

Decommissioning of nuclear facilities includes all actions taken from shut down of the facility with the ultimate goal of unrestricted release or radioactive reuse of the site. The decommissioning strategy to be developed is influenced by several factors such as availability of dismantling and waste treatment technologies, availability of a repository, and regulatory and licensing requirements as well as economic aspects. All these will differ from site to site and to some extent from country to country, and will largely affect the decommissioning costs.

Deferred dismantling is usually applied in nuclear facility decommissioning if radioactive decay of short-lived radionuclides promises to facilitate the handling procedures or to reduce the amount of radioactive waste. In contrast, immediate dismantling may be preferred if an early release of the site for reuse or the availability of knowledgeable staff are of major concern.

Dismantling techniques are available and state-of-the-art in decommissioning of conventional facilities. They have been adapted to radiation safety and waste management requirements, and have successfully been applied in the dismantling of nuclear facilities, as reported in [IAEA, 1994]. Some conclusions can be drawn from these experiences:

- Mixing of radioactive with non-radioactive waste should be avoided as far as possible during dismantling in order to minimise the amount of radioactive waste. For future facilities this can be supported by a proper design, e.g. a sandwich structure of the shielding.
- The generation of airborne particulate radioactivity should be minimised to facilitate the radiation protection measures for the personnel.
- If possible, unconditioned or conditioned recycling should be preferred to disposal for economical as well as ecological reasons.

One example for conditioned recycling is the Siempelkamp foundry in Germany, where cut metallic components from the decommissioning of nuclear installations are melted and recycled for nuclear container manufacturing[17] [Sonck, 2000].

Finally, the radioactive waste must be disposed off in a well designed repository in order to be safely isolated from the biosphere for a sufficient time span. The repository may be constructed near the surface, typically for the emplacement of short lived, low and intermediate level waste. For long lived and high level waste, a repository with engineered and natural barriers at depths up to several hundred meters in a geologically stable formation is preferred. Germany has decided to dispose all kinds of radioactive waste in a deep geological repository.

To ensure safe operation of a repository, both personnel and the environment must be protected, i.e. the dose limits set by the national authorities must be met. The safety of a repository is to be demonstrated by a site-specific safety analysis taking into account the (hydro-) geological situation, the technical concept of the disposal facility and the waste

[17] Use of this recycled iron for ESS target shielding is in discussion, too.

Tab. 8.7.1.1: Nuclides with activity > 0.1 GBq in one ESS target, 100 years after shutdown [Lensing, 2004] (*based on SNS-calculations)

	Nuclide	Activity/ GBq	$t_{1/2}$ / h,d,y	Energy of decay products/MeV
α-DECAY				*α –particles*
	^{148}Gd*	$2*10^4$	75 y	3.2
	^{150}Gd (daugther)	1.5	$2*10^6$ y	2.7
	^{154}Dy	1.2	$3*10^6$ y	2.9
ß-DECAY				*Average for $\beta^{\pm}$ / Max. for γ*
	^{3}H*	$2.9*10^3$	12 y	0.006
	^{14}C	24	5730 y	0.05
	^{32}Si	$1.7*10^3$	172 y	0.07
	^{32}P (daugther)	$1.8*10^3$	14 d	0.7
	^{36}Cl	0.7	$3*10^5$ y	0.25
	^{39}Ar	$2.2*10^3$	270 y	0.22
	^{42}Ar	$1.8*10^3$	33y	0.23
	^{42}K (daugther)	$1.8*10^3$	12 h	1.43 / 2.4
	^{60}Fe	0.5	$1.3*10^6$ y	0.05 /0.06
	^{60}Co	0.6	5.2 y	0.1 / 2.5
	^{63}Ni	$5.5*10^3$	100 y	0.02
	^{79}Se	3.1	$6.5*10^5$ y	0.05
	^{90} Sr	1500	29 y	0.2
	^{90}Y (daugther)	1500	64 h	0.94 / 2.2
	^{93}Zr	1500	$1.5*10^6$ y	0.02 / 0.03
	^{94}Nb	23	$2.3*10^4$ y	0.14 / 0.87
	^{98} Tc	0.2	$4.2*10^6$ y	0.12/ 0.75
	^{99} Tc	5	$2.1*10^5$ y	0.09 / 0.09
	^{107}Pd	0.2	$6.5*10^6$ y	0.01
	$^{113\,m}$Cd	0.6	14 y	0.1/ 0.26
	^{121}Sn	50	27 h	0.12
	^{151}Sm	390	90 y	0.02 / 0.02
	^{154}Eu	0.8	9 y	0.22 / 1.9
γ-EMITTER				*Max. for γ / Average for $\beta^{\pm}$*
	^{41}Ca	1.6	$1*10^5$ y	X-rays
	^{44}Sc (daugther)	45	4 h	3.3 / 0.6
	^{44}Ti	45	49 y	0.15 /0.06
	^{60m}Co	0.5	0.15h	2.2 /0.56
	^{81}Kr	80	$2*10^5$	0.28 / 0.15
	^{91}Nb	600	680 y	X-rays /0.005
	^{93m}Nb	50	16 y	X-rays/0.03
	^{93}Mo	55	$4*10^3$ y	X-rays/0.04
	^{97}Tc	0.2	$2.6*10^6$ y	X-rays/0.01
	^{121m}Sn	60	55 y	X-rays/0.01
	^{133}Ba	90	10.5 y	0.28/022
	^{137}La	100	$6*10^4$ y	0.6 / 0.2
	^{145}Pm	$4.5*10^3$	18 y	X-rays / 0.1
	^{146}Pm	0.3	5.5 y	X-rays / 0.08
	^{150}Eu	$6*10^3$	37 y	1.8 /0.2
	^{158}Tb	$6*10^4$	180 y	1.2 /0.05
	^{163}Ho	$1.9*10^3$	4570 y	X-rays
	^{193}Pt	$4*10^5$	50 y	X-rays/0.013
	^{194}Au (daugther)	$1.4*10^5$	38 h	2.4 / 0.03
	^{194} Hg	$1.4*10^5$	520 y	X-rays

packages properties. This safety analysis must include the operating as well as the post-operational phase for both normal operation and incidents/accidents. The safety assessment studies result in requirements that must be fulfilled by each waste package to be accepted for disposal.

Table 8.7.1.1 contains nuclides most relevant for ESS waste disposal with their halve lives and their radiation energies. The data used are those of [Lensing, 2004], multiplied by a factor 1.6. For H-3 and Gd-148, SNS results [SNS, 2000] are taken for the reason to remain conservative. Criterion of this selection is that the activity 100 y after shutdown is > 0.1 GBq.

The total activity 100 y after shutdown is $8 \cdot 10^5$ GBq for each ESS mercury target, see chapter 4-3. As shown in Tab 8.7.1.1, the dominant activity> 10^5 GBq is resulting from isotopes heavier than W, indicating that a solid tungsten target will lead to a total activity of only 10 % compared to a mercury one. There are no long-lived unstable W-isotopes. The tritium production rate for a solid W-target will be higher than for a liquid Hg one, which needs particularly careful examinations on its disposal.

