Preliminary safety analysis of the Gorleben site

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Introduction

The safety requirements governing the final disposal of heat-generating radioactive waste in Germany were implemented by the Federal Ministry of Environment, Natural Conservation and Nuclear Safety (BMU) in 20101. The Ministry considers as a fundamental objective the protection of man and the environment against the hazards of radioactive waste. Unreasonable burdens and obligation for future generations shall be avoided. The main safety principles are concentration and inclusion of radioactive and other pollutants in a containment-providing rock zone. Any release of radioactive nuclides may increase the risk for men and the environment only negligibly compared to natural radiation exposure. No intervention or maintenance work shall be necessary in the post-closure phase. Retrieval/recovery of the waste shall be possible up to 500 years after closure.

The Gorleben salt dome has been discussed since the 1970s as a possible repository site for heat-generating radioactive waste in Germany. The objective of the project preliminary safety analysis of the Gorleben site (VSG) was to assess if repository concepts at the Gorleben site or other sites with a comparable geology could comply with these requirements based on currently available knowledge (Fischer-Appelt, 2013; Bracke, 2013). In addition to this it was assessed if methodological approaches can be used for a future site selection procedure and which technological and conceptual considerations can be transferred to other geological situations.

The objective included the compilation and review of the available exploration data of the Gorleben site and on disposal in salt rock, the development of repository designs, and the identification of the needs for future R&D work and further site investigations.

Structure of the VSG

The VSG was composed of four main working topics:

1. Fundamentals: This topic included the description of the geological site and its future evolution over one million years; further, an inventory of the waste that could presumably be emplaced in a repository at the Gorleben site according to the current situation in Germany with its phase-out of nuclear energy (June 2011), and, finally, generation of a concept to accomplish radiological safety and to demonstrate its compliance with the safety requirements (BMU, 2010).
2. Based on these fundamentals, repository concepts were developed aiming toward operational safety, long-term safety and retrieval/recovering of the waste. Two emplacement variants, namely storage of spent fuels in drifts or in boreholes,

different types of canisters (POLLUX®, CASTOR®, BSK3R) and one optional variant emplacement for non-heat-generating waste were projected.

3. The analysis of the repository system was based on these concepts. The features, events and processes were compiled and described, then used to derive scenarios and to assess the probability of the evolution of the system. Geomechanical analyses investigated the integrity of the geological barrier (containment-providing rock zone) for 1 million years considering external and internal events and processes such as glaciation, decay heat or gas generation. Similarly, the seals for shafts and drifts were designed and analysed. The radiological consequences were analysed by numerical models for the transport of the liquid and gas phase (two-phase transport) in the long-term safety analysis.

4. The feasibility of the repository system as a containment system for radionuclides and the methodology to show compliance with the safety requirements were assessed. The uncertainties, which e.g. result from the incomplete geological exploration of the Gorleben site and which require additional R&D, were shown (Fischer-Appelt, 2013). The methodological approaches were discussed for their suitability to compare repository sites and their technical transferability to repository sites in other geological formations.

Geology

The Gorleben salt dome is 4 km wide and nearly 15 km long (Bornemann, 2011). It is composed of different salt rock types of the Zechstein (Upper Permian) series and extends to the Zechstein basin to a depth of more than 3 km. In the course of salt dome formation, the salt was moving several kilometres. During the uplift of the salt the initially plane-bedded strata of the Zechstein series were extensively folded. In the core of the salt dome the “Hauptsalz” sequence, which is characterised by a particularly high creep capacity, forms a homogeneous halite body with a volume of several cubic kilometres. The Hauptsalz contains gaseous and liquid hydrocarbons in separated zones of decimetre to metre dimensions. The overall hydrocarbon content is far below 0.01 weight-%. At the flanks, the salt dome consists of salt rocks with lower creep capacities. Brine reservoirs with fluid volumes in the range of litres to hundreds of cubic metres may exist in certain regions of this part of the salt dome. The water content of the Hauptsalz is below 0.02 weight-%. Interconnected pores do not exist in the salt rock outside of fluid-bearing or fractured areas, i.e. the salt rock is impermeable. The exploration of the Gorleben site as a potential site for a HLW repository started in 1979.

Based on this data a prognosis of the future evolution of the site was performed (Mrugalla, 2011). Geological and climatic features, events and processes were considered. The tectonic and volcanic activity, diapirism, subrosion, hydrology and climate were described and grouped into probable and less probable evolutions. Possible sequences of future glaciations were deduced from geological history.

Waste

The heat-generating radioactive waste will be composed of irradiated fuel elements from power reactors, vitrified reprocessing waste and irradiated fuel elements from research and prototype reactors. As an option, negligible heat-generating waste was also considered to be disposed to assess the feasibility of joint disposal in a separate area of the repository. A hypothetical amount and composition of this waste was assumed. It included depleted uranium tails from enrichment (about 35 000 m^3), graphite (about 1 000 m^3) and mixed waste (about 15 000 m^3) (Peiffer, 2011).

Safety concept

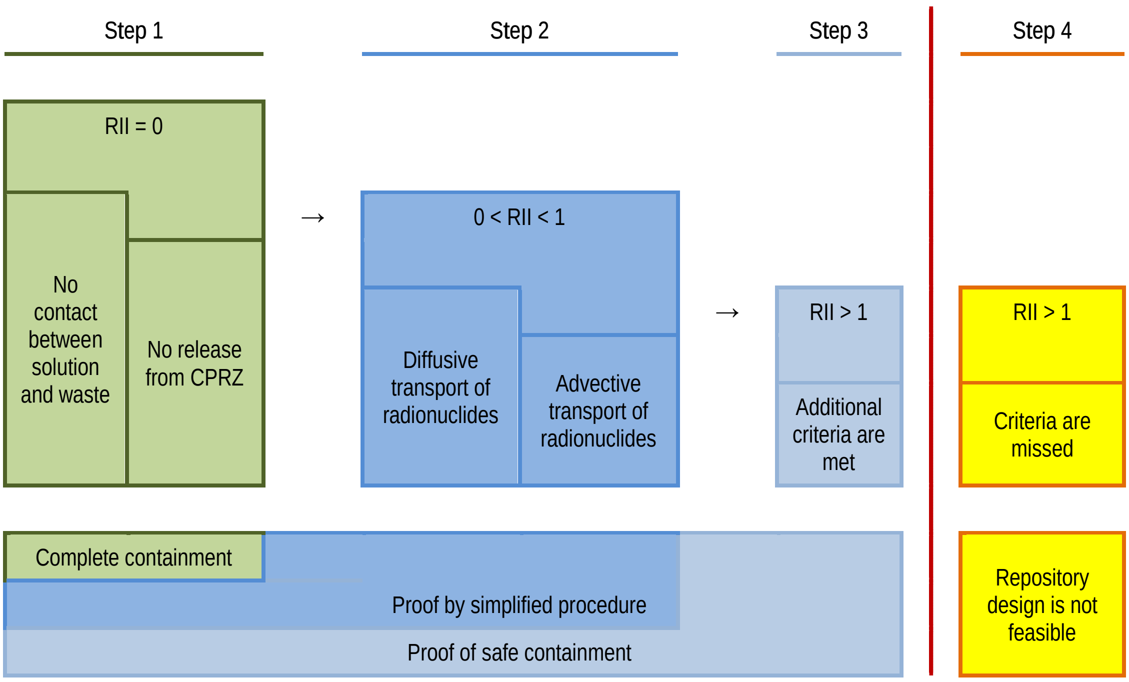
In due consideration of the German safety requirements (BMU, 2010) the safety concept for the project VSG was based on the following principles (Mönig, 2012):

- Radioactive waste must be contained in a containment-providing rock zone (CPRZ), i.e. a part of the host-rock enclosing the repository jointly with geotechnical barriers.
- Containment shall be effective immediately after closure.
- Containment must be provided by the repository system permanently and maintenance-free.
- Intrusion of brine to the waste forms shall be prevented or limited.

According to the safety requirements a site for disposal of heat-generating radioactive waste is only suitable if a sufficiently large containment-providing rock zone is available. Its integrity must be ensured for 1 million years and a robust, staggered, maintenance-free multi-barrier system from technical components (container, drift seal, shaft seal,...) must be developed, which prevents an unacceptable release of radionuclides over the short and long terms. This principle is also applicable if a barrier fails partially. Safe containment has to be demonstrated for probable and less probable evolutions of the site, while evolutions with very low probability (less than 1% over the demonstration period of 1 million years) need not to be considered. Criticality must be excluded in all phases of the repository development.

The evaluation of the safety of the system includes the assessment of probabilities and consequences. The proof of compliance with the safety requirements considers four steps and applies an indicator called the Radiological Insignificancy Index (RII) (Figure 1). The total release of radionuclides from the containment-providing rock zone and technical barriers is used in a generic model for radiation exposure. The RII is then calculated as a ratio to a radiation dose, which is considered insignificant (Mönig, 2012).

Figure 1: Radiological Insignificancy Index (RII)



The four steps considered in the RII are as follows:

- Complete containment is provided if there is no contact of the waste with solutions and no gaseous radionuclides are released from the containment-providing rock zone (RII = 0).
- Safe containment of radionuclides is achieved if the RII is greater than 0 and is less than 1. A simplified procedure is sufficient proof. The assessment distinguishes between a release by diffusion or advection.
- If the RII is greater than 1 additional criteria must be met. Additional criteria refer to the individual radiation dose and are related to the probability of the evolution of the system (scenarios). A more detailed procedure is required.
- If these additional criteria are unfulfilled the designed repository is not feasible. Safe containment cannot be provided by the repository concept. The design of the repository has to be changed and assessed once again. If all possible measures are optimised and safe containment still cannot be demonstrated, the site is not suitable.

Uncertainties and assumptions

There are some uncertainties for the Gorleben site, which cannot currently be further reduced or even eliminated due to the present status of knowledge, e.g.:

- The total lateral size of the salt dome is not yet known.
- The features of the salt rock are known only for the explored area.
- The extension of the Hauptsalz may not be large enough for all designed repository concepts, including the required safety distance to adjacent rock layers.

Therefore assumptions have been made which should be verified in the future. The main assumptions are:

- The lateral size of the salt dome is in accordance with the geological sketch of Bornemann (2011).
- The known features from the salt rock currently under exploration can be extrapolated to the entire area necessary for the whole repository.
- The extension of the Hauptsalz is sufficiently large for all designed repository concepts including the required safety distance to adjacent rock layers.

Geotechnical measures shall provide long- and short-term barriers. The long-term barrier is the backfill with salt grit. Its initial high porosity and permeability is reduced continuously by compaction. This is a time-dependent process and re-establishes the features of the undisturbed rock salt within the lifetime of the short-term barriers.

The short-term barriers are drift and shaft seals. These are composed by layers of different material providing diversity and redundancy. The failure of a drift or shaft seal is regarded as a less probable scenario.

Additional uncertainties concern data, parameters and models. These are dealt with in deterministic model calculations using bandwidths.

Repository concepts

The repository concepts for the Gorleben site are described for three emplacement variants (Bollingfehr, 2012):

- *Variant A*: As an option, non-heat-generating radioactive waste was emplaced in a separate area of the repository. This variant was combined with the following variants for spent fuel (Figure 2).
- *Variant B*: Emplacement of heat-generating radioactive waste (spent fuel and vitrified waste) in self-shielding waste containers (POLLUX[ã] casks) in horizontal drifts (Figure 2). As an alternative, the emplacement of heat-generating radioactive waste in transport and storage casks (CASTOR[ã]) in horizontal boreholes was considered, although this required an enhanced technical design for shafts and underground transportation.
- *Variant C*: Emplacement of heat-generating radioactive waste in multi-purpose conical overpacks (Figure 3) in deep vertical boreholes.

Figure 2: Repository design and layout (combination of Variants A and B)

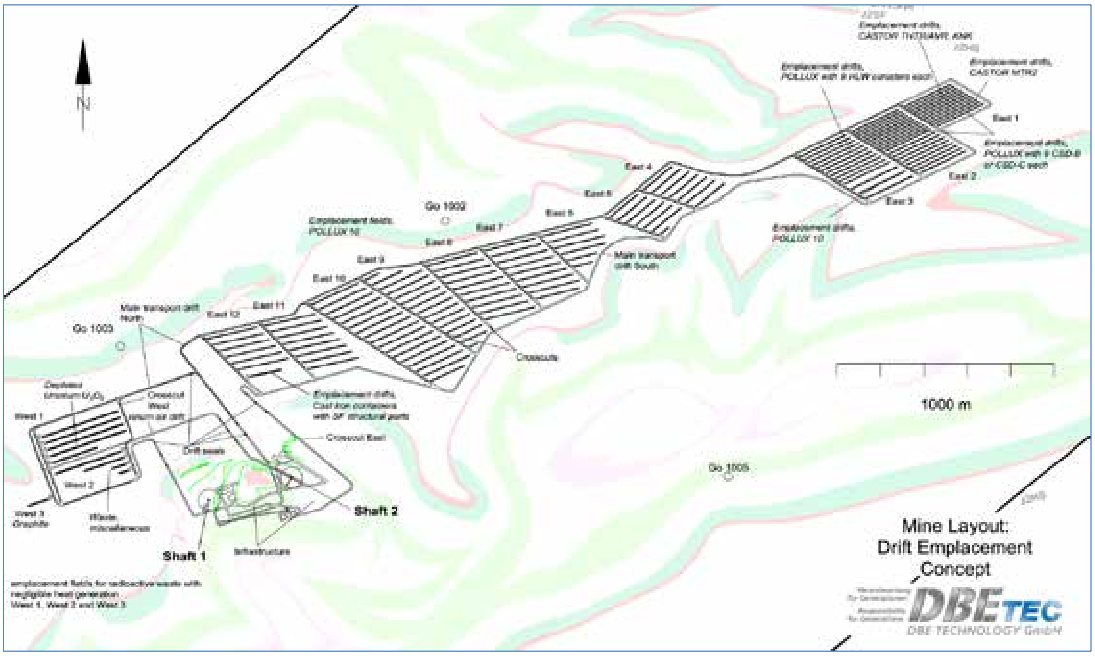
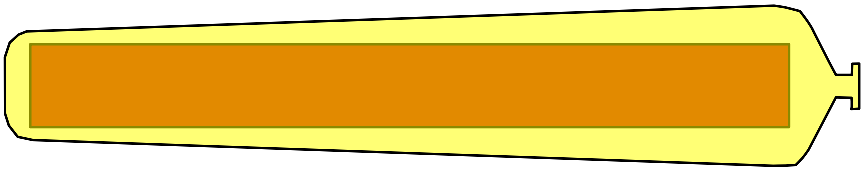


Figure 3: Conical overpack (Variant C)



The overall layout was optimised to minimise the size of the repository but also to comply with temperature criteria. The technical installation and casks/containers were selected to ensure manageability, radiation protection and operational safety. Disposal was planned using retreat working.

The technical solutions for retrieval of overpacks in Variant C were projected for the first time. A steel tubing (liner) of 300 m length was foreseen for disposal of the conical overpacks in boreholes. The void space in the liners was planned to be filled with dry quartz sand. The shape of the overpacks for spent fuel and flasks were designed conically to facilitate retrieval using vibration.

The sealing system for drifts uses sorel concrete and salt grit as backfill. Salt grit with a higher moist content is used as backfill for the main drifts to enhance compaction while for the emplacement fields salt grit with natural moist content is foreseen. The shaft seals are composed of bentonite, salt concrete and sorel concrete.

Scenario analysis

The site and the repository system will undergo exactly one evolution, which will be governed both by climatic and geological processes at the site and processes induced by the repository construction and the emplacement of heat-generating waste. This evolution cannot be predicted in all details.

A novel scenario development methodology was developed in the project VSG910. It aims at deriving one reference scenario for each repository design (horizontal drift/ borehole emplacement) and a number of differing alternative scenarios. At large, the scenarios shall comprehensively represent the range of possible repository system evolutions. The methodology allows the straightforward assignment of probability classes to the scenarios according to the regulatory framework (BMU, 2010). The individual scenarios are described by features, events and processes (FEP) that determine the future evolution of the final repository system at Gorleben. FEP may initiate or influence other FEP, be influenced by or result from other FEP (Wolf, 2013). These interdependencies were used to derive scenarios systematically. The reference scenarios were derived from probable FEP and basic assumptions. The alternative scenarios were generated from violation of assumptions, from less probable FEP and from probable FEP with less probable parameter values. The human intrusion scenario was reviewed separately (Beuth, 2013b).

System analysis

The system analysis addressed the following general questions:

- Will the integrity of the geological salt barrier remain intact under the expected loads, e.g. like heat generation or glaciation?
- Is there any flow of brine to the emplacement areas?
- Are radionuclides released from the containment providing rock zone?
- If so, what radiological consequences have to be expected?

Based on these scenarios an analysis of geomechanical and geotechnical integrity was performed (Kock, 2012; Müller-Hoeppe, 2012). A demonstration of integrity is required for probable scenarios (BMU, 2010). The integrity must be checked for less probable scenarios and their radiological consequences analysed. The final step was the assessment and synthesis of the results.

The dilatancy criterion and the fluid pressure criterion are the main criteria to assess the geomechanical integrity of the rock salt barrier. The dilatancy criterion specifies that no damage to the rock fabric (e.g. induced cracking or the interlinking of intercrystalline pore space) may occur in response to deviatoric stresses. The damage process is associated with dilatancy, i.e. an increase in volume caused by the development of micro-cracks and crack accumulations.

The fluid pressure criterion specifies that the smallest formation stress (considering compressive stresses as positive) in the barrier, plus any tensile strength which may be present, must be larger than the fluid pressure at a given depth. If this criterion is satisfied, fracturing of the host rock by fluid-pressure-driven penetration of fluids into the rock can be excluded.

The integrity of the salt barrier is only ensured if both criteria are satisfied in a sufficiently large zone around the underground workings of the repository. Linked flow paths from the water-bearing horizons in the overburden down to the emplacement zone, as well as release of hazardous substances from the repository itself (e.g. due to generation of a gas pressure) can then be excluded from a geomechanical point of view.

The mechanical and thermo-mechanical simulations carried out using a range of codes and material laws produced the following results and conclusions:

- The emplacement of heat-generating waste heats up the salt dome over a large volume, but the thermally-induced stresses and deformations do not generate any continuous migration paths.
- The highest thermo-mechanical stresses affecting the salt barrier occur within the first hundred years after sealing the geologic repository. Any loss of integrity of the barrier becomes even less likely in the subsequent time period. Mechanical damage caused by exceeding the dilatancy limit only affects the rock zones directly adjacent to the underground cavities within a few decimetres up to 3 metres and rock zones, which are restricted to the distant top salt zone. These rock zones at the salt top are of no importance with respect to the integrity of the salt barrier, which constitutes the containment-providing rock zone around the emplacement fields.
- The thermo-mechanical stresses calculated for the borehole emplacement design are higher than those calculated for the drift emplacement concept because the heat is released in a smaller and differently shaped volume.

The integrity of the geotechnical barriers (drift and shaft seal) was demonstrated by numerical calculations concerning geological, thermal and geochemical impacts during their lifetime and by providing redundant and varying types of sealing systems in combination.

Long-term safety assessment

For the analysis of the radiological consequences (Larue, 2013) radionuclide transport was modelled using a two-phase model, TOUGH2 (Pruess, 1999), and a one-phase model, MARNIE (Martens, 2002). The layout of the repository for the drift emplacement concept was transferred into a 3-D grid for TOUGH2 and a 1-D grid for MARNIE. This included simplification steps due to constraints of the codes.

Even a failure of a single seal did not result in advective flow of brine within or into the emplacement fields. Furthermore the salt grit is compacted in a relatively short time when applying conservatively selected parameters according to experimental data for modelling of the compaction process.

A conservative assumption was used for the final salt grit porosity after compaction. Using the 1-D grid no radionuclide transport by advection was detected in the liquid phase beyond the containment-providing rock zone (CPRZ). As a consequence any radionuclides in the liquid phase were transported by diffusion only. The release of radionuclides is higher through the eastern drift seal (Figure 2, close to cross-cut east) than through the western seal since it is closer to the disposal fields for spent fuel. The absolute value is a few Bq/a.

The radionuclide transport and release via the gas phase from the CPRZ is relevant up to some hundred years after closure in two-phase model calculations with TOUGH2. The RII at the drift seal exceeds in some model cases (BMU, 2010). The compaction of salt grit and metal corrosion with gas generation are driving forces on the transport and release of gaseous radionuclides (e.g. ^{14}C as methane from structural parts) through a drift seal. No ^{14}C was released via transport in the gas phase through a shaft seal.

Although many of the parameters for the MARNIE and TOUGH2 calculations were conservatively selected, they should be improved by future R&D work to improve the coverage of non-linear interactions. These are, e.g.:

- the compaction rate;
- the advective and diffusive transport parameters of radionuclides at low salt grit porosities;
- the diffusion coefficient in high compacted salt grit;
- the solubility limits for some radionuclides;
- the release of radionuclides into salt grit;
- the formation of gaseous radionuclides.

Synthesis of the project results

The safety concept, generated during the course of the project, was suitable to demonstrate its compatibility with the safety requirements. The generated design of the repository system was feasible and also complied with safety requirements. Nevertheless some assumptions were necessary. The assumptions refer to the status of geological exploration, the reliability of construction and some inherent uncertainties. If these assumptions are met, the designed repository system is assessed to be robust.

Optimising strategies regarding the repository design are conceivable. A repository layout such as placing the structural components farther away from the drift seals would likely result in a lower ^{14}C flow through the drift seals. Furthermore, implementing a void volume as a sink (e.g. an infrastructure area backfilled with gravel) might hinder any gas flow through the shaft seals. Further, the use of gas-tight casks for the structural components (like POLLUX®) could confine volatile radionuclides for decades or centuries.

Some conclusions were:

- The possible release of gaseous radionuclides, the two-phase flow processes and the subsequent model for radiation exposure require additional R&D.
- The containment-providing rock zone can be minimised by an iterative process but has to be preliminarily assessed as an initial guess.
- The handling of combinations of less probable but interdependent FEP has to be improved.
- The mobilisation of other pollutants and the heating of groundwater should be observed and requirements should be defined.
- The complete, though preliminary, safety analysis for the Gorleben site identified important tasks for research and development. This would not have been possible with a generic safety analysis.
- A safety analysis should be repeated at selected time intervals.
- Indications were given that a repository in salt rock is feasible.

Applicability to a site selection process

During the course of the project the request emerged in politics to establish a transparent, stepwise site selection process in Germany. Alternative sites should be identified and explored in addition to the Gorleben site. A bill was introduced for a site selection process and adopted in July 2013. This bill foresees a development of site selection criteria and standards for site comparison, which should be developed for different steps of the process. Safety analyses are foreseen to evaluate the results of surface and sub-surface explorations of possible sites.

The results of the VSG are valid for a repository site in salt rock. The methodology can be transferred to sites in other geological formations but requires a concrete concept for disposal and closure to perform a safety assessment and comparison of sites.

References

Beuth, T., *et al.* (2013a), *Szenarienentwicklung: Methodik und Anwendung, Bericht zum Arbeitspaket 8*, GRS-284, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-60-1.

Beuth, T., *et al.* (2013b), *Untersuchungen zum menschlichen Eindringen in ein Endlager, Bericht zum Arbeitspaket 11, Vorläufige Sicherheitsanalyse für den Standort Gorleben*, GRS-280, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-56-4.

Bollingfehr, W., *et al.* (2012), *Endlagerauslegung und –optimierung, Bericht zum Arbeitspaket 6*, GRS-281, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-57-1.

Bornemann, O., *et al.* (2011), *Description of the Gorleben Site, Part 3: Results of the Geological Surface and Underground Exploration of the Salt Formation*, Bundesanstalt für Geowissenschaften und Rohstoffe, Hannover, ISBN 978-3-9813373-6-5.

Bracke, G., K. Fischer-Appelt, B. Baltes (2013), "Preliminary Safety Analysis of the Gorleben Site: Overview – 13298", *WM2013 Conference*, Phoenix, AZ, USA, 24-28 February 2013.

Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU) (2010), *Sicherheitsanforderungen an die Endlagerung wärmeentwickelnder radioaktiver Abfälle (Stand: 30.09.2010)*, BMU, Bonn, Germany.

Fischer-Appelt, K., *et al.* (2013), *Synthesebericht für die VSG, Bericht zum Arbeitspaket, 13, Vorläufige Sicherheitsanalyse für den Standort Gorleben*, GRS-290, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-66-3.

Kock, I., *et al.* (2012), *Integritätsanalyse der geologischen Barriere, Bericht zum Arbeitspaket 9.1*, GRS-286, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-62-5.

Larue, J., *et al.* (2013), *Radiologische Konsequenzenanalyse, Bericht zum Arbeitspaket 10, Vorläufige Sicherheitsanalyse für den Standort Gorleben*, GRS-289, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-65-6.

Martens, K-H., H. Fischer, P. Romstedt (2002), *Beschreibung des Rechenprogrammes MARNIE. 135 Seiten*, GRS-A-3027, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany.

Mönig, J., *et al.* (2012), *Sicherheits- und Nachweiskonzept, Bericht zum Arbeitspaket 4*, GRS-277, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-53-3.

Mrugalla, S. (2011), *Geowissenschaftliche Langzeitprognose, Bericht zum Arbeitspaket 2*, GRS-275, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-51-9.

Müller-Hoeppe, N., *et al.* (2012), *Integrität geotechnischer Barrieren –Teil 1: Vorbemessung. Bericht zum Arbeitspaket 9.2, Vorläufige Sicherheitsanalyse für den Standort Gorleben*, GRS-287, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-63-2.

Peiffer, F., *et al.* (2011), *Abfallspezifikation und Mengengerüst, Bericht zum Arbeitspaket 3*, GRS-278, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-54-0.

Pruess, K.O. and G. Moridis (1999), *TOUGH2 User's Guide, Version 2.0*, LBNL-43134, Earth Sciences Division, Lawrence Berkeley National Laboratory, University of California, Berkeley, California, USA.

Wolf, J., *et al.* (2013), *FEP-Katalog für die VSG – Konzept und Aufbau, Bericht zum Arbeitspaket 7, Vorläufige Sicherheitsanalyse für den Standort Gorleben*, GRS-282, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Cologne, Germany, ISBN 978-3-939355-58-8.