BGZ Gesellschaft für Zwischenlagerung mbH

THINKING AHEAD
INTERIM STORAGE

BGZ's research programme

3rd edition

Contents

Foreword

Dear reader,

This is the third edition of our research programme. A lot has happened since the first edition in 2021. BGZ's research programme has become an integral part of the scientific discourse, both nationally and internationally. We present our work at numerous conferences to discuss the latest developments in science and technology in the field of interim storage with experts. For example, we collaborate with scientists and national laboratories in the USA as part of the Extended Storage Collaboration Program and the Used Fuel and High-Level Waste Program.

University research in Germany plays a key role in implementing the research programme. This is particularly evident in Garching: BGZ.lab is located on the campus of the Technical University of Munich, right between the Nuclear Engineering and Physics faculties, the Radiochemistry Institute of Munich, and the FRM II research reactor. BGZ.lab bridges the gap between university research and research aimed at achieving protection goals. As a company wholly owned by the federal government and tasked with ensuring the safe interim storage of highly radioactive waste, BGZ's research cannot be an end in itself. All of the projects that we run with national and international partners demonstrate that safe interim storage is possible for periods well beyond 40 years. We attach paramount importance to good scientific practice. Our ISO 9001 certification is testament to the high quality standards we uphold in our work: Standardised procedures for quality assurance and improvement form an integral part of these standards.

A particular highlight of the research conducted over the past two years is undoubtedly the LEDA project, which involves artificially ageing fuel rods in the hot cells of Studsvik, a Swedish nuclear service provider. This project is a key component in demonstrating the behaviour of fuel elements over several decades in transport and storage casks. The first heat treatment campaign, which lasted seven months, was successfully completed. The experimental data is currently being analysed. We reached another milestone in 2025 with the transportation of high burn-up fuel rods from the Swiss Gösgen nuclear power plant to the Studsvik laboratories. The LEDA project can now analyse dozens of fuel rods with different cladding tube materials and burn-ups (including from German nuclear power plants). This scope is also unparalleled internationally.

Significant progress has also been made in other research projects, including investigation into the long-term behaviour of metal seals (MSTOR). Although the project will continue until at least 2031, the data collected so far is already providing a unique foundation on which to develop prediction models for the long-term behaviour of the seals. Like LEDA, MSTOR is a project in which many national and international partners are involved – a characteristic feature of our work that we are particularly proud of.

Other research projects such as MuTomCa, SPIZWURZ, DCS-Monitor II, SCIP IV and the EPRI Thermal Benchmark have already completed their work and are providing important input for future BGZ research.

What can we expect in the years ahead? The scientific community's current dialogue is vital for our research and will be expanded further. SCIP V marks the start of the fifth round of the international OECD/NEA research project, in which BGZ is once again involved. Fuel rod tests are also being performed at the European Commission's Joint Research Center in Karlsruhe. An exciting collaboration is also underway with MINES Paris – PSL, a renowned engineering university in Paris, to simulate sealing behaviour. And in the foreseeable future, the first research results will be incorporated into the safety analyses: The environmental impact assessment procedure for extended interim storage has begun at our Gorleben site, which means that the application for the authorisation procedure is getting closer.

I hope you enjoy reading our research programme.

Yours, Wilhelm Graf
Technical Managing
Director

1. BGZ's research mission

The strategy for the responsible and safe disposal of spent nuclear fuel and 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 facilities before being finally placed in a repository. The process of searching for and selecting a site for a repository for high-level radioactive waste is far from complete and forecasts are difficult [3]. Irrespective of this, the interim storage periods of up to 40 years assumed and approved to date will not be sufficient to cover the period until a repository for spent nuclear fuel and heat-generating radioactive waste is commissioned and the storage facilities are completely cleared (see Figure 2) [1, 3, 4].

As the operator and licence holder, BGZ is required to demonstrate that the transport and storage casks are placed in a safe manner and that the safety objectives for extended interim storage are met (see Figure 1), irrespective of the duration of such storage and in accordance with the state of the art in science and technology.

The research programme identifies the research that is required for extended interim storage and provides an overview of BGZ's research strategy and activities. Published for the first time in 2022, the programme is constantly updated and adapted to the evolving state of science and technology and changing boundary conditions.

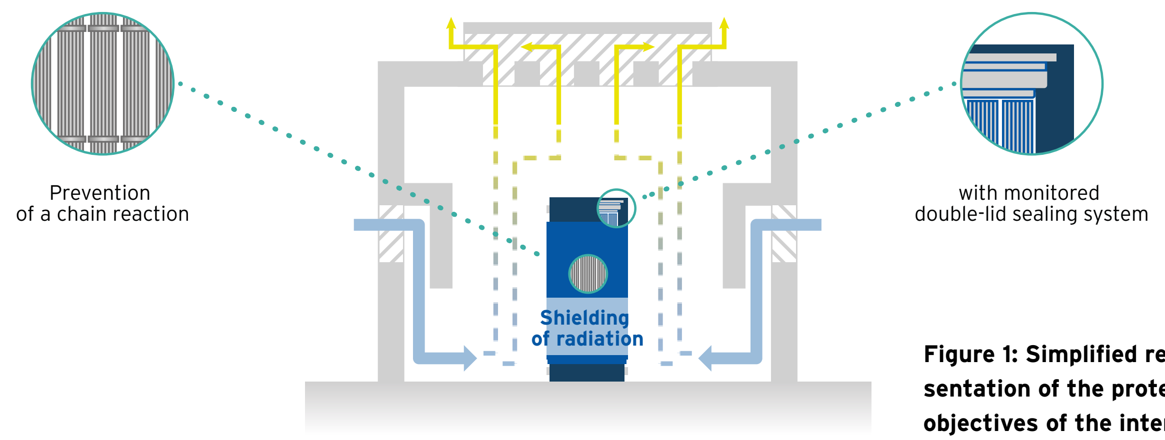

Figure 1: Simplified representation of the protection objectives of the interim storage concept

1 The Brunsbüttel on-site interim storage facility is currently still 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.

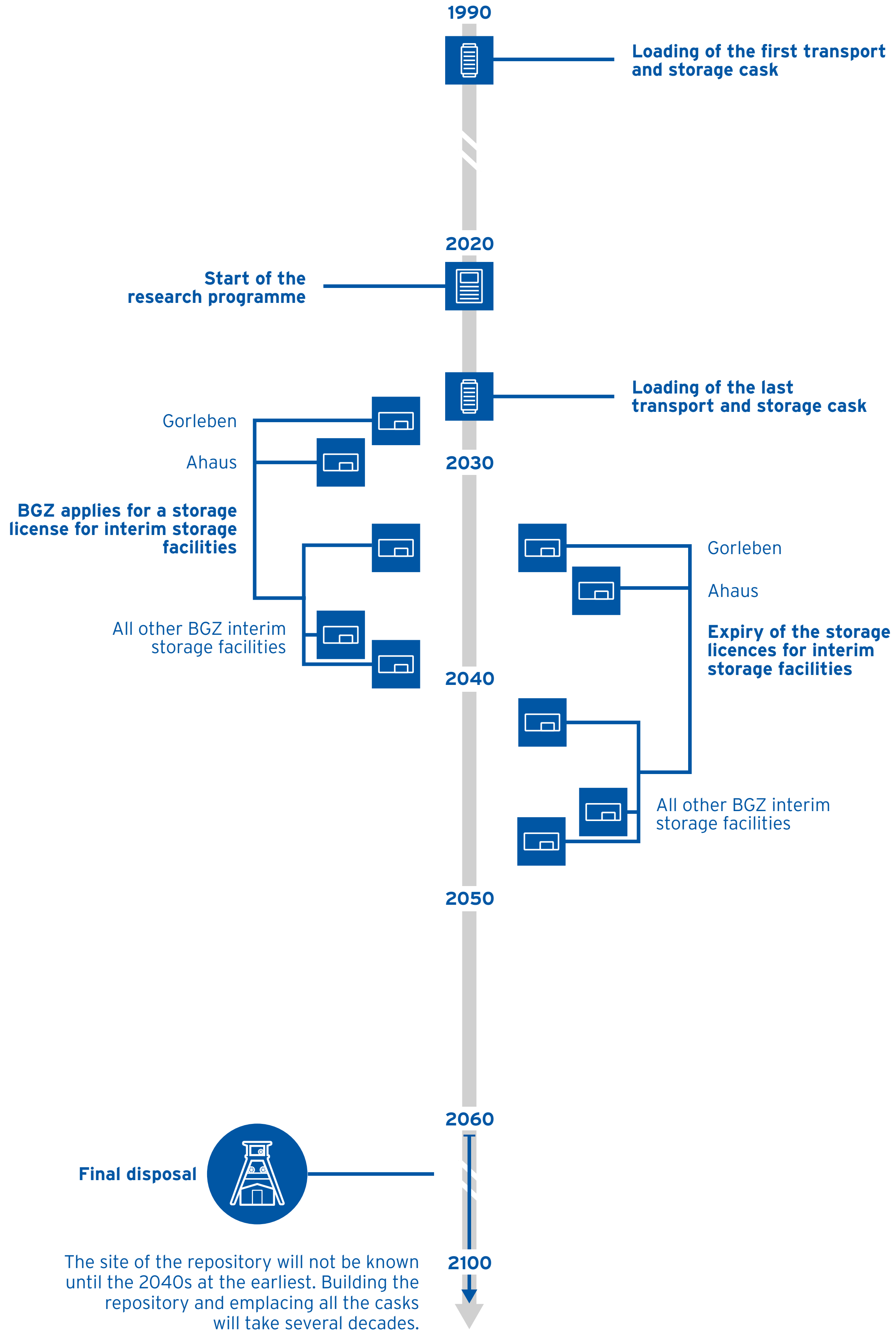

Figure 2: Simplified representation of the time sequence from initial cask loading through to emplacement in a repository

2. Dry interim storage in Germany

In Germany, after being used in a nuclear reactor and subsequently stored in reactor pools for decay, spent fuel elements are packed into dual-purpose casks (DPC) and kept in dry interim storage facilities.

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 transport cask storage facility of the Interim Storage Facility North (ZLN) of EWN Entsorgungswerk für Nuklearanlagen GmbH in Lubmin (Mecklenburg-Western Pomerania). Spent nuclear fuel (SNF) and waste containing SNF from the operation of nuclear power plants in the former GDR and from federal nuclear facilities are temporarily stored here.

The Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH (JEN), which also belongs to the EWN Group, operates the cask storage facility of the Jülich Experimental Reactor Working Group (AVR cask storage facility) in Jülich (North Rhine-Westphalia) 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 spent nuclear fuel 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 still 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 Germany's national 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 law 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 final phase out of nuclear power for electricity generation on 15 April 2023, the last loadings of LWR-FA are expected in 2028.

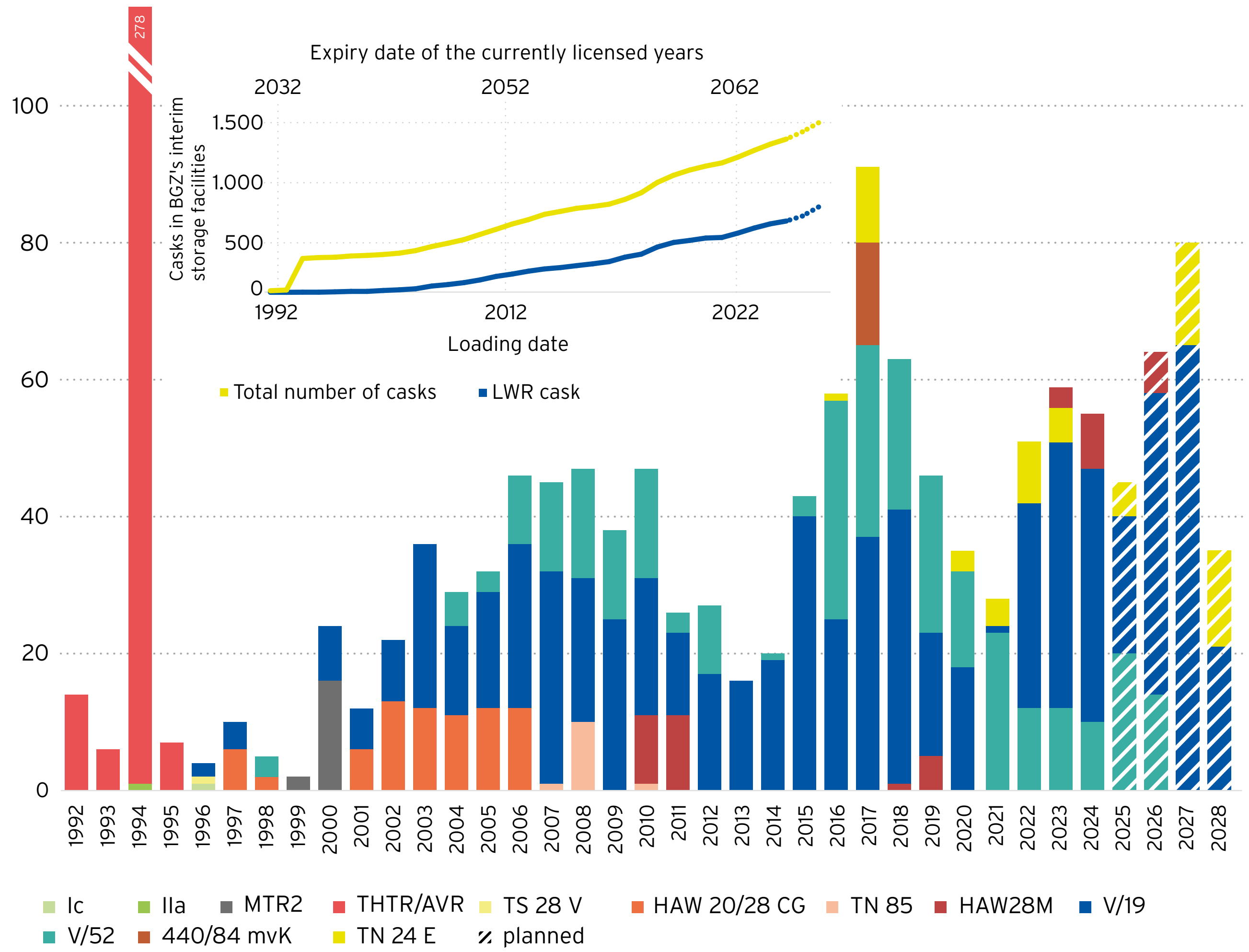

Figure 3: Temporal development of cask loads by type (forecast without CASTOR® MTR3 and without CASTOR® THTR-AVR from the AVR cask storage facility), as of June 2025

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: TS 28 V (1996), CASTOR® HAW 20/28 CG (1997 to 2006), TN® 85 (2008 to 2010) and CASTOR® HAW28M (2010 and following years). 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, Philippsburg, Isar and Brokdorf interim storage facilities were then selected to receive the outstanding reprocessing waste. The return of six casks to Biblis was accomplished in 2020, followed by four casks to Philippsburg in 2024 and seven casks to Isar in 2025. Currently planning envisages completing the return of all HAW canisters from reprocessing with the emplacement of seven casks in Brokdorf in 2026. The last LWR fuel assemblies will then be stored at BGZ in 2028, meaning that BGZ will then have almost 1,500 transport and storage casks in its interim storage facilities.