8.7.2 Disposal of the target

Special attention must be paid to the treatment and disposal of the irradiated mercury target due to its potential conventional (bio-toxic) as well as radiological hazards in a repository. Some major aspects are discussed hereafter, based upon the acceptance requirements of the German Konrad repository mentioned above [BfS, 1994].

Requirements on target waste form
The following general basic requirements must be met by all kinds of waste packages:

- Waste forms must be in solid form.
- Waste forms must neither rot nor ferment.
- Waste forms must not contain, with the exception of residue levels achieved by reasonably to be expected effort
  - neither liquids nor gases in ampoules, bottles or other containers,
  - neither freely mobile liquids, nor release such liquids under normal storage and handling conditions,
  - neither self-igniting nor explosive materials.

The following additional basic requirements apply to waste forms manufactured using an immobilisation material (e.g. cement, concrete, bitumen or plastic):

- Reactions between the radioactive waste, the immobilisation material and/or the packaging must be reduced to a permissible rate from a safety point of view.
- The immobilisation material used must completely be set or solidified.
- Sealing of radioactive wastes or void spaces between inner packages must be done with suitable free-flowing immobilisation materials consolidated, if necessary, by means of technical measures (e.g. vibration).
- Immobilisation materials used for the sealing of radioactive wastes or void spaces between inner packagings may also be mixed with contaminated liquids if the quality requirements of the waste form group in question are fulfilled and compatibility with those materials to be cast is guaranteed. Radionuclides contained in the contaminated liquids must be taken into account in activity data.

Immobilisation

Due to the general basic requirements wastes must not contain liquids. Consequently, the liquid mercury target has to be converted into a solid form. Two possibilities seem to be applicable:

1) Conversion of the liquid mercury into a solid chemical compound. Due to their stability, inorganic compounds like amalgams, HgCl or HgS seem to be promising candidates. The conversion should be followed by an immobilisation of the Hg-compound, e.g. by cementation.
2) Direct immobilisation of the liquid mercury by reaction with or encapsulation in a matrix like cement.

Based upon existing knowledge, the first alternative seems to be more advantageous, because the reaction of the mercury with the different cement phases is not yet totally understood. Further studies to this problem are urgently required, because the proof of an adequate decommissioning/final disposal concept (including sufficient funding) will be requested within the ESS licensing procedure.

Activity limitations

In Germany, a preliminary set of acceptance requirements is derived for the Konrad repository, an abandoned iron ore mine provided for the disposal of radioactive waste with negligible heat generation, which is planned to start operation within a few years if all claims against the license are dismissed. They include general requirements on waste package as well

Table 8.7.2.1: Target inventory and activity limits for selected radionuclides potentially relevant for the irradiated mercury disposal.

Nuclide	Half life [d]	Target [1)] inventory [Bq]	Target [2)] inventory [Bq]	Activity limits [3)] [Bq]
H-3	4.50E+03	1.68E+15	6.08E+12	3.0E+09
Cd-109	4.64E+02	5.73E+11	0	9.3E+13
Sb-125	9.96E+02	2.69E+12	6	3.6E+13
I-125	6.01E+01	6.69E+11	0	2.1E+10
I-129	5.73E+09	5.23E+06	5.23E+06	1.9E+07
Ba-133	3.84E+03	4.50E+13	3.17E+10	1.4E+13
Pm-147	9.56E+02	3.84E+12	13.9	6.4E+15
Sm-151	3.24E+04	5.98E+11	2.75+11	1.2E+16
Eu-152	4.86E+03	8.82E+12	4.95E+10	4.4E+12
Eu-154	3.14E+03	1.68E+12	5.52E+08	6.3E+12
Eu-155	1.81E+03	8.64E+11	7.83E+05	2.1E+14
Ta-182	1.15E+02	1.11E+11	0	8.6E+13
Hg-203	4.66E+01	2.21E+14	0	4.1E+14

1) 10 MW, 40 years irradiation, 1 year cooling time [SNS, 2000], safety factor of 1.6

2) As 1) but for 100 y cooling time

3) Data given per container, applicable for cementitious waste form (waste product group 05) and waste container without specified leak-tightness.

as specific ones on waste forms and packagings and limitations for activities of individual radionuclides. A total of 11 container types is planned for the disposal in the Konrad repository, divided into cylindrical concrete or cast-iron containers with volumes between 0.7 and 1.3 m^3, and into box-type containers with gross volumes between 3.9 and 10.9 m^3. Details are given in [BfS, 1994].

Permissible activities for radionuclides and radionuclide groups (non-specified alpha and beta/gamma emitters) result from safety assessments for the operational and post-operational phases of the repository. Table 8.7.2.1 shows a list of radionuclides of irradiated mercury, for which activity limits for the Konrad repository exist. Radionuclides with halve life shorter than or equal to ten days were neglected. Most of the target radionuclides like long-lived Hg-194, Ho-163 and others with high activities (see table 8.7.2.1) cannot be taken into account, because these radionuclides are not quantified in the Konrad repository acceptance criteria. This underlines the need for extension of regulations for ESS purposes, also in the field of waste disposal.

From the table it can be derived, that most examined radionuclides are already after 1 year of cooling time well within the activity limits for a deep geological repository in Germany, with the exception of iodine-125 and tritium, and to a minor extent barium-133 and europium-152. Whereas the iodine, barium and europium problems could easily be overcome by relatively short decay storage of a few years, the tritium inventory would require the storage of the target for many decades to meet the activity limit derived from normal operation, as the data for 100 y of cooling time in table 8.7.2.1 indicate. This underlines an urgent need to examine the strongly different H-3 contents generated in targets, calculated at SNS respectively ESS (see chapter 8.5.1, footnote 9). Future investigations must also deal with other measures mitigating this problem, e.g. recovery of tritium from the mercury with following solidification in a metallic matrix, which would raise the activity limits for H-3 by two orders of magnitude. However, it must be pointed out that these activity limits differ from site to site and from country to country.

Chemical toxicity has not yet been explicitly addressed in safety assessments for the disposal of radioactive waste in Germany. However, the licence (plan-approval decision) for the Konrad repository [NMU, 2002] which is actually subject to claims, as a first of its kind addresses the limitation of chemo-toxic substances in radioactive waste, in order to comply with the limits of the Groundwater Ordinance. Based upon that site specific model (dissolution in 1,000,000 m^3 deep groundwater and following dilution by a factor of 10,000 during transport to the surface groundwater, limit for Hg in surface groundwater: 0.5 µg/litre), the total mass of mercury disposable in the Konrad repository would be limited to 5 tonnes. Having the total amount of mercury of 30 tonnes in ESS in mind, there is obviously some need for further examinations.

Target hull

The target hull has to be exchanged up to several times a year and, accordingly, represents a significant volume of waste. As demonstrated in [Filges, 2003], the total activity generated in the target windows is for 10 y after irradiation even about the same as in the mercury itself, but decays much faster than the latter (100 y after irradiation: The window activity declines to a value of about a factor 500 smaller than found in mercury). Nevertheless, future work has to deal with storage/disposal of the target hull, too.

8.8 SAFETY ANALYSES

8.8.1 General

Preliminary safety analyses for ESS were performed based on respective analyses for SNS, documented in the PSAR (Preliminary Safety Analysis Report) [SNS, 2000], [SNS, 2002], with the main aims:

- ✓ Preparation of a sufficient basis for the ESS safety report and for other documents required within the ESS licensing procedure
- ✓ Support for the optimization of the ESS design with respect to safety features

The ESS safety analysis is a stepwise, partly iterative process. As a first step, it contains the identification of safety relevant accident-initiating events [Bongartz, 2003] and as second step a preliminary risk assessment for selected relevant events has to be performed [Bongartz, 2003a]. As dose criterion the effective doses for the public with some frequency consideration behind was taken into account, as is usual practice in nuclear related facilities. In this update report only the general methodology of safety analyses performed can be described, together with a summary of main results; all details of these safety analyses are found in [Bongartz, 2003], [Bongartz, 2003a].

8.8.2 Comparison of dose criteria for SNS and ESS

Comparing PSAR/SNS dose and consequence estimations with ESS conditions the following differences have to be taken into account:

- ➢ Greater distance between the SNS facility and the public (1375 m, compared to about 250-300 m expected for ESS): Doses decrease roughly with up to the inverse square of the distance,
- ➢ Larger radiotoxic inventories in the ESS targets due to the higher proton beam power,
- ➢ More stringent safety limits in European countries (because of higher population densities in most parts of Europe); e.g. a 50 mSv effective dose is the maximum allowed for DBA in Germany, compared to 250 mSv in USA,
- ➢ Use of a ‘realistic’ consequence model in PSAR/SNS instead of a formal, highly conservative dose estimation as required in ESS commissioning,
- ➢ No consideration of the ingestion pathway in PSAR/SNS consequence estimations.

Altogether, fractional releases of the ESS inventory have to be about 2 orders of magnitude smaller than those for SNS accidents, whose consequences are near to dose limits. Accordingly the depth of safety analyses in PSAR/SNS is not sufficient for ESS conditions and additional steps in the iterative safety analysis process have to be undertaken at least for certain relevant accident sequences.

8.8.3 Identification of safety relevant events

The SNS target is similar to the ESS target; in PSAR/SNS a broad spectrum of events, which may lead to an accident connected with a release of toxics into environment, is taken into account. These "accident initiating events" are arranged into one of the following five categories:

a) Loss of confinement function by internal initiation
b) Internal explosions (by hydrogen/air mixtures, caused by moderator failure)

c) External events by civilization induced reasons (fire, explosion of a gas cloud, airplane crash)
d) Natural hazards (earthquake, flooding)
e) Internal fire

As a first step, the safety relevant events within of this large collection of > 150 events have to be identified by screening (without taking into account any measures and/or systems for accident mitigation or control). This identification of safety relevant events in ESS was performed in an analogous manner to PSAR/SNS [Bongartz, 2003], based on risk binning. Table 8.8.3-1 contains the selection criteria. The arrangement of these events into one of the 8 risk bins results from combination of consequences and frequencies, both of which are subdivided into different classes.

Table 8.8.3.1: Matrix for ESS risk bins

Frequency: → **Consequences: ↓**	**Beyond Extreme Unlikely** **($< 10^{-6}$/y)**	**Extreme Unlikely** **($10^{-6} \leq f < 10^{-4}$/y)**	**Unlikely** **($10^{-4} \leq f < 10^{-2}$/y)**	**Anticipated** **($\geq 10^{-2}$/y)**
High **(>50 mSv)**	**Risk bin 7** *{7}*	**Risk bin 5** *{4}*	**Risk bin 3** *{15 (+1)}*	**Risk bin 1** *{15}*
Moderate **($0.3 \leq x \leq 50$ mSv)**		**Risk bin 6** *{1}*	**Risk bin 4** *{4}*	**Risk bin 2** *{38 (+31)}*
Negligible **(<0.3 mSv)**	**Risk bin 8** *{46}*			

{number of events in brackets (+ potentially safety relevant events)}

As required by European conditions dose criteria identifying the consequence class were changed compared to PSAR/SNS: The dose limits were taken from nuclear licensing process in Germany. The consequence class 'High' was used for events resulting in effective doses for the public at the fence of the facility > 50 mSv. The consequence class 'Moderate' covers consequences with doses of 0.3 mSv - 50 mSv and the class 'Negligible' holds for doses < 0.3 mSv.

Concerning frequency classes, the same subdivision as in PSAR/SNS are used (Anticipated, Unlikely, Extreme Unlikely and Beyond Extreme Unlikely).

Relevant risk bins (selected by the risk = consequence x frequency) are marked red in table 8.8.3.1, these are numbered 1, 2, 3 and 5.

Frequencies for events are mainly taken from PSAR/SNS. Also, consequence estimations are based on doses presented in PSAR/SNS, however, ESS doses are higher for the reasons explained in 8.8.2 by up to 2 orders of magnitude, which was taken into account.

Without consideration of mitigating/control measures dose limits are exceeded for ESS in 72 of 162 events considered (see numbers given in {}-brackets in risk binning table 8.8.3.1; some events, whose association to a certain risk bin was uncertain, were counted several times). Altogether 32 safety relevant accident-initiating events were detected, from which finally the 'design basis accidents' for ESS have to be selected. These 32 events belong to:

- Loss of Confinement function: 15
- Internal explosions: 4
- External events (civilization induced): 3
- Natural hazards: 4
- Internal fire events: 6

Selection of design basis accidents is required for the proof of a sufficient safety design of ESS, including proof of adequate accident mitigating measures, during the licensing process.

Most relevant are such events, where the target system with its remarkable radiotoxic inventory is involved. Maximum consequences are obtained, if the mercury is heated up in course of the accident and (partly) evaporates into the environment. In case of ESS, these extreme cases may lead to a pronounced exaggeration of dose limits. Representative accidents of this type are on the one hand side those with failure of target cooling combined with a failure of the proton beam shutdown system. On the other hand, heat-up of the mercury by explosion, fire (independent explosion/fire or caused by an earthquake) belongs to these types of accidents, too. For both cases, protecting design measures are required for ESS.

For loss of confinement function without mercury boiling the doses are estimated to be about 2 orders of magnitude smaller than for the before mentioned cases, but are still higher than dose limits, and accordingly, mitigating measures are necessary in contrast to SNS; the same holds for accidents in ancillary or auxiliary systems with altogether small radioactive inventory (e.g. hydrogen release out of the moderator system).

Altogether it may be concluded, that technical or administrative measures, which are foreseen in ESS to reduce consequences or accident frequencies, are strongly required.

8.8.4 Risk assessment for selected events

After selection of the safety relevant events more detailed studies are required within of the ESS safety analysis.

Event tree analyses
Some of the identified 32 safety relevant events, for which sufficient conceptual design data were available, were analysed more in detail using the event tree method [Bongartz, 2003a]. In detail the event tree method is used as follows: Starting with an initiating event, the availability of measures and/or systems for accident mitigation or control is examined. A simplified example of such an event tree is presented in figure 8.8.4.1 (mitigating measures/systems are coloured in green).

Depending on success (yes) or failure (no) of these measures or systems different branches are obtained; each particular branch represents an individual event sequence. The event sequence consisting only from 'yes'-branches for availability of measures and/or systems for accident mitigation or control symbolises the complete realisation of the particular safety concept; for all sequences containing 'no'-branches in success of safety measures/systems the consequences have to be estimated.

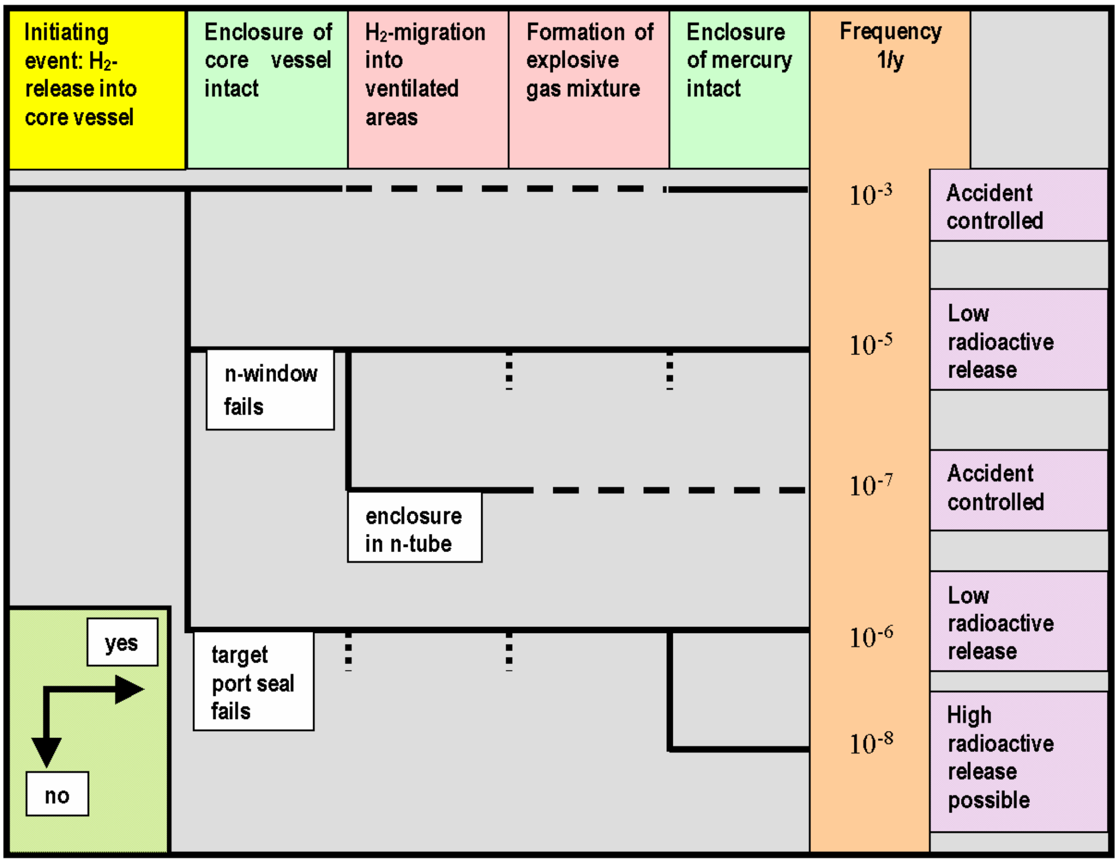

Figure 8.8.4.1: Part of an ESS event tree (simplified) for H_2-release into the core vessel (ESS design variant with overpressure in the core vessel)

The frequencies of individual event sequences are calculated here by multiplication of the frequencies of the initiating event and those for failure of systems and measures for accident mitigation/prevention. For the initiating events statistical examination of failure experience is used for estimation of frequencies. In case, that there is no sufficient experience available, estimations based on an engineering judgement have to be applied; for ESS frequencies from PSAR/SNS [SNS, 2000] were taken into account.

Probabilities for failures of measures and systems for accident mitigation/prevention are based for complex systems on estimations with help of the fault tree method. For more simple systems, experience is directly used as basis for these probabilities. For ESS there are often sufficiently detailed design data not yet available; accordingly, these frequencies and probabilities are estimated based on data and experience for similar systems, and in case, that this is not possible, conservative engineering judgement was applied.

Estimation of the risk of an accident as a whole requires an additional step: Accident sequences belonging to the same initiating event and having similar consequences have to be comprehended. In order to obtain a complete risk figure of the whole plant the frequencies of all accidents with similar consequences have to be summed up.

Accident classification

Estimation of relevance of ESS accident sequences and accidents requires a more detailed classification system than that, used for risk binning before. The following classification system for frequencies was used:

Table 8.8.4.1 Frequency classification

Frequency class	Frequency f (1/y)
High	$f > 10^{-2}$
Moderate	$10^{-4} < f \leq 10^{-2}$
Small	$10^{-5} < f \leq 10^{-4}$
Very small	$10^{-6} < h \leq 10^{-5}$
Extreme small	$10^{-8} < f \leq 10^{-6}$
Negligible	$f \leq 10^{-8}$

Basis for classification of accident consequences is the radiation dose for the public at the ESS fence, compared with the dose limits of 50 mSv and 0.3 mSv used already in risk binning; these dose limits are frequency related 'whole body'[18] dose limits, which have to be met in Germany in accidents with nuclear facilities [KTA, 1989]. Again, SNS dose calculations are used as basis for estimations of doses in ESS accidents. In contrast to risk binning, the before mentioned consequence classification holds for accidents with frequencies $< 10^{-2}$/y only, for higher frequencies a dose limit of 0.3 mSv has to be applied (see table 8.8.4.2). Typical accidents are the same as those mentioned in course of risk binning in chapter 8.8.3.

Table 8.8.4.2: Consequence classification

Consequence class	Description
High	The dose limit of 50 mSv (respectively 0.3 mSv for $f > 10^{-2}$/y) is substantially exceeded at ESS fence.
Moderate	Doses at the ESS fence are in the range of the dose limit of 50 mSv (respectively 0.3 mSv for $f > 10^{-2}$/y).
Small	The dose limit of 50 mSv (respectively 0.3 mSv for $f > 10^{-2}$/y) is surely met at ESS fence.

By combination of their consequences and frequencies the risk relevance of ESS accidents was determined. Here, frequency-related goals for protection of the public, as demonstrated in table 8.8.4.4, were taken into account (see also chapter 8.1). This safety assessment was performed for Loss of Confinement function and H_2-explosions, except those events, where ancillary or auxiliary systems are involved. The latter could not be analysed more detailed because of a lack of design data. The same holds for other events (e.g. fire), where besides more detailed design data deterministic pre-analyses (e.g. concerning fire scenarios etc.) are required, which could not yet be performed.

The classification of the risk relevance for ESS accidents follows the scheme, given in table 8.8.4.3.

[18] Instead of the 'whole body dose', the effective dose is used in calculations for ESS; the 'effective dose' concept has completely displaced in Europe the former 'whole body dose'.

Table 8.8.4.3: Classification of accident risk relevance

Accident frequency class (f [1/y])	Consequence class	Resulting risk relevance class
High ($f > 10^{-2}$)	High Moderate Small	Very High High Moderate
Moderate ($10^{-2} < f \leq 10^{-4}$)	High Moderate Small	High Moderate Small
Small ($10^{-4} < f \leq 10^{-5}$)	High Moderate Small	Moderate Moderate Small
Very small ($10^{-5} < f \leq 10^{-6}$)	High Moderate Small	Small Small Very small
Extreme small ($10^{-6} < f \leq 10^{-8}$)	High Moderate Small	Very small Very small Negligible
Negligible ($f < 10^{-8}$)	High/Moderate /Small	Negligible

Table 8.8.4.4: Risk classification scheme for ESS-accidents related to protection goals

ACCIDENT FREQUENCY			FREQUENCY-RELATED GOALS FOR PROTECTION OF THE PUBLIC: Risk relevance (classified accidents)		
Characterization	**Class**	**f** (y^{-1})			
• Accident has to be expected	High	10^{-0}	Operation and incidents	*'High'*	*'Very high'*
• Accident is not expected, but cannot be excluded	Mode-rate	10^{-2}	Design basis accidents		*'High'*
• Accident cannot be completely excluded	Small	10^{-4}			*'Moderate'* (3.2.2)
	Very small	10^{-5}	'Intermediate range'		*'Small'* (3.1.1/3.1.2/3.2.1/3.2.3/3.2.4)
• Accident can be excluded, but accident sequence is credible	Extreme small	10^{-6}	Beyond design basis (hypothetical) range		*'Very small'* (3.2.5/3.2.6/3.2.7/3.2.8)
• Accident is not credible	Negli-gible	10^{-8}			*'Negligible'*
= goals for public protection not met			0.1	1 | 10 ***0.3 mSv***	100 ***50 mSv***
			Accidental dose at the ESS fence		

The final results of these accident assessments for ESS are presented in table 8.8.4.5. It should be noted, that the altogether small risk relevance of 'Loss of confinement' accidents and of H_2-explosion accidents is mainly due to their small frequencies. Differences in their relevance concerning risk are caused by differences in their consequences. Accidents with 'High' consequences are combined with so small frequencies, that the risk relevance remains nevertheless small. The largest risk relevance results from an accident (leak in target container) with 'moderate' consequences and 'small' frequencies.

Table 8.8.4.5: Overview on ESS accident classification and risk assessment

ESS Accident[5]	Conse-quences	Frequency (1/y)	Risk relevance
Loss of Confinement function			
Complete loss of mercury flow (3.2.4)	High[1]	Very small ($1.3 \cdot 10^{-6}$)	Small
Partial loss of mercury flow (3.2.5)	High	Extreme small ($3.5 \cdot 10^{-7}$)	Very small
Loss of target heat sink (3.2.6)	High	Extreme small ($6 \cdot 10^{-7}$)	Very small
Proton beam misalignment (3.2.1)	Moderate[2]	Very small ($1.2 \cdot 10^{-6}$)	Small
Leak in target container (3.2.2)	Moderate	Small ($1.2 \cdot 10^{-5}$)	Moderate
Hg-leak in target cell (3.2.3)	Moderate	Very small ($1.1 \cdot 10^{-6}$)	Small
Crane load drop onto Hg-circuit (3.2.7)	Moderate	Extreme small ($2 \cdot 10^{-7}$)	Very small
Crane load drop onto target cell (3.2.8)	Moderate	Extreme small ($1.5 \cdot 10^{-7}$)	Very small
Release from ancillary and auxiliary systems	≤Moderate	-[3]	-[4]
Explosion of H_2/air mixtures			
H_2-release into core container, *old design* (3.2.1)	High	Very small ($1.5 \cdot 10^{-6}$)	Small
H_2-release into core container, *new design* (3.1.1)	Moderate	Very small ($2.1 \cdot 10^{-6}$)	Small
External events			
e.g. Explosion of a gas cloud	Moderate	-[3]	-[4]
Natural hazards			
e.g. earthquake followed by an explosion and a fire	High	-[3]	-[4]
Internal fires			
e.g. fire in target cell	High	-[3]	-[4]

[1] Substantial transgression of 50 mSv dose limit at the ESS fence
[2] Transgression of 50 mSv dose limit at the ESS fence cannot be excluded
[3] No assessment because of lack of detailed design data
[4] Classification depends on the extent of mitigating/emergency measures taken
[5] Accident identification numbers in brackets, for details see [Bongartz, 2003a]

Accidents, which could not be analysed in detail because of lack of design data, do probably not change the risk figure demonstrated above: The risk relevance of an internal fire for example can by an adequate design reduced to the same low level as found for H_2-explosions. The total risk of ESS will probably be determined by extremely severe, beyond design earthquakes; this however does not necessarily mean, that catastrophic consequences have to be expected, at least, if standard design measures against earthquakes are undertaken.

Concerning design optimisation, two different activity enclosure concepts for the core vessel was examined, both having certain advantages from the safety point of view: Overpressure of the vessel atmosphere (*new design* in table 8.8.4.5) and sub-atmospheric pressure (*old design* in table 8.8.4.5). It was shown, that concerning the risk relevance both concepts are feasible.

It should be noted, that (as seen from table 8.8.4.4) accidents with frequencies > 'extreme small' have to be treated as DBA, what means, that consequences, calculated with highly conservative assumptions, have to remain ≤ small. This is not fulfilled for all accidents examined. However, this safety study revealed sufficient potential for a reduction of both, consequences and frequencies by design measures. Improvement of tools and methods for safety analyses, which is possible in many ways, may lead to the same result. Accordingly, table 8.8.4.5, summarising the results of this preliminary risk analysis, should not be interpreted as indication for problems of ESS to meet DBA criteria, but as a guideline, where ESS design or safety analysis methods should be improved.

8.8.5 Concluding remarks

Compared with existing nuclear fission facilities with similar amount of toxic inventory (research reactors etc.) the risk of ESS is estimated to remain small, mainly due to the very small probability of accidents with high consequences. A further gain concerning risk reduction is expected by an additional optimisation of the ESS safety design. Altogether, there is no doubt, that the present ESS design is able to get licensed (present licensing conditions assumed) from the point of view of radiological safety.

Taking into account, that radiological licensing requirements are becoming continuously more stringent, there is sufficient potential detected for a facility like ESS to fulfil these requirements. A safety goal similar to that for advanced fission reactors (small, safety optimized systems, seems to be achievable by ESS without drastic changes in design: These safety goals require for internal accidents to met the restrictive DBA dose limits even as deep into the range of 'hypothetical' accidents, as credible safety assessments will allow, i.e. down to frequencies of 10^{-7}/y. Achievement of these safety goals requires the application of more passive or inherent safety features, e.g. for the proton beam shutdown system[19].

[19] The development of passive beam shutdown systems is already under way for accelerator driven systems (ADS) for nuclear waste transmutation [Wider, 2000]; this work might be useful for ESS, too.

8.9 PUBLIC RELATION ORIENTED SAFETY WORK

8.9.1 General

For information of the public about safety and licensing items of ESS a safety and licensing fact sheet is required. As basis for this fact sheet relevant information (given in the next subchapter) was collected and transferred to PR people for working out the fact sheet. This safety fact sheet itself, being readable by laymen too, will be found in the public part of the ESS website [ESS, 2003].

8.9.2 Collection of information for an ESS safety fact sheet

Introduction
ESS must meet very high safety standards, because a substantial amount of radioactive material will be generated and because ESS contains a sizeable amount of conventionally toxic mercury. These high safety standards are guaranteed by safety regulations, whose fulfilment will be proven within the strict licensing or authorization process required for ESS. Although safety regulations are partly different in detail in various European countries, this does not significantly affect the resulting safety standards. Besides questions about fulfilment of formal safety regulations this fact sheet will cover also all other items relevant for the public debate about ESS safety.

Major issues pertinent to the public debate on safety
Noting the issues the public wants to be informed about when the production of radioactive material together with the presence of conventionally toxic mercury and/or nuclear installations are at hand, there are seven major points to know:

1. There are no fissile or fertile[20] materials and no criticality or chain reactions in ESS; the spallation stops, if the proton beam is shut down.
2. Ultra long lived, fissile actinides elements are absent in ESS; this facilitates final waste handling/disposal.
3. There are no proliferation[21] problems combined with ESS.
4. The power density of the decay heat in a mercury target is so small, that even in complete loss of forced cooling a boiling of mercury by decay heat is excluded[22].
5. There is no chemical reaction between mercury and the main cooling material water or between mercury and air[23].
6. The radioactive inventory in the ESS targets (2.5×10^{8} GBq at the end of life of the ESS) and its radiotoxic potential[24] are similar to that of typical neutron generating research reactors. The targets contain about 30 t of volatile and toxic mercury; this amount however is below the limit of 50 t, which forces a conventional safety analysis to this aspect (EU Seveso-II-guideline)
7. Liquid targets do not have a release barrier by the crystal lattice as solid state targets have.

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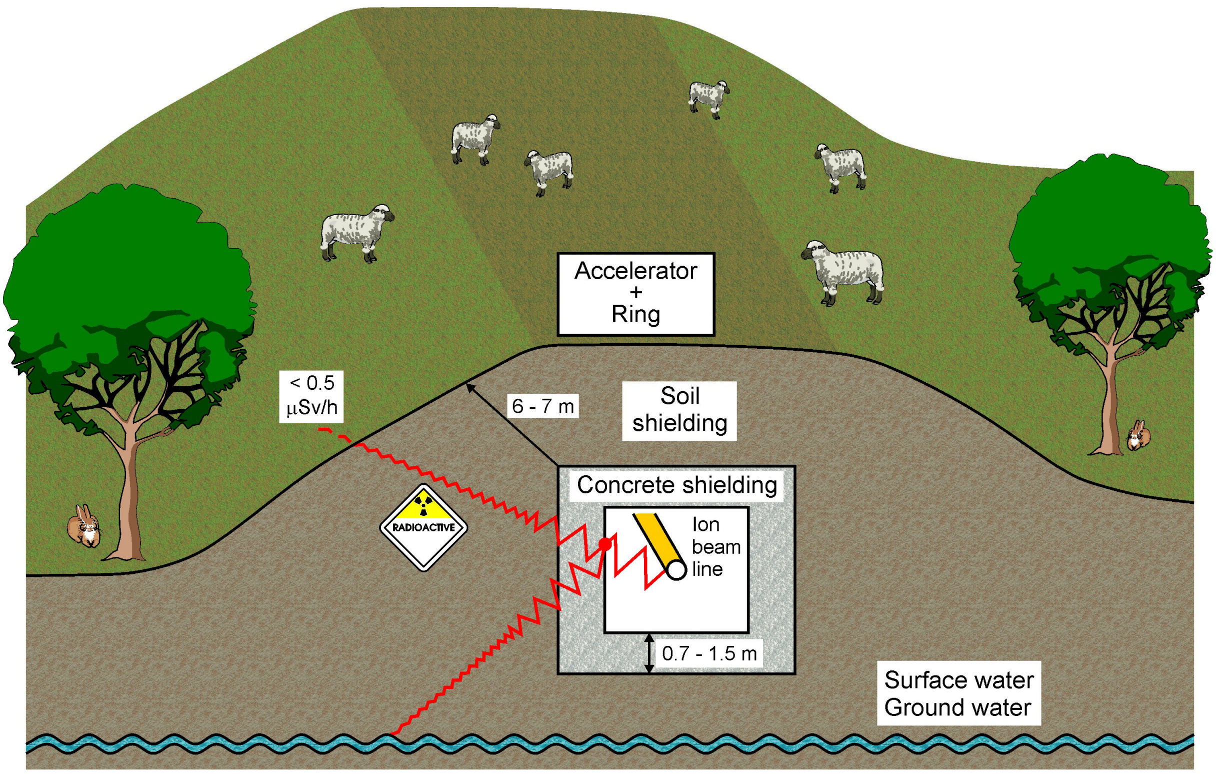
[20] Fertile materials = nuclides like U-238 or Th-232, which produce fissile materials by capture of slow (thermal) neutrons
[21] Proliferation = distribution of fissile material suitable for nuclear weapon production
[22] This means, that accidents by loss of control on decay heat analogous to core melt-down in nuclear facilities are not possible
[23] In contrast to that, non noble target materials (W, Ta, U) may react severely with water or air, which enhances radioactivity release rates
[24] The radiotoxic potential of ESS requires probably an emergency plan with involvement of the public

More detailed safety aspects concerning normal operation, accidents and final waste handling/disposal

Concerning the safety and environmental impact of ESS, normal operation, accidents and final waste handling/disposal have to be treated separately:

For the radioactive burden to the public under ***normal operation*** and in such abnormal conditions, which have to be expected to occur within the lifetime of the facility, EU rules tolerate not more than a dose of 1 mSv/year for a reference person. This maximum tolerable dose has to be compared with the average dose a person obtains by natural sources (1 – >20 mSv/year in Europe, depending on site; average: 2 mSv/year) and civilization induced sources (average: 1 mSv/year in Europe, mainly medical diagnostics). Direct (neutron) radiation from ESS ('skyshine', mainly from accelerator and ring) and emissions via air and water (significant contributions also from target) have to be taken into account within ESS-licensing/authorization in the required proof, that the dose limit is met. Because conservative assumptions in dose calculation for the reference person are accumulated (like the assumption, that this person lives for the whole year unsheltered on the highest exposed place, drinks the highest contaminated water and eats only the highest contaminated food, the expected (realistic) dose for the public is far below 1 mSv/year. Moreover, all radioactive sources in the respective area (except for certain medical facilities) have to be summed up. In addition the ALARA (as low as reasonably achievable) principle has to be fulfilled, which means, that for the reference person the maximum dose may not even reach 1 mSv/year, if that can be avoided by adequate measures. Such measures to be used for fulfilling necessary safety standards are e.g. (see figures 8.9.2.1 and 8.9.2.2):

- shielding against skyshine and against activation of soil/groundwater (see figures 8.9.2.1 and 8.9.2.2)
- gas tight enclosures and filters for reduction of emissions of toxic matter (see figure 8.9.2.2).



(drawing by S. Reiche-Begemann, FZJ)

Figure 8.9.2.1: Shielding of accelerator and ring

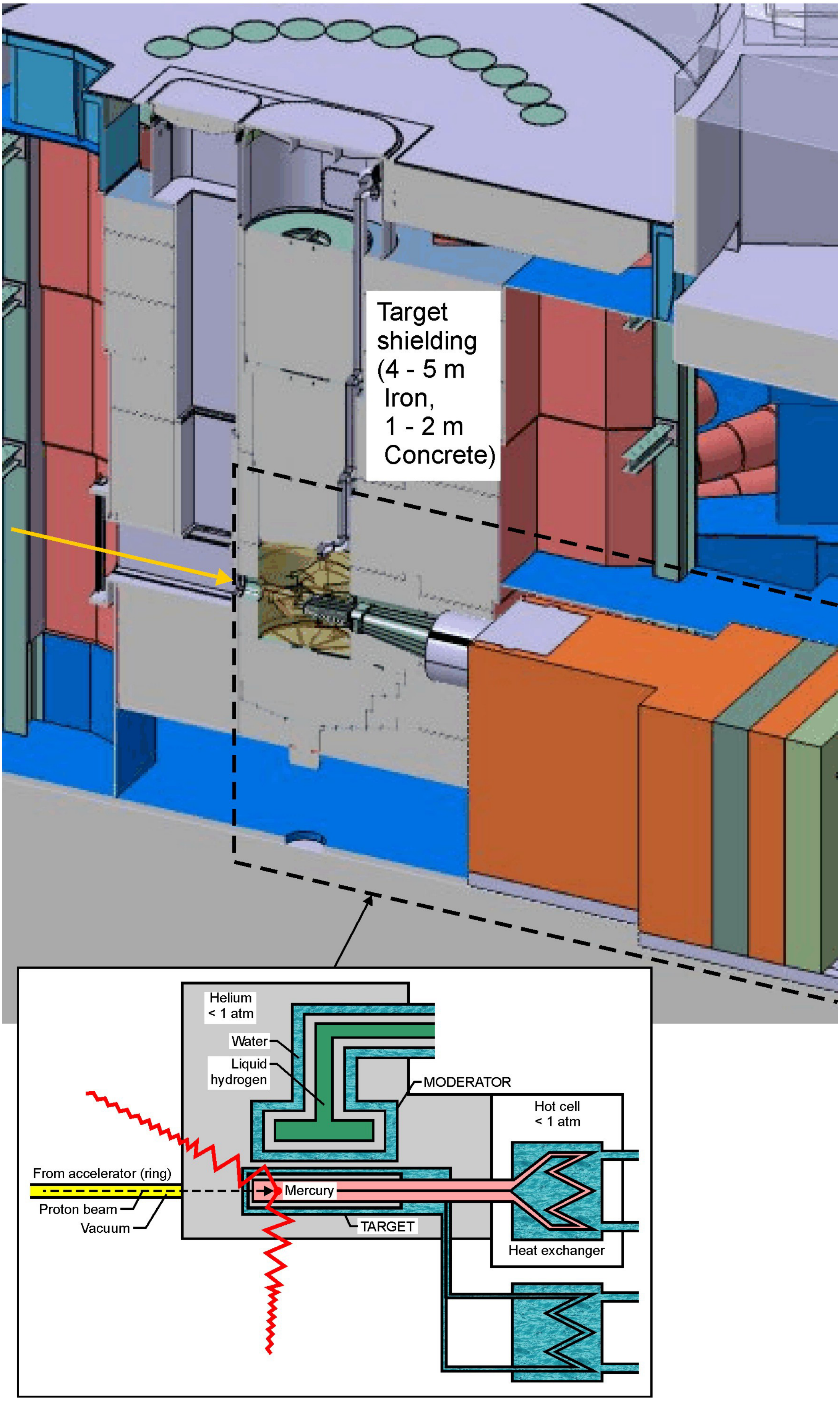


Figure 8.9.2.2: Shielding and radioactivity enclosure measures for the ESS target range

Accidents, which may occur in ESS are restricted mainly to the target; on the one hand they may be initiated by events within the facility itself, like loss of cooling, proton beam mismatch, leaks within target hull or moderator enclosure, internal fire. On the other hand they may be connected to external events like earthquake, airplane crash, meteorite, external fire or gas cloud explosion. Initiating internal events require for evolution of an accident (which means non negligible release of the toxic inventory, non negligible consequences) follow-up failures of safety relevant systems (e.g. proton beam shutdown system, malfunction of ventilation systems/filters); for certain external events of very low probability (like severe earthquakes, greater meteorite) this is not necessarily the case.

Licensing/authorization of ESS in EU requires, that for selected, more frequent accidents (so called design basis accidents DBA)[25] dose limits for the public. These dose limits and the dose calculation methods are still different within EU. For Germany, one important dose limit is 50 mSv effective[26] dose; the dose calculation has to be performed in a similar conservative manner as described above for normal operation. For accidents with pronounced low frequencies (so called 'hypothetical' accidents) no dose limits exist in most EU countries; here, an emergency plan has to be worked out, which defines details of emergency measures for protection of the public (sheltering, evacuation, relocation, food ban etc.).

In order to establish a very high safety standard, ESS goes beyond these guidelines; its safety goals require a design, which fulfills for internal accidents the restrictive DBA dose limits even as deep into the range of 'hypothetical' accidents, as credible safety assessments will allow[27]. This is achieved mainly by passive safety features e.g. by multiple enclosure of toxic/burnable inventories, sufficient fire protecting walls, improvement of the mechanical strength of the shielding in the whole target area for protection against airplane crashes; design of safety relevant active systems (proton beam shutdown) is based on redundancy[28] and diversity[29]; this leads - in combination with absence of criticality and large decay heat power - to the non-appearance of credible internal mechanisms in ESS, which might lead to a complete destruction of the enclosure of toxic materials. This safety goal has to be proven by an adequate probabilistic safety assessment.

Concerning external accidents the same holds for certain events like airplane crashes, external fires and gas cloud explosions. However for natural hazards like very strong earthquakes a proper protecting design is not possible with adequate measures and not even reasonable: Such events cause severe damage of large areas, which is only slightly increased by consequences resulting from ESS. So the risk added by ESS to the total risk of the respective site area remains small.

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[25] At present, DBA cover a frequency range of 10^{-2}/year[25] down to 10^{-5} - 10^{-6}/year (a frequency 10^{-x}/year means: The respective event sequence is expected one time within 10^{x} years). DBA for ESS will be defined by the licensing authorities

[26] The artificial 'effective dose' is composed of all relevant organ doses, which are multiplied with specific weighing factors

[27] Which means in detail down to frequencies of 10^{-7}/year, i.e. for accidents with more than an order of magnitude smaller frequency than required in licensing. For comparison it has to be noted, that catastrophic releases in operating nuclear power plants by core melt-down (combined with costs of up to 6000 billion €) are expected already with frequencies of 10^{-6} – 10^{-7}/year. The frequency limit of 10^{-7}/year for fulfillment of DBA dose criteria was selected for ESS, because it marks the limit for credible probabilistic assessments; it does not mean, that accidents with severe consequences will occur in ESS already at slightly lower frequencies.

[28] Redundancy = several safety systems of the same kind are used in parallel

[29] Diversity = safety systems of different physical principles are used

With respect to ESS ***final waste handling/disposal*** the situation is as follows:

- High active waste[30] results from the target ranges only and is of limited volume (several m³)[31].
- The activity of the target material decreases by about 3 orders of magnitude within 1000 years after ESS shut down[32]; after 3000 years, the activity is mainly due to Holmium-163, which is a minor radiotoxic nuclide.

  Because the content of severe radiotoxic α-emitters is small in the ESS waste and very long-lived and fissile actinides are not present, we are convinced, that the problem of final handling/storage of the ESS target can be solved with adequate effort[33].
- There are considerable amounts of low and medium active waste from ESS to be disposed, too. However, because of the comparatively small lifetime of the respective activity, this is a problem that can easily be solved within established procedures.

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30 High active waste = requires cooling during storage

31 This holds, if after ESS shut down the usual decommissioning period of 5 - 10 years takes place, before dismantling starts.

32 This decay rate is small compared with the fission product decay by 5 – 6 orders of magnitude in the same time period found for waste of research or power reactors. However, the main problem concerning waste disposal of fission reactor systems results not from fission products, but from the presence of ultra long lived, highly toxic (α-emitting) and (partly) fissile actinides. These actinides dominate the waste activity of fission reactor systems for > 500 - 1000 years after shut down, when most of the fission products are gone

33 R&D to final handling/storage of ESS target waste is under way (see 8.7 and 8.10)

8.10 OUTSTANDING AND FUTURE WORK RELATED TO ESS SAFETY

Due to decreasing effort spent to nuclear safety related fields (and nuclear items in general), knowledge conservation in this area is of particular relevance for a future ESS facility. Fortunately, this is achievable to some extent by participation in similar fields like ADS (accelerator driven systems for nuclear transmutation) or EURISOL (European isotope separation). Besides that, some major safety related open questions remain to be answered.

The following topics are intended to be continued in order to allow for a conservation of knowledge concerning ESS safety and to improve the knowledge in some particular safety relevant areas; the first three items have the highest priority:

- ✓ Initiated by the site proposers the national rules concerning licensing of an innovative spallation source like ESS have to be checked with respect to their completeness and, if required, the process for extension of these rules has to be started.
- ✓ Better knowledge about the nuclide vector within proton irradiated mercury is required. Accordingly, experimental examinations on SINQ irradiated mercury are planned in collaboration within PSI. Here, long-lived nuclides will be determined by ICPMS and by γ–spectrometry. Nuclides to be quantitatively determined are (with halve-life in brackets): Ba-133 (10.5 y), La-137 (60000 y), Pm-145 (17.7 y), Gd-148 (74.5 y), Eu-150 (37 y), Ho-163 (4500 y), Hf-172 (1.8 y), Pt-193 (50 y), Hg-194 (512 y), Au-195 (0.5 y). Similar examinations will be performed on irradiated tungsten, which is an alternative target material, too. Besides that, theoretical studies concerning the completeness of the respective nuclide vectors will be performed and discrepancies between SNS and ESS calculations on the tritium amount in the targets have to be examined.
- ✓ Final disposal of irradiated mercury requires imperatively its solidification, perhaps after separation of the low volatile spallation products. Altogether, this is a major, costly task, whose chemical engineering aspects should be examined in detail beforehand as part of the FP6; separation of low volatile nuclides by distillation before of final disposal will be examined, too.
- ✓ The examinations on ground water/soil activation described in detail in chapter 8.4 have to be finished. This will be done as part of the FP6 contract SAFERIB (Safety of radioactive ion beam facilities), and perhaps in FP6 EURISOL, too. This task is of advantage for all kinds of ion accelerators.
- ✓ Realistic estimations of ESS accident consequences are part of a FP6 proposal; in detail consequence estimations for innovative nuclear related systems (Modular HTR, Fusion reactor, ADS, ESS), using the established EURATOM code COSYMA, are planned; this requires an upgrading of the COSYMA-Code.
- ✓ Conservation of knowledge concerning tools and procedures of safety analyses for ESS is aspired by participation of members of the ESS safety task within the safety examinations for ADS systems and for EURISOL (respective EU FP6 proposals)

Several safety related subtasks should be initiated before an ESS licensing process starts. These subtasks were not selected to be performed within the limited effort of the safety task or to be started within EU 6th framework program (FP6), because these items are not ESS specific and/or deep knowledge is already available or large resources are required:

- ✓ Complete Probabilistic Risk Analysis (PRA) for ESS, using the consequence estimation code described below
- ✓ Studies on ESS behaviour in severe external accidents (severe earthquake, airplane crash)

- ✓ Design studies for ESS safety systems:
  - o Radiation safety interlock system (radiation monitors and machine interlocks), for the Accelerator Personnel Safety System, for the Target rooms Personnel Safety System and for the Experimental Stations Personnel Safety System
  - o Definition of a dosimetry system (environmental, personnel, operational)
  - o Design of active air/water, filtration, containment and conditioning systems.
  - o Design of active handling systems for maintenance and repair of machine components and conditioning of routine/decommissioning wastes.
  - o Design of a gas handling system for (radioactive) gases produced in the target
- ✓ Calculations on activation of cooling water and air in accelerator and target building and resulting doses (Similar to ISIS as beam losses will be the same)
- ✓ Activation studies on accelerator compounds (including dose rates and doses to workers) and selection of suitable materials (Similar to ISIS as beam losses will be the same)
- ✓ Calculations on the formation of chemical toxic gases (ozone, nitrogen oxides) in presence of irradiation

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