Requirements and Concepts for Waste Containers for Heat-Generating Radioactive Waste and Spent Fuel in Rock Salt, Claystone, and Crystalline Rock (acronym: KoBrA) – WM21211a

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ABSTRACT

With the Repository Site Selection Act (StandAG) as of summer 2017 a new procedure for siting a final repository for high-level radioactive waste (HLW) including spent nuclear fuel has been established in Germany. In this context, all three potential host rock formations (rock salt, claystone and crystalline rock) will be taken into account.

A safety concept for a HLW repository relies on the containment and retardation capabilities of natural barriers (host rock, overburden), geotechnical barriers (drift and shaft seals), and technical barriers (containers). The containers have to provide safety functions (containment, shielding, sub-criticality and sufficient decay heat dissipation of the radioactive inventory) during all handling procedures until disposed is completed. Subsequently, containers have to provide these safety functions depending on the geologic boundary conditions and on the design criteria, as well as during possible future retrieval and recovery operations. In June 2017, the German Ministry for Economic Affairs and Energy (BMWi), and the Project Management Agency Karlsruhe (PTKA) thus launched the R&D project "Requirements and Concepts for Waste Containers for Heat-Generating Radioactive Waste and Spent Fuel in Rock Salt, Claystone and Crystalline Rock" (acronym: KoBrA) as a common undertaking of the Bundesanstalt für Materialforschung und-prüfung (BAM) and BGE TECHNOLOGY GmbH.

Based on a tailored top-down approach, the KoBrA project delivered systematically derived requirements for HLW containers. Existing legislation and technical guidelines were taken into account as well as the international state of the art in HLW waste container design. In addition, the main safety functions of a container were taken into account during repository operation on the surface and underground and in the long term. Furthermore, exemplary container concepts were derived under consideration of the host rock-specific requirements and impacts using the methodological approach developed in the KoBrA project.

The presented results of the KoBrA project are summarised and published in a synthesis report [11] and the outcomes from the work packages ascribed to the project's sub-goals are described and published in detail in four appendices [12], [13], [14], and [15]. The synthesis report is available in English as well [16].

MOTIVATION AND OBJECTIVES

According to the Site Selection Act of 2017 [1], rock salt, claystone, and crystalline rock are to be considered as potential host rocks when selecting a site for a repository for high-level radioactive waste (heat-generating radioactive waste and spent fuel assemblies) in Germany. The selection of a specific repository site also depends of the design and evaluation of different repository systems that appear suitable for the respective host rocks. A central component of these repository systems are the waste containers, which have to provide both containment of the radioactive waste and radiation protection, and thus have a decisive influence on the demonstration of operational and long-term safety. In addition, the container design determines the boundary conditions for the transport and emplacement technologies. When developing container concepts, it is thus necessary to systematically derive all the requirements to be met.

Until the early 2000s, the development of repository concepts and waste containers for heat-generating radioactive waste and spent fuel in Germany was carried out with a focus on Gorleben as potential with rock salt as the host rock formation. The waste container development was performed within the framework of R&D work carried out by the Federal Ministry for Economic Affairs and Energy (BMWi) in co-operation with the nuclear industry. Thus, a self-shielding, so-called POLLUX® cask, was developed. A prototype was built, and the associated transport and emplacement equipment manufactured and successfully tested in demonstration experiments within the scope of the R&D project "Direct Disposal of Spent Fuel Assemblies (DEAB)" [2]. While maintaining the same level of safety, an alternative waste container was developed, called BSK-3 (unshielded fuel rod canister). The BSK-3 was designed for emplacement into deep vertical boreholes. Again, the transport and emplacement equipment for this container was designed, manufactured, and successfully tested in demonstration experiments using a dummy. Both container types were used as a design basis for the repository concept developments, mine working layouts, and preliminary safety analyses carried out in Germany to date. Furthermore, within the scope of a feasibility study for the nuclear industry, investigations showed that the emplacement of dual purpose casks of the CASTOR® type (for transport and interim storage) is technically feasible. In several research projects on repository designs in claystone and crystalline rock, the container concepts developed for a repository in rock salt (POLLUX® and BSK-3) have so far been used as a design basis since the 2000s. Thus, a need to develop waste container concepts tailored for the other two considered types of host rock was obvious.