Beyond 2028, only CASTOR® MTR3 type casks are planned for emplacement 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 Fuel Storage Facility (BZA).

			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
Emplacement completed				•	•		•	•				•			•	•
	Cask type	**Inventory**	**Stored cask types**													
CASTOR® casks	Ic	SWR-BE		•												
	IIa	DWR-BE		•												
	V/19	DWR-BE	•	•	•	•		•	•		•		•	•	•	•
	V/52	SWR-BE	•				•			•	•	•			•	
	440/84 mvK	DWR-BE												•		
	HAW 20/28 CG	HAW		•												
	HAW28M	HAW		•	•	(X)					•				•	
	MTR2	FR-BE	•													
	MTR3	FR-BE	(X)													
	THTR/AVR	THTR-BE	•													
TN® casks	TN® 24 E	DWR-BE									•			•		
	TN® 85	HAW		•												
	TS 28 V	HAW		•												

* BGZ accession to the re-licensing procedure in January 2019
(X) Storage of cask type and inventory applied for

Table 1: Overview of the DPC types (applied for) and corresponding inventories located in BGZ interim storage facilities

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 are systematically and critically questioned with regard to extended interim storage. The research programme also draws on ongoing 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 with respect to extended interim storage (see Figure 4). 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.

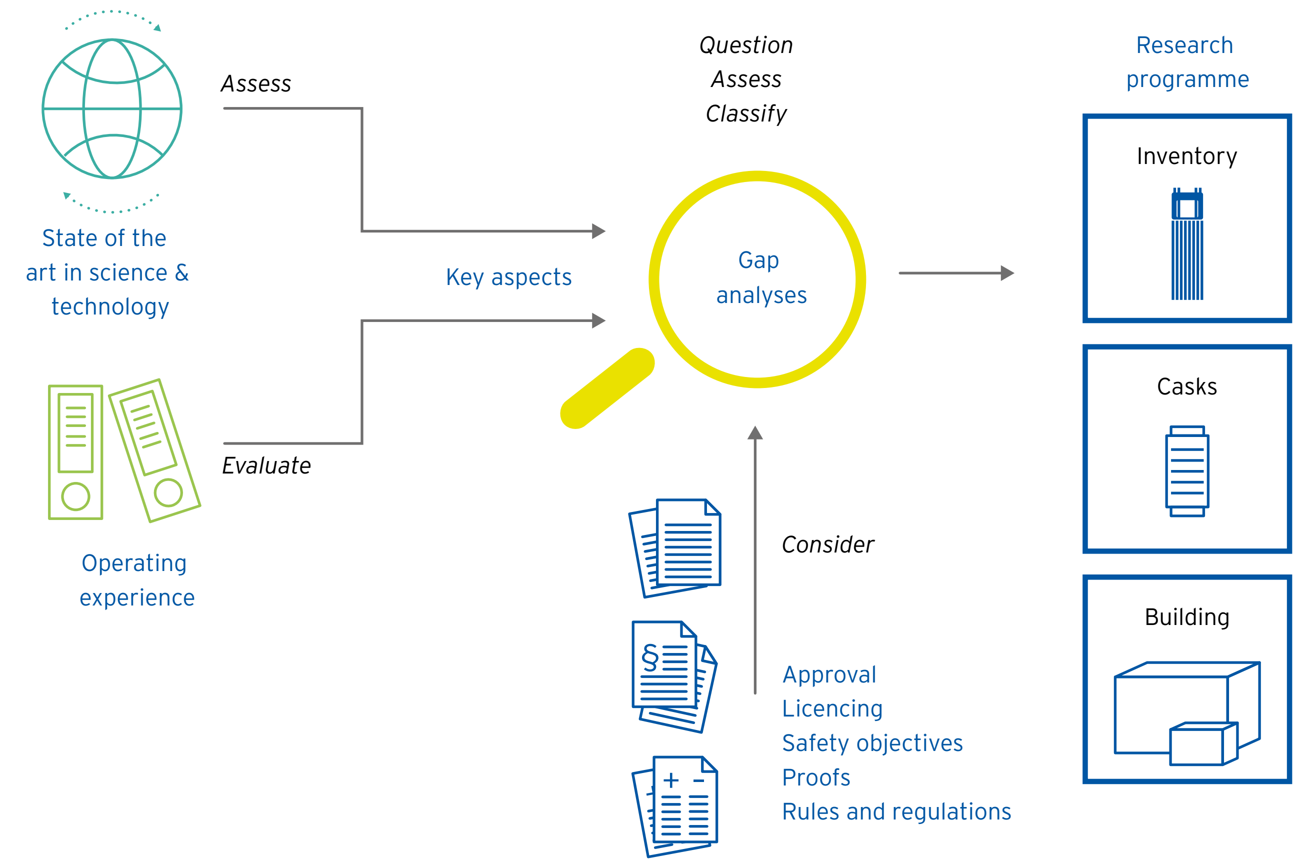

Figure 4: Systematic review and derivation of research needs in the context of protection-oriented research

4. Communicating progress and outcomes

The research programme and progress on as well as partial and final results of the research projects are communicated on several levels by different actors and with content adapted for the respective target groups. This communication work is carried out by scientists themselves, as well as by press and public relations departments, targeting the general and professional public and the scientific community.

BGZ regularly uses events at its locations as well the series of events held as the "Interim Storage Forum" to answer and discuss essential issues relating to safe interim storage with representatives of citizens' initiatives and interested citizens. In addition to the various issues relating to the safe storage of radioactive waste, the focus of open dialogue and discussion is also repeatedly on extended interim storage and current research. The Interim Storage Forum's digital question forum [6] also offers the opportunity to ask individual questions about safe interim storage in general and extended interim storage, as well as specific questions about research. All questions and the corresponding answers are published in the question forum [7].

Progress and results from the individual projects within the programme are regularly presented to an expert audience and the scientific community and discussed with participants at the biannual "Expert Workshop on Interim Storage". Direct and ongoing exchanges also take place at the scientific and professional 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 BGZ's 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).

The Research section of BGZ's website [8] provides further information on BGZ's research activities, current progress and publications in addition to the research programme.

5. National and international cooperation

In order to carry out its research tasks, BGZ engages in 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 cooperate with all important and relevant partners in the field of nuclear waste management. Collaboration on specific projects is outlined in Chapter 11 for each project.

In addition to collaborating on specific research projects, BGZ cooperates with strategic partners in the field of nuclear waste management, participating in programmes, organisations and committees. These are presented in brief separately below.

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 transport cask storage facility near Lubmin, which will be replaced by the ESTRAL storage facility by the end of the 2020s. Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH (JEN), a wholly owned subsidiary of EWN, also operates the AVR cask storage facility in Jülich.

In addition, EWN's wholly owned subsidiary, Kerntechnische Entsorgung Karlsruhe GmbH (KTE), has stored nuclear fuel in the form of vitrified radioactive waste from the reprocessing of spent fuel assemblies in the ZLN. The resulting common interests and focal points are the subject of regular consultations. BGZ also undertakes joint research projects with the EWN Group.

Bundesgesellschaft für Endlagerung
BGZ has another important strategic partner at the national level in the state-owned BGE Bundesgesellschaft für Endlagerung. One of BGE's tasks is to find, construct and operate a site for a repository for highly radioactive waste. This includes packaging highly radioactive waste in a manner suitable for final disposal. In addition to the temporal interdependence of the two companies' tasks, there are also crucial technical overlaps in the transition from interim storage to final disposal of radioactive waste. Consultations between BGE and BGZ take place on a regular basis.

Cooperation in DIN standards committees
Within the DIN Standards Committee NA 062 Materials Testing (NMP), BGZ sends several permanent members to three working committees of Division 7 "Nuclear technology and radiation protection":

- NA 0620754 AA "Criticality safety and decay power": The working committee draws up and updates relevant standards and holds important discussions on topics of criticality safety and the decay power of SNF.
- NA 0620741 AA "Safety of transport and storage casks for radioactive materials": Among other things, the working committee is responsible for the development of standards for the safe transport and storage of casks for radioactive materials from the nuclear fuel cycle and research.
- NA 0620743 AA "Components made of concrete, reinforced concrete, prestressed concrete and steel in nuclear facilities and nuclear waste management facilities": The committee works on structural engineering aspects in nuclear facilities and nuclear waste management facilities.

Gesellschaft für Nuklear-Service mbH and
Orano Nuclear Packages and Services
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 (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
BGZ has important strategic partners in Switzerland – ZWILAG Zwischenlager Würenlingen AG and its owners BKW Energie AG, Kernkraftwerk Gösgen Däniken AG, Kernkraftwerk Leibstadt AG and Axpo Power AG – who are also preparing for extended interim storage. At ZWILAG, spent nuclear fuel and vitrified radioactive waste from the reprocessing of spent fuel elements are also stored in dual-purpose casks (DPCs) of the CASTOR® type. With the CASTOR® Ic DIORIT, which has now been in operation for over 40 years, the longest-serving CASTOR® type cask is also stored here. The shared interests and key tasks in this context provide the basis for regular mutual exchange. Research projects are also carried out in cooperation with the Swiss operators.

Paul Scherrer Institute (PSI)
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 largescale facilities such as the Swiss Synchrotron Light Source (SLS) and the Swiss Spallation Neutron Source (SINQ).

Technical University of Munich
To address issues arising from the research programme in the field of inventories, BGZ has set up its own research group (BGZ.lab) on the research campus of the Technical University of Munich (TUM) in Garching. The TUM has many years of extensive expertise in the field of nuclear technology, which began with the commissioning of Germany's first reactor plant at the Garching site, the FRM, in 1957. Today, the research campus in Garching in particular has an infrastructure in the nuclear sector that is now unique in Germany: In addition to a broad theoretical training programme in specialist areas relevant to BGZ, such as nuclear and reactor technology, reactor physics, nuclear chemistry, engineering sciences on material and material issues, simulation and data analysis, the site offers a research reactor facility, the FRM II, and extensive radiochemical laboratories including hot cells. These already have the necessary licences to handle nuclear fuels.

Electric Power Research Institute
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 spent nuclear fuel 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, 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.

BGZ is also a member of the EPRI's "Used Fuel & High Level Waste (UF&HLW)" programme. The UF&HLW programme researches the scientific and technical basis for the safe handling, interim and long-term storage, transport and disposal of high-level radioactive waste. Its work involves developing knowledge, guidelines, tools and technologies that minimise operational and radiological risks, increase efficiency and flexibility, and support international decision-makers in closing potential knowledge and technology gaps. Current work focuses on the ageing and performance assessment of fuel rods and casks, criticality safety and the modelling of decay heat, thermal and radiation doses.

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 [9] provides nuclear data and computer programmes for participating countries.

BGZ is a member of the Working Party on Nuclear Criticality Safety (WPNCS) [10] 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 [11, 12].

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.

BGZ regularly participates in the meetings of various Coordinated Research Projects (CRP) and Technical Meetings (TM).

6. Operational experience and ageing management

Important conclusions can be drawn for the licensing 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. In ageing management, all systems and components that are relevant for the safe and reliable operation of the interim storage facility over the entire interim storage period are inspected and evaluated at regular intervals. The aim is to identify potential ageing mechanisms at an early stage and to prevent their effects in a targeted and effective manner to ensure safe and reliable operation in the long term.

As well as evaluating its own operating experience, BGZ also works closely with EWN Entsorgungswerk für Nuklearanlagen GmbH. Experience is also regularly exchanged at an international level. For example, this takes place with the operators of the Swiss interim storage facility ZWILAG Zwischenlager Würenlingen AG and 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]. The operation of interim storage facilities is governed by laws and regulations such as the Radiation Protection Act [13] and the Radiation Protection Ordinance [14] as well as the guidelines on dry interim storage issued by the Nuclear Waste Management Commission (ESK) [15]. To date the ESK has issued a discussion paper on extended interim storage [16]. The federal and state governments are currently working together with the Federal Office for the Safety of Nuclear Waste Management (BASE) on an independent set of regulations for extended interim storage. In addition, regular discussions take place between BGZ as the applicant and BASE as the approval authority in connection with the pending applications for storage licences.

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 at any time.

Approvals are based on the international regulations of the IAEA [17] and their transposition into European or national regulations, such as the European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR) [18], the Regulation concerning the International Carriage of Dangerous Goods by Rail (RID) [19], the European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways (ADN) [20] or the Ordinance on the Transport of Dangerous Goods by Road, Rail and Inland Waterways (GGVSEB) [21].

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) [22], the dangerous goods regulations of the Federal Institute for Materials Research and Testing (BAMGGR) [23] 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 themselves 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 that does not rely on any storage facility components. 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 thickwalled 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 percent 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 5).

The casks have a metallic 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) [24], the casks have a monitored doublelid 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.

Figure 5: CASTOR® cask in storage configuration (example: CASTOR® V/19)

At the time applications will be made for extended interim storage, all the casks for LWR-FA and HAW canisters 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 makes it possible to determine the thermal load history and operational history of each individual cask.

Before 2006, a thermal load of 25 kW was rarely exceeded when loading LWR casks. With the continuous disposal of SNF arising from the operation of the NPPs by way of dry interim storage, the thermal load of the inventory also increased. This was because the decay times depended on the utilization of the cooling ponds and 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. It is evident that loading in the first few years tended to be carried out at 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 utilized in casks for LWR-FA at the time of loading, corresponding to 20 kW, and to date only 9% 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.

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 the individual HAW canisters 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 9% of the cask loadings reach a thermal load of between 45 and 50 kW (see Figure 7).

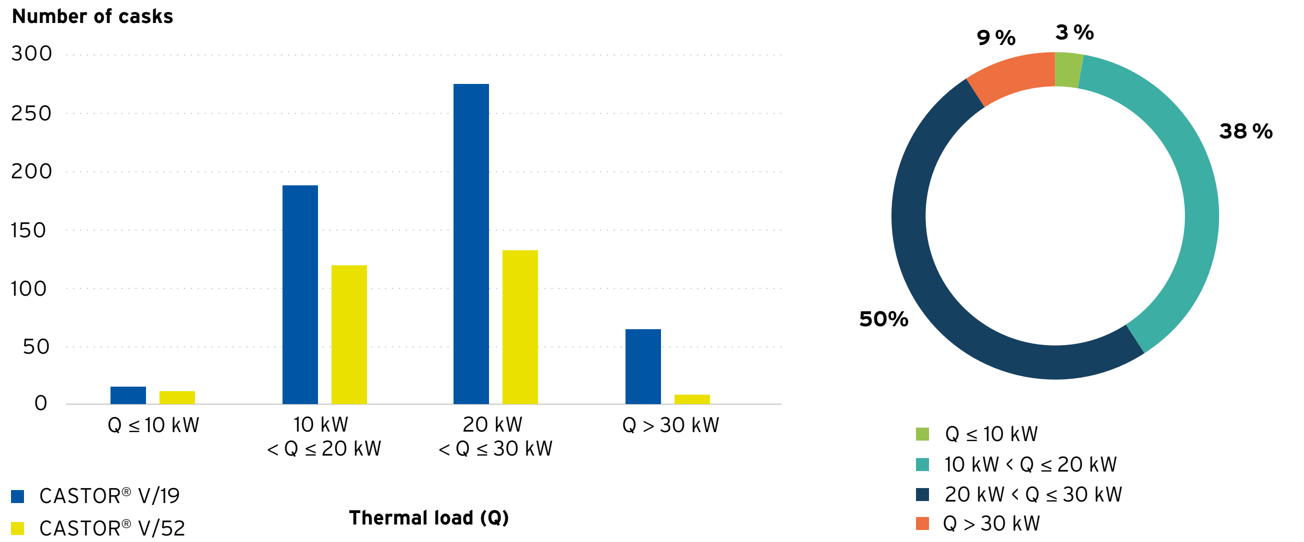

Figure 6: Distribution of the thermal load of CASTOR® casks (LWR-FA) at the time of loading, as of 2024

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.

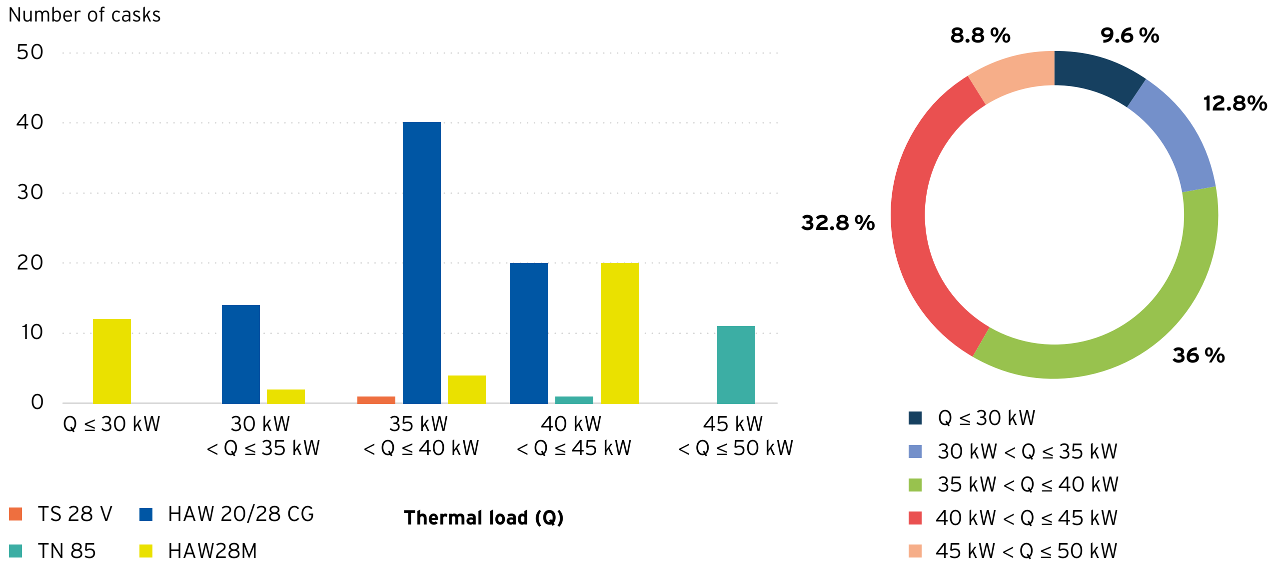

Figure 7: Distribution of the thermal load of casks for HAW canisters at the time of loading, as of 2024

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 power 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 cause any relevant changes in the properties of the structural materials used. Extending the 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, and therefore has no influence on the safe maintenance of subcriticality. Corrosion processes within the tight enclosure (cask shaft, inter-lid space) come to a complete standstill early on during storage due to the limited amount of residual water – assuming they are able to occur at all given an inert atmosphere and the material pairings. The outer cask areas are also checked as part of ageing management. Any inspection findings can be remedied at any time by taking maintenance measures. There is therefore no identified need for research on these ageing mechanisms.

The situation is different for metal seals and lid screws as these are the key components for ensuring the protection objective of safely containing 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. 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 screw connections is possible in principle.

Other ageing mechanisms which may impact compliance with a protection 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, dose rate and temperature measurements can contribute to the integral assessment of the functionality of the casks with regard to the fulfilment of the protection objectives (shielding and heat dissipation).

Additional need for action arises from ageing management and BGZ's operating experience with regard to pressure switches. Even if, on the basis of current findings, no doubts exist regarding the continued usability of the installed pressure switches during extended interim storage, optimisation of this component should be sought due to a certain accumulation of defects in the contact bushings of the pressure switch.

9. Need for research on inventory

The inventories stored in 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), 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 BGZ interim storage facilities all have the same basic design: Several fuel rods are arranged in a quadratic lattice (see Figure 8). The number of rods in a fuel element 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, so-called guide and instrumentation tubes. The LWR fuel assemblies can also be assigned to various other categories, i.e. according to reactor type (PWR/BWR), cladding material, fuel or also according to characteristics based on the individual irradiation history such as burn-up and decay power.

Mixed oxides (MOX) and uranium from reprocessing (reprocessed uranium, RepU) were used as fuel in addition to UO_2. The enrichment and burn-up of the fuel assemblies increased over the operating time, i.e. the enrichment increased to approximately 5 weight-percent U-235 and the nominal fuel element target burn-ups averaged 65 GWd/tHM. The fuel was filled in the form of pellets into fuel rod cladding tubes made of various zirconium alloys. For PWR reactors, these 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 tubes with and without liners were used for the BWR reactors.

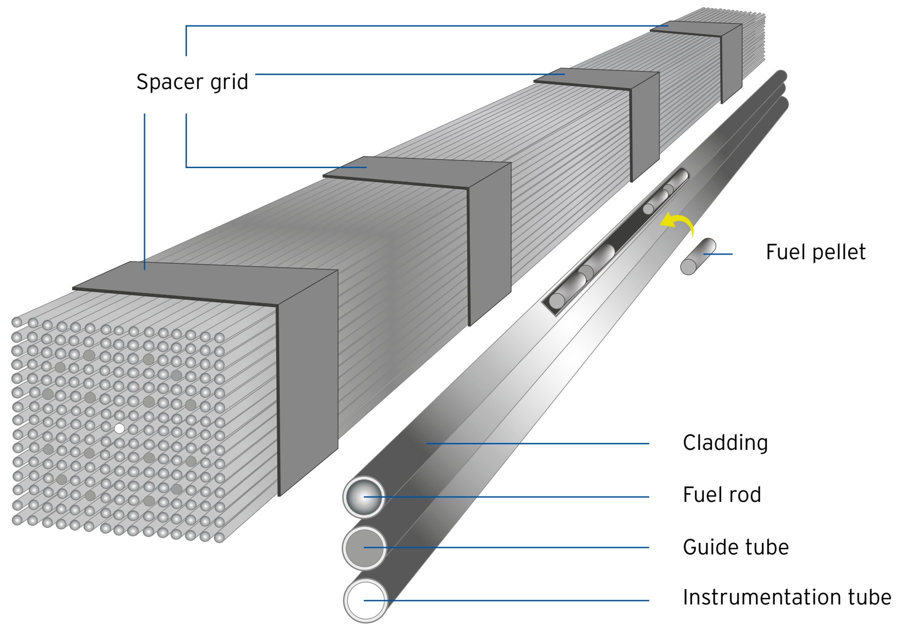

Figure 8: Schematic representation of a PWR fuel element

Fuel assemblies and fuel rod cladding fulfil safety-related functions in the safety concept of 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 the LWR fuel assemblies will have been loaded and stored in casks in the interim storage facilities at the time that applications are made for extended interim storage. The real boundary conditions for loadings and inventories will therefore be known. Currently, there are about 25,000 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 still in the cooling ponds are expected to be completed in 2028. After final emplacement, BGZ expects a total of about 28,000 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 2028 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 of between 40 and 55 GWd/tHM and 16% with burn-ups of over 55 GWd/tHM. Zry-2 with and without liner are used as cladding tube 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 2028, Figure 9 also shows the respective share of burn-ups.

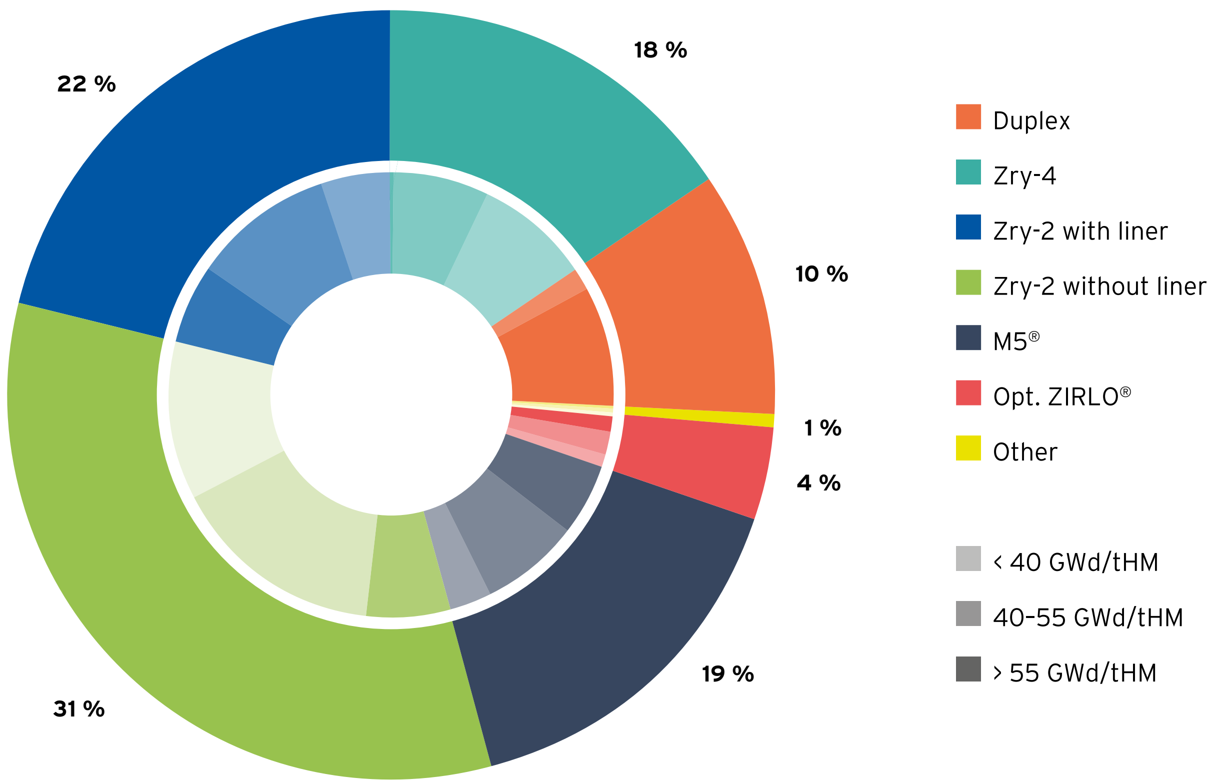

Figure 9: Share of cladding materials with associated burn-up for the entire inventory probably stored by 2028

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. 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. Subsequently, no further casks with LWR fuel assemblies were delivered to Gorleben. 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.

There are a total of six LWR fuel element casks 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 fuel element cask in an on-site interim storage facility took place in Lingen in December 2002. Fuel element casks were then emplaced in further 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. The cumulative number of LWR fuel assemblies stored in BGZ interim storage facilities is shown in Figure 10.

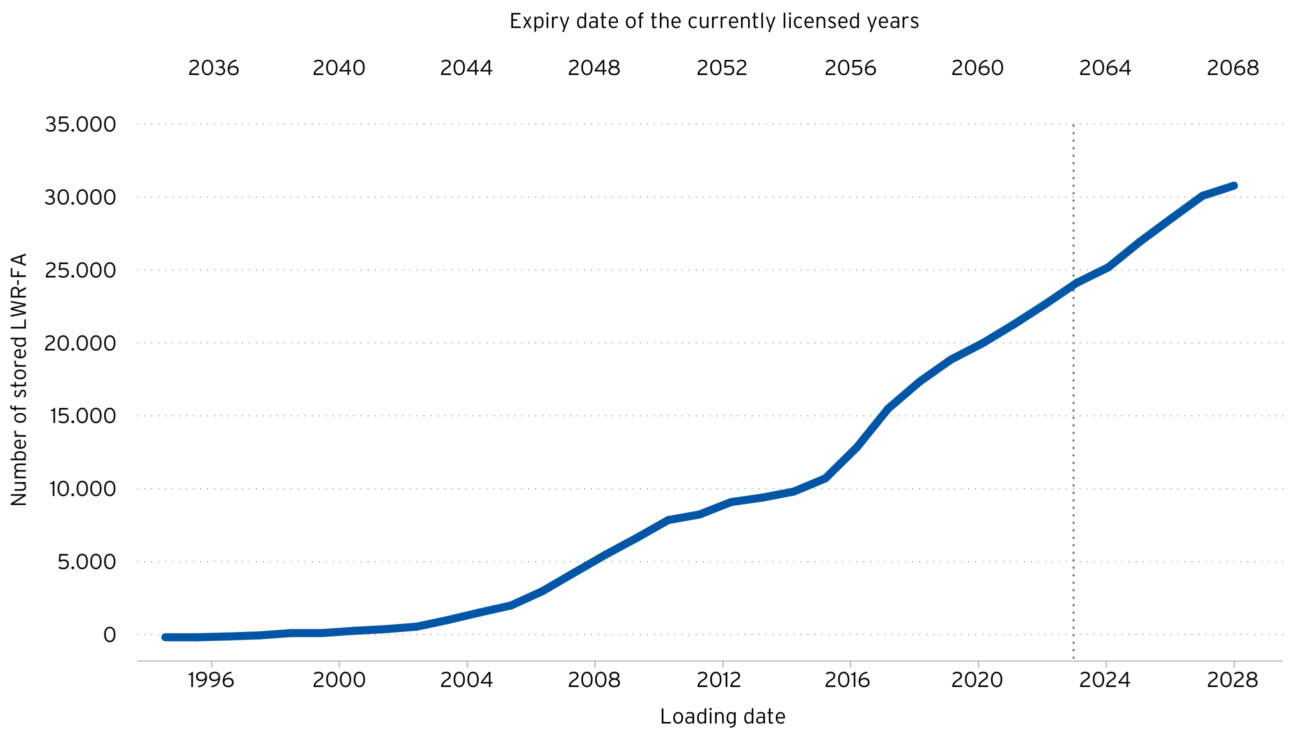

Figure 10: Number of LWR fuel assemblies currently stored and still expected from the power reactors in the interim storage facilities operated by BGZ, as of June 2025.

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 [26, 27] and international studies and publications [28, 29, 30] as well as with the gap analyses commissioned by BGZ from fuel element 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 elongation 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 by 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 (depending on the use of the fuel assembly in the reactor) 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.

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 [30, 31]. 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 in casks for periods of 40 years or more. 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 as well as possible experimentally.

A consistent and validated means of prediction involving calculating and simulating the behaviour of the fuel rods is needed in order to evaluate the relevance of individual effects for cladding tube integrity. Progress varies on the development of the individual calculation steps that are needed. 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 tube mechanics [32]. 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 2, 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 other averaged burn-ups of the almost 300 fuel assemblies are below (and in some cases significantly below) 40 GWd/tHM. 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 [33]. 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 [34], in which 32 fuel assemblies from power reactors in the USA were packed into a TN®32 cask. After a ten-year storage period, the cask is expected to be transported to a suitable laboratory in 2027 for a detailed analysis of the fuel elements. 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 percent 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 or studies of fuel element manufacturers such as Westinghouse [27]. The evaluation of the data and its transfer to actual stored inventories have high priority for BGZ.

A small proportion of fuel assemblies stored in 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 averaged over the fuel element for UO_2 and MOX fuels are rarely envisaged internationally. This means that existing international research activities cannot be applied or only to a limited extent to extended interim storage for these high burn-ups and especially not to MOX fuels. BGZ has therefore identified the need for research on the fuel assemblies stored at on-site interim storage facilities.

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 spent nuclear fuel of THTR-300, which was shut down in 1989, is 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 mm 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 11). 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 percent with U-235.

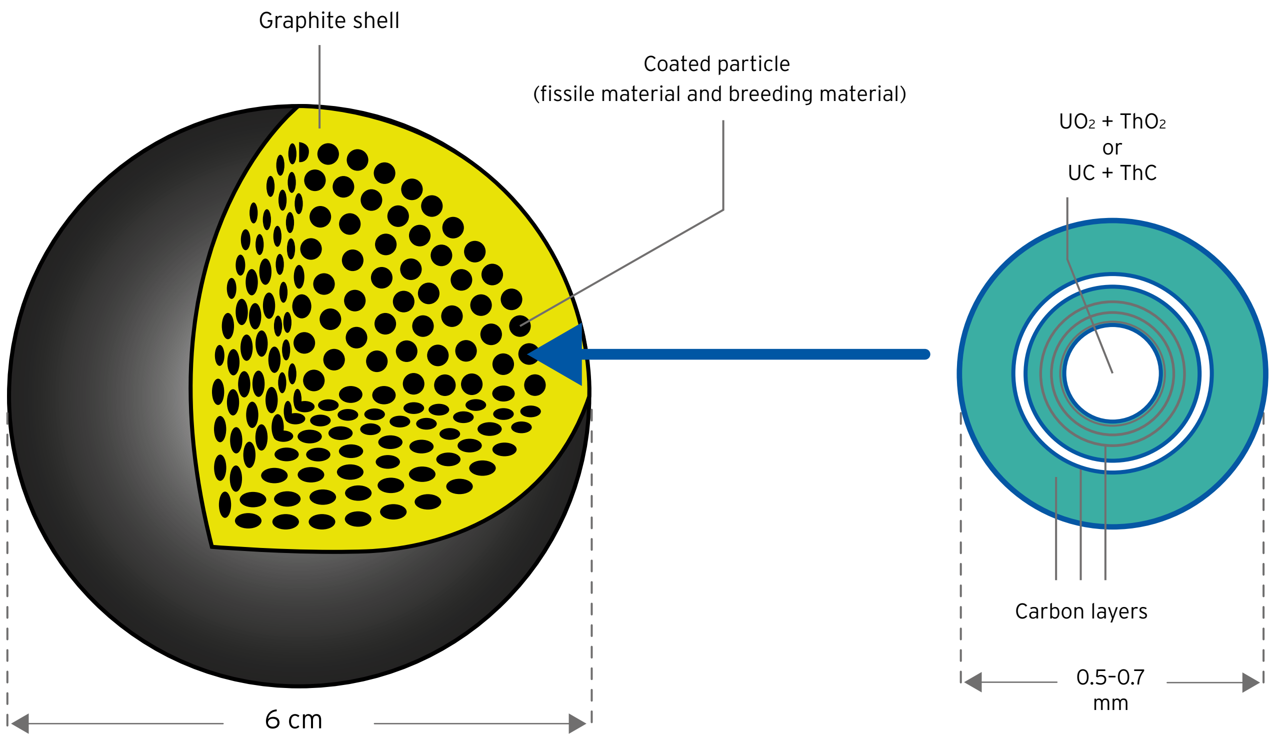

Figure 11: Diagrammatic representation of the THTR fuel pebbles

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 of the burn-up reactor belonging to the THTR are composed of rectangular plates made of an alloy containing 80 weight percent of aluminium and 20 weight percent 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 spent nuclear fuel of the RFR has 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 cask with 16 individual rods of the same fuel element type. The respective fuel is UO_2 with an initial enrichment of 36% by weight (WWR-M/M2) or 10% by weight ^{235}U (EK-10). RFR fuel elements 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 casks, 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 power, the temperature level is well below critical values that could have an influence on the behaviour of material. Along with the relatively low burn-ups and the associated low activity inventory, the level of neutron radiation is also significantly lower than when casks are 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.

Nonetheless, 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 CSDV (Colis Standard de Déchets Vitrifiés; standard waste casks – vitrified) or HAW canisters (see Figure 12).

The reprocessing of the LWR fuel assemblies and the vitrification of the waste was performed in the reprocessing plants of La Hague (France) and Sellafield (England). For reprocessing, the irradiated nuclear fuel is initially reduced

Source: GNS

Figure 12: Loading of a cask with HAW canisters (left), single canister (right)

mechanically and chemically dissolved. This is followed by separating the recyclable isotopic fuel components (uranium and plutonium) from the waste for reuse in MOX and RepU 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 or moulds at a temperature of about 1,000°C. After cooling to approximately 500°C and the associated 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 so that the vitrified waste can be stored for several thousand years.

In addition to the glass matrix, the 5 mm-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 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 any need for further research on HAW canisters for the purpose of demonstrating compliance with the safety objectives for extended interim storage.

Nonetheless, international experience with the transport and storage of HAW canisters will continue to be compared with the status presented here by following the current state of the art in science and technology.

10. Need for research on interim storage buildings

In Germany, dual-purpose casks for spent nuclear 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, in the event of 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 13) 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 these sites until they were sent for reprocessing. Therefore, each of these sites provides a storage capacity for 420 positions, and application has been made for a total thermal load from the stored casks of 16 MW (Gorleben) and 17 MW (Ahaus). As a result of the change in disposal strategy, only 113 (Gorleben) and 52 (Ahaus[2]) 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, which have increased wall thicknesses compared to conventional buildings in order to meet radiological requirements and to comply with designs for earthquakes and external impacts as well as security functions. 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 constructions 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 the necessary 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. The storage area is fitted with heavy steel gates to ensure radiation protection and control access.

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

Overhead cranes are used to handle the casks. 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 designed according to the Steag concept (see Figure 14). As the 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 15). 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.

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 operations loads also 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 the building's stability and serviceability. This may also involve further monitoring, repair or renovation measures.

In addition, international experience in condition assessment and monitoring for the assessment of the technical service life of buildings and the developing state of the art in science and technology will continue to be compared with the state of the art presented here.

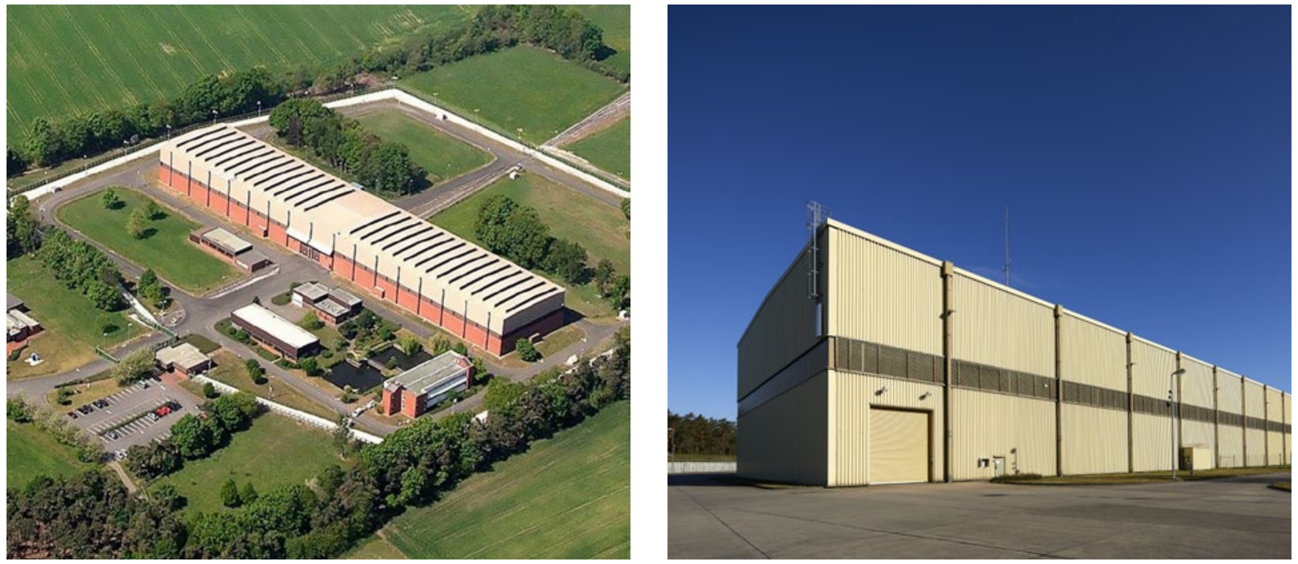

Figure 13: Photos of the central interim storage facilities – Ahaus (left) and Gorleben (right)

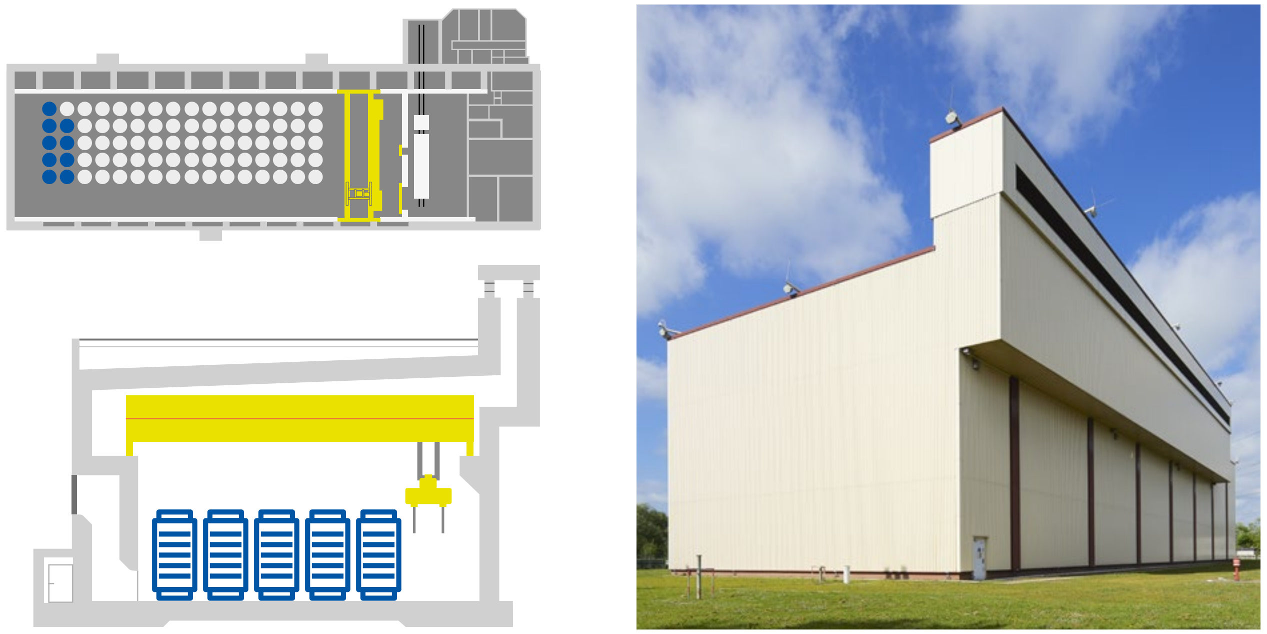

Figure 14: Simplified representation of the Steag concept – left: schematic top and cross view, right: interim storage building in Brokdorf (example)

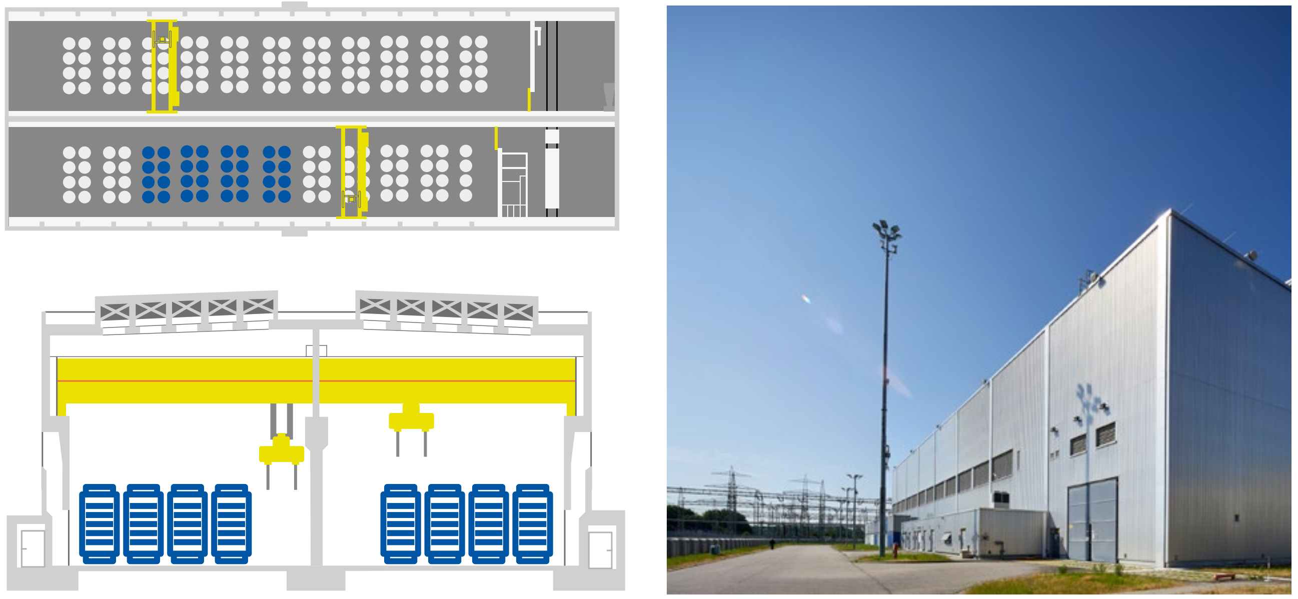

Figure 15: Simplified representation of the WTI concept – left: schematic top and cross view, right: interim storage building in Philippsburg (example)

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 Gesellschaft für Nuklear-Service mbH (cask manufacturer, Germany) » Technetics Group (seal manufacturer, France) » EWN Entsorgungswerk für Nuklearanlagen GmbH (store operator)
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 16) 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 m³/s. The stipulated standard He leakage rate of 10^{-8} Pa m³/s is therefore also a system-specific quality criterion for long-term functionality and is not a radiologically based tightness requirement, as this could also be met at higher leakage rates.

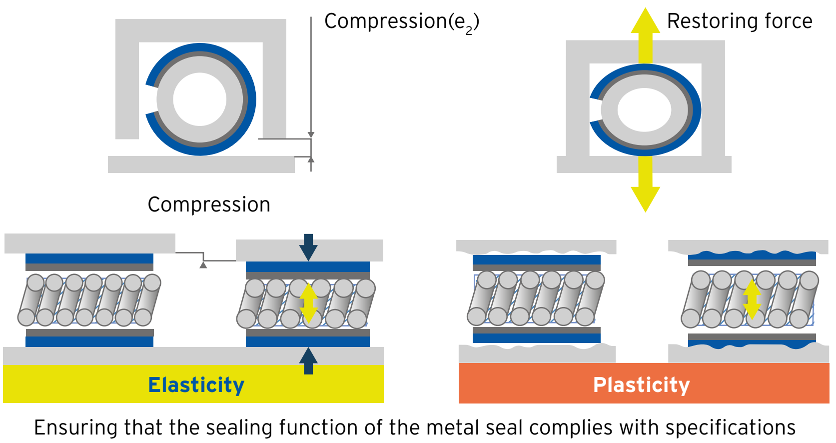

Figure 16: Functionality of Helicoflex® metal seals

During initial compression of the metal seal, a force-deformation curve is obtained as shown in Figure 17. 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 17). 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 17, 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} also 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.

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 more than 40 years. However, in order to validate and quantitatively predict the sealing behaviour

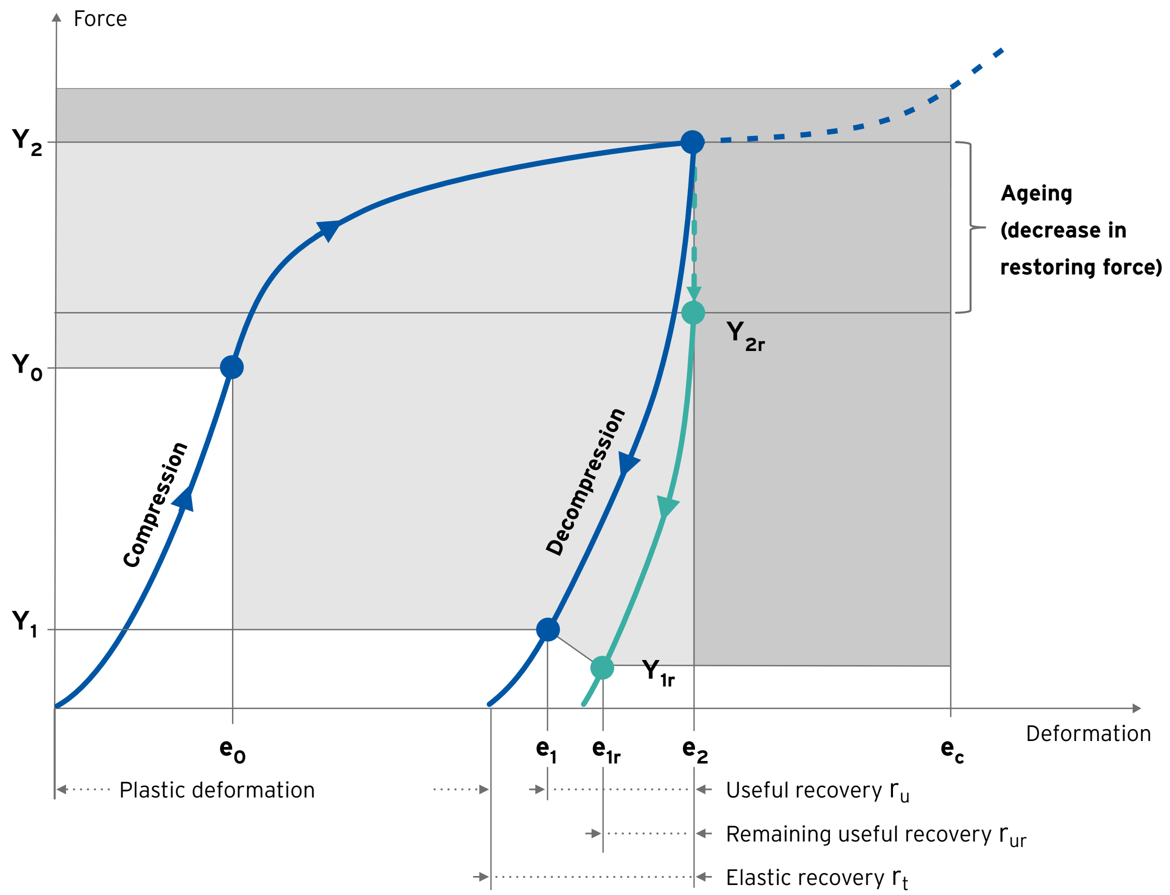

Figure 17: Change in characteristic values of metal seals due to ageing

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, 130°C and 150°C in order 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 planned in order 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 the transferability to the use of another sealing surface/lid combination. This is a combination of uncoated 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 18):

- **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 (at least 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 main lid seals of 150°C (1 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.

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 allow 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°, 100°, 80° 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 [35, 36].

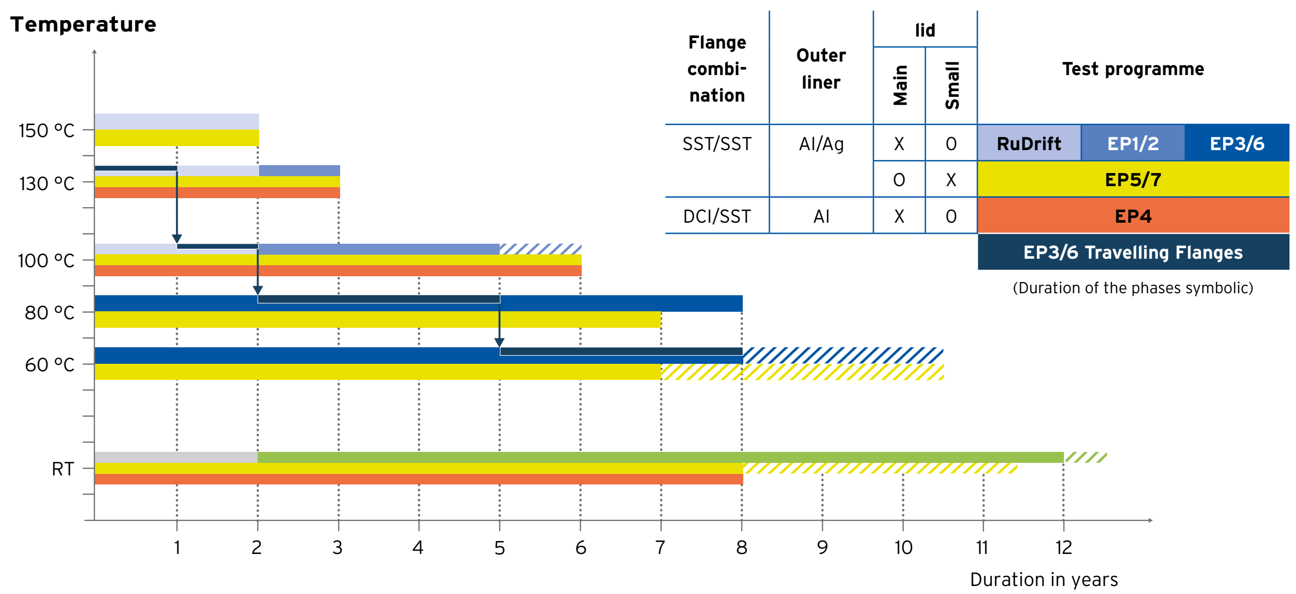

Figure 18: Overview of the RuDrift/MSTOR test programme

11.1.2 MShift – Leak rate of aged metal seals with transverse lid displacement

Subject matter:	MShift – Leak rate of aged metal seals with transverse lid displacement » Performance of tests on aged metal seals » Derivation of design leakage rates after a generic incident with dynamic lateral displacement
Project partners:	» GNS Gesellschaft für Nuklear-Service mbH (cask manufacturer, Germany) » ESU (energy supply company) » BAM (Federal Institute for Materials Research and Testing)
Project period:	2022 to 2026

Compliance with the specified leakage rate of 10^{-8} Pa m³/s is only ensured within the operating range of the seal, i.e. if the contact between the outer liner and the sealing surface, as set at the time of installation of the seal, is maintained unchanged. In particular, the effect of transverse forces on the seal, caused by a radial relative movement between the lid and cask, is not taken into account. Under normal operating and transport conditions, the seal is in a static state; the pre-tensioning of the screw connections ensures that no relative movements, i.e. changes in position, occur in the lid system.

Under hypothetical accident or incident conditions, however, changes in the position of the lid system can no longer be completely ruled out. In order to include the (removal) transport after extended interim storage in the ageing considerations, a fall in any direction from a height of 9 m onto an unyielding foundation must be assumed in accordance with the regulations. The resulting loads can cause the main lids to slip, for example in the event of a horizontal fall due to their high inertia. This transverse displacement can lead to changes in the contact surface between the seal and the flange and/or to deformation of the seal as a result of the transverse forces.

As the sealing effect is based on a complex interaction in the microscopic range between the ductile outer liner of the metal seal and the contact surface of the lid or cask, this cannot be derived from the seal characteristic curve, but must be determined experimentally for the respective load case. This is done by measuring the change in helium leak rate as a function of lid displacement.

In a drop test with a prototype cask whose lid system would be surrounded by a shock absorber as in the original, it is not possible to measure the leak rate at the time of the impact event due to the drop height and the inaccessibility of the lid system. In order to still obtain valid measurement results, the cask is not dropped from 9 m, but the "lid" is displaced by external forces. Similar to the MSTOR tests, a pair of flanges is clamped to the seal to be tested. A guided falling mass then hits one of the two flange halves from a defined height, causing it to move transversely, while the other half, to which the leak rate measurement equipment is connected, remains stationary. In this way, the leak rate can be determined over the entire duration of the dynamic transverse displacement as a function of the displacement movement. In addition, this generic test setup allows the transverse displacement to be set independently of the specific boundary conditions when testing a single design.

In line with MShift's objective of determining the leak rate of aged metal seals under dynamic transverse displacement, artificially pre-aged seals are used in the tests. In analogy to MSTOR, ageing is carried out by artificial ageing of the seals already compressed in the flange pair, whereby the ageing period and temperature are adapted to the storage periods to be taken into account. The MShift project tests both silver-coated and aluminium-coated metal seals, each with the cross-sectional dimensions of the original metal seals used in the CASTOR® cask main lids. In addition to tests at room temperature, tests are also carried out at -40°C to additionally cover the influence of temperature. Following the transverse displacement, leak rate measurements are continued for up to two weeks in order to provide an indication of the long-term performance of the seal following an incident. Due to the pre-tensioning of the screw connections, which ensures that the flange halves continue to be compressed tightly, and the remaining elastic recovery of the metal seal, the leak rate is expected to reach a maximum during dynamic displacement and to decrease again until it has reached a steady-state value.

11.1.3 MLift – Leak rate after recompression of aged metal seals

Subject matter:	MLift – Leak rate after recompression of aged metal seals » Performance of tests on aged metal seals » Derivation of design leak rates after a generic accident with lifting of the lid
Project partners:	» GNS Gesellschaft für Nuklear-Service mbH (cask manufacturer, Germany) » Technetics Group (seal manufacturer, France)
Project period:	2023 to 2031

The MLift project also investigates the performance of aged metal seals under accident or incident conditions. While the MShift project considers the load case of a transverse displacement of the lid, the MLift project described here focuses on the load case of a brief axial lift of the lid.

Gaps in the sealing area and transverse displacement of the lid are excluded under normal operating conditions and represent a generic scenario for accident or incident conditions. Generic because the mechanical design of the casks is basically such that, even under maximum accident or incident loads, the lifting of the lid is limited to values that remain below the useful elastic recovery of an aged metal seal. To cover longer periods of time, a scenario where the aged metal seal completely loses contact with the sealing surface during load application is considered here as a limiting case. The screw design ensures that the gap created under load is completely closed by the remaining screw clamping force and the seal is recompressed. The recompression of the seal is ensured by the fact that the screws of the lid system may essentially only be elastically loaded even under accident conditions in accordance with the design requirements (BAM GR 012 Guideline on the assessment of the lid systems and load attachment systems of transport casks for radioactive materials, edition 2020-12), so that sufficient residual clamping force remains after the impact phase to pull the lid back tightly.

As the aim of the research project is to determine a covering leak rate for the load case described, the contact between the seal and the lid is artificially removed during the tests by completely opening the flange with the aged seal. One of the two flange halves is lifted off after the screw connection has been loosened and, after a relative rotation (by one screw pitch circle), is then replaced and screwed back on. Twisting ensures that a completely new contact between the seal and the sealing surface must be made during compression.

As only main lids are potentially affected by the load case – the incident/accident loads acting on small lids are orders of magnitude lower – only metal seals for the primary and secondary lids are tested accordingly. This means that ten silver-coated and ten aluminium-coated seals with corresponding cross-sections are compressed in the flanges and artificially aged in line with the findings of the MSTOR project. After ageing, the flanges are opened and reassembled as described above. The leak rate is measured for at least one week following recompression. In this way, in addition to the maximum leak rate immediately after recompression, a statement can also be made about the development of leakage following a hypothetical incident.

In addition to the seals/flanges to be procured specifically for this project, it is planned to include the test flanges used in the MSTOR project in the MLift programme once the respective test series have been completed. This increases the statistical quality of the measurement programme and allows the effects of different ageing conditions or temperatures on the seal behaviour during recompression to be evaluated.

11.1.4 MSim – Numerical simulation of the long-term behaviour of metal seals during long-term interim storage

Subject matter:	MSim – Numerical simulation of the long-term behaviour of metal seals » Microstructural material analyses » Creation of material models
Project partners:	» Technetics Group (seal manufacturer, France) » MINES Paris – PSL
Project period:	2025 to 2028

The decisive factor for the application of the results from the experimental programmes on metal seals (MSTOR, MLift, MShift) in the safety proofs is the establishment of reliable prediction models that establish the relationship between temperature and time with regard to the ageing of metal seals for a specific application. As the prediction models are largely derived phenomenologically from the test results, they do not provide a direct physical explanation for the observed behaviour. To validate the prediction models developed in MSTOR, additional investigations and numerical simulations of the long-term behaviour of the metal seals are carried out.

The numerical simulation of the long-term behaviour of Helicoflex® metal seals is a holistic approach in which the interaction between the individual seal components (outer liner, intermediate liner and spiral spring) is explicitly taken into account. The advantage of a numerical model is that it depicts expected ageing processes based on physical relationships, with feedback from time-varying stress, strain and temperature distributions directly included in the ageing rate. This requires special material models, the creation of which requires additional creep tests on material samples of the sealing components and which are therefore part of the project.

The sealing parameters determined in the MSTOR project form the database that is to be used to check the numerical simulations. In addition, the numerical simulations can be used to explain the macroscopic changes in properties (restoring force, useful elastic recovery) that occur during long-term use and that were experimentally recorded in the MSTOR project, taking into account the spatial stress/strain state of the sealing components.

The material models are created in close collaboration with the seal manufacturer Technetics; the academic partner is the materials science institute of the Paris engineering university MINES Paris – PSL.

11.1.5 OBSERVE – Dose rate and temperature measurement programme

Subject matter:	OBSERVE – Dose rate and temperature measurement programme on loaded casks » Comparison of calculated expected values with measurement values on selected casks at different points in time during storage » Checking the shielding/heat dissipation properties of the casks (protection objectives)
Project partners:	» WTI Wissenschaftlich-Technische Ingenieurberatung GmbH
Project period:	Phase I (feasibility study): completed Phase II (development, production and commissioning of the measuring device): by 2028

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 19 and Figure 20). 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.

Activity and thus heat generation both decrease during storage as a result of 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 the ageing behaviour has been correctly assessed. In this way the measurement programmes can make an active contribution to ageing management.

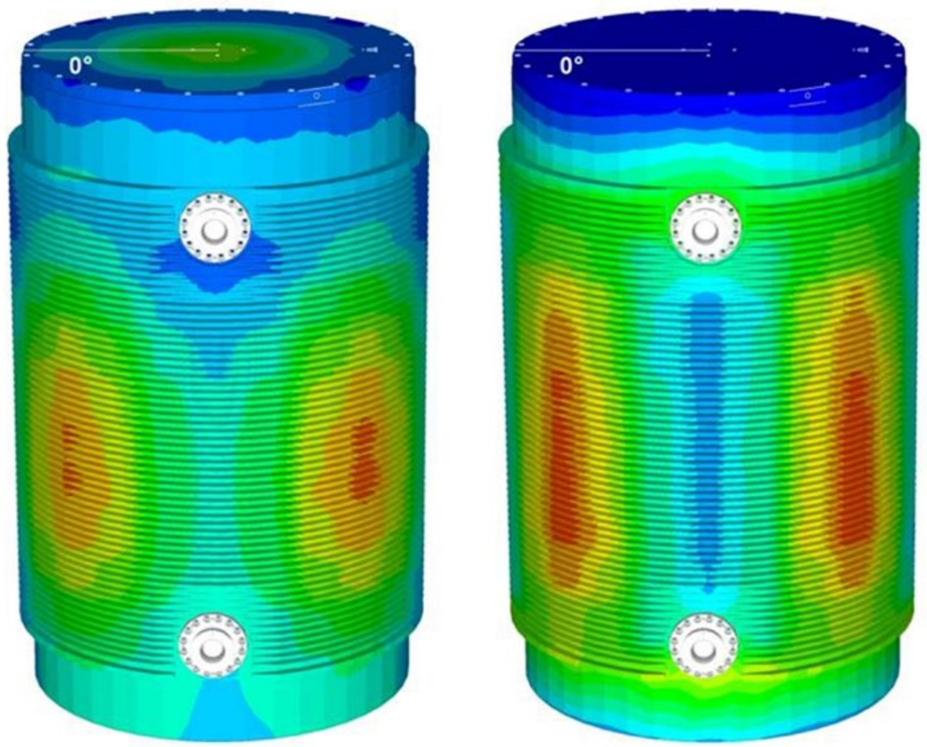

Figure 19: Distribution of dose rates on the surface of a CASTOR® cask (left: neutron dose rate, right: gamma dose rate)

In the first stage of the project, feasibility studies were carried out to check the extent to which dose rate/temperature measurements are suitable for drawing conclusions about the condition of cask components and the inventory [37]. Observations have shown that, to a certain extent, both changes in the fuel arrangement and gaps in the neutron shielding can be detected by close-meshed dose rate measurements on the cladding surface. It can therefore already be said that the OBSERVE measurement programme is in principle suitable for supplementing existing ageing management measures and for assessing the condition of the casks by measurement. In Phase II, the requirements for performing the measurements (equipment, measurement grid, measuring period) are derived and a corresponding measurement programme is developed.

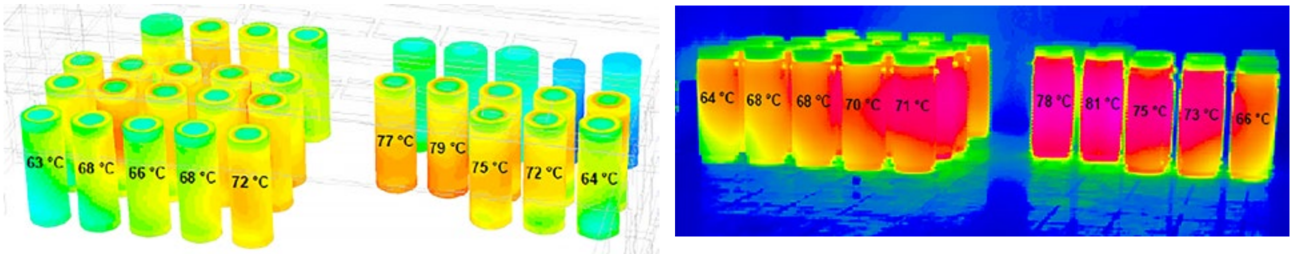

Figure 20: Comparison of calculated and measured maximum cask temperatures in an interim storage facility (left: CFD calculation, right: thermography)

11.1.6 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:	2021 to 2029 including BAM qualification

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 21 and Figure 22), 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". There have only been a very limited number of defects – so-called pressure switch events – in the more than 1,500 pressure switches that are installed worldwide. Some of these 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 23 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 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. After successful completion of the internal qualification, a qualification of the glass feedthrough is planned at BAM in order to be able to use the optimised design in the future production of pressure switches.

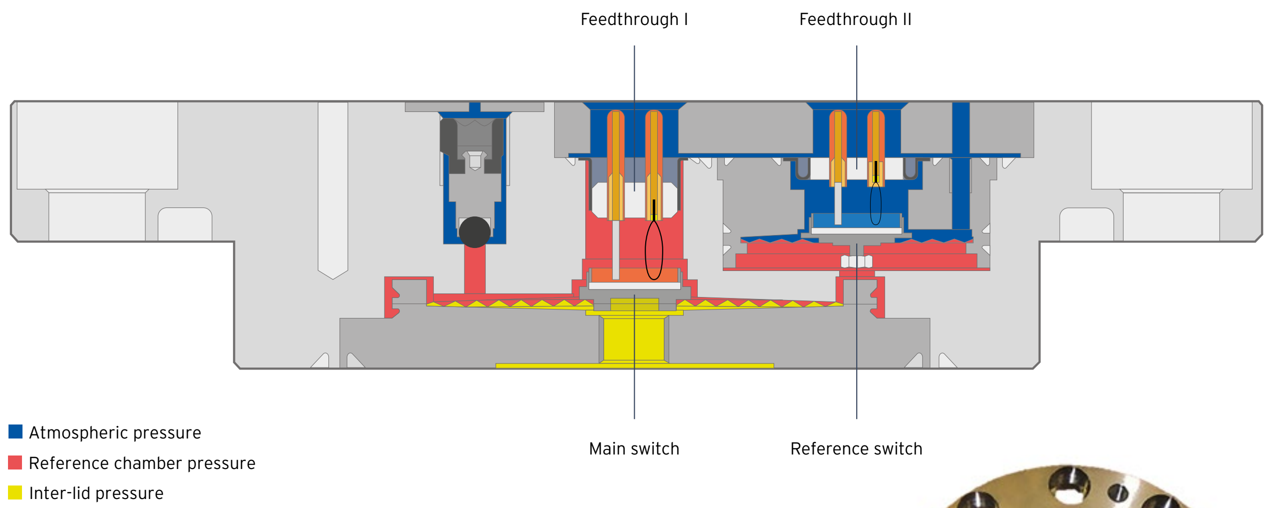

Figure 21: Basic design of the pressure switch

Figure 22: Pressure switch – top view without cover plate

Source: GNS

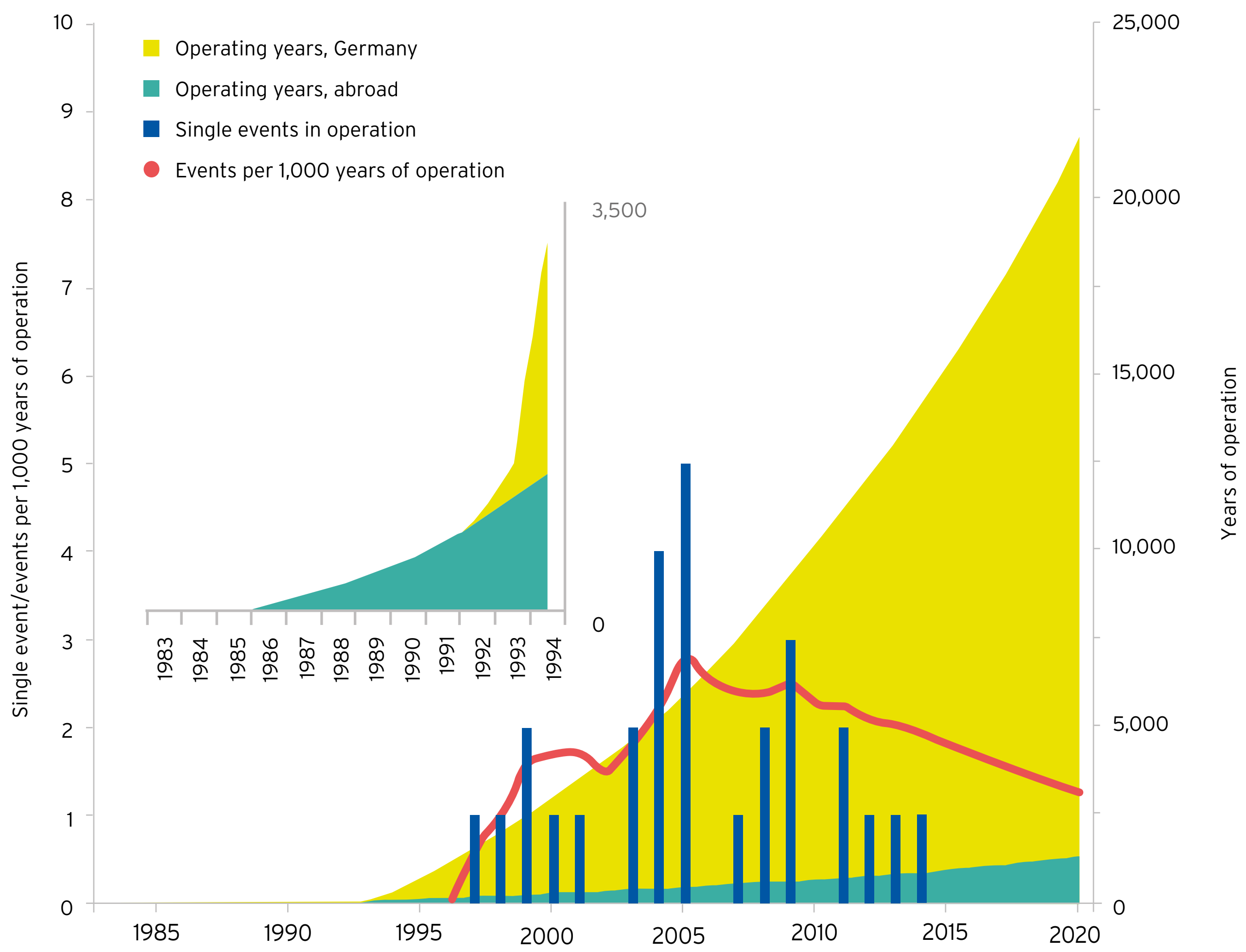

Figure 23: Comparison of years of operation and pressure switch events

11.2 Inventories

11.2.1 SCIP IV & V – Studsvik Cladding Integrity Project

Subject matter:	SCIP IV & V – Studsvik Cladding Integrity Project, Phase IV und V » 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
Project period:	Ongoing project with five-year phases, BGZ has been a participant since Phase IV (July 2019 to June 2024) Current Phase V: July 2024 to June 2029

The OECD/NEA Studsvik Cladding Integrity Project (SCIP) is concerned with the behaviour of cladding materials and the integrity of irradiated fuel rods. The SCIP projects are organised by the OECD-NEA and carried out in the hot cells of Studsvik Nuclear AB in Sweden. In Phase IV of the project (SCIP-IV, 2019-2024), systematic investigations into the behaviour of cladding tube materials under conditions of dry interim storage were carried out for the first time.

The results from SCIP-IV are used at BGZ to develop new models that improve the understanding of cladding tube behaviour in interim storage. The results from SCIP-IV were also incorporated into the development of the experimental programme for the current project phase SCIP-V (2024-2029). SCIP-V is an international research project comprising 44 organisations from 15 countries. To continue the investigations from SCIP-IV, the test programme provides for further basic and safety research on cladding tube behaviour under the conditions of dry interim storage. The focus here is on the creep behaviour of the cladding tubes under intermediate storage conditions, the hydrogen behaviour and the possible resulting hydrogen fracture mechanisms in cladding tube material. In addition, processes are investigated that have a bearing on hydrogen and creep behaviour. The test programme addresses a large number of the research questions identified by BGZ. BGZ actively contributes to the design of SCIP-V by submitting experimental proposals and discussing and interpreting the results. This involves regular discussions about the test design, test conditions and sample material for experiments to be carried out, as well as the evaluation and interpretation of the results obtained.

11.2.2 EPRI-ESCP Dose Modeling Task Group

Subject matter:	EPRI ESCP Dose Modeling Task Group » Blind benchmark for dose calculation of a loaded cask » Modelling, comparison of calculation results with each other and with measured data
Organisation:	» Subcommittee of the EPRI ESCP Modeling & Benchmark Group » International blind benchmark with participants from Europe, the USA and Asia » BGZ participates together with WTI
Project period:	2022 to 2026

The objective of the EPRI ESCP Dose Modeling Task Group is to model a loaded cask and calculate the dose, with subsequent comparison of the results between the participants and with the measurement results, which will be known only after the calculations.

The dose measurement on the cask was carried out on 22 September 2018, but the results are not known to the participants. This is to prevent the modelling from being influenced by the measured results.

A US cask (Holtec International MPC-68 with Holtec International HI-STORM 100 overpack) loaded with 68 BWR fuel assemblies with burn-ups of around 20 to 40 GWd/tHM will be modelled. The dose rate is calculated based on the irradiation history and the cask loading plan.

Participants' results will be compared in terms of deviations in results, modelling and implementation of the benchmark specifications. The calculation results will also be compared with the measured values to allow conclusions to be drawn about the accuracy of generalised prediction models. The results will be published in an EPRI report.

11.2.3 EPRI-ESCP Decay Heat Task Group

Subject matter:	EPRI-ESCP Decay Heat Task Group » Review of new measurement data on decay power and data uncertainties » Derivation of missing parameter spaces » Recommendations for new calorimetric investigations on spent fuel elements
Organisation:	» Subcommittee of the EPRI ESCP Modeling & Benchmark Group » International expert panel with participants from Europe, the USA and Asia » The task group is led by the Swedish SKB
Project period:	2022 to 2026

In the EPRI ESCP Decay Heat Task Group, current measurement data on the decay power of spent fuel elements are discussed. Particular attention will be paid to the evaluation of uncertainties. Among other things, the committee will discuss what measurements are still missing, such as fuel type, burnup history or storage time, and whether and how these measurements can be made.

An accurate determination of the decay power for long term storage, and the determination of its inaccuracy, is important both for extended interim storage and for subsequent disposal for several reasons. Material temperatures such as those of the cladding tubes or the outer wall of the cask can be better predicted with more accurate data on decay power. In this project, BGZ is directly involved in the evaluation of current measurement data and in the recommendation and planning of new experiments to complete and extend the existing data on decay power. A detailed technical report analysing the measurement data is publicly available [38].

11.2.4 LEDA – Long-Term Experimental Dry Storage Analysis

Subject matter:	LEDA – Long-term experimental analysis of cladding performance » 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 and implementation with partners from industry and science
Project period:	2022 to 2030

An experimental campaign is being conducted at Studsvik in Sweden to investigate questions relating to cladding performance under conditions of dry interim storage and, in particular, 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 for dry storage in Germany. Typical conditions are set and monitored via online monitoring. Furthermore, suitable investigations for the pre- and post-characterisation of the cladding tube 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 will be reviewed taking account of (long-term) hydrogen behaviour for storage times of over 40 years, and their completeness will be analysed.

The long-term measurement campaign will be carried out with different irradiated fuel rod segments under prototypical boundary conditions. This involves an integral approach under drying process and dry interim storage conditions.

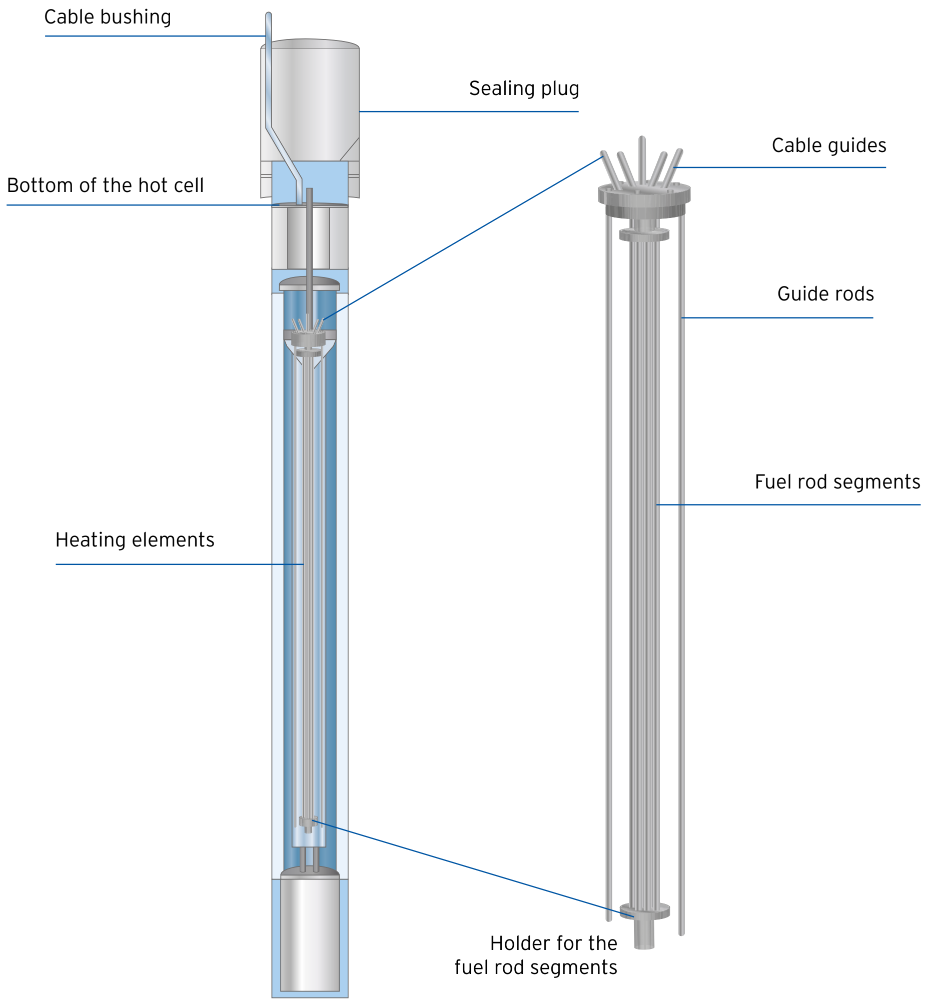

Figure 24: Cross-section of the test apparatus after integration into the hot cell (left) with the enlarged section of the holder for the fuel rod segments (right). The test apparatus is located in a shaft in the floor of the hot cell. Up to eight fuel rod segments can be exposed to a temperature transient at the same time.

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 with respect to extended interim storage in Germany.

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, which has now been extensively modified, is shown in Figure 24 and also in [39]. 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 is maintained and then successively lowered, similar to the drop in temperature in the fuel assemblies in interim dry storage after the cask has been sealed.

At least three test campaigns with up to eight fuel rod segments each are currently planned. Each campaign includes an extensive pre- and post-characterisation phase during which mechanical and material parameters are determined. The current plans include investigations of a total of eleven fuel rod segments and ten entire fuel rods. The latter are each divided into four segments so that they can be analysed in the test stand.

The segments and rods have been selected to best represent the inventory held by BGZ. The burn-up of the fuel rods and segments is between 20 and 80 GWd/tHM and the cladding tube materials consist of the alloys M5®, DUPLEX, Optimized ZIRLO®, Zry-4 and Zry-2 with and without liner.

11.2.5 LAGER – Laser Ablation of Gadolinium Evolution Radially

Subject matter:	LAGER – Laser Ablation of Gadolinium Evolution Radially » Experimental investigation of a gadolinium-containing BWR fuel rod with a rod burn-up of only 9 GWd/tHM
Organisation:	» Multilateral project, led by Studsvik AB » Project partners from industry, science and institutes from Europe, the USA and Asia » BGZ is an active project partner
Project period:	2022 to 2026

The LAGER project is a multilateral project to better understand and quantify the radial distribution of gadolinium isotopes in a fuel rod with low and relevant burn-up. Gadolinium is added to the fuel of individual rods in modern fuel element designs to reduce the reactivity of the fresh fuel element. Comparable data is currently scarce and current models use indirectly validated estimates instead.

The aim of the LAGER project is to obtain detailed data on the Gd distribution at selected axial positions and to make the data available for code validation. In addition, source term data will be determined and the modelling of burn-up calculations will be further developed. The experimentally determined data are needed to further develop accurate models for Gd burn-up and to enable prediction of reactivity and energy distribution within the fuel assemblies, especially at low burn-ups in the range of maximum reactivity. This data is of great value to BGZ, as the rod under analysis is an important part of the inventory. In the course of the German nuclear moratorium in 2011 and the subsequent amendment of the Atomic Energy Act, BWR fuel assemblies with very low burn-up were also placed in dry interim storage in Germany.

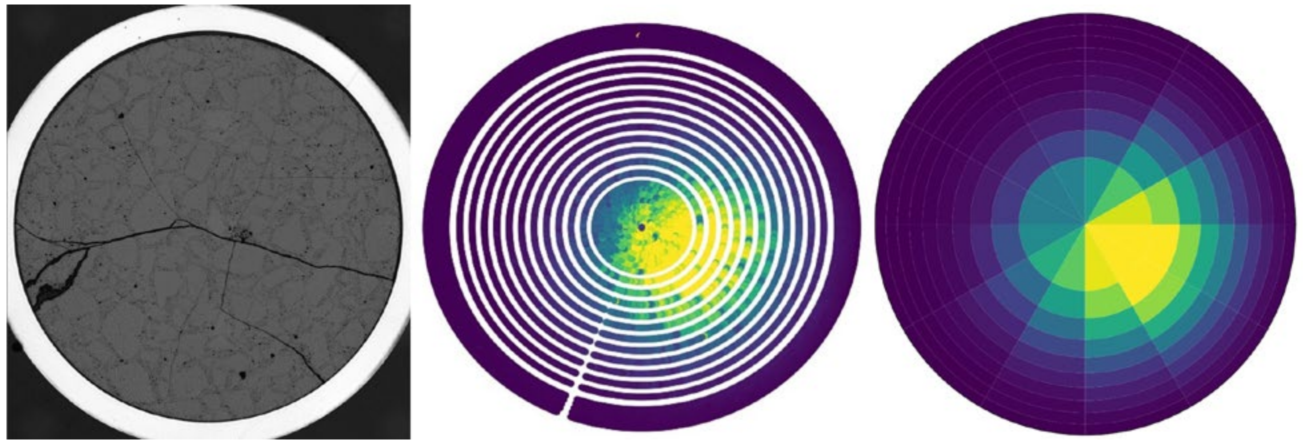

Figure 25: Laser ablation mass spectrometry sample from the LAGER project: Light microscope image of the pellet cross-section, measurement of the ^{155}Gd and simulation results for the ^{155}Gd concentration (from left to right)

The fuel rod investigated in LAGER is an intact BWR rod from a 10 x 10 fuel assembly with 3.66 weight percent ^{235}U enrichment and 4 per cent initial Gd content. The average rod burn-up is approximately 9 GWd/tHM with axial values between 6 and 11 GWd/tHM. The irradiation history is typical of BWR irradiation for fresh fuel without a history of a control rod, but with a neighbouring part-length fuel rod. The latter leads to an axial power divergence of 5 to 15 per cent. The operator, Vattenfall, has provided the detailed operating history for the project, which enables accurate comparative calculations to be made. The LAGER project is being carried out at Studsvik's hot cell facilities, using both industry-standard post-irradiation analysis methods and unique techniques for quantifying local pellet properties. The axial positions for the local measurements were selected based on modelling and non-destructive testing. The LAGER project partners were actively involved in selecting the axial samples and the experimental setup, and decisions were made by consensus. BGZ accompanies the investigations with its own detailed simulations and is thus able to check the quality of the existing models and simulation programmes and identify any need for improvement.

At the same time, a deeper understanding of the local irradiation processes is achieved. This in turn enables a more precise computational determination of the radionuclide composition and the residual reactivity of the irradiated fuel and helps to quantify the uncertainties in the models.

The results will be discussed and compared within the framework of accompanying workshops in the international expert group. The measurement campaign is expected to be completed in 2025. An initial overview of BGZ's calculations can be found in [40].

11.2.6 HYDAX – Hydrogen and Hydrides Distribution in Axial Cladding Direction

Subject matter:	HYDAX – Hydrogen and Hydrides Distribution in Axial Cladding Direction » Experimental investigations of hydrogen and hydrides behaviour in the axial cladding direction
Organisation:	» BGZ in collaboration with the Paul Scherrer Institute (PSI)
Project period:	2023 to 2028

In the HYDAX project, BGZ and the Paul Scherrer Institute in Switzerland are investigating the axial behaviour of hydrogen and hydrides in various cladding tube materials. Experimental investigations are carried out on irradiated and non-irradiated cladding tube samples with and without liner. Specifically, the kinetics of hydrogen diffusion in the axial cladding tube direction and hydride formation will be investigated, taking into account the material texture. This will enable differences in hydrogen diffusion to be identified compared to the radial direction. Temperature and its rate of change also play a role in diffusion. By varying these, the effects on hydride formation can also be studied experimentally. Another effect is the stress and strain in the cladding tube. By using irradiated and non-irradiated cladding tube materials, the effects of radiation damage to the material on diffusion and hydride formation properties can also be observed and studied. An overview of the initial work can be found in [41].

11.2.7 Bend & Break – Investigations into fuel release during fuel rod bending tests

Subject matter:	Investigations into fuel release during fuel rod bending tests » 3-point bending tests on irradiated fuel rods with M5® cladding tube materials » Determination and description of the fuel releases
Organisation:	» BGZ in cooperation with Framatome GmbH and the Gösgen-Däniken nuclear power plant (CH) » Experimental investigations in the hot cells of the European Commission's Joint Research Centre in Karlsruhe
Project period:	2025 to 2026

The aim of Bend & Break is to determine the type, quantity and size distribution of released particles through systematic static tests on irradiated M5® cladding tubes with fuel from reprocessed uranium (RepU/ERU).

The package design approvals are based on the conservative assumption that, under accident transport conditions, large quantities of fuel will escape from the cladding tubes under the assumed mechanical load of irradiated fuel rods. In the mid-2010s, static 3-point bending and dynamic hammer drop tests were carried out on irradiated fuel rods with duplex and Zry-4 cladding tubes in the hot cells of the Joint Research Centre (JRC) of the European Commission in Karlsruhe and the release rates and particle size distributions were documented in detail. The Bend & Break project extends the results with the corresponding tests on M5® cladding pipes. The tests will be carried out in the JRC's hot cells together with the project partners Framatome and the Gösgen nuclear power plant. The project will precisely record the type, quantity and size distribution of the fuel fragments released in the event of failure and compare the data obtained with the results already available for Duplex and Zry-4.

11.2.8 VisCas – Visualisation of fuel rods in loaded casks

Subject matter:	VisCas – Visualisation of fuel rods in loaded casks » Investigation of the theoretical resolution of experimental realisations of muon tomography » Creating and analysing suitable simulation models
Organisation:	» Research project with the Chair of Nuclear Technology at the Technical University of Munich (TUM) » Modelling and simulation of the interaction with Geant4 » The objective is to theoretically assess various detectors and experimental setups and to analyse the data with a view to visualising individual fuel rods
Project period:	2023 to 2026

The focus of the research work is on modelling the interaction of muons with the inventory of the most frequently used storage and transport casks for spent nuclear fuel from pressurised water reactors in Germany, the CASTOR® V/19. The specific questions are:

- Can individual fuel rods be analysed using a suitable experimental setup in combination with suitable evaluation algorithms?
- Can the position, shape and integrity of individual fuel rods be assessed, for example by visualising the three-dimensional density distribution of the fuel?
- Can it be recognised or ruled out that fuel has escaped?

To answer these questions, the work is divided into two parts: a theoretical part for the modelling, simulation and data analysis framework and an application part where the knowledge gained is used to evaluate, design and describe future experimental implementations.

The modelling and simulations are carried out using the Geant4 program [42]. One aim of the theoretical part is to adapt the previously used, simplified loading data of an existing model to real data. An analysis will then be made of the extent to which the model can be simplified to reduce computational time and the volume of result data without compromising the accuracy of the results.

The experimental data and results obtained in the MuTomCa project will also be analysed.

Based on the findings and further Geant4 simulations on detector positions and resolution as a function of the measurement duration, a design for an improved experimental setup for visualising individual fuel rods will be developed. The interim results are published in [43, 44].

11.2.9 SKELETON – Determination of the material behaviour of irradiated fuel assembly structural parts

Subject matter:	SKELETON – Determination of the material behaviour of irradiated fuel assembly structural parts » Experimental and theoretical investigations of material samples of irradiated BWR and PWR fuel element structural parts » Creating and analysing suitable simulation models
Organisation:	» Management by BGZ, implementation with other partners » Experiments are carried out in laboratories of Framatome GmbH
Project period:	2024 to 2028

The integrity of a fuel element (FE) depends crucially on the load-bearing capacity of the structural components. These include the FE head and base, spacers, water channels (for BWR-FE) and control rod guide tubes (for PWR-FE), which form the load-bearing structure. After the fuel rods, the structural parts receive the highest radiation doses and are exposed to particular stresses due to their function. The SKELETON research project evaluates the mechanical integrity of structural parts of irradiated light water fuel elements (LWR-FE).

SKELETON combines the processing of existing test data with new investigations on irradiated FE structural components. First of all, the results of experiments that have already been carried out are systematically evaluated in order to create a reliable database on the behaviour of PWR control rod guide tubes. In addition, samples from a water channel of a BWR-FE used in a German nuclear power plant are analysed in the hot cells of Framatome GmbH. This includes measurements of fracture toughness, complete stress-strain curves and determination of the hydrogen content. Heat treatment simulates the conditions of dry interim storage, and the resulting material changes are analysed and recorded.

The characteristic values obtained experimentally are incorporated into the development of physically consistent material models, which are implemented in a numerical simulation environment. By comparing the simulation results with the measurement data obtained and with established fuel performance codes, a validated tool is created that realistically depicts typical load scenarios during transport, interim storage and handling.

11.3 Interim storage buildings

11.3.1 ZuMoBau-ZL – Condition assessment and monitoring for the evaluation of the technical service life of structural facilities of interim storage facilities for high-level radioactive waste

Subject matter:	ZuMoBauZL – Condition assessment and monitoring for the evaluation of the technical service life of structural facilities of interim storage facilities for high-level radioactive waste » Identification of suitable monitoring and measurement techniques for recording the relevant ageing processes » Development of concepts for service life prediction based on monitoring data
Organisation:	» BMUKN research funding (funding codes RS1586A, 1501609B) » Joint project of GRS, BAM and TU Braunschweig (iBMB) » BGZ has a seat on the technical monitoring committee
Project period:	2023 to 2026

The joint project, coordinated by the iBMB at the Technical University of Braunschweig, aims to develop concepts for service life prediction using appropriate measures and measurement techniques in the field of non-destructive testing. Many of the findings and methods that have been tried and tested on conventional structures can be applied to interim storage structures. The project will combine these findings with the additional requirements for an interim storage facility to provide a realistic overview of ageing processes and their effects on an interim storage facility, taking into account real operating experience. Subsequently, using an exclusion methodology to be developed, the non-relevant ageing processes and effects are excluded from further consideration.

In addition to identifying the relevant damage processes, the aim is to use models to describe and evaluate such processes, which will make it possible to predict the service life of components. The model parameters identified in the process are then analysed with regard to their relevance to the operational safety and protective function of the interim storage facilities. The condition assessment methods for existing structures applicable to engineering structures according to DIN 1076 or to buildings according to VDI 6200 are also analysed for possible adaptation, taking into account the special requirements of interim storage facilities.

Once the global and local parameters for describing the relevant damage mechanisms have been identified, various sensors and measurement techniques will be analysed to determine their suitability for more accurate and continuous determination of these parameters.

Concepts for the digital storage and management of the data will then be developed for the monitoring and condition data generated for service life prediction. Requirements are also defined for the connection to suitable building management systems, i.e. interfaces to a BIM (Building Information Modelling) concept or the implementation of the "digital twin" concept.

12. Completed projects

Projects from the research programme will also be available on BGZ's website in the Research section after their completion. Reports, scientific publications, theses and conference papers produced as part of the projects are available there, where legally possible.

12.1 Inventories

SPIZWURZ – Stress-induced hydrogen rearrangement in fuel rod cladding (2020 – 2024)
In the BMWi/BMUKN-funded joint project of GRS and KIT (funding codes RS1568A [GRS] and 1501609B [KIT]), experimental and model-based investigations were carried out into how hydrogen is distributed in zirconium-based cladding tubes under long-term stress conditions. The work expanded the data basis for hydrogen diffusion at micro and macro level and provided data on the hydride structure of non-irradiated cladding tubes. BGZ was involved as an observer and was able to incorporate the findings into its own research projects with irradiated fuel rods; the progress of the project is documented in [45].

EPRI-ESCP Thermal Modeling Benchmark (2019 – 2024)
As part of an international benchmark of the Electric Power Research Institute [41, 42], BGZ – together with GNS and WTI – compares thermal calculation methods on an instrumented TN®32 cask that was loaded at the North Anna nuclear power plant (USA) in 2017. The aim is to estimate cladding tube temperatures more realistically and to quantify the sensitivity of different modelling approaches. The first modelling phase has been completed and published [46]. Phase 2 deepens the uncertainty and sensitivity analyses of the various modelling approaches and feeds into the best-estimate proofs for future storage licences. Partial results are published in [47]. A detailed final report had not yet been published at the time of going to press.

DCSMonitor II – Non-invasive monitoring of CASTOR® casks (2020 – 2024)
In the BMWi/BMUKN-funded joint project of the Dresden University of Technology (TUD, project coordinator), the Zittau/Görlitz University of Applied Sciences and the Helmholtz-Zentrum Dresden-Rossendorf e. V. (funding codes 1501606A, 1501606B), a radiation field-based diagnostic system was developed that combines gamma, neutron and muon measurements to reliably detect inventory changes in transport and storage casks. In addition to numerical sensitivity studies, automated measurement systems are being set up and tested on real CASTOR® casks in EWN's interim storage facility. BGZ supports the project as an associated partner and uses the findings to further develop its monitoring and safety concepts in the OBSERVE project. Reports and interim results are available in [48].

MuTomCa – Muon Tