BGZ Gesellschaft für Zwischenlagerung mbH

THINKING AHEAD
INTERIM STORAGE

BGZ's research programme

Contents

List of abbreviations

Abbreviation	Explanation
a	Year (time unit)
AVR	Jülich Experimental Reactor Consortium
AVR-FE	AVR reactor fuel element (multi-layer spherical fuel pebble)
BWR	Boiling water reactor
BWR-FA	Boiling water reactor fuel assembly
BZA	Ahaus Interim Storage Facility for SNF
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (Alternative Energies and Atomic Energy Commission), French Research Centre for Nuclear Energy
°C	Degree Celsius (temperature unit)
DPC	Dual-purpose cask for the transport and storage of HAW
ESK	Nuclear Waste Management Commission
EWN	Entsorgungswerk für Nuklearanlagen GmbH
FA	Fuel assembly
GNS	Gesellschaft für Nuklear-Service mbH
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH
GWd/tHM	Gigawatt days per metric tonne of heavy metal, i.e. amount of thermal energy generated per tonne of heavy metal used. Measure of the burn-up (consumption) of nuclear fuel during use in the reactor.
HAW	High active waste
HAW canister	Stainless steel canister filled with HAW in the form of vitrified residues from reprocessing
IAEA	International Atomic Energy Agency
ISF	Interim storage facility
JEN	Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH
K	Kelvin (absolute temperature, 273.15 K = 0°C)
kW	Kilowatt (unit of measurement for heat generation, 1 kW = 1,000 watts)
LWR-FA	Light-water reactor fuel assembly (synonymous with power reactor FA)
MPa	Megapascal, here: unit of measurement for mechanical stress
MSTOR	Metal seals during long-term storage
MW	Megawatt (unit of measurement for heat generation 1 MW = 1,000 kW)
NEA	Nuclear Energy Agency
NPP	Nuclear power plant
OECD	Organisation for Economic Co-operation and Development
Orano NPS	Orano Nuclear Packages and Services
PWR	Pressurised water reactor
PWR-FA	Pressurised water reactor fuel assembly
RR-FA	Research reactor fuel assembly
SNF	Spent nuclear fuel
THTR	Thorium high-temperature reactor
THTR-FE	Thorium high-temperature reactor fuel element (multi-layer spherical fuel pebble)
Weight %	Weight percent (percentage by mass in the mixture of substances)
ZLN	Interim Storage Facility North *(Zwischenlager Nord)*
$\frac{\mathbf{Pa \cdot m^3}}{\mathbf{s}}$	Standard helium leakage rate Measure for the quantity of helium (as test gas) escaping per unit of time, with Pa – pascal (unit of pressure), m^3 – cubic metre (unit of volume), s – second (unit of time)

1. The BGZ research mission

The strategy for the responsible and safe disposal of spent nuclear fuel and other heat-generating radioactive waste is stipulated in the German government's National Waste Management Programme [1]. BGZ Gesellschaft für Zwischenlagerung mbH's mandate in the national waste management strategy is derived from the Act on the Reorganisation of Responsibility in Nuclear Waste Management [2]. BGZ is a company organised in private legal form that is wholly owned by the Federal Government. BGZ was founded to ensure the reliable and safe operation of interim storage facilities for low, intermediate and high-level radioactive waste. Since 1 January 2019, the interim storage facilities for high-level radioactive waste have included not only the Ahaus and Gorleben interim fuel storage facilities but also the Biblis, Brokdorf, Grafenrheinfeld, Grohnde, Gundremmingen, Isar, Krümmel, Lingen, Neckarwestheim, Philippsburg and Unterweser sites[1]. High-level radioactive waste is stored at interim storage facilities until it is transferred to a repository. The procedural periods for searching and selecting a site for a repository for high-level radioactive waste are established in the German Site Selection Act [3]. The intended and approved interim storage period of up to 40 years will almost certainly not be long enough. It might take longer to commission a repository for the spent nuclear fuel (SNF) and other heat-generating radioactive waste and to clear the storage facilities [1, 4]. As shown in Figure 1, the time limits of the existing storage licences for interim storage facilities will expire between 2034[2] and 2047. As the operator and licence holder, BGZ is required to provide ongoing proof of where the dual-purpose casks remain as well as of compliance with the safety objectives for extended interim storage in accordance with the state of the art in science and technology.

The research programme developed here identifies the research that is required and provides an overview of the BGZ's research strategy and activities. The research programme is constantly updated and adapted in line with the evolving state of the art in science and technology.

1 The Brunsbüttel on-site interim storage facility is currently operated by Kernkraftwerk Brunsbüttel GmbH & Co. oHG; transfer to BGZ will take place as soon as the ongoing procedure for reissuing the storage licence is completed and the licence can be utilised.

2 The period specified in the storage licences for a period of 40 years begins with the closing of the DPC after it is loaded. This period will expire for the first DPC as early as 2032.

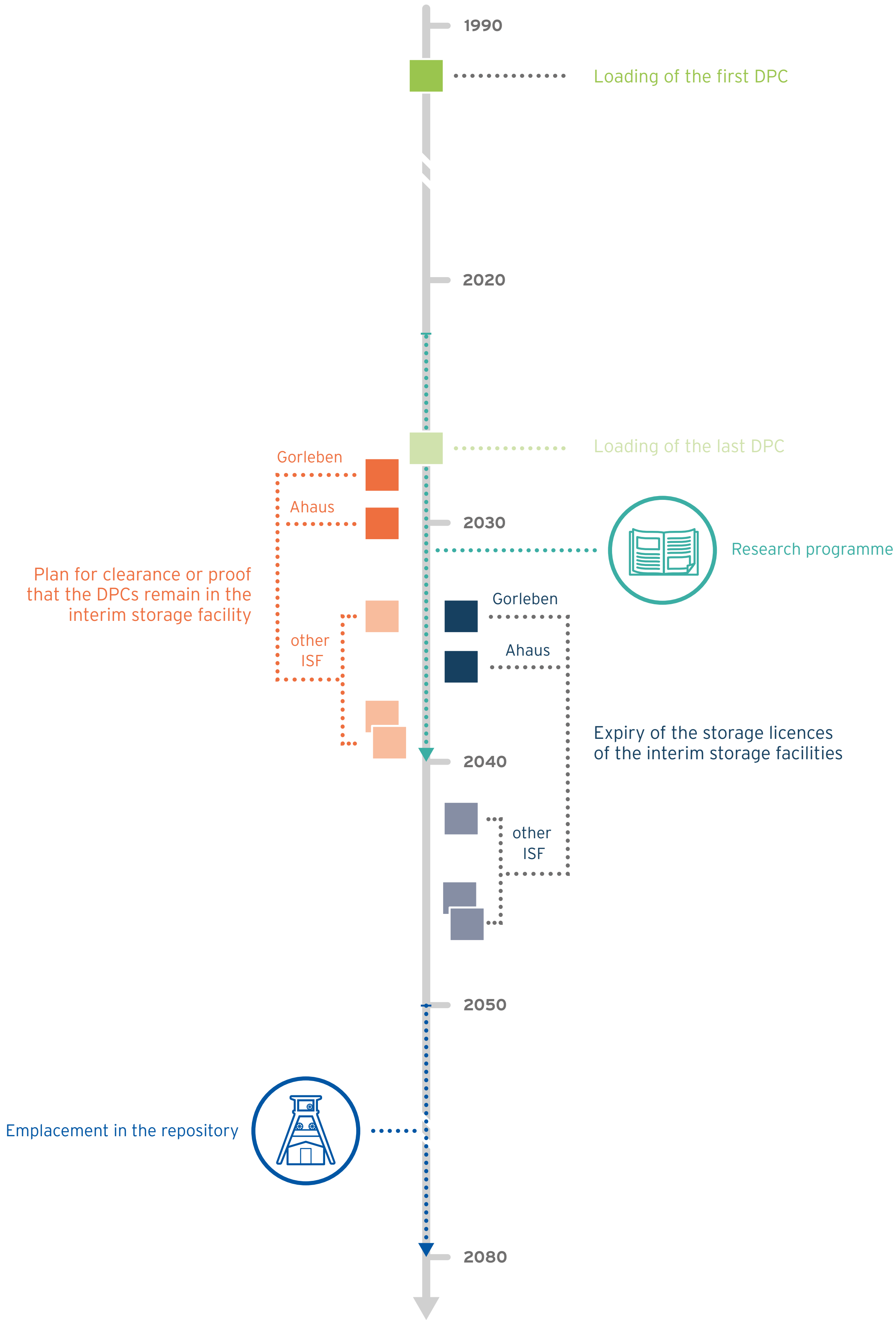


Figure 1: Simplified representation of the time sequence from initial cask loading through to emplacement in a repository

2. Dry interim storage in Germany

After they have been used in a reactor, fuel assemblies are stored in spent fuel pools before being packed in dual-purpose casks (DPC) for the transport and storage of HAW for dry interim storage.

BGZ operates the two central interim storage facilities in Gorleben and Ahaus as well as all the interim storage facilities built at nuclear power plant sites. An exception to this is the interim storage facility at the Brunsbüttel nuclear power plant. Here, the interim storage facility for high-level radioactive waste will only be transferred to BGZ once the ongoing procedure to reissue the storage licence has been completed.

The remaining two interim storage facilities for high-level radioactive waste are operated by companies of the EWN Group, which are financed from public funds. One of these is the Interim Storage Facility North (ZLN) that is operated by the EWN Entsorgungswerk für Nuklearanlagen GmbH. This is where SNF and waste containing SNF from the operation of the nuclear power plants of the former GDR as well as from nuclear facilities of the federal government are temporarily stored. The second of these interim storage facilities, the so called AVR cask storage facility in Jülich, is operated by Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH, which also belongs to the EWN Group, for the interim storage of fuel element pebbles from the AVR experimental reactor.

The inventory in the interim storage facilities operated by BGZ consists largely of spent light-water reactor fuel assemblies (LWR-FA) from German energy utilities and vitrified high-level radioactive waste from their reprocessing. Only the Ahaus interim storage facility also stores SNF from research and prototype reactors.

The two central interim storage facilities in Gorleben and Ahaus were built in the 1980s. The planning for these interim storage facilities was based on the original German disposal strategy, which primarily envisaged the reprocessing of spent fuel assemblies. In accordance with the principles for the handling and disposal of spent fuel valid at that time, no DPCs were loaded with spent fuel assemblies from power reactors for dry interim storage before 1994. After decay storage, the spent fuel assemblies were transported directly from the nuclear power plants to France (La Hague) or the UK (Sellafield) for reprocessing.

In Germany, the first casks intended for dry interim storage were loaded with fuel assemblies from the prototype thorium high-temperature reactor (THTR) between 1992 and 1994. In 1994, the CASTOR® IIa was the first cask to be loaded with fuel assemblies from a commercial power reactor. By 1999, eleven casks had been loaded with spent light-water reactor fuel assemblies (LWR-FA). The preferred disposal strategy continued to be reprocessing. Due to a paradigm shift in the German government's energy policy at the beginning of the 2000s and the associated amendment of the Atomic Energy Act (AtG) [5], transport for reprocessing was banned by the legislator from 1 July 2005. It was legally regulated that the operators of nuclear power plants must store the LWR-FA in the vicinity of the respective power plants. The number of cask loadings with spent LWR-FA subsequently increased continuously after twelve on-site interim storage facilities had been commissioned (in 2002, 2006 and 2007). With the decision to phase out the use of nuclear energy for electricity generation by the end of 2022, the last loadings of LWR-FA are expected in 2027.

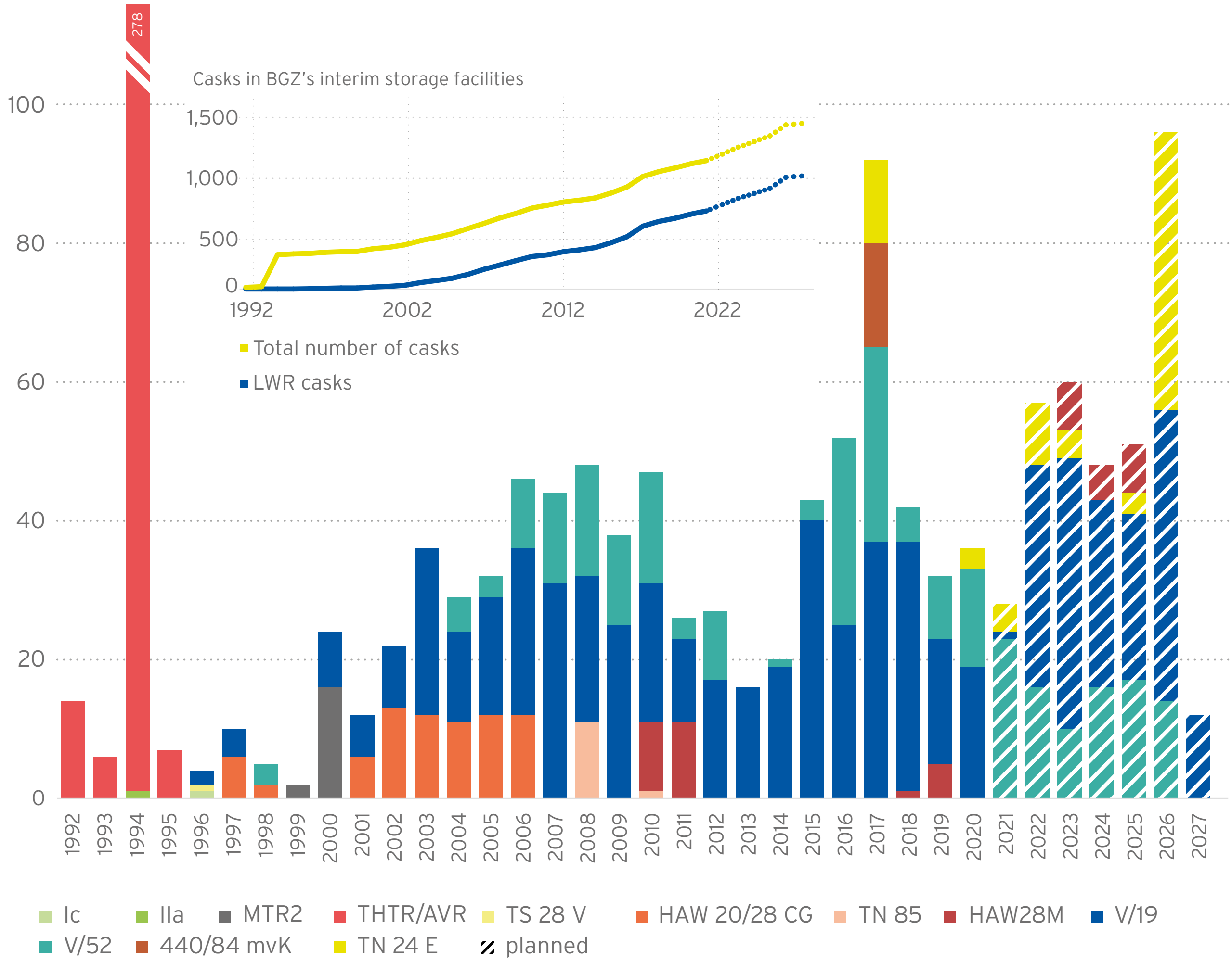


Figure 2: Development over time of cask loadings according to cask types End of 2020
(without CASTOR® MTR3 and CASTOR® THR/AVR with AVR-SNF)

The return and storage of vitrified high-level radioactive waste from reprocessed spent fuel assemblies began in 1996. Waste was returned in four cask types as follows: 1996 TS 28 V, 1997 to 2006 CASTOR® HAW 20/28 CG, 2008 to 2010 TN® 85 and from 2010 CASTOR® HAW28M. Until 2011, the casks with HAW canisters were stored exclusively in the central interim storage facility at Gorleben. On 28 June 2013, the German Bundestag amended the Atomic Energy Act as part of the adoption of the Article Law on the first Site Selection Act. Energy supply companies were thereafter required to store the remaining radioactive waste from reprocessing in on-site interim storage facilities. The Biblis, Isar, Brokdorf and Philippsburg interim storage sites were then selected to receive the outstanding reprocessing waste. Six casks with HAW canisters were stored in Biblis in 2020. Currently planning envisages completing the return of all HAW canisters from reprocessing as well as the packaging of LWR-FA into DPCs and their emplacement in interim storage facilities by 2027. As of 2027, BGZ is expected to store almost 1,500 dual-purpose casks in the interim storage facilities.

Beyond 2027, only CASTOR® MTR3 type casks are expected to be placed into storage. These will contain SNF from German research reactors – similar to the predecessor CASTOR® MTR2 type, 18 of which, containing fuel assemblies from the Rossendorf research reactor, are already being stored in the Ahaus Interim Storage Facility (BZA).

	Cask type	Inventory	Ahaus	Gorleben	Biblis	Brokdorf	Brunsbüttel*	Grafenrheinfeld	Grohnde	Gundremmingen	Isar	Krümmel	Lingen	Neckarwestheim	Philippsburg	Unterweser
Site			BGZ interim storage facilities													
Type of storage facility			Central interim storage facilities		Decentralised on-site interim storage facilities											
					WTI	Steag	Steag	WTI	Steag	WTI	WTI	Steag	Steag	Tunnel	WTI	Steag
			Stored cask types													
CASTOR® casks	Ic	BWR-FA		•												
	IIa	PWR-FA		•												
	V/19	PWR-FA	•	•	•	•		•	•		•		•	•	•	•
	V/52	BWR-FA	•				•			•	•	•			•	
	440/84 mvK	PWR-FA												•		
	HAW 20/28 CG	HAW canisters		•												
	HAW28M	HAW canisters		•	•	(X)					(X)				(X)	
	MTR2	RR-FA	•													
	MTR3	RR-FA	(X)													
	THTR/AVR	THTR-FE	•													
		AVR-FE	(•)													
TN® casks	TN® 24 E	PWR-FA									•			•		
	TN® 85	HAW canisters		•												
	TS 28 V	HAW canisters		•												

* BGZ accession to the relicencing procedure in January 2019

- • Cask type and inventory stored in BGZ interim storage facility
- (•) Storage licence granted in 2016. The owner of the AVR fuel elements, JEN, is responsible for deciding where the casks will remain in future.
- (X) Storage of cask type and inventory applied for

Table 1: Overview of the DPC types and corresponding inventories located in BGZ interim storage facilities
April 2022

3. Development of the research programme

The development of the research programme follows a holistic approach to safety objectives. In this context, the safety proofs on which the storage licences and package design approvals are based were systematically and critically questioned with regard to extended interim storage. The research programme also draws on operational experience with storage buildings, casks and inventories. International exchange and the review of current research findings that define the state of the art in science and technology also play an important role in determining the research needs that are required on extended interim storage (see Figure 3). Effects and possible countermeasures are always key factors to be taken into account when assessing the need for research. The underlying concept of dry interim storage and the associated functions of the casks, the inventory and the storage building, must also be included. The aim is to identify any need for action at an early stage in order to be able to use the remaining time until the application is submitted for corresponding research programmes and thus open up additional options for action if necessary.

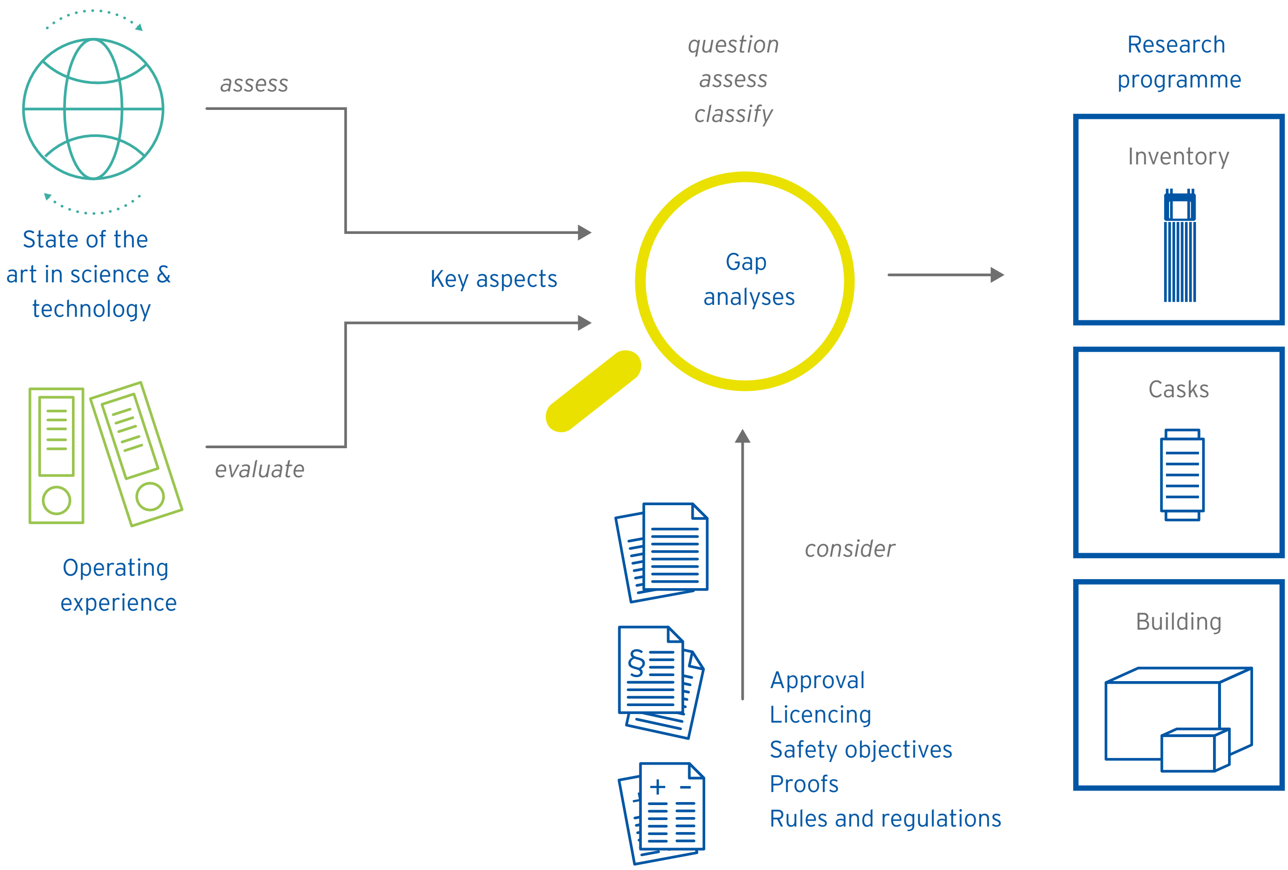


Figure 3: Procedure for the development of the research programme


max. Tragfähigkeit je Bühnenhälfte 4t (440kg/m²)

4. Communicating progress and outcomes

Results and progress of the research programme are communicated on several levels by different actors and with content adapted to the respective target group. This communication work is undertaken by individual scientists, by specialist departments and in the form of press and public relations work at various levels, including addressing the general public, independent experts, and the scientific community.

BGZ is already using the series of events being held as the "Interim Storage Forum" to answer and discuss essential issues relating to safe interim storage with representatives of citizens' initiatives, public authorities, and scientific institutions as well as interested citizens. In this framework, dialogue and discussions cover various aspects of the safe storage of radioactive waste as well as current research. The internet platform of the Interim Storage Forum [6] is designed to foster the exchange of information on the interim storage of radioactive waste. The forum can be used to direct specific questions to BGZ and to provide prompt responses. All questions and the corresponding answers are published in the question forum [7].

The current status and results of the research programme are presented and discussed with the participants at a biennial "Expert Workshop on Interim Storage". The majority of workshop participants are from universities, research institutions, public authorities, expert bodies, and relevant committees or are representatives of the industry in Germany and Switzerland.

Direct and ongoing exchanges take place at the scientific level with national and international project partners from research institutions and industry, as well as in the various national and international committees (see also Chapter 5). Progress and the results generated by the BGZ research programme are also presented to and discussed by the broad international and national scientific community in the form of regular presentations at specialist meetings, conferences, and workshops. The research programme and selected results are also published in conference proceedings and the relevant peer-reviewed scientific journals. Whenever possible, contents are subsequently made freely available to the general public (so-called open access).

5. National and international cooperation

BGZ's work on the research tasks presented in this report involves extensive professional exchange at national and international level. Within the framework of specific research projects, collaboration takes place with partners involved in the manufacture of casks and fuel assemblies as well as partners in research institutes, universities and other relevant companies. BGZ strives to transparently involve all important and relevant partners in the field of nuclear waste management in its research. Collaboration on specific projects is outlined in Chapter 11 for each project.

As well as collaborating on specific research projects, BGZ also cooperates with strategic partners in the field of nuclear waste management and participates in programmes, organisations and committees. These are presented separately in brief in the following.

EWN Group

EWN Entsorgungswerk für Nuklearanlagen GmbH, which is also a state-owned company that is preparing for extended interim storage, is an important strategic partner for BGZ at the national level. EWN operates the ZLN storage facility near Lubmin, which will be replaced by the ESTRAL storage facility by the mid/late 2020s. The Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH (JEN), which is affiliated with EWN, also operates the AVR storage facility in Jülich.

In addition, the Kerntechnische Entsorgung Karlsruhe GmbH (KTE), which is also affiliated with EWN, has stored nuclear fuel in the form of vitrified radioactive waste from the reprocessing of spent fuel assemblies in the ZLN. The shared interests and key tasks in this context provide the basis for regular mutual exchange. There are also plans for joint research projects with the EWN Group (see Chapter 11.1).

Cooperation in DIN standards committees

BGZ sends a permanent member to the working group NA 062-07-54 AA "Criticality safety and decay power" of the DIN Standards Committee Materials Testing (NMP). The working committee draws up and updates the relevant standards and holds important discussions on topics of criticality safety and the decay power of SNF. Cooperation in additional DIN standards committees that are relevant to interim storage is planned.

Cask manufacturers – GNS and Orano NPS

The dual-purpose casks (DPCs) used to store spent fuel assemblies and vitrified waste from reprocessing in BGZ interim storage facilities are from the German-based Gesellschaft für Nuklear-Service mbH (GNS) or from Orano Nuclear Packages and Services (Orano NPS) in France. Both companies have decades of experience in the field of nuclear waste disposal, especially in the development, approval and production of dual-purpose casks for highly radioactive waste. The CASTOR® cask type developed by GNS accounts for more than 90% of the DPCs stored by BGZ; the remaining are TN® type casks from the company Orano NPS. The cask manufacturers GNS and Orano NPS hold the package design approval for DPCs and are thus important partners of BGZ.

Partnerships with Swiss institutions

ZWILAG Zwischenlager Würenlingen AG and BGZ have been in regular contact with each other for many years. In view of the extended interim storage period, shared interests and key tasks, there are plans to extend this mutual exchange to the Swiss nuclear power plant operating companies BKW Energie AG, Kernkraftwerk Gösgen-Däniken AG, Kernkraftwerk Leibstadt AG and Axpo Power AG in the future.

The Paul Scherrer Institute (PSI) is the largest research institute for natural and engineering sciences in Switzerland and has a long tradition in energy research. The PSI is an important partner for BGZ in research on the safe disposal of spent fuel. The PSI has an extensive research infrastructure that includes the operation of hot cells in which entire fuel rods can be studied, as well as other large-scale facilities such as the Swiss Synchrotron Light Source (SLS) and the Swiss Spallation Neutron Source (SINQ).

Extended Storage Collaboration Program
The Electric Power Research Institute (EPRI) is an independent non-profit organisation in the USA that conducts research on electrical power supply. The research is mainly financed by the members and participants, who are made up of approximately 1,000 organisations from 40 countries worldwide. In 2009, the Extended Storage Collaboration Program (ESCP) was founded with the aim of expanding the technical basis for ensuring the extended interim storage of irradiated fuel assemblies and their subsequent transport. Among other things, the focus is on defining common goals, exchanging information and strengthening international cooperation. Approximately 600 participants from 19 countries take part in the ESCP programme and more than 150 participants in each of the bi-annual meetings.

BGZ chairs the modelling and benchmark working group here, which is primarily concerned with preparing and using experimental data for the validation of existing computational programmes, as well as determining and quantifying uncertainties in the forecast models (see also Chapter 11.2.3).

Nuclear Energy Agency
The Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) – OECD/NEA – provides a framework in which governments can compare policy experiences, seek answers to common questions, identify best practices and work to coordinate national and international strategies. The NEA's specific areas of competence include safety and regulation of nuclear activities, radioactive waste management, radiation protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public relations. The NEA Data Bank [8] provides nuclear data and computer programmes for participating countries.

BGZ is a member of the Working Party on Nuclear Criticality Safety (WPNCS) [9] and provides one member of the delegation from the Federal Republic of Germany. The WPNCS addresses technical and scientific issues relevant to criticality safety. This also includes the transport and storage of fuels.

International Atomic Energy Agency
The International Atomic Energy Agency (IAEA), which has its headquarters in Vienna, promotes the safe and peaceful use of nuclear energy. It was founded in 1957 as the "Atoms for Peace" organisation of the United Nations and has 172 member states today. The focus of its work is on nuclear safety and the safeguarding and monitoring of fissile nuclear materials. The IAEA promotes research and technology for the application of ionising radiation in medicine, food safety, agriculture and environmental monitoring.

The internationally required safety level is developed by the IAEA and defined in the Safety Standards [10]. These cover all aspects of reactor safety, radiation protection, the transport of nuclear goods and the disposal of radioactive waste. The Commission on Safety Standards manages the continuous development of technical committees that are made up of experts from the member states. BGZ is represented in the Transport Safety Standards Committee (TRANSSC, Transports of Nuclear Goods). BGZ also participates in Coordinated Research Projects (CRP) and Technical Meetings (TM) of the IAEA.

6. Operational experience and ageing management

Important conclusions can be drawn for the licencing of extended interim storage from past and future operating experience and the lessons learned from the ageing management of a total of 14 interim storage facilities. As well as evaluating our own operating experience, we also exchange experience with EWN Entsorgungswerk für Nuklearanlagen GmbH on a regular basis. Regular contact is also maintained at the international level with the operator of the Swiss interim storage facility ZWILAG Zwischenlager Würenlingen AG. In the future, operating experience will also be discussed with the Swiss nuclear power plant operating companies BKW Energie AG, Kernkraftwerk Gösgen-Däniken AG, Kernkraftwerk Leibstadt AG and Axpo Power AG.

7. Licences, approvals, regulations

All interim storage facilities operated by BGZ and the casks stored in them, including their respective inventories, have a valid storage licence pursuant to Section 6 of the Atomic Energy Act [5]. In addition to the Atomic Energy Act, the operation of interim storage facilities is governed by laws and regulations such as the Radiation Protection Act [11] and the Radiation Protection Ordinance [12] as well as the guidelines on dry interim storage issued by the Nuclear Waste Management Commission (ESK) [13]. To date the ESK has only issued a discussion paper on extended interim storage [14].

The dual-purpose casks stored by BGZ function as both transport and storage casks. For this reason they hold a storage licence under Section 6 of the Atomic Energy Act as well as a package design approval as type B(U) package under the legal provisions on the transport of hazardous goods. The package design approvals are retained permanently to ensure that the stored casks can be transported away. Approvals are based on the corresponding international regulations of the IAEA [15] and their transposition into European or national regulations, such as the European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR [16]), the Regulation concerning the International Carriage of Dangerous Goods by Rail (RID [17]), the European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways (ADN [18]) or the Ordinance on the Transport of Dangerous Goods by Road, Rail and Inland Waterways (GGVSEB [19]).

A subordinate body of rules and regulations also partly applies in both fields of law. These include the rules and regulations of the Nuclear Safety Standards Commission (KTA) [20], the dangerous goods regulations of the Federal Institute for Materials Research and Testing (BAM-GGR) [21] as well as national and international standards and guidelines of the German Institute for Standardisation (DIN), the International Organization for Standardization (ISO) and the Association of German Engineers e.V. (VDI).

8. Need for research on dual-purpose casks

8.1 Cask review

Only dual-purpose casks (DPCs) are used for the dry interim storage of highly radioactive waste (HAW) in Germany. This means that the transport configuration of these casks must meet the requirements for type B(U) packages in the regulations for transport of radioactive material. At the same time, the interim storage concept also stipulates that the casks must meet the requirements for a storage licence under Section 6 of the Atomic Energy Act. The DPCs must therefore ensure safety during storage and transport – both under normal operating conditions and under hypothetical accident conditions.

The casks fulfil their safety functions as a fully passive system. All cask components are suitable for long-term use, in line with safety philosophy in nuclear waste management. There are therefore no plans for their systematic replacement. The shielding effect of the thick-walled cask body means that the DPCs are always accessible. Inspection or maintenance work can thus be carried out as necessary. The cask design also facilitates replacement of many of the components that do not directly serve the purpose of containment. A total of 13 cask types are used in BGZ interim storage facilities (see Table 1).

More than 90% of the DPCs in BGZ interim storage facilities are CASTOR® type casks. All casks are subject to the same requirements. These requirements arise from the applicable legal transport, storage and associated regulations, handling interfaces (e.g. for loading in the NPPs) and the properties of the radioactive inventory. For this reason all stored cask types have basically similar design features (see Figure 4).

Figure 4: CASTOR® cask in storage configuration (example: CASTOR® V/19)

The casks have a metallic cask body. The thickness of the cask walls is determined primarily by shielding requirements. In line with the recommendations of the Nuclear Waste Management Commission (ESK) [22], the casks have a monitored double-lid sealing system for storage. Both lid systems (primary and secondary lid system) are screwed to the cask body independently of each other.

All cask types use highly durable Helicoflex® metal seals with an outer jacket of silver or aluminium. A basket is used to position the inventory inside the cask. Most of the cask types are also equipped with moderator materials to improve neutron shielding.

At the time applications will be made for extended interim storage, all the casks for LWR-FA and HAW canisters (stainless steel canisters filled with HAW in the form of vitrified residues from reprocessing) will have been loaded and stored in the interim storage facilities. This was not the case when the original 40-year storage licences were applied for. This means that the real boundary conditions for casks and their inventories as well as the occupancy of the respective interim storage facilities are known. The interim storage facilities are therefore in static operation mode without new emplacements. This enables to determine for each individual cask its thermal load history and its operational history.

This disposal and loading strategy will determine the situation in 2050, the date on which commissioning of the repository is planned (see Figure 5). By this date, fewer than 50% of the stored casks will have exceeded the previously considered and licenced storage period of 40 years. The oldest casks will have an operating period of 58 years. However, many of the older cask types were loaded with fuel assemblies from research and prototype reactors without any significant thermal load. In these cases, thermally driven ageing effects will be less pronounced. If only those casks with a significant thermal load at the time of loading are considered, fewer than 30% will have an operating period of more than 40 years by 2050 and only 2% an operating period of longer than 50 years.

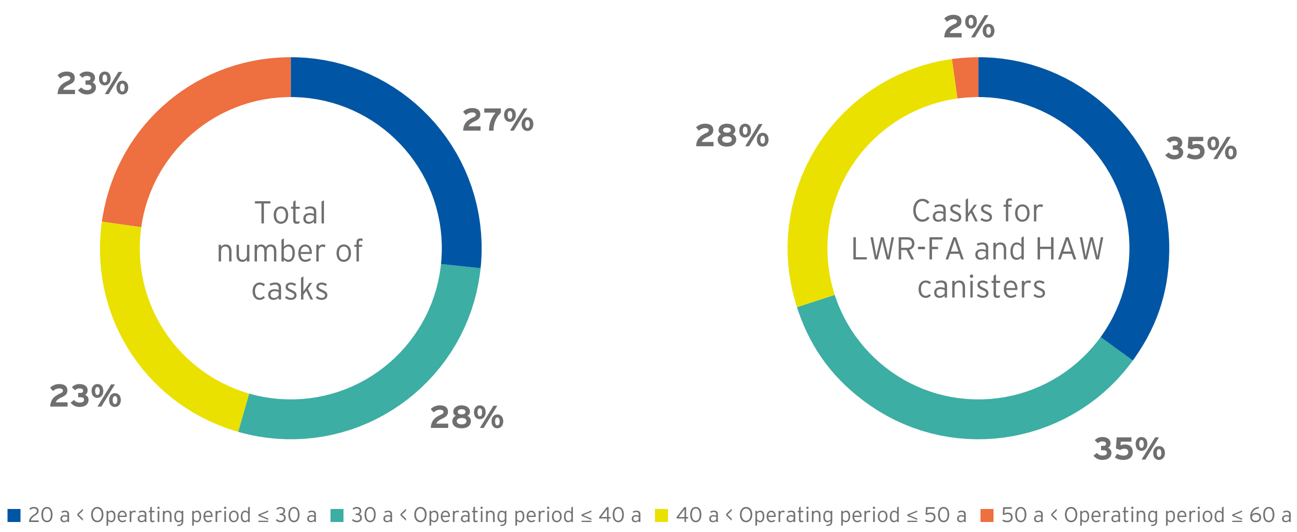


Figure 5: Operating period in years of stored casks with regard to the year 2050

In terms of actual cask loadings and their thermal load, the overall picture can be made more precise. The thermal load of casks with LWR fuel assemblies at the time of loading was about 20 kW on average, although it tended to be lower for the first cask loadings. Before 2006, a thermal load of 25 kW was rarely exceeded. With the continuous disposal of SNF arising from the operation of the NPPs, the thermal load of the inventory also increased. This was because it was no longer possible to benefit from long-decayed inventory to the same extent. This applies in particular after the shut-down of NPPs, as the final reactor core loading also needs to be disposed of as soon as possible in order to be able to start dismantling at an early stage. In the first few years, loading tended to take place with lower thermal loads, so that older casks in particular were exposed to lower temperatures from the outset and temperature-driven processes consequently have a less pronounced ageing effect. On average, only about 50% of the approved design thermal load is utilised in casks for LWR-FA at the time of loading and to date only 7% of loadings have been carried out with a thermal load of more than 30 kW. These rather large discrepancies between approved and actual thermal load result from the fact that, in the interest of disposal safety at power plants, designs must be suitable to cover even the most unfavourable loading requirements. In reality, loadings have been made according to the dose rate minimisation concept in such a way as to ensure the greatest possible limit margin.

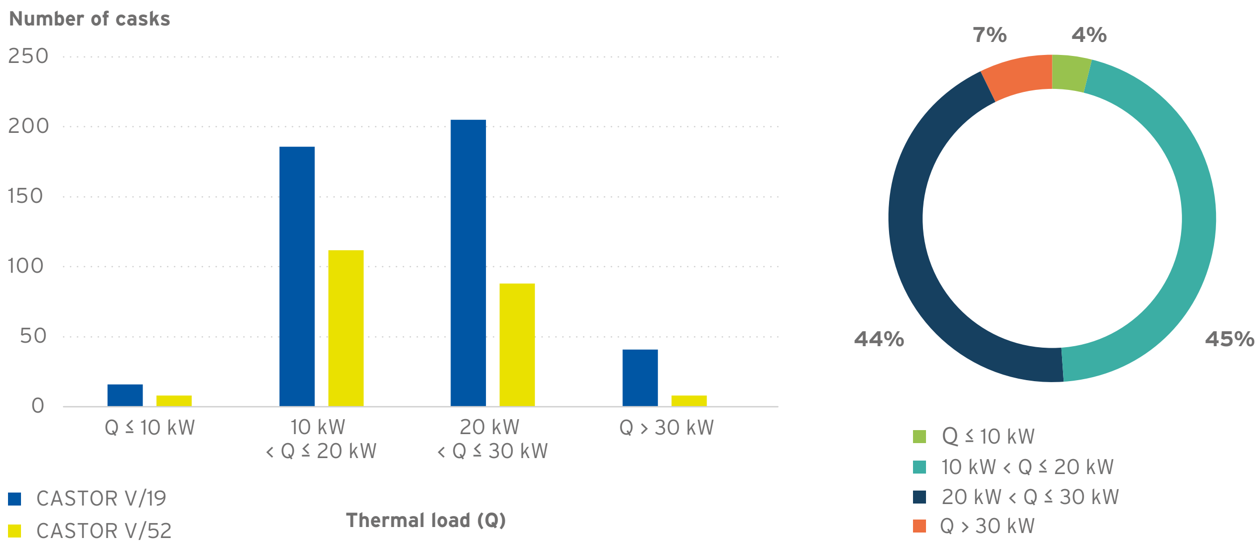


Figure 6: Distribution of the thermal load of CASTOR® casks (LWR-FA) at the time of loading, end of 2020

On the other hand, a much narrower thermal load range applies to casks for HAW canisters. The reason for this is that the HAW canisters are produced with predefined activity in a controlled manufacturing process. The differences between the loadings of individual HAW casks therefore result from the permissible thermal load for each cask type and the time period between the production (casting) of the glass and the loading date of the cask. About 80% of the HAW casks' thermal design load is therefore utilised on average whereby only 10% of the cask loadings reach a thermal load of between 45 and 50 kW (see Figure 7).

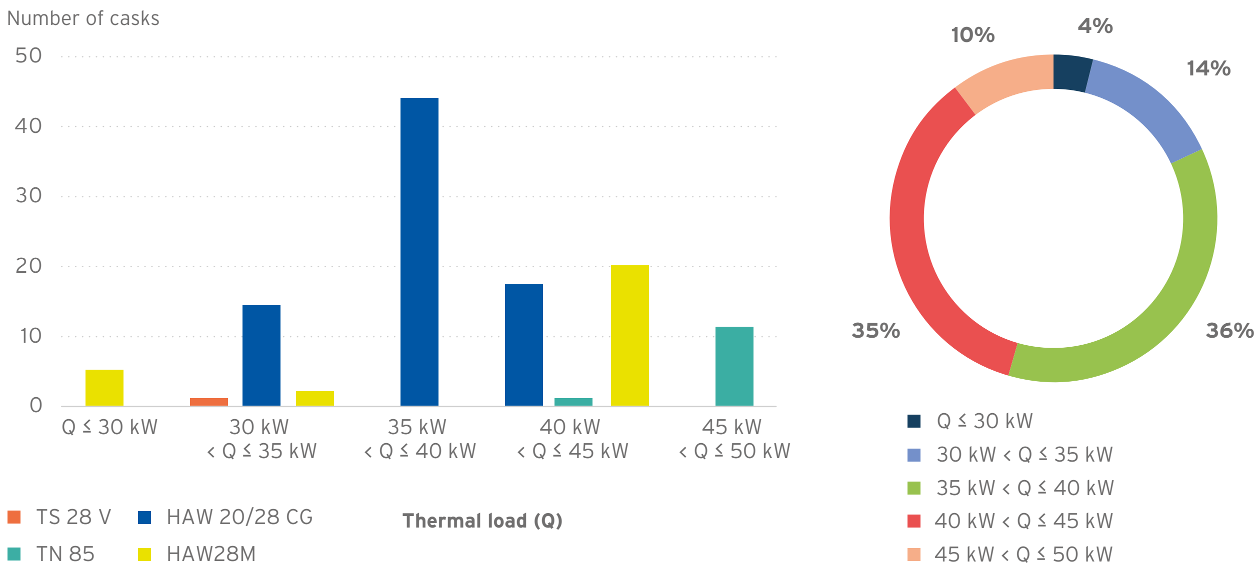


Figure 7: Distribution of the thermal load of casks for HAW canisters at the time of loading, end of 2020

As the thermal load of individual casks and the overall thermal load of the interim storage facilities are significantly lower than that for which they are designed, the real component and inventory temperatures in the storage period, which are decisive for the ageing processes, are significantly lower than originally assumed. This applies in particular to the casks with the longest operating time, as their thermal load was rather low at the time of loading and they were placed in initially empty storage facilities. The assessment of the ageing behaviour of casks (and inventories) can therefore capitalise on this fact when determining research requirements.

8.2 Determining the need for research

Focusing on the safety objectives, BGZ concludes that temperature-driven ageing mechanisms in particular become less significant during extended interim storage as decay heat diminishes during storage. No fundamental changes may be expected compared to the previous assessment for 40 years. Much the same applies to ionising radiation. The gamma energies and neutron flux deposited over the storage period are orders of magnitude too low to lead to relevant changes in the properties of the structural materials. An extended storage period is not expected to lead to any change in the safety assessment. The consumption of neutron absorbers by neutron capture is negligible compared to the available quantity, even over long storage periods. This has no influence on the safe maintenance of subcriticality. If, bearing in mind the inert atmosphere and the material used, corrosion processes occur at all within the containment (cask cavity, inter-lid space), they stop completely already at the beginning of the storage due to the limited amount of residual water. In addition, the outer cask areas are inspected as part of the ageing management programme. Any finding can be remedied at any time by taking maintenance measures. Therefore, there is no identified need for research on these ageing mechanisms.

The situation is different for metal seals and lid screws as these are the central cask components for ensuring the safety objective of containment of radioactive materials. Both components are subject to relaxation processes that lead to a reduction in the resilience of the metal seal or to a potential decrease in the screw preload. Additional studies of the long-term sealing behaviour are regarded as important and are therefore included in the research programme (see Chapter 11.1.1). Low temperature levels and associated low creep are expected to cause only minimal screw relaxation changes, which have been taken into account in the design of the screwed connection. On the other hand, retightening of the screwed connections is possible in principle.

Other ageing mechanisms which may impact compliance with a safety objective concern the moderator material. The shielding properties of this material may deteriorate as a result of creep processes and associated changes in arrangement, as well as of thermal and radiation-induced decomposition processes. However, ageing behaviour has already been described in full within the framework of the current licence procedure and can be extrapolated accordingly to the extended interim storage period. In addition, the decrease in activity associated with further decay of the radioactive inventory counteracts any possible deterioration in shielding properties. There is therefore no need for specific research on moderator behaviour. However, the dose rate and temperature measurement programme (see Chapter 11.1.2) focuses explicitly on the integral assessment of shielding properties.

Additional need for action arises from ageing management and BGZ's operating experience with regard to pressure switches. Despite the extremely low failure rate so far, work is being undertaken to optimise this component in the light of a certain accumulation of defects at the contact bushings of the pressure switch. Even though, on the basis of the current findings, no doubts exist regarding the continued usability of the installed pressure switches during extended interim storage, a test programme is currently underway (see Chapter 11.1.3, DPOPT) with the aim of approving and using an improved design of pressure switches in the future.

9. Need for research on inventory

The inventories stored in the BGZ interim storage facilities can be divided into fuel assemblies and waste from reprocessing. The fuel assemblies can be further categorised into those from light water reactors (LWRs), i.e. from power reactors (both pressurised water reactors (PWRs) and boiling water reactors (BWRs)), and the fuel assemblies from German prototype and research reactors.

9.1 LWR fuel assemblies

9.1.1 Inventory review

The LWR fuel assemblies stored in the BGZ interim storage facilities share the same basic design: a defined number of fuel rods are arranged in a quadratic lattice (see Figure 8). The number of rods in a fuel assembly varies from 8x8 to 18x18. Depending on the reactor and the fuel assembly manufacturer, the design of the fuel assemblies includes rods that only contain fuel over a certain length (so-called partial-length fuel rods) and non-fuel rods. The LWR fuel assemblies can further be assigned to different categories, i.e. according to reactor type (PWR/BWR), cladding material, fuel type or also according to characteristics based on the individual irradiation history such as burn-up and decay heat. Mixed oxides (MOX) and uranium from reprocessing (enriched reprocessed uranium, ERU) were used as fuel in addition to U02. The enrichment and burn-up of the fuel assemblies increased over the operating time, i.e. the enrichment increased to approximately 4.9 weight % of U-235 and the nominal fuel assembly target burn-up to a maximum of 65 GWd/tHM. The fuel was filled in the form of pellets into fuel rod cladding tubes made of various zirconium alloys. PWR fuel assemblies can essentially be divided into Zry-4, different duplex variants and the materials ZIRLO® and Optimized ZIRLO® from Westinghouse and M5® from Framatome. Zry-2 cladding with and without liners were used for the BWR reactors.

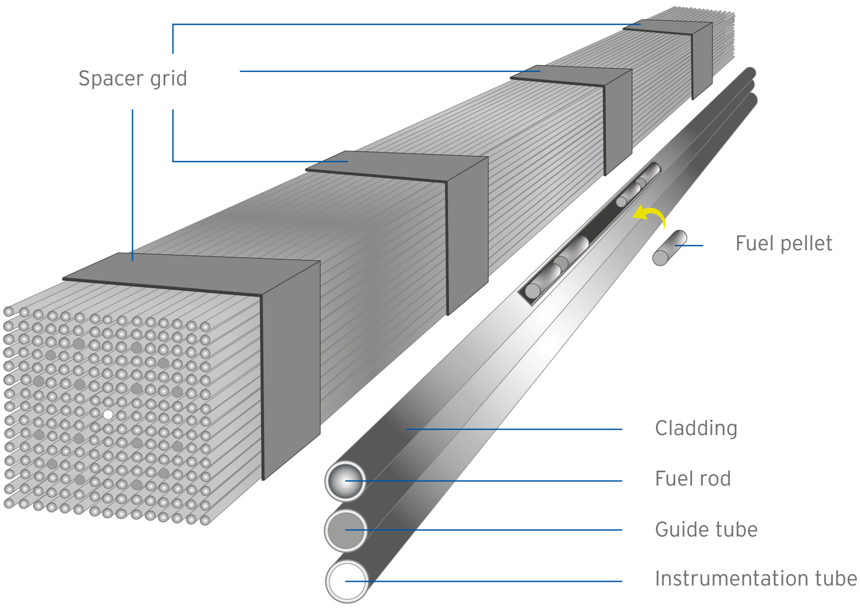


Figure 8: Schematic representation of a PWR fuel element

Fuel assemblies and fuel rod cladding fulfil safety-related functions in the safety concept for the transport and storage casks. The fuel rod cladding tubes and their combination into fuel assemblies ensure that the fuel always remains in the same geometrical arrangement. The cladding is the primary barrier for SNF that prevents the uncontrolled dispersion of radioactive substances into the cask interior.

Unlike the initial applications for 40-year storage licences, all fuel assemblies will have been loaded and stored in casks in the interim storage facilities when future applications are made for extended interim storage. The real boundary conditions for loadings and inventories will therefore be known. Currently, there are about 19,000 LWR fuel assemblies in BGZ-operated interim storage facilities. Based on the disposal and loading strategy referred to above, the last loadings of LWR fuel assemblies that are currently in the operating reactors and the cooling ponds are expected to be completed in 2027. This means that about 35% of all the LWR fuel assemblies are currently still in the cooling ponds or reactor cores of the nuclear power plants. After final emplacement, BGZ expects a total of about 29,000 SNF assemblies from power reactors to be stored in more than 1,000 casks.

Less than 7% of the fuel assemblies expected to be stored by 2027 will contain MOX fuel. The burn-ups of the inventories stored up to that point consist of: 31% below 40 GWd/tHM, 53% with burn-ups between 40 and 55 GWd/tHM and 16% with burn-ups above 55 GWd/tHM[3]. Zry-2 with and without liner are used as cladding materials for the BWR fuel assemblies. Zry-4 and M5 as well as various duplex variations and Optimized ZIRLO® are the main cladding materials used for the stored PWR fuel assemblies. In addition to the share of cladding materials for the fuel assemblies stored up to 2027, the shaded inner circle of Figure 9 also shows the respective share of burn-ups.

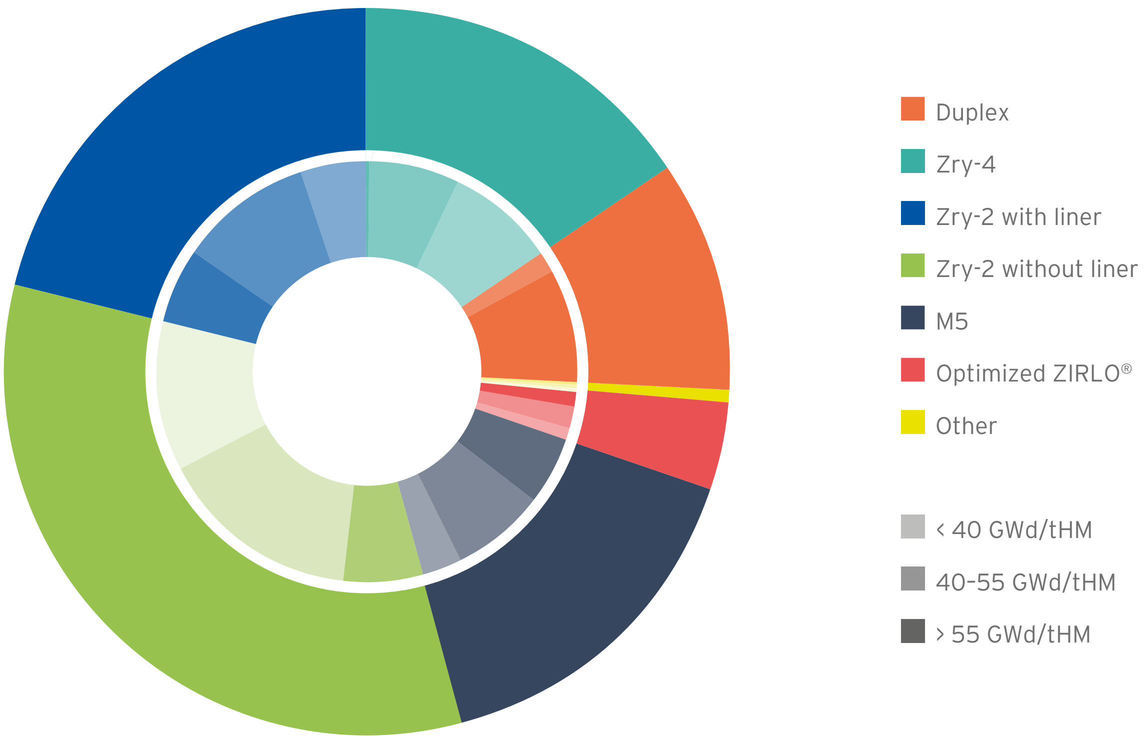


Figure 9: Share of cladding materials with associated burn-up for the entire inventory probably stored by 2027

3 Some changes may still occur as there are still some fuel assemblies in the reactor cores of reactors that are still in operation.

The different colours represent the different cladding materials and the circle segments show their frequency of occurrence. The shading in the inner circle shows three burn-up classes: up to 40 (lightest shading), between 40 and 55 (light shading) and greater than 55 GWd/tHM (solid colour).

The first licences to expire are those for the central interim storage facilities in Gorleben and Ahaus. These are also the inventories that, by 2050, will have been in dry interim storage for more than 40 years. The inventories stored there are described in more detail in the following. The first casks for dry interim storage of irradiated LWR fuel assemblies from German power reactors were placed in the Gorleben interim storage facility. The casks stored are one CASTOR® Ic with UO_2 BWR fuel assemblies, one CASTOR® IIa with UO_2 PWR fuel assemblies and three CASTOR® V/19 with UO_2 PWR fuel assemblies. All five casks were emplaced up to 1997. No further casks with LWR fuel assemblies were delivered to Gorleben after that date. The burn-ups of the fuel assemblies are comparatively low, at around 30 GWd/tHM. The maximum burn-up is around 34 GWd/tHM, the minimum 22 GWd/tHM. The cladding tube materials are Zry-4 for the PWR fuel assemblies and Zry-2 for the BWR fuel assemblies.

A total of six casks containing LWR fuel assemblies are stored in the Ahaus interim storage facility: three of type CASTOR® V/19 with PWR fuel assemblies and three of type CASTOR® V/52 with BWR fuel assemblies. The burn-ups here are also comparatively low; only two of the 213 fuel assemblies are slightly above 40 GWd/tHM. The cladding tube materials are also exclusively Zry-4 and Zry-2.

The first emplacement of a cask containing SNF in an on-site interim storage facility took place in Lingen in December 2002. Further casks were afterwards emplaced in interim storage facilities at the Gundremmingen, Grohnde, Biblis, Grafenrheinfeld, Krümmel and Neckarwestheim sites in 2006. The interim storage facilities at the Isar, Brokdorf, Philippsburg and Unterweser sites began operation in 2007.

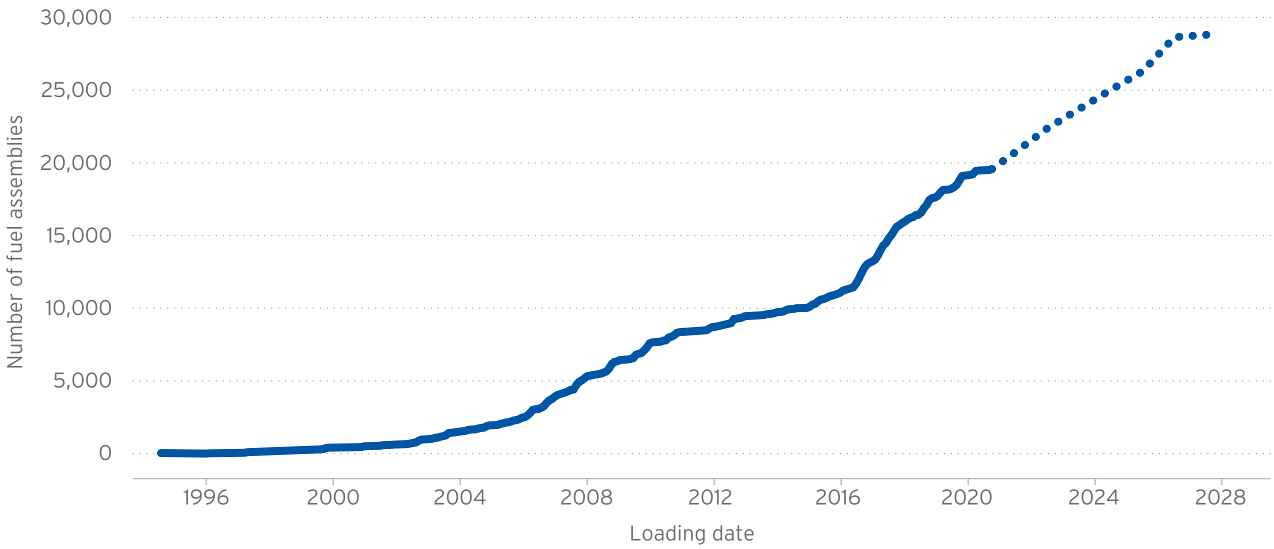


Figure 10: Number of LWR fuel assemblies currently stored and still expected from the power reactors in the interim storage facilities operated by BGZ

9.1.2 Determining the need for research

BGZ's strategy concerning its research requirements is based on a timeline resulting from the expiry of the storage licences and the duration of dry interim storage, as well as the classification and coverage of the stored inventory by existing national and international research programmes.

BGZ has come to the conclusion that further research work needs to be undertaken on describing fuel rod behaviour and demonstrating fuel rod integrity. This assessment is consistent with national [23, 24] and international studies and publications [25, 26, 27] as well as with the gap analyses commissioned by BGZ from fuel assembly and cask manufacturers.

The current criteria for maintaining fuel rod integrity were developed in order to limit damage mechanisms and to exclude a systematic loss of cladding tube integrity. These mechanisms include cladding creep due to internal gas pressure, delayed hydride induced cracking (DHC) and cladding corrosion due to oxidation. To limit these effects and thereby guarantee the integrity of the fuel rod cladding tubes during dry interim storage, the cladding tube temperature was limited to a maximum of 370°C, the circumferential stress to a maximum of 120 MPa and the circumferential strain to 1%. Limiting the maximum cladding tube temperature during handling and storage to 370°C prevents annealing of the radiation hardening (thereby limiting creep rates). It also limits the dissolution of existing hydrides in cladding tubes with higher hydrogen absorption (such as Zircaloy-4). For the fuel rod cladding tubes, the maximum temperature after reactor use occurs during the drying process in the cask. During subsequent storage the temperatures decrease in line with the decreasing decay heat of the fuel assemblies. The temperature distribution in the sealed casks varies over time as well as in axial and radial alignment and depends on the actual loading scheme (i.e. the decay heat of the fuel assemblies and their arrangement in the cask) as well as the influence of neighbouring casks. The conservatism of previous verification calculations overestimated temperatures in the cask. Knowledge of a more realistic temperature distribution is necessary for the evaluation of further effects that may be of significance for extended interim storage beyond the 40 years.

Limiting the maximum tangential stress in the cladding tube to 120 MPa prevents excessive creep rates from occurring, which could result in cladding tube failure. The tangential stress is also one of the boundary conditions for hydride reorientation during and after drying. Limiting the maximum tangential stress also limits this effect.

Limiting the maximum tangential strain of the cladding tube during interim storage to 1% prevents excessive creep elongation of the cladding tube as a result of the tangential stress in the cladding tube wall. Relevant creep rates require a temperature of at least 300°C, which is only present in the first months to years due to decreasing decay heat.

These criteria have been stipulated for the current licence for a period of 40 years. The objective of BGZ research is to expand the criteria for extended interim storage to enable safe interim storage and subsequent transport. Research will also be undertaken to determine whether other damage mechanisms can lead to cladding tube failure, such as:

- **Hydrogen migration and redistribution**: The hydrogen absorbed during irradiation accumulates locally during storage leading to loss of ductility.
- **Hydride reorientation**: The hydrogen dissolved during the drying process increasingly precipitates in radially oriented hydride platelets resulting in loss of ductility.
- **Delayed hydride cracking**: Hydrogen deposits at an existing crack tip and leads to a local increase in volume due to hydride formation, which can result in further crack growth and a possible loss of integrity.
- **Stress corrosion cracking** leading to loss of integrity

As referred to above, these effects depend on the cladding tube material as well as the individual irradiation history and thermal development.

Verification for the purposes of extended interim storage requires the generation of additional experimental data and the implementation and validation of analytical procedures, methods and models in the required analytical prediction methods. Many of the relevant topics are already the subject of international research and development, although not always in the parameter ranges representative of LWR fuel assemblies from German power plants. Numerous studies have been undertaken on the behaviour of hydrogen during dry interim storage when temperatures decrease too quickly, for example. The conclusions drawn from these studies may differ significantly from those undertaken with realistic cooling rates [28, 29]. Research geared towards safety goals should, whenever possible, use results that are based on assumed boundary conditions prototypical for Germany. The assumption of non-prototypical boundary conditions always requires subsequent transfer of the results into the relevant prototypical parameter ranges. The same applies to the investigation of individual effects in comparison to integral tests on irradiated fuel rods. Interactions with other single effects must always be taken into account when results from single effect tests are transferred to the real situation. For the further procedure, BGZ concludes that in principle experimental work should be pursued as integral tests in prototypical parameter ranges.

Certain long-term effects cannot be investigated directly because the corresponding material is not available: there are no irradiated fuel rods with prototypical boundary conditions that have been in dry interim storage for 60 or 70 years. Effects such as the release of fission gas in the sealed cladding tube over the entire storage time can only be described theoretically and must be verified experimentally as far as possible, for example by means of examining individual rods.

A consistent and validated procedure of prediction involving calculating and simulating the behaviour of the cladding tube is needed to evaluate the relevance of individual effects for cladding integrity. Far more development work has been done on the isotopic composition of the fuel, as well as the thermal development of the cladding tube temperatures, than on predicting cladding mechanics [30]. There is still a lack of models, procedures and, above all, experimental data in the corresponding calculation programmes for validation in the parameter ranges relevant for Germany. Further experimental work in the relevant parameter ranges can make a decisive contribution to the reliability of theoretical prediction of compliance with safety objectives.

The expiry dates of each licence and the inventory affected must be taken into account to determine the need for research with regard to the inventories actually stored by BGZ. As shown in Figure 1, the first licences to expire will be those for the central interim storage facilities in Ahaus and Gorleben. Research and development to validate cladding integrity has the highest priority for these inventories. The inventories stored there consist of BWR and PWR fuel assemblies with low burn-ups. With the exception of two fuel assemblies in Ahaus with 40 and 43 GWd/tHM, the averaged burn-ups of the almost 300 fuel assemblies are (in some cases significantly) below 40 GWd/tHM. In 2050, these fuel assemblies will have been in dry interim storage for more than 40 years. The fuel assemblies are similar in their properties with respect to cladding material and in their irradiation history to those described in the CASTOR® V/21 PWR Spent Fuel Storage Cask Performance Test [31]. In the test programme, an already loaded CASTOR® V/21 cask was opened after 14 years of storage and the stored fuel assemblies were examined experimentally. The data obtained from the experiments and investigations have not demonstrated any significant damage. As long-term effects, such as the build-up of fission gases and the associated increase in internal pressure, play a subordinate role in these comparatively low burn-ups according to the current state of the art in science and technology, the data obtained in the test programme are considered extremely important.

The majority of the fuel assemblies stored in BGZ interim storage facilities have a burn-up of 40 to 55 GWd/tHM averaged over the fuel assembly. Some national and international research has been carried out on fuel assemblies with corresponding burn-ups, such as the High Burnup Dry Storage Research and Development Project in the USA [32], in which 32 fuel assemblies from power reactors in the USA were packed into a TN®-32 cask. The fuel assemblies will be removed from the cask in 2027 (that is after ten years of storage) and examined in detail. The inventory stored in the TN®-32 includes PWR fuel assemblies with Zry-4 (2 fuel assemblies), M5® (18) and ZIRLO® (12) cladding types. Burn-ups range between 50 and 55.5 GWd/tHM. Initial enrichments range from 4.2 to 4.55 weight % of U-235. The Zry-2 BWR cladding tube materials with and without liner as well as PWR cladding tubes made of Duplex or Optimized ZIRLO® are not being investigated. In these cases, reference can be made to studies from programmes such as SCIP IV or also studies of fuel element manufacturers such as Westinghouse [29]. The evaluation of the data and its transfer to actual stored inventories has highest priority for BGZ.

The smallest proportion of fuel assemblies stored in the BGZ interim storage facilities has an average burn-up of more than 55 GWd/tHM (see Figure 9). Comparatively high burn-ups of 65 GWd/tHM for UO_2 and MOX fuels are, with the exception of Switzerland, not envisaged internationally. This means that existing international research activities cannot be applied to extended interim storage for these high burn-ups and especially not to MOX fuels. BGZ has identified need for research on the fuel assemblies stored at on-site interim storage facilities.

The actual period of time of dry interim storage must be taken into account in order to comply with the safety objectives. The oldest fuel assemblies, i.e. those with the longest storage time in dry interim storage, are the early loaded, low burn-up fuel assemblies. Until 2006, no fuel assembly with burn-ups above 50 GWd/tHM were packed in casks. Until 2010, few individual fuel assemblies with burn-ups above 60 GWd/tHM were packed, but the majority had burn-ups below 55 GWd/tHM. Assuming that a repository is in operation in the year 2050, these fuel assemblies will have been in dry interim storage for more than 40 years. A percentage distribution of the dry interim storage time of the fuel assemblies as of 2050 is shown in Figure 11. Around 25% of the fuel assemblies will then have been in dry interim storage for more than 40 years at this point.

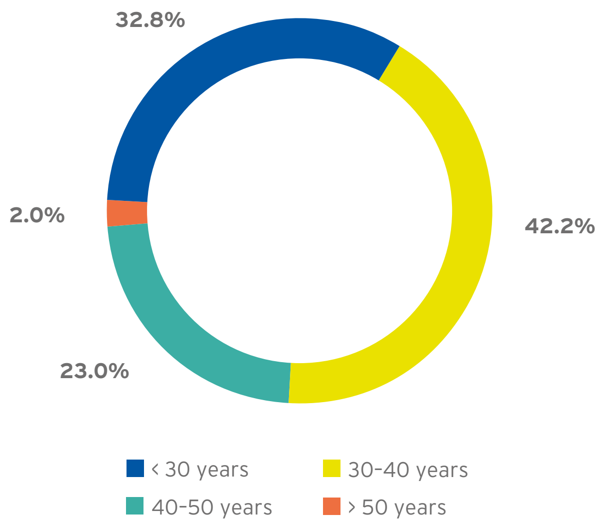
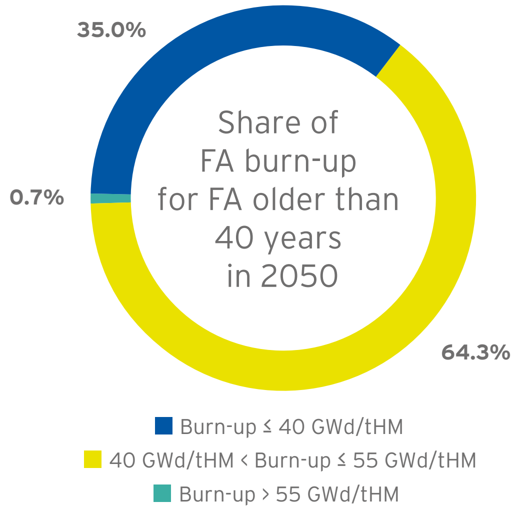


Figure 11: Distribution of the storage time of the stored LWR fuel assemblies up to the year 2050

Figure 12 shows the distribution of burn-ups for the proportion of fuel assemblies in interim storage for over 40 years. The majority of the fuel assemblies have an average burn-up of between 40 and 55 GWd/tHM. Only 0.7% of the fuel assemblies have a burn-up greater than 55 GWd/tHM.



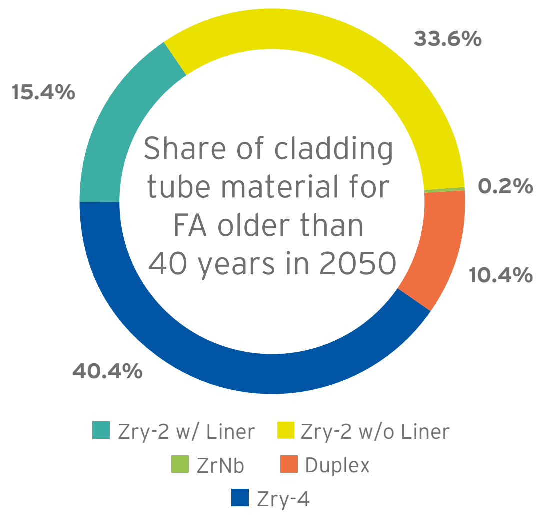


Figure 12: Distribution of burn-ups (left) and cladding materials (right) of fuel assemblies in dry interim storage for longer than 40 years in 2050

According to the current state of the art in science and technology, the current licence criteria are sufficient for a period of 40 years [25]. These 40 years would be exceeded for the fuel assemblies referred to above. Work to extend the current state of the art in science and technology to exclude systematic cladding tube failure is therefore of huge importance for these fuel assemblies.

The remaining inventory will have been in the casks for less than 40 years by 2050. The fuel assemblies last packed in 2027 will only have been packed in the casks for 23 years. However, it takes several years to move the casks with LWR fuel assemblies to the receiving storage facility for the repository. BGZ anticipates that it will take at least 30 years to transport all 1,500 casks. The figures and ratios referred to may change depending on retrieval, conditioning and storage processes. This is taken into account as part of the continuous evaluation of research priorities in the research programme. BGZ initiated several projects, such as LEDA (see Chapter 11.2.4), in which it is investigating the main issues relating to extended interim storage, including also LWR fuel assemblies that will not have been stored for over 40 years by the year 2050.

9.2 Fuel assemblies from research, experimental and test reactors

Currently, the SNF from the Thorium High Temperature Reactor (THTR-300) and the Rossendorf Research Reactor (RFR) are located in interim storage facilities operated by BGZ.

The irradiated fuel elements of the THTR-300, which was shut down in 1989, are stored in 305 CASTOR® THTR/AVR casks in the Ahaus interim storage facility. The inventories that contain nuclear fuel are spherical fuel elements or plate-shaped fuel elements from the burn-up measurement reactor of the THTR.

The spherical fuel elements have a diameter of about 60 millimetres and consist of a graphite matrix in the centre, which contains the fuel in the form of coated particles, and an outer fuel-free graphite shell (see Figure 13). The fuel particles consist of a central fuel core and a coating system to retain the fission products. The multi-layer coating system is composed of a low-density buffer layer and a high-density outer layer – each made of pyrolytically deposited carbon – separated by an intermediate layer of silicon carbide. The fuel core consists of either carbide or oxide uranium and/or thorium compounds, and the uranium may be enriched up to 93 weight % with U-235.

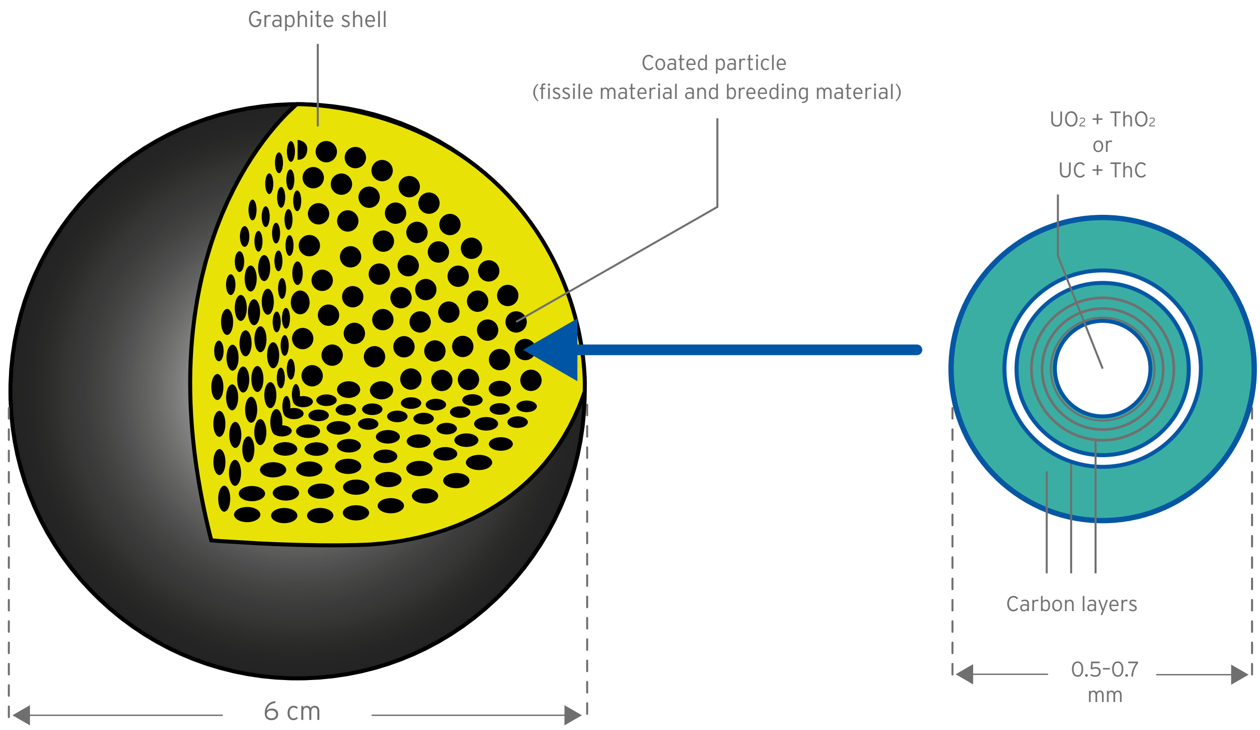


Figure 13: Diagrammatic representation of the THTR fuel pebbles

The spherical fuel elements are placed in a tightly welded canister made of stainless steel for interim storage in the CASTOR® THTR/AVR casks. One canister contains a maximum of 2,110 fuel elements, with an average burn-up per canister of 0.114 GWd/kgU. Already at the time of loading in 1992, the thermal load did not exceed 1 kW per canister or cask after a minimum decay period of three years.

The fuel elements from the burn-up measurement reactor belonging to the THTR are composed of rectangular plates made of an alloy containing 80 weight % of aluminium and 20 weight % of enriched uranium. The average burn-up of 0.13 MWd/kgU is also extremely low. The total of 767 fuel elements of this type were packed in a total of two casks.

The irradiated fuel assemblies of the RFR have been stored in 18 CASTOR® MTR2 casks in the Ahaus interim storage facility since 2005. The inventory consists of 803 type WWR-M and M2 fuel assemblies and 147 EK-10 fuel assemblies and a canister with 16 individual rods of the same fuel assembly type. The respective fuel is UO_2 with an initial enrichment of 36 weight % (WWR-M/M2) or 10 weight % (EK-10) of 235-U. RFR fuel assemblies of type WWR-M/M2 have a fuel matrix of UO_2-Al enclosed in Al casing. The fuel matrix is contained in two nested cylindrical tubes and an enclosing hexagonal tube. Type EK-10 RFR fuel assemblies contain fuel rods with a fuel matrix of UO_2-Mg surrounded by an Al cladding tube. Individual fuel rods of dismantled EK-10 RFR fuel assemblies are also bundled in fuel rod canisters, the dimensions and nuclear properties of which correspond to EK-10 fuel assemblies. The thermal load of the fuel assemblies was a maximum of 103 watts per cask when loaded.

All the fuel element types from research and prototype reactors described here are subject to relatively low thermal and radiological loads during interim storage. Due to low decay thermal load, the temperature level is well below the critical values that could have an influence on the behaviour of the material. Along with the relatively low burn-ups and the associated low activity inventory, the level of neutron radiation is also significantly lower than for casks loaded with LWR-FA.

At present, BGZ has not identified any need for research on fuel assemblies from research and prototype reactors for the purpose of demonstrating compliance with the safety objectives for extended interim storage.

Notwithstanding this, international experience and the developing state of the art in science and technology in the transport and storage of similar fuel assemblies will continue to be compared with the state of the art presented here.

9.3 Vitrified waste

The vitrified waste consists of highly radioactive waste from the reprocessing of irradiated LWR fuel assemblies from German power reactors. This HAW is already conditioned for final disposal in the form of so-called CSD-V (Colis Standard de Déchets Vitrifiés, respectively standard waste package – vitrified) or HAW canisters (see Figure 14).

Source: GNS

Figure 14: Loading of a cask with HAW canisters (left), single canister (right)

The reprocessing of the LWR fuel assemblies and the vitrification of the waste took place in the reprocessing plants La Hague (France) and Sellafield (England). For reprocessing, the irradiated nuclear fuel is initially crushed mechanically and dissolved chemically. This is followed by separating the recyclable isotopic fuel components (uranium and plutonium) from the waste for reuse in MOX and ERU fuel assemblies. The remaining concentrate of actinides and fission product residues is bound in borosilicate glass during the vitrification process. The resulting glass melt is poured into standardised stainless steel canisters at a temperature of about 1,000°C. After cooling to approx. 500°C and the solidification of the HAW glass, the canisters are sealed with a welded-on steel lid. The aim of vitrification is essentially to immobilise the highly radioactive waste in the glass matrix for final disposal, i.e. so that the vitrified waste can be stored for several thousand years.

In addition to the glass matrix, the 5-millimetre-thick stainless steel canister is a further barrier for the retention of radioactive particles. The canister also enables safe handling of the HAW glass, for example when loading the dual-purpose casks for temporary storage.

The long-term behaviour of HAW canisters has already been extensively investigated and the glass production process has been optimised in the framework of research on repositories. The low thermal, mechanical and radiological loads during interim storage rule out significant radionuclide mobilisation.

Based on the current state of the art in science and technology, BGZ does not see an urgent need for further research on HAW canisters for the purpose of demonstrating compliance with the safety objectives for extended interim storage.

Nonetheless, the lessons learned internationally regarding the transport and storage of HAW canisters will continue to be compared with the status presented here in line with the developing state of the art in science and technology.

10. Need for research on interim storage buildings

In Germany, dual-purpose casks for spent fuel or vitrified HAW from reprocessing are stored in interim storage buildings made of reinforced concrete. The primary safety objectives, such as criticality safety, shielding, containment and heat dissipation, are met by the dual-purpose casks themselves. The storage buildings contribute indirectly to ensuring compliance with the safety objectives by fulfilling the following safety functions:

- protecting the DPCs from environmental influences,
- additional shielding of ionising radiation,
- ensuring heat dissipation from the casks to the environment,
- preventing damage arising from accidents or beyond-design-basis events.

The storage building ensures safety during normal operation, accidents, and natural events, such as earthquakes or floods, and also contributes to security. Interim storage buildings do this by providing protection against disruptive action or other interference by third parties aimed at theft or the release of nuclear fuel stored in the interim storage facility.

The two central interim storage facilities (see Figure 15) were designed in the 1970s when it was assumed that irradiated LWR fuel assemblies from the operation of German nuclear power plants would be temporarily stored at their sites until they were sent for reprocessing. Therefore, a storage capacity at these sites for 420 positions and application has been created for a total thermal load from the stored casks of 16 MW (Gorleben) or 17 MW (Ahaus). As a result of the change in disposal strategy, only 113 (Gorleben) and 56 (Ahaus[4]) storage positions have actually been occupied in the central interim storage facilities since they were commissioned in the 1990s. At no time has the total thermal load of the stored casks exceeded 5 MW (Gorleben) or 0.1 MW (Ahaus). The decentralised on-site interim storage facilities were taken into operation between 2002 and 2007. The planned storage capacities of the on-site interim storage facilities vary between 80 and 192 positions and between 2 MW and 7.4 MW total thermal load, depending on the demand for disposal identified at the time.

Interim storage buildings are essentially storage halls made of reinforced concrete that, in order to meet radiological requirements and to comply with designs for earthquakes and external impacts as well as security functions, have increased wall thicknesses compared to conventional buildings. In terms of their design, the interim storage facilities can be divided into the two central interim storage facilities and the on-site interim storage facilities, which have been constructed either according to the WTI concept or according to the Steag concept. The WTI concept is based on the design of the central interim storage facilities. One exception in terms of its construction is the interim storage facility in Neckarwestheim, which was designed as an underground tunnel storage facility due to its location in a former quarry.

All interim storage facilities have a reception or shipping area, a maintenance area and a storage area. Incoming casks are unloaded from the transport vehicle and checks are carried out in the reception or shipping area prior to storage. The casks also leave the interim storage facility via this area. Repair work and cask inspections can be carried out in the maintenance area if necessary. The dual-purpose casks are stored upright in the storage area. For permanent monitoring of their sealing function, the casks or, more precisely, their pressure switches are connected to the cask monitoring system of the interim storage facility. For radiation protection reasons, the storage area is locked with heavy access-controlled steel gates. Casks are handled using overhead cranes. Only one overhead crane is needed to access all areas of the single-bay storage buildings, i.e. for the central interim storage facilities and the on-site interim storage facilities according to the Steag concept (see Figure 16). As the

4 Several smaller casks (e.g. CASTOR® MTR or CASTOR® THTR/AVR) can be temporarily stored in the BZA for each approved storage space for large casks (e.g. CASTOR® V casks).

WTI interim storage facilities have a two-bay design, two overhead cranes for cask handling are arranged on parallel tracks. This means that the maintenance area can only be reached by one of the two cranes (see Figure 17). An additional transport vehicle is therefore needed to ensure that the maintenance station can be used for casks from both bays and that casks can be transferred from one storage bay to the other.

Figure 15: Photos of the central interim storage facilities – Ahaus (left) and Gorleben (right)

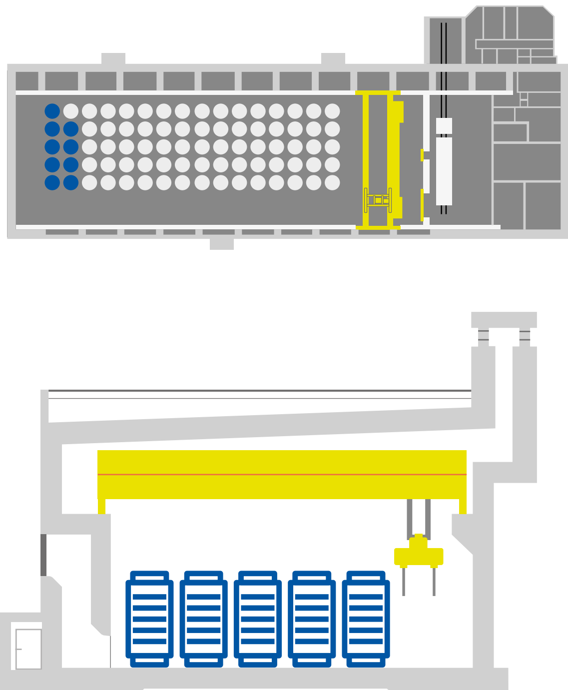


Figure 16: Simplified representation of the Steag concept – left: schematic cross and top view, right: interim storage building in Brokdorf (example)

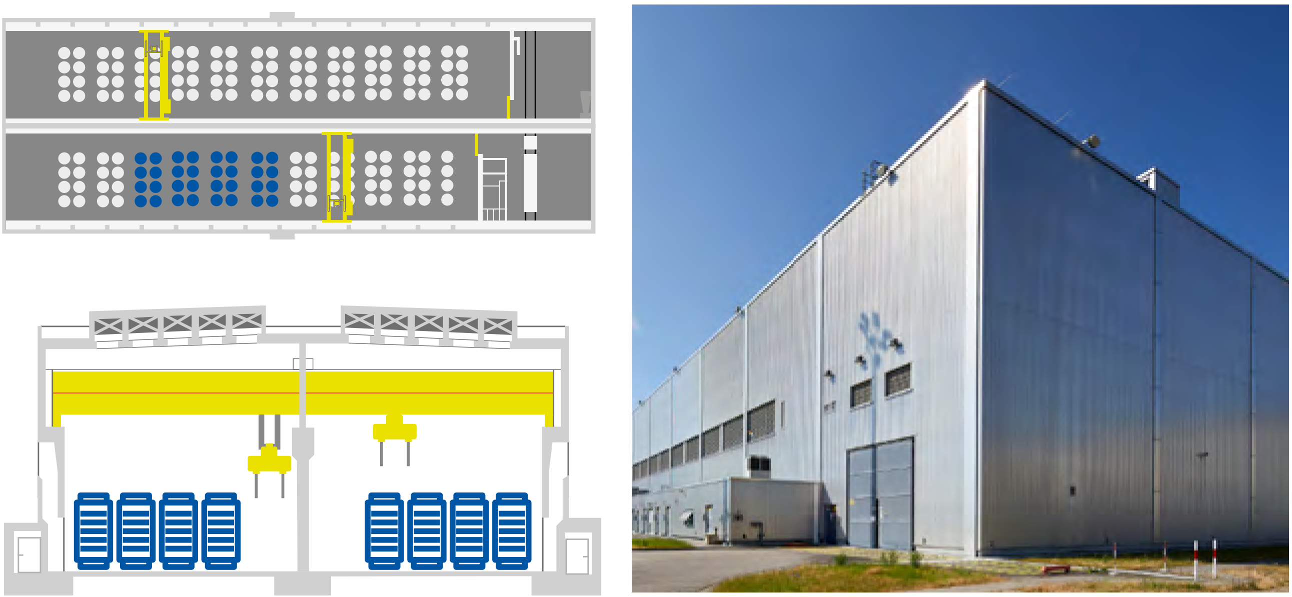


Figure 17: Simplified representation of the WTI concept – left: schematic cross and top view, right: interim storage building in Philippsburg (example)

The dimensioning of the air inlets and outlets in the interim storage buildings and the positioning of the casks ensure that the heat emitted by the casks is dissipated sufficiently to the environment. This means that, even at maximum occupancy, the temperatures of all load-bearing structures remain below critical values. So even the resulting maximum temperatures occurring in the building will not accelerate the ageing or weakening of the supporting structure. Furthermore, the shielding effect of the casks reduces the ionising radiation emitted by the inventory to such an extent that the energy doses applied in the building structures have only a negligible damaging effect. This means that the ageing mechanisms affecting interim storage buildings are much the same as in conventional buildings. BGZ therefore does not need to carry out any specific research in this respect. The dimensioning of the buildings for accident loads leads to relatively low stresses during normal operations. Consequently, ageing effects due to operational loads play at most a minor role.

Nonetheless, the building is regularly checked as part of cross-site ageing management. If necessary, the findings of such checks are used to take further measures to ensure the verification and long-term preservation of serviceability. This may also involve repair or renovation measures.

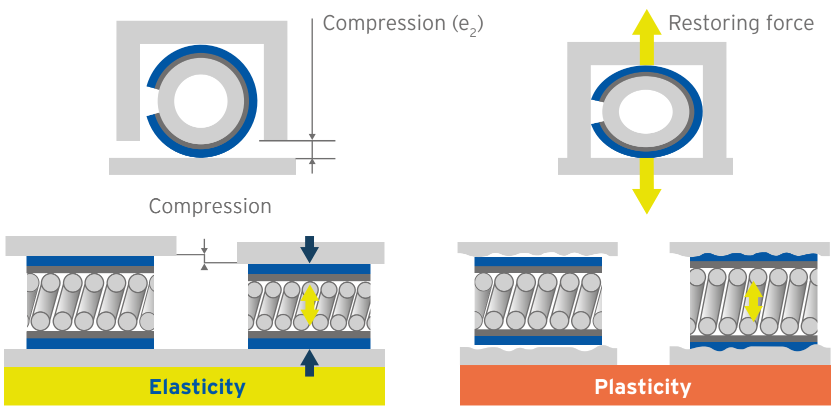
11. Research activities

11.1 Casks

11.1.1 MSTOR – Long-term behaviour of metal seals

Subject matter::	MSTOR – Metal seals during long-term storage » Extension of the existing experimental basis for the temperature-dependent ageing behaviour of metal seals » Development of a forecast model for sealing parameters
Project partners:	» GNS (cask manufacturer, Germany) » Technetics (seal manufacturer, France)
Project period:	2021 to 2031, longer if necessary

Safe confinement of radioactive materials during storage is ensured by the double-lid sealing system with compressed metal seals. Only Helicoflex® metal seals made by the French manufacturer Technetics are used in the sealing barrier. Helicoflex® type seals consist of a helical spring core and a stainless steel jacket covered by an outer liner made of aluminium or silver. The functioning of the seals is based on the elasticity of the helical spring (see Figure 18) that, when compressed, generates the restoring force that is necessary to maintain contact between the outer liner and the sealing surfaces. The plasticity of the outer liner also ensures that the surface of the seal is optimally adapted to the structure of the sealing surface. During compression, the liner material fills the flange imperfections in order to achieve a high degree of tightness with standard He leakage rates of less than 10^{-8} Pa* m^3/s (see Figure 18). The stipulated standard He leakage rate of 10^{-8} Pa* m^3/s is therefore also a system-specific quality criterion for required long-term functionality and is not a radiologically based tightness requirement, as this could also be met at higher leakage rates.



Ensuring that the sealing function of the metal seal complies with specifications

Figure 18: Functionality of Helicoflex® metal seals

During initial compression of the metal seal, a force-deformation curve is obtained as shown in Figure 19. At deformation e_0, the required tightness is achieved for the first time, but only at deformation e_2 is the seal at its operating point. The seal is compressed to the operating point thanks to the depth of the groove in which the seal is inserted in the lid, as the difference between the groove depth and the torus diameter of the seal corresponds exactly to the optimum compression e_2.

If the seal is decompressed as a result of external loads, i.e. if a gap emerges between the lid and the mating surface, the required tightness is maintained until e_1 is reached. The permissible gap between the sealing surface and the lid corresponds to the useful elastic recovery r_u (see Figure 19). In particular, this parameter and the associated force Y_1 are essential for evaluating the behaviour of the metal seal under operating and accident conditions.

In the assembly state, mechanical stress and temperature exposure lead to creep processes in the metal seal. These become noticeable in the form of relaxation. As shown in Figure 19, relaxation decreases the restoring force at the operating point to Y_{2r} with unchanged deformation. Although the minimum force required to maintain the specified tightness Y_{1r} decreases upon decompression, the deformation e_{1r} decreases accordingly, so that the remaining useful recovery r_{ur} for an aged seal decreases considerably compared to the assembly state.

Knowledge of the characteristic values Y_{1r} and r_{ur} of an aged seal is therefore of great importance to reliably evaluate the long-term behaviour of the metal seals. In particular, the remaining useful recovery r_{ur} is a measure for assessing the robustness of the sealing barrier, i.e. the extent to which the seal is able to maintain the required tightness even under external mechanical influences.

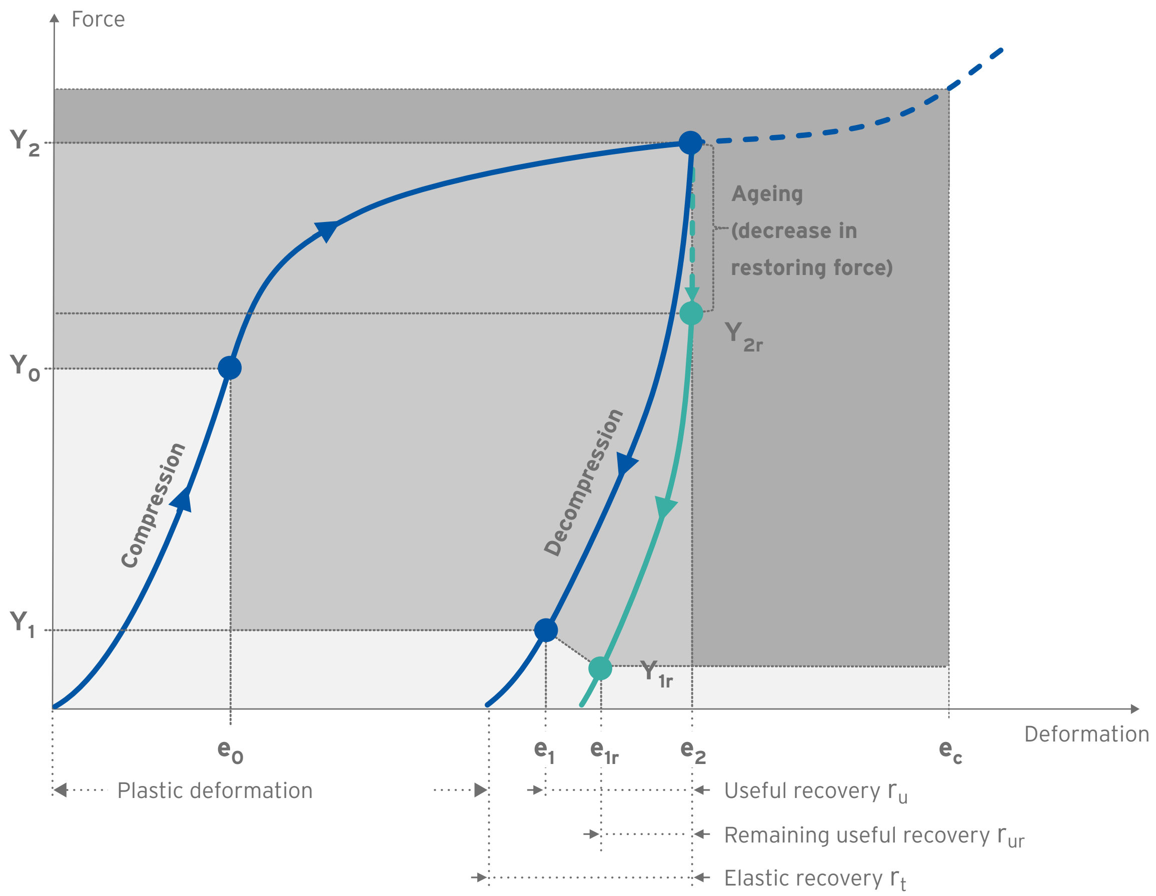


Figure 19: Change in characteristic values of metal seals due to ageing

The long-term tests carried out so far and the operating experience of more than 25 years demonstrate that the high tightness requirements are also met in the long term by the metal seals used. This can be expected to remain the case for interim storage of 40 years and longer. However, in order to validate and quantitatively predict the sealing behaviour over longer periods of time, further investigations are required to determine the change in sealing characteristics under the influence of temperature and time.

A study on the ageing behaviour of metal seals was carried out between 2013 and 2016 under the direction of GNS with RuDrift. The tests were carried out in the joint laboratory of Technetics and CEA at the Pierrelatte site (France). The stainless steel (SST) test flanges developed for the tests represent the standard sealing surface/lid combination (nickel-plated cask body/martensitic lid material). Aluminium- and silver-coated metal seals were used in the test flanges. These metal seals are of the kind used in the primary and secondary lids, the so-called main lids. The compressed metal seals used in the test flanges were stored for two years at temperatures of 100°C and 130°C and for one year at 150°C to predict sealing behaviour, particularly for the maximum design temperatures. Reference flanges were stored in parallel at room temperature. At different times, the test flanges were removed from the furnaces and the decompression curves were recorded with respect to Y_{1r} and r_{ur}. The results describe the quantitative behaviour of the metal seals for a temperature level that corresponds to the maximum design thermal load.

In fact, seal temperatures during interim storage are significantly lower than the test temperatures at RuDrift. This means that the design temperatures are not reached even at the time of loading and continue to drop due to the decreasing heat generation. Additional tests at lower temperature levels with a correspondingly extended test duration are also planned. These will be performed to obtain valid statements on the behaviour of metal seals at the sealing temperatures expected during extended interim storage. The study will also cover seals with a smaller torus diameter. These seals are used in the so-called small lids (closure lid, protective cap and pressure switch). There are also plans to study transferability to the use of another sealing surface/lid combination. This is a combination of uncoated ductile cast iron (DCI) with stainless steel (SST), which is only relevant for main lids with aluminium-coated metal seals. The test programme (EP) under MSTOR is therefore as follows (see Figure 20):

- **EP1/2**
  Continuation of the artificial ageing of aluminium- and silver-coated metal seals in main lids at 130°C (plus one year) and 100°C (plus three years), which was started with RuDrift. The aim is to improve prediction accuracy in the selected temperature range and to carry out further reference testing of seals stored at room temperature (plus eight years).

- **EP3/6**
  Expansion of the database for aluminium- and silver-coated metal seals in main lids at ageing temperatures of 60°C and 80°C, in each case over a period of at least eight years.

- **EP4**
  Creation of a database for aluminium-coated metal seals in main lids for the flange combination DCI/SST at representative ageing temperatures of 130°C (one year), 100°C (six years) and at room temperature (six years) for reference purposes.

- **EP5/7**
  Creation of a database for aluminium- and silver-coated metal seals in small lids for the flange combination SST/SST at ageing temperatures analogous to the large lid seals of 150°C (one year), 130°C (three years), 100°C (six years), 80°C (seven years) and 60°C (at least seven years) as well as at room temperature (at least seven years) for reference purposes.

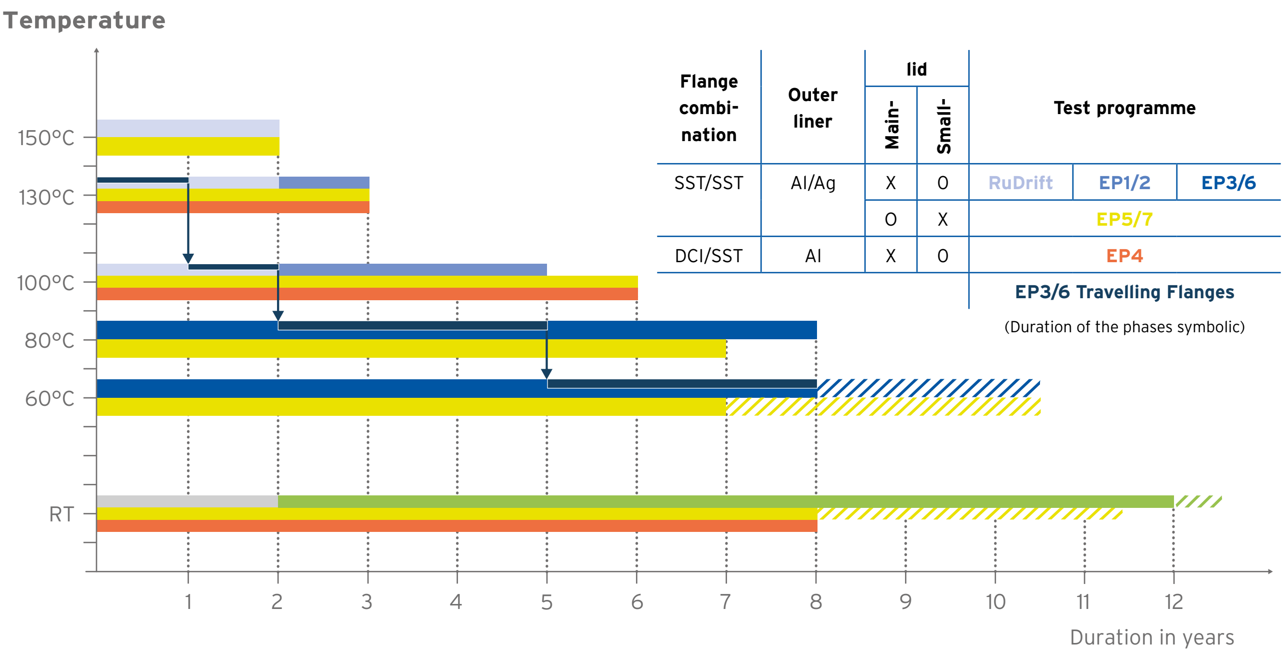


Figure 20: Overview of the RuDrift/MSTOR test programme

Models for predicting the change in the characteristic values of the metal seals are generated based on the temperature and time-dependent measurement results. These models enable the long-term behaviour of metal seals to be evaluated, taking into account real thermal loads and decay behaviour. The results can also be used to specifically pre-age metal seals by artificial ageing, for example for further studies on aged seals. To verify the calculations, additional flanges (so-called travelling flanges) with silver- and aluminium-coated metal seals are aged at various temperatures (130°C, 100°C, 80°C and 60°C) in the EP3/6 test programmes. The respective measurement results are compared with the temperature- and time-dependent calculations. The aim is to demonstrate that the prediction model found is able to correctly calculate ageing under realistic operating conditions at decreasing temperatures.

11.1.2 OBSERVE – Dose rate and temperature measurement programme

Subject matter:	OBSERVE » Dose rate and temperature measurement programme on loaded casks » Comparison of calculated and measured values on selected casks at different points in time during storage » Verification of the shielding/heat dissipation performance of the casks (safety objectives)
Project partners:	WTI
Period:	Phase I: mid-2021 to end of 2022, followed by measurement programme (Phase II)

The distribution of dose rate and temperature on the cask surface depends on the arrangement and characteristics of the radioactive inventory and cask components (see Figure 21 and Figure 22). Measurements are routinely taken before casks are placed in storage to confirm compliance with the applicable dose rate and temperature limits for storage in the respective storage facility. These measurements are carried out at representative measuring points specifically defined for each cask type in order to obtain reliable information about the mean value or to find maximum values.

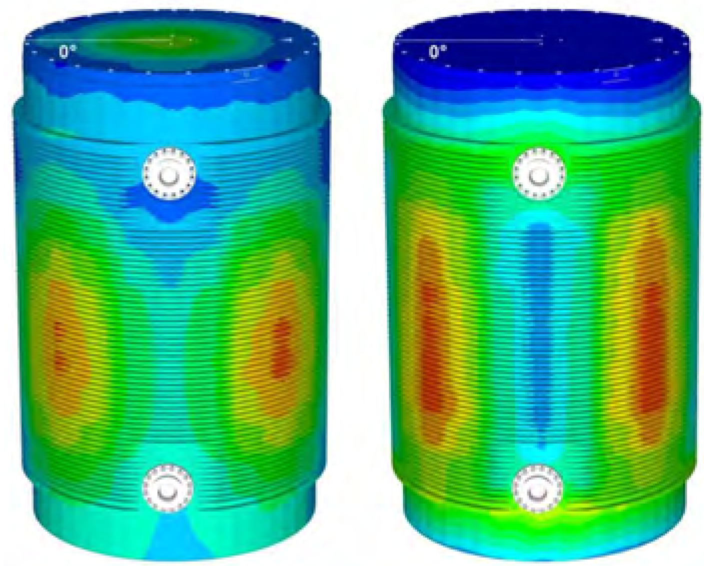


Figure 21: Distribution of dose rates on the surface of a CASTOR® cask (left: neutron dose rate, right: gamma dose rate)

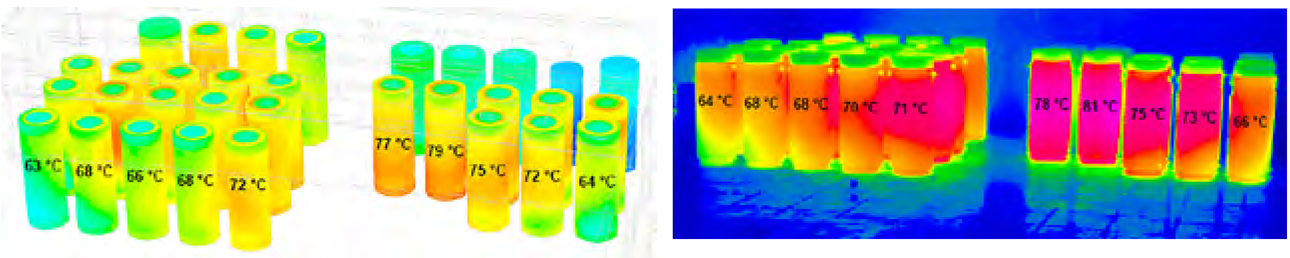


Figure 22: Comparison of calculated and measured maximum cask temperatures in an interim storage facility (left: CFD calculation, right: thermography)

Activity and thus heat generation both decrease during storage due to the radioactive decay of the inventory. Likewise, there are temperature-dependent and radiation-induced changes in the properties of the materials used (for example, thermal expansion, density changes and radiolysis of the moderator material), which in turn effect shielding and heat dissipation.

The planned dose rate and temperature measurement programme OBSERVE aims to measure the entire surface of selected casks intermittently over a long period of time. Measurements provide immediate and integral information on the ageing behaviour of the casks or the inventory. The measured dose rate and temperature curves must be compared with the predicted values for this purpose. These comparisons and the respective distribution can be used to draw conclusions about actual shielding and heat dissipation properties. If the measured values are within the expected range, this confirms that no unexpected changes have occurred and that ageing behaviour has been correctly assessed. In this way the measurement programmes can make an active contribution to ageing management.

In the first stage of the project, sensitivity studies will be carried out to check how suitable dose rate/temperature measurements are for drawing conclusions about the condition of cask components and the inventory. The results of the sensitivity analyses will be used to determine the requirements for carrying out the measurements (equipment, measurement grid, measurement location) and to develop a corresponding measurement programme.

11.1.3 DPOPT – Optimisation of the pressure switch

Subject matter:	DPOPT » Optimisation of the pressure switch » Qualification of a production-optimised component
Project partners:	» GNS (cask manufacturer, Germany) » HBM (pressure switch manufacturer, Germany)
Project period:	From early 2021 to 2022

The sealing function of the double-lid sealing system is monitored by means of a pressure switch mounted in the outer sealing barrier of the double-lid sealing system and connected to the inter-lid space. The pressure switch in turn is connected to the cask monitoring system. If, during storage, the pressure in the inter-lid space drops below the pressure level in the reference chamber of the pressure switch due to a defect in one of the two barriers, the contact of the main switch is opened and the cask monitoring system reports "Inter-lid pressure low".

The pressure switch is a complex component (see Figure 23 and Figure 24), which is manufactured to high quality standards. However, random defects cannot be completely ruled out. The pressure switch therefore has a self-monitoring function in the form of an additional switch for the reference chamber. If pressure in the reference chamber drops below the specified switching pressure due to a pressure switch defect, the corresponding contact is opened and the cask monitoring system reports "Reference chamber pressure low".

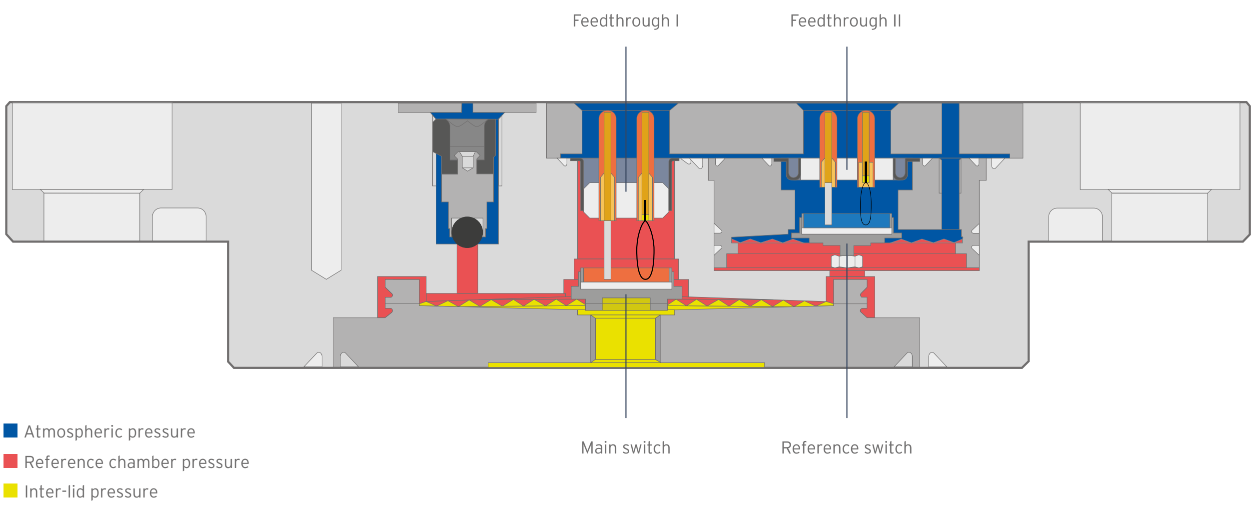


Figure 23: Basic design of the pressure switch

Figure 24: Pressure switch – top view without cover plate I

Source: GNS

There have only been a very limited number of defects – so-called pressure switch events – in the more than 1,400 pressure switches that are installed worldwide. Some of these 1,400 switches have been in operation for more than 30 years, adding up to a total operating time of more than 20,000 years. The failure probability of a pressure switch is in the range of less than 10^{-6} per year.

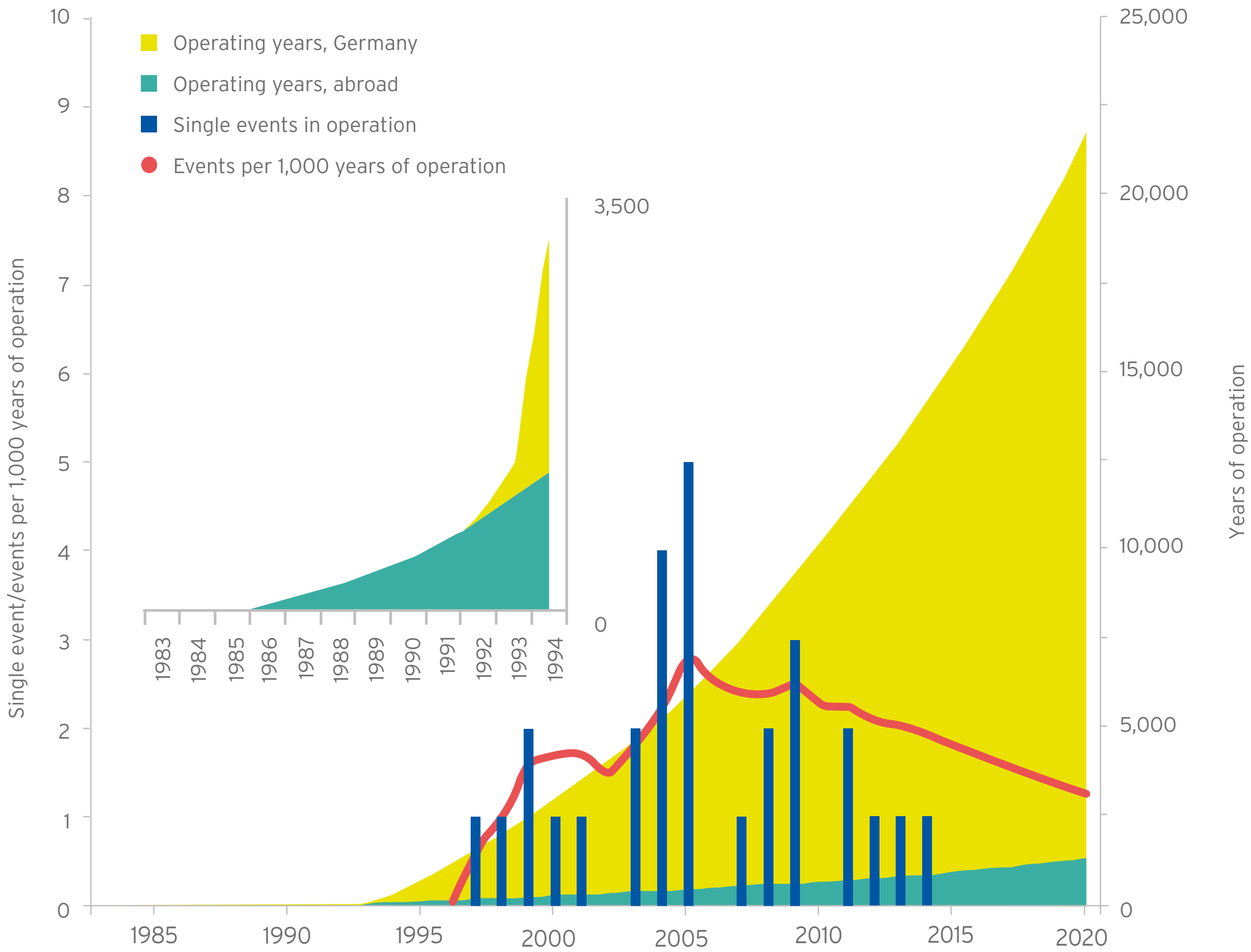


Figure 25: Comparison of years of operation and pressure switch events

Figure 25 shows that pressure switch events do not correlate with the total number of operating years. It can therefore be assumed that events occur randomly, and it cannot be deduced from their distribution over time that a systematic increase in failures is associated with extended interim storage. The decrease in the frequency of failures is also related to incorporation of the lessons learned during operation or during failure evaluation into the handling of the pressure switches.

Specific evaluations of pressure switch events show that the vast majority of events were due to leakage at the feedthroughs of the contact pins. This resulted in a pressure drop in the reference chamber but not in a failure of the containment function of the lid system. However, BGZ is working with GNS to optimise the pressure switch for interim storage purposes. Preliminary studies by the manufacturers HBM and GNS show that using glass feedthroughs instead of the ceramic feedthroughs previously used is a promising solution to the problem, as glass feedthroughs do not require additional soldering material.

The DPOPT research programme will subject the new contact feedthrough design to several tests covering operating conditions. The contact pins will be subjected to different mechanical stresses, which are considered to be the cause of the previous defects. These include pure weight loads and the simulation of repeated plugging and unplugging. The new switch will also be subjected to thermal loads to demonstrate that the glass feedthroughs are suitable for the entire temperature range envisaged. Load tests will be followed by helium leak tests in which the specified maximum leak rate must not be exceeded.

11.2 Inventories

11.2.1 SCIP IV – Studsvik Cladding Integrity Project

Subject matter:	SCIP IV – Studsvik Cladding Integrity Project IV » Expansion of the existing experimental basis for cladding performance under conditions of extended interim storage » Derivation of models for predicting cladding performance
Organisation:	» OECD-NEA international project » Participants from Europe, Japan, USA, China and Korea » BGZ as consortium partner together with GRS
Project period:	July 2019 to June 2024

The OECD/NEA Studsvik Cladding Integrity Project (SCIP IV) is the fourth research programme of a cooperation between OECD/NEA and various organisations from 15 countries. The project was launched in July 2019 and will be implemented by Studsvik AB over a period of five years with a total budget of 14 million euros. The project will include basic and safety research on cladding performance during loss-of-coolant accidents, overheating, operational power transients and performance under interim and final storage conditions. The content of the current research programme builds on the previous programmes and specifically addresses questions pertaining to interim storage facilities. These include the creep behaviour of cladding tubes under interim storage conditions, the hydrogen fracture mechanisms in cladding tube material and the associated hydride reorientation, the interaction between fuel pellets and the cladding tube (pellet cladding interaction, fuel swelling), the permeation of gases in the fuel, the microstructure in high burn-up fuels, the mechanical behaviour of irradiated fuel assemblies and the fragmentation behaviour of different irradiated fuel types [33]. The current experimental programme addresses a large number of the research questions identified by BGZ, some of which are also being carried out in parameter ranges relevant to Germany. BGZ is actively participating in the research programme. This includes regular discussions about the design, procedure and sample material of the experiments, as well as the evaluation and interpretation of the results obtained.

11.2.2 SpizWurZ – Stress-induced hydrogen rearrangement in fuel cladding during long-term interim storage

Subject matter:	SpizWurZ – Stress-induced hydrogen rearrangement in fuel cladding during long-term interim storage » Expansion of the existing experimental basis for cladding performance under conditions of extended interim storage » Derivation of models for predicting cladding performance
Organisation:	» BMWi research funding for nuclear safety as part of the 7th Energy Research Programme (funding codes RS1586A, 1501609B) » GRS and KIT (IAM, INE) collaborative project » BGZ has observer status
Project period:	June 2020 to June 2023

BGZ is engaged as an observer in the collaborative project of GRS (funding code RS1568A) and KIT (funding code 1501609B) funded by the BMWi and managed by GRS.

The collaborative project focuses on experimental and theoretical studies on the behaviour of hydrogen in fuel rod cladding materials under long-term interim storage conditions. In principle, hydrogen exerts an embrittling effect on zirconium-based materials. In dissolved form, the distribution of the hydrogen stored in the cladding tube changes by diffusion under the impact of temperature, concentration and stress. There is no complete description in the literature of the hydrogen flow taking into account all relevant parameters. However, this would be necessary for a reliable assessment of fuel rod integrity. The project extends the qualitative and quantitative understanding of hydrogen diffusion on a macroscopic and microscopic level to predict the formation of hydride structures in zirconium-based cladding tube materials.

One focus of the project is the experimental determination and description of the solubility and diffusion of hydrogen in cladding tube materials under longer-term interim storage conditions. The qualitative and quantitative description of hydrogen diffusion at the macroscopic and microscopic level will be studied to improve the prediction of hydride structures forming in zirconium-based cladding tube materials and the resulting material embrittlement. The results of individual effects will then be combined to produce a consistent description for theoretical modelling purposes of real cladding tube materials under conditions of longer-term interim storage with reference to irradiation and slow cooling rates.

One of the aims is to determine the chemical potential and diffusion coefficients of hydrogen in elastically stressed zirconium alloys. The project will also produce and perform a blind benchmark for the evaluation of the existing computational code for the simulation of fuel assemblies in transport and storage casks.

Specifically, bundle experiments will be carried out in the QUENCH facility at the Karlsruhe Institute of Technology (KIT) to determine macroscopic hydrogen flow in the cladding tube. For this purpose, different cladding tube materials (Zry-4, ZIRLO®, Duplex) will be loaded with hydrogen and individual internal pressures and hydrogen concentrations set. Subsequently, the unirradiated cladding tube samples will be cooled from temperatures of around 370°C at a cooling rate of about 1 K/d in an experiment lasting several months. The hydrogen distribution in the cladding tubes will then be determined by means including neutron radiography.

The microscopic diffusion-induced hydrogen flow as a function of orientation, morphology and mechanical stress in zirconium alloys will also be investigated and quantified. For this purpose, investigations on hydrogen diffusion on cladding tube materials of a material texture previously analysed in detail will be carried out on pre-oxidised material samples. In addition, an in-situ and ex-situ investigation of the impact of stress on the concentration distribution in Zircaloy will be carried out (by means of tensile tests).

The real elastic strain in the fuel rod cladding tube after more than 30 years of storage will be determined experimentally in the hot cells of the Institute for Nuclear Waste Disposal (KIT-INE) by separating out the fuel and measuring the cladding tube diameter. A Zircaloy-4 cladding tube sample with UO_2 (sample burn-up of 50.4 GWd/tHM) from the Obrigheim PWR is used as the sample. The irradiation history of the samples irradiated in the 1980s is very well known. The initial period of the project, which began in 2020, is three years.

11.2.3 Thermal Modelling Benchmark

Subject matter:	Thermal Modelling Benchmark » Determining and comparing precise fuel rod cladding tube temperatures » Uncertainty and sensitivity analysis of the different calculation and modelling approaches
Organisation:	» International Thermal Modelling Benchmark of the EPRI-ESCP » Participants from Europe, Asia, USA » BGZ in collaboration with GNS and WTI
Project period:	July 2019 to June 2024

As almost all degradation effects of the cladding tube are temperature-dependent, it is critical to have as much precise knowledge as possible about the cladding tube temperature and its development in order to determine the integrity of the cladding tube. After reactor deployment, cladding tubes have a temperature of the decay pool of below 50°C. Following the decay period, the fuel assemblies are packed into and dried in casks. During the drying process, the cladding tubes reach their maximum temperature (outside the reactor), which subsequently decreases according to the heat conduction in the cask and the decay heat of the fuel. Long-term temperature development during storage essentially follows a decreasing exponential function.

The time span from the drying process through to several months later is critical for the majority of the hydrogen-dependent effects that apply to longer-term interim storage. The rapid rise in temperature during the drying process and the way in which the cladding tube temperature decreases again significantly determine the behaviour of the hydrogen absorbed in the cladding tube.

An instrumented type TN® 32 standard cask was loaded with PWR fuel assemblies at the North Anna power plant in 2017 and subjected to a measurement programme [32]. Temperature measurements were an important part of the test procedure given that knowledge and accurate prediction of the temperature distribution in the cask and the cladding tubes has a significant influence on the ageing behaviour of the components. These are used to verify the thermal calculation methods. In the current project phase, the American project coordinator "Electric Power Research Institute" (EPRI) also invited interested international institutions to participate in a sub-project of the HBU Data Project: the "International Thermal Modelling Benchmark Study". The aim is to obtain an international overview of the methods used and their sensitivity in comparison to experimentally obtained results.

BGZ is participating in this benchmark together with GNS in order to gain further insights into safety-related verifications based on best estimate analyses for the upcoming reapplication for nuclear storage licences under Section 6 of the Atomic Energy Act.

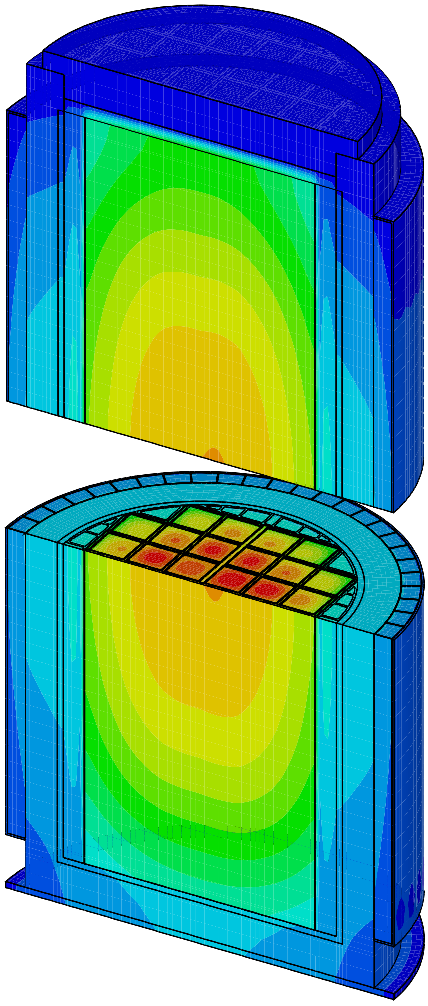


Figure 26: Calculated temperature distribution of the loaded cask based on the information from the benchmark description

The model of the cask was created in the first phase and the calculations carried out according to the benchmark description (see Figure 26). The results were sent to the organisers for evaluation. Evaluations are currently being undertaken and the specifications for the second phase are being defined in parallel. The second phase will involve more detailed study and quantification of the uncertainties in the modelling and their effects on the calculated cladding tube temperatures. BGZ will contribute results to the benchmark in addition to the results arrived at in association with GNS and WTI.

11.2.4 LEDA – Long-Term Experimental Dry Storage Analysis

Subject matter:	LEDA – Long-Term Experimental Dry Storage Analysis » Expansion of the existing experimental basis for cladding performance under conditions of extended interim storage » Derivation of models for predicting cladding performance
Organisation:	» Managed by BGZ » Experiments will be performed in Studsvik laboratories in Sweden. » Joint planning with partners (Framatome GmbH, GRS gGmbH) » Implementation with other partners
Project period:	2022 to 2026

In the laboratories of Studsvik in Sweden, BGZ is planning an experimental campaign to answer questions about cladding performance under conditions of dry interim storage and, in particular, to investigate the significance of hydrogen. The experimental studies will take the form of "integral effect tests", i.e. tests and investigations on different fuel rod segments with prototypical boundary conditions representative for dry storage in Germany. Typical conditions are set and monitored. Furthermore, suitable investigations for the pre- and post-characterisation of the cladding materials are also required. The objectives of the LEDA test programme are the integral study of the behaviour of fuel rod segments representative for Germany under typical conditions of dry interim storage. The experimental data generated in LEDA will be used to extend and validate analytical models and methods for predicting fuel rod integrity for verification purposes in line with protection goals. The previous criteria for the exclusion of systematic cladding failure and their completeness will be reviewed and analysed taking account of (long-term) hydrogen behaviour for storage times of over 40 years.

The long-term measurement campaign will be carried out with different irradiated fuel rod segments under prototypical boundary conditions. An integral approach under drying process and dry interim storage conditions is taken. The implementation of adequate pre- and post-characterisation of the fuel rod segments and the focus on hydrogen-induced effects, in combination with the use of fuel rod segments representative for Germany, in contrast to individual effect tests, allows results to be interpreted directly.

The fuel rod segments that will be studied are as similar as possible to the fuel rods used in Germany in terms of cladding tube materials, fuels and irradiation histories. This will ensure that consideration is given as far as possible to the prototypical conditions that apply to cask loads.

The tests will be carried out in the hot cells at Studsvik in Sweden in a test rig suitable for this purpose. The latter was developed and manufactured as part of the Halden Reactor Project (HRP) to investigate fuel rod behaviour under dry interim storage conditions. After the operating licence for the Halden plant expired, the Halden Board agreed to the test equipment being transported to the Studsvik laboratories in Nyköping to ensure that it could continue to be used by BGZ. A sketch of the test rig is shown in Figure 27. Eight fuel rod segments can be simultaneously exposed to a typical temperature transient in the test rig, varying axially and over time. Typical axial temperature profiles are set for the tests. Over a period of several months, the temperature will be successively lowered, similar to the drop in temperature in the fuel assemblies in interim dry storage after the cask has been sealed.

The ongoing steps within the project LEDA include preparing the test rig at Studsvik's laboratories and selecting and preparing the fuel rod samples for testing.

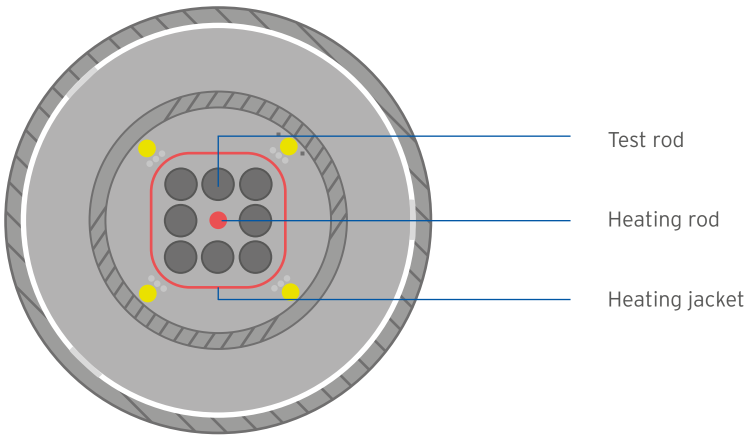


Figure 27: Sketch of the test rig developed within the HRP and used further in LEDA. The test rig can accommodate up to eight fuel rod segments. The temperature is regulated via the heating jacket and heating rod.

11.2.5 DCS Monitor II

The aim of the collaborative project, which is coordinated by Dresden Technical University, is to study approaches to radiation field-based diagnostis for gamma radiation, neutron flux and muons in greater depth. This will be done in the form of simulations and experiments aimed at developing a validated and applicable monitoring procedure for CASTOR® casks. This will include field studies on real casks in interim storage facilities for the first time. The project is dedicated to the in-depth analysis and experimental evaluation of radiometric detection methods for larger, predominantly geometric changes in the cask inventory based on gamma, neutron and muon fields. The project builds on generic feasibility studies on various potential diagnostic methods for non-invasive monitoring of CASTOR® casks, which were carried out as part of a previous project (funding codes 1501513A, 1501513B). The project explicitly does not extend to research on change and damage mechanisms.

Subject matter:	DCS Monitor II – Development and testing of methods for the non-invasive analysis of the inventory condition for transport and storage casks in extended interim storage
Organisation:	» BMWi research funding for nuclear safety as part of the 7th Energy Research Programme (funding codes 1501606A, 1501606B) » Collaborative project of the Dresden University of Technology (TUD, project coordinator), Zittau/Görlitz University of Applied Science and the Helmholtz Centre Dresden-Rossendorf » BGZ has the status of an associated project partner.
Project period:	April 2020 to March 2023

A detector based on straw tube technology provided by Forschungszentrum Jülich will be commissioned and an in-house detector concept will be developed to study the applicability of muon imaging for a monitoring procedure for dual-purpose casks (DPCs) for high-level radioactive waste. With the support of GNS and BGZ, the detectors will be used as far as possible to carry out measurements on large-scale dummies and later on DPCs in the interim storage facility. The aim is to experimentally demonstrate the feasibility of mapping fuel distribution in the DPCs using muon imaging within the framework of practical measurement times. In parallel, extensive simulation studies will be carried out to arrive at a suitable measurement concept and to generate suitable algorithms for volume reconstruction from the measurement data of the muon detectors. This work should deliver a suitable strategy for one-off and recurring cask scans using muon imaging.

The project will also investigate in more depth the use of measurements of the gamma and neutron radiation field of loaded DPCs for non-invasive monitoring. A numerical sensitivity study will be carried out to qualify the gamma and neutron measurement technology. The data generated will include information on how differences in cask loading affect the gamma and neutron field of a CASTOR®. A semi-automated radiation measurement system for measuring the gamma and neutron radiation emitted by the DPCs will also be designed and built. The radiation measurement system will be used in a specially designed experimental campaign on CASTOR® casks in the interim storage facility. The focus will be on reliably determining signature differences in the radiation field with known differences in the loading or burn-up of single fuel assemblies. Accompanying ongoing research will also be undertaken on current research results regarding potential cladding tube damage and nuclear fuel distributions during extended interim storage.

BGZ supports the preparation and implementation of the project as an associated project partner.

11.2.6 Muon radiography research network

Cosmic muons are produced in large numbers by cosmic rays in the upper atmosphere. Their properties are similar to those of electrons except that they are much heavier. The material-dependent interactions of cosmic muons with structures on Earth are already used in established methods in archaeology, geology, volcanology and fission material monitoring using radio- and tomographic imaging. Non-invasive studies of the cask inventory using cosmic muons is still a relatively young and dynamic field of research. While initial work in 2003 focused on proliferation aspects, such as the identification of missing fuel assemblies, more recent efforts have focused on visualising individual fuel rods [34]. Some national research projects already exist in this field. These are funded from a variety of grant sources.

The BMWK, for example, is currently funding the development and testing of procedures within the DCS Monitor II research project (see Chapter 11.2.5) for the non-invasive analysis of the inventory condition for transport and storage casks during extended interim storage. As well as theoretical modelling, the project also includes experimental work on muon tomography in loaded casks.

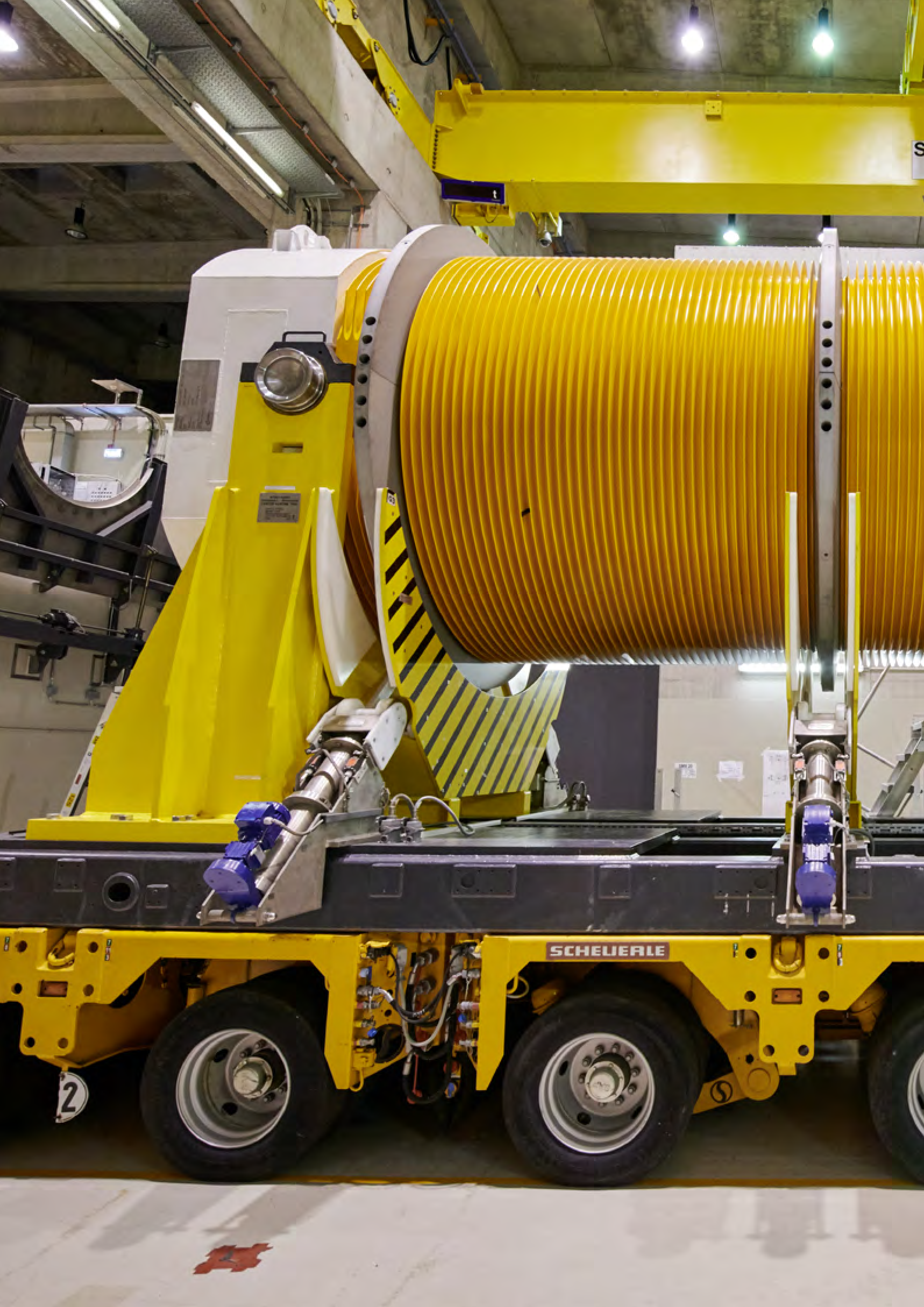
The Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany's central expert organisation in the field of nuclear safety, is investigating and developing the theoretical modelling of imaging procedures using atmospheric muons as part of an in-house research project on aspects of extended interim storage. This project is funded by the BMUV.

What these projects have in common is that they deal with issues arising in the context of extended interim storage and study imaging procedures using muons on CASTOR® V/19 casks.

There are plans to establish a joint network with the participation of BGZ to undertake focused studies on the potential of the imaging methods and in particular the data evaluation algorithms as well as future implementation and application of the technology in Germany. One aim of the network is to build up broad knowledge and to further develop specific expertise. The various focuses of the work funded in each of the projects and the specific know-how of the participants means that research will cover the entire field of the application of imaging methods using atmospheric muons to the study of loaded transport and storage casks.

The planned establishment of this network will lay the groundwork for bundling national research resources and a platform for the exchange of ideas and knowledge in the field of muon tomography.

BGZ is further participating in the MuTomCa project (Muon Tomography for Shielded Casks) [35] on fissile material monitoring. The cooperation framework includes Istituto Nazionale di Fisica Nucleare (INFN, Italy), EURATOM and the Forschungszentrum Jülich and will enable synergy effects to be exploited in the network.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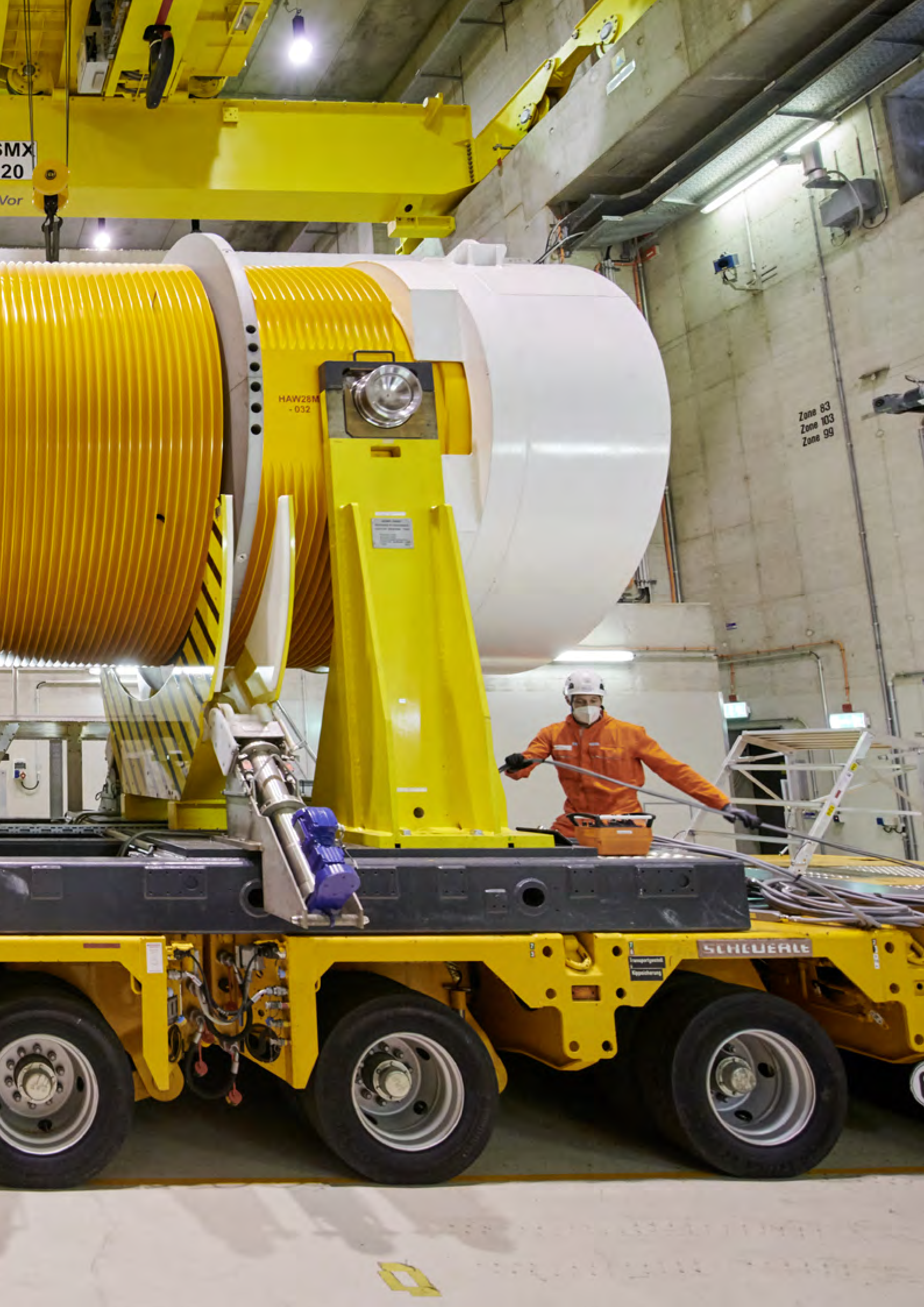
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