In June 2017 the German Ministry for Economic Affairs and Energy and PTKA consequently launched the R&D project KoBrA as a common undertaking of BAM and BGE TECHNOLOGY GmbH.
The main objective of KoBrA was to systematically derive requirements for waste containers for the disposal of heat-generating radioactive waste either in rock salt, claystone, or in crystalline rock for the very first time in Germany. The following sub-goals were defined:

- Review of the international state of the art on requirements, concepts, and demonstration methods for waste containers
- Systematic determination of container-relevant boundary conditions and load parameters
- Systematic derivation and compilation of requirements for waste containers
- Development of proposals for possible waste container concepts

METHODOLOGICAL APPROACH FOR THE DERIVATION OF CONTAINER REQUIREMENTS

The methodological approach to identify qualitative as well as quantitative waste container requirements applied in the KoBrA project was a holistic one. By means of a top-down approach (the structure and process for which is shown in Figure 1), the requirements for waste containers and the functions necessary for their fulfilment were derived.

In a first step, the relevant national and international laws and regulations were taken as a basis. From these regulatory requirements, generic, i.e. abstract or general container requirements, were developed for the distinguishable phases of utilisation for the waste containers in the repository. The result was the specification of use- or phase-dependent generic container requirements. From these, in turn, the container functions necessary to fulfil the phase-dependent requirements were derived. For the three host rocks to be considered, the possible impacts on the containers for all phases of the disposal process were determined and represented as phase-dependent load parameters and boundary conditions. Taking into account the possible phase-dependent impacts, generic container concepts were developed that provide the container functions required to meet the phase-dependent requirements.

In the second step of the top-down approach, generic repository systems and associated safety and safety demonstration concepts were included for the three host rock formations in question. This way, both the container functions necessary for meeting the requirements and the impacts for the respective host rock could be specified in more detail. This way, specific container concepts for the individual host rocks could be developed based on the generic container concepts from the previous step.

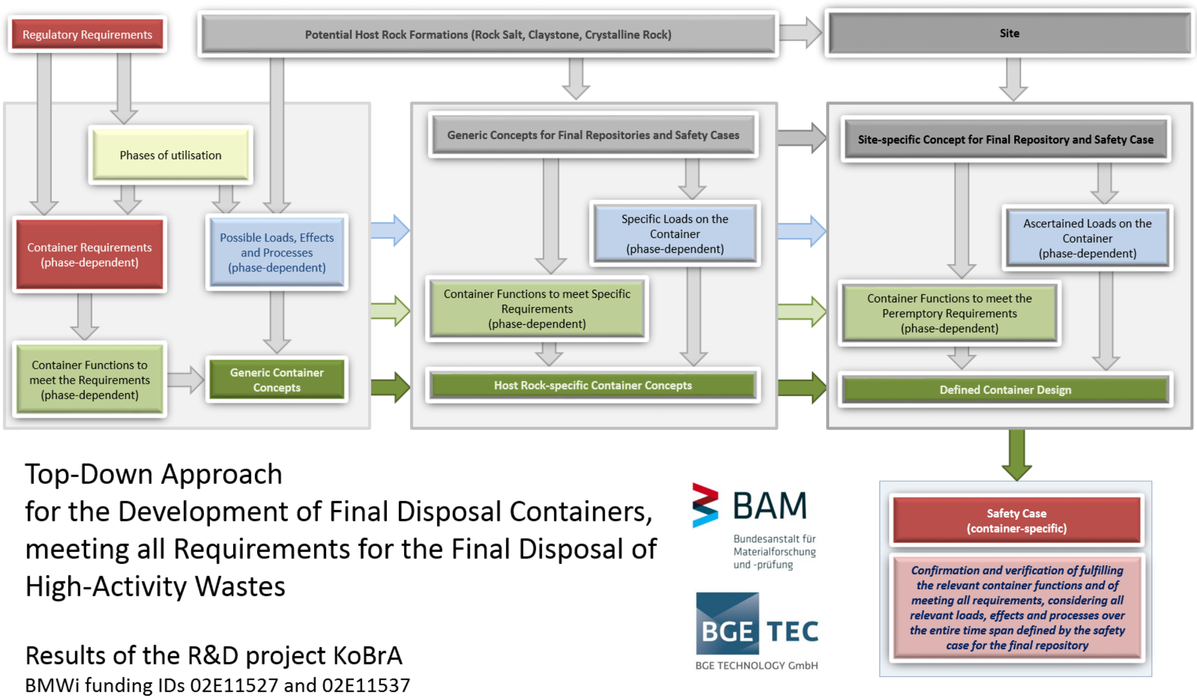


Fig. 1. Top-down Approach for the Development of Containers for High-Level Radioactive Waste [11]

Finally, in the third and last step of the top-down approach, the repository site identified in the site selection procedure with its specific properties and characteristics will have to be taken into account. Based on the specific safety and safety demonstration concept to be developed for this site, the quantified, site-related impacts and the container functions necessary to meet the quantified, site-related requirements, the actual container design for the repository site can be developed. Subsequently, a container-specific safety demonstration for the container design has to be carried out. Here, the compliance of all relevant container functions to meet all requirements must be demonstrated for the relevant impacts over all phases of the repository lifetime.

WASTE CONTAINERS IN INTERNATIONAL REPOSITORY PROGRAMMES

One pillar of the KoBrA-project was a comprehensive literature review in order to compile the national and international status of waste container requirements and concepts for containers for high-level radioactive waste in different host rock formations. The relevant information was recorded, evaluated, and compiled in a set of documents. These documents served as information and reference source for the entire project and beyond. Both the categories and the source base for the literature search were selected according to a top-down approach. The main criterion for the selection of documents was the content of waste container-relevant information. More than 7,200 documents were collected, of which more than 2,100 were directly relevant to waste containers. As source basis for this document collection, the nuclear energy programmes of altogether more than 50 countries were considered, of which 45 countries are implementing or planning waste disposal programmes. The respective level of progress of these programmes also varies between starting the geologic data collection (e.g. Mexico) to implementing the licensing procedure (e.g. Sweden) to constructing a repository (e.g. Finland). Accordingly, the information available regarding container requirements and concepts varies largely. However, most of the documents found, approximately 6,700 files, are from 11 countries with advanced waste management programmes and were analysed in more detail (see Table 1). A more detailed review on the status of waste container development in the national disposal programmes – the existence of waste containers or waste container concepts – was selected accordingly. For these programmes, which cover the three potential host rocks considered in Germany, detailed information on the container requirements and concepts were summarised. In addition to the highly advanced programmes in France, Finland, and Sweden with already selected repository sites, there are programmes with ongoing site selection (Germany, Canada, Switzerland, and Czech Republic) and currently still generic programmes (United Kingdom).

In the United States of America, two parallel programmes have to be considered; a waste disposal programme which led to the Yucca Mountain Project (YMP) in 1987 (licence application 2009 [3]). This programme was no longer financed from 2010 onwards. Instead, a commission (2010-2012) was established to reassess the disposal programme [4]. As a result of this reassessment, it was decided to restart the site selection procedure in the United States. This included new exploration activities. A new site decision is now to be made by 2026 [5]. Finally, the licensing procedure for the YMP was decreed to continue in 2017, but the necessary financing has not yet been released [6].

Host Rocks	Country	Location	Site Selection	Building Permit	Start of Repository Operation	Operator	Supervisor/ Regulatory Body
Crystalline	Sweden	Forsmark	**2009**	≈ 2020	≈ 2030	SKB	SSM
	Finland	Olkiluoto	**2001**	**2015**	after 2022	Posiva Oy	STUK
	Canada	5 candidates	not limited in time			NWMO	CNSC
	Rep. Korea	open	not specified		after 2040	KHNP	NCS/ KINS
	Czech Rep.	7 candidates	2025	until 2050	2065	SÚRAO	SÚJB
Claystone	Belgium	open	open			ONDRAF / NIRAS	FANC / AFCN
	France	Bure	**2013**	2022	2025	Andra	ASN
	Switzerland	3 candidates	2030	after 2024	2060	Nagra	ENSI
Crystalline Claystone Rock salt	Germany	open	2031	open		BGE	BASE
	United Kingdom		no schedule yet			NDA	ONR, Environm. Agencies
Tuff	United States of America	Yucca Mountain (1987-2010)	**1987**	open	open	U.S. DOE	U.S. NRC
open		open	2026	2042	2048		

Table 1. Relevant International Waste Management Programmes [11]
(completed steps of these programmes in **bold**)

The international waste container requirements in the waste management programmes taken into account can be divided into protection-oriented and operation-oriented requirements. The former are very similar or even identical due to the internationally largely harmonised protection goal definitions in the respective waste management programmes. Requirements regarding the exclusion of criticality or adequate radiation shielding for the protection of the other barriers are regulated internationally in a comparable manner. Temperature limits at the surface of the waste container for the protection of the multi-barrier system, on the other hand, are defined depending on the host rock or repository concept, whereby the use of claybased geotechnical or geological barriers leads in almost all cases to a limitation to a maximum of 100 °C. In Germany, a host rock-independent temperature limit of 100 °C is considered as provisional safety requirement for waste container concepts [1].

In all international disposal programmes, operation-oriented requirements result, on the one hand, from the handling of the waste containers, from which maximum permissible limits for temperatures and local dose rates are derived, or from the necessity of technical aids, such as transfer casks. On the other hand, requirements result from manufacturing, loading, transport, and storage of the containers and the associated quality assurance. These requirements include e.g. the complete documentation and identifiability of the containers, maximum container masses and dimensions, as well as materials used, components for handling (e.g. trunnions for lifting) and the availability of manufacturing technologies (e.g. welding technologies).

The majority of the waste container concepts considered internationally consist of containers that are not sufficiently shielded for handling and transport but are nevertheless mechanically robust. Together with the waste form, they form the waste package but depend on additional transfer casks for handling operations during transport and storage. An example for this is the Swedish-Finnish KBS-3 concept, which consists of an inner cast-iron structure for accommodating the fuel assemblies and an outer 50-mm-thick copper shell. This is also a model for other container concepts. Container concepts in crystalline rock are consistently designed for long utilisation phases up to the entire country-specific reference period. The safe containment of the radioactive inventory has to be demonstrated in a corrosive environment for up to 100,000 years in the international concepts considered. A reference demonstration period of 1 million years, as required in Germany [1], has not yet been considered anywhere in the world. The containers usually consist of an iron-based solid inner shell and a corrosion-resistant outer shell or coating. A bentonite buffer is used as a geotechnical barrier protecting the near-field of the waste containers. For container concepts in claystone, shorter utilisation phases of several thousand years are required depending on the barrier effect of the host rock. So far, waste container concepts consist of steel, where a limited corrosion is accepted. The cask concepts for the disposal of high-level radioactive waste in rock salt – self-shielding double-shell casks and unshielded canisters – which have so far only been considered in Germany, have been designed for a utilisation phase of only up to about 500 years.

RELEVANT BOUNDARY CONDITIONS AND IMPACTS FOR WASTE CONTAINERS

A waste container is exposed to a variety of mechanical, thermal, radiological, chemical, and biological impacts and effects in a repository for heat-generating radioactive waste and spent nuclear fuel assemblies. Depending on the safety demonstration concept, the waste container, if necessary in combination with geotechnical barriers (e.g. buffers), must provide safety-functions under these impacts for a concept-specific length of time of up to 1 million years, which is the reference period in Germany. The site-specific and concept-specific impacts on the waste container and the respective loads result from the boundary conditions of the repository site and the repository concept.

On the one hand, there are boundary conditions that provide stationary impacts on the waste container over long periods of time, e.g. the host rock geology at the repository site and the corresponding mechanical, hydraulic, chemical, and thermal rock properties at the emplacement level. In Germany, the properties of three different host rock formations have to be considered: rock salt, claystone, and crystalline rock. Another invariant boundary condition is the type and amount of waste to be disposed of. The inventory consists of spent fuel assemblies from nuclear power reactors, spent fuel assemblies from prototype and research reactors, and waste from the reprocessing of spent fuel assemblies-.

On the other hand, some impacts are variable and can be adjusted in a safety-oriented manner within the waste container and repository design phases, e.g. the envisaged depth of the repository mine, the mechanical impacts from transport, emplacement, and retrieval technology as well as operational disturbances or accident scenarios to be considered. However, there are also boundary conditions and actions that can be influenced by the waste container itself, such as the container materials selected, which influences the geochemical environment in the near field. Thus, many different boundary conditions and impacts have to be taken into account when developing the waste container concept and design.

DERIVATION AND COMPILATION OF REQUIREMENTS FOR WASTE CONTAINERS

As already mentioned at the beginning of the paper, the methodology of a top-down approach was developed for the systematic derivation of requirements for waste containers and the container functions necessary to fulfil them.

According to this top-down approach, the currently valid legal regulations and framework conditions relevant to the design and demonstration of waste containers were reviewed and compiled in tables. Among all examined regulations, the BMU's Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste as of 2010 [7] were found to comprehensively cover all fundamental requirements for waste containers. This set of regulations was updated by the BMU in the period 2019/2020 and transformed into a Repository Safety Requirements Ordinance [8] in autumn 2020. This ordinance now is an essential reference for the site selection procedure in Germany, including container requirements to be considered. These fundamental requirements are initially independent of the host rock and comprise the four basic protection goals:

- Confinement of the radioactive inventory,
- Shielding of ionising radiation,
- Exclusion of criticality,
- Temperature limitation of the waste packages,
- Limitation of corrosion and gas production,
- Handleability.

In accordance with the safety requirements of BMU [3, 4], the entire period of a repository can be divided into four different time phases, for which there are partly different requirements. These are the following container-specific utilisation phases:

Phase 1:	Emplacement phase (provision of the container for emplacement until emplacement of the container is completed)
Phase 2:	Retrievability phase (completion of emplacement of the container until start of decommissioning of the repository)
Phase 3:	Recoverability phase (start of decommissioning of the repository until 500 years after closure of the repository)
Phase 4:	Later post-operational phase (500 years after closure of the repository until the end of the reference period)

The basic container requirements for the different time phases of the disposal process were specified in more detail based on the safety and safety demonstration concepts for repository concepts, which have been elaborated in previous feasibility studies in Germany. Table 2 shows the basic requirements for waste containers during a container lifetime (utilisation phases). These basic container requirements were derived for all three potential host rocks, rock salt, claystone, and crystalline rock. Moreover, the basic requirements and container functions need to be quantified. Based on existing regulatory specifications, quantified container requirements and functions can be derived.

The main container functions for the six basic requirements are the following:

- In principle, the containment function must be ensured in such a way that a release of radioactive nuclides into the biosphere is excluded to such an extent that the radiological limits stipulated the Radiation Protection Act [9] and the Radiation Protection Ordinance [10] are complied with.

- For shielding of the ionising radiation emitted by the radioactive inventory, the waste container must provide sufficient shielding functions under the impacts in the repository.
- In all safety concepts, exclusion of criticality is required as a fundamental protection goal during all utilisation phases of the repository and must be demonstrated accordingly.

<table>
<tr><th rowspan="2">Basic Requirement</th><th colspan="4">Time phase</th></tr>
<tr><th>Emplacement</th><th>Retrievability</th><th>Recoverability</th><th>Later post-operation</th></tr>
<tr><td>Confinement of radioactive inventory</td><td colspan="2">To be fully ensured</td><td>To be sufficiently ensured for developments that are probable according to [3] or can be expected according to [4]</td><td></td></tr>
<tr><td rowspan="2">Shielding of ionising radiation</td><td colspan="3">To be sufficiently ensured for repository operation</td><td></td></tr>
<tr><td></td><td colspan="3">To be sufficiently ensured to avoid radiologically- or radiolytically-induced damage to the barrier host rock</td></tr>
<tr><td>Exclusion of criticality</td><td colspan="4">To be fully ensured for the most reactive arrangement of the nuclear fuel including assumed water ingress</td></tr>
<tr><td rowspan="2">Temperature limitation</td><td colspan="2">To be sufficiently ensured for repository operation</td><td>To be sufficiently ensured for developments that are probable according to [3] or can be expected according to [4]</td><td></td></tr>
<tr><td></td><td colspan="3">To be sufficiently ensured to avoid thermal damage to the barriers and the host rock</td></tr>
<tr><td>Limitation of corrosion and gas production</td><td></td><td colspan="3">To be sufficiently ensured to prevent unacceptable corrosion and gas pressures
in case of access of aqueous solutions</td></tr>
<tr><td>Handleability</td><td colspan="2">To be sufficiently ensured</td><td>To be sufficiently ensured for developments that are probable according to [3] or can be expected according to [4]</td><td></td></tr>
</table>

Table 2. Basic Requirements of Waste Containers during Container Utilisation Phases

- The surface temperature of the waste container has to be limited in order to ensure, on the one hand, handleability during the phases emplacement, retrievability, and recoverability and, on the other hand, to exclude safety-relevant damage to inventories and technical and geological barriers with regard to their barrier functions.
- Materials should be selected that, under the boundary conditions of the future repository site, are expected to minimise corrosion.

- In order to meet the requirements with regard to handling during the time phases emplacement, retrievability and recoverability, the waste containers must be designed in combination with a suitable handling technology.

The container functions necessary to meet the requirements are affected by various influences on the containers. The following basic effects can be distinguished and have to be considered as impacts accordingly. Table 3 shows the classification of relevant impacts on waste containers. A comprehensive and detailed description of the quantified impacts determined thus far for the different repository concepts in the different host rocks has been compiled and described in detail.

- Mechanical impacts: including static and dynamic loads
- Thermal impacts: based on the heat released by the radioactive decay processes in the inventory
- Radiological impacts: from the radiation field generated by the decay processes in the inventory
- Chemical and biological impacts: including all impacts caused by the hydro-chemical or geochemical environment and by microbes or their metabolic products.

Impact class	Nature of the relevant impact		Possible consequences
Mechanical	static	Rock pressure, swelling pressure, water pressure (isotropic, anisotropic)	Material or component failure; Danger to the safety functions
		Shear stress	
		Assembling; Handling	
	dynamic	Accelerations, vibrations, shocks from handling and repository operation	
		Accident scenarios, e.g. container crashes	
Thermal	Decay heat of radioactive waste		Thermal damage to technical components, geotechnical and/or geologic barriers
	Rock temperature		
	Accident fire		
Radiological	Gamma and neutron radiation		Material embrittlement or decomposition; Radiolysis gas formation
Chemical and Biological	Corrosive media Free water Microorganisms		Corrosion Material decomposition Hydrogen embrittlement Stress corrosion cracking Gas formation

Table 3. Classification of Relevant Impacts on Waste Containers [11]

Compiling all the necessary quantified requirements, the safety functions, and the expected impacts for a waste container for a selected repository and emplacement concept in a selected host rock provides the prerequisites for the waste container design. Eventually, within the framework of the site-specific licensing procedure, the waste container-specific safety demonstration for the specific container design has to be carried out. Thus, compliance of the relevant container functions and fulfilment of all requirements for the relevant impacts over the entire reference period have to be demonstrated.

IDEAS OF POSSIBLE WASTE CONTAINER CONCEPTS

The investigations on possible waste container concepts revealed on the one hand to what extent waste container concepts already developed in national and international waste management programmes can meet the requirements derived for Germany. On the other hand, an approach for deriving generic waste container concepts for the three potential host rocks rock salt, claystone, and crystalline rock has been developed. The evaluation of existing container concepts and the development of new container concepts was carried out exclusively on the basis of requirements, i.e. in a safety- and function-oriented manner based on the overall top-down approach for the derivation of (generic) container concepts meeting the requirements (see Figure 1).

Consequently, the same overall top-down approach (Figure 1) is the basis for the methodology to investigate existing container concepts with regard to a possible transferability to future repository concepts in Germany. Accordingly, container functions are derived from the requirements for the waste containers. Taking into account the impacts on the waste containers, these container functions serve to fulfil the requirements and form the basis for developing a container concept. To ensure comparability of different container concepts, these requirements and impacts must therefore be checked first, by means of a comprehensive comparison, to establish to what extent a transferability of existing container concepts to other repository concepts is possible. The methodology developed is shown in Figure 2.

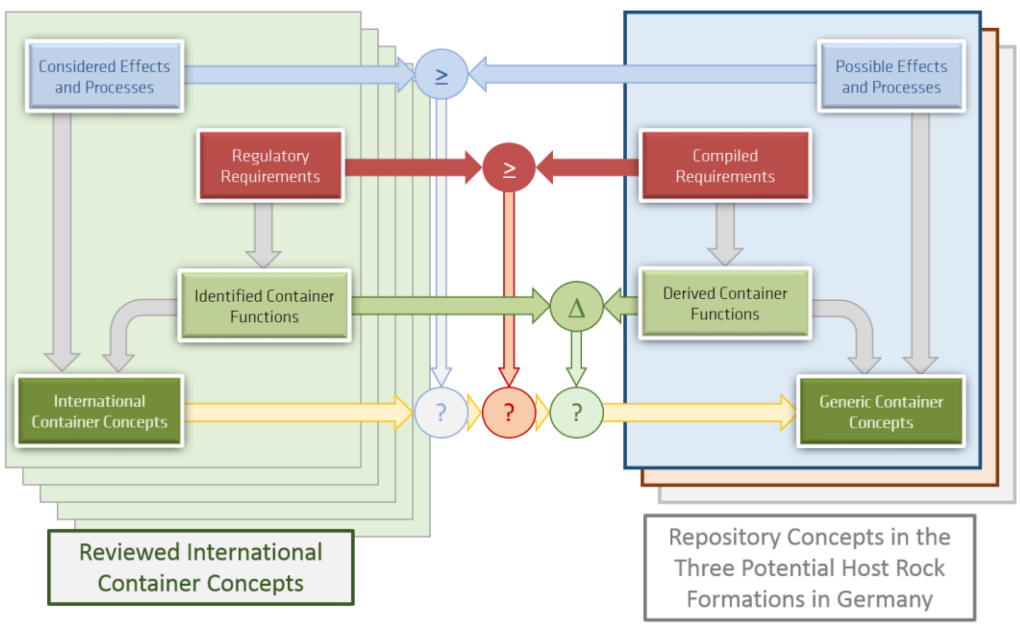
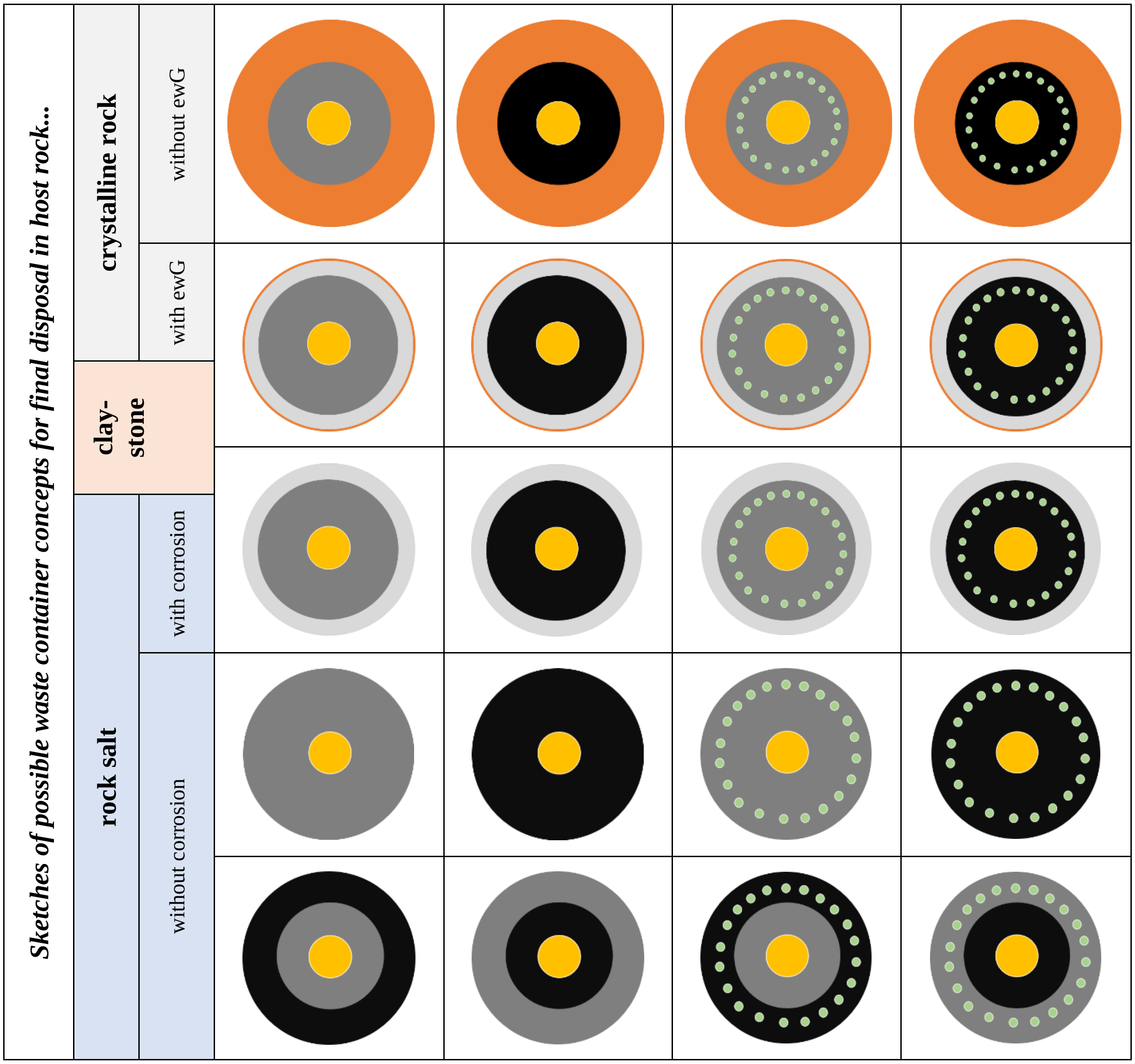


Fig. 2. Methodology for Investigating the Transferability of National and International Existing Waste Container Concepts to Possible Repository Concepts in Germany [11]

The relationships on both sides of the diagram (existing waste container concepts on the left, transferable container concepts derived for the three potential host rocks on the right) correspond directly to the top-down approach shown in Figure 1. An essential aspect for the investigation carried out in the project was the assessment whether the impacts and requirements of the national and international waste container concepts cover the impacts and requirements to be expected in a future high-level waste repository in Germany. If differences were found, a comparative consideration of the container functions was necessary to be able to make a statement for a possible transferability. The investigations carried out in the KoBrA project can therefore primarily provide indications of the strengths and weaknesses of the waste container concepts and, in addition, identify further research and development needs. Furthermore, the investigations of the different waste container concepts have shown that, apart from a few host rock- or even site-specific exceptions, mainly generic container concepts and repository concepts have been developed so far from which information on impacts could be obtained. Quantitative statements can thus only be made to a very limited extent, and qualitative statements are also based on hypotheses and argumentative considerations. The investigation of possible waste containers did include, in addition to existing international container types, container concepts developed in Germany like the POLLUX® cask, which was designed and manufactured as a prototype for a repository in rock salt. These container types will still be kept as an option.

Based on the top-down approach (see Figure 1), a methodology for the development of generic container concepts meeting the requirements was developed. For this purpose, a solution matrix was created to derive generic container concepts that meet the requirements. The required waste container functions were divided into sub-functions, which can each be assigned to the generic components of a waste container. For all sub-functions, possible technological or physical solutions can be compiled and be implemented in the respective container component, thus enabling the container components to meet the attributed sub-functions. The impacts to be expected for the waste container, depending on the chosen host rock formation and repository concept therein, serve to select and quantify the sub-functions of the respective container components. Finally, when the container components meet all sub-functions, a generic container concept can be drawn up. Several ideas and suggestions for generic container concepts were elaborated based on the implementation of the requirements derived by means of an exemplary generic solution matrix. Table 4 shows sketches of possible waste container concepts derived for disposal in the three potential host rocks.



Colour legend				
	cast iron	(cast) steel	stainless steel	corrosion protection (copper, titanium, nickel alloy)
	inventory	moderator rod (exemplary moderator element)		

Table 4. Diagram of Generic Waste Container Concepts for Different Host Rocks [11]

CONCLUSIONS AND RECOMMENDATIONS

The investigations in the KoBrA project provided a new top-down approach (Figure 1) that enables a systematic derivation of qualitative and quantitative requirements for waste containers for a repository for high-level radioactive waste (HLW) in the three potential host rocks rock salt, claystone, and crystalline rock. Based on this top-down approach, a methodology for the assessment of the transferability of existing national and international waste container concepts and for the development of generic container concepts that meet the requirements was developed as well. However, it should be noted that there is a limit to taking advantage of already existing waste container concepts because of differences in the set of waste container requirements. For instance, international requirements for retrievability vary and are very limited. Requirements for recoverability do not exist at all. In this respect, neither the international nor the previous national container concepts, including the POLLUX® cask, which was manufactured as a prototype and developed to within licence application readiness for the host rock salt, have been designed with regard to these requirements.

The investigations also revealed that, apart from safe containment and the radiological requirements, the fulfilment of the thermal requirements with a general temperature limitation for all host rocks to 100 °C according to the Site Selection Act [1] would lead to a considerable verification and adaptation effort for the majority of the container concepts considered. Even more important is the requirement regarding recoverability, according to the Site Selection Act [1] reduced to the partial requirements regarding identifiability, safe enclosure, and handleability. This requirement is still connected with open questions for all container concepts. In some cases, it is even decisive for the design, for example with regard to the demonstration of long-term resistance.

However, generic container concepts were derived by way of examples, which can be expected to meet the basic container requirements or associated container functions in the different potential host rocks. The methodology of a solution matrix for technological solution principles of partial functions still shows the richness of variants of a possible design space despite several restrictive basic assumptions. Thus, for the three potential host rocks, a total of 20 generic concepts for self-shielding robust containers were derived, each of which provides all the basically required container functions. These container concepts have to be designed in more detail based concrete site-specific data. Accordingly, the number of generic container concepts must ultimately be further reduced by selection and evaluation criteria to be derived.

Here it must be taken into account that this development and decision process must take place in the ongoing site selection procedure with regard to container concepts for all three host rocks in question. In order to keep the associated expenditure within reasonable limits, the systematic and comprehensible optimisation and restriction of suitable container concepts is of utmost importance.

In summary, the KoBrA research project revealed the following main recommendations for future waste container development in Germany:

(1) Continuous monitoring and evaluation of the development of international waste container concepts;

(2) Concretisation of the site-specific requirements for waste containers;

(3) Improvement of the geological database for the host rocks and potential repository sites to determine host rock- and site-specific impacts on waste containers;

(4) Concretisation of the operation-related impacts on the waste containers in interaction with the development of suitable repository concepts and transport and emplacement techniques;

(5) Derivation of a methodology for defining selection and decision criteria for suitable repository concepts with regard to the site selection procedure in Germany;

(6) Preference is given to robust container concepts utilising proven materials and technologies that have already been developed and tested as far as possible and that can be manufactured reliably, to the required quality standards, and on schedule in the required large quantities;

(7) Rapid development of container concepts for claystone and crystalline rock in line with requirements and further development of the container concepts already existing for rock salt.

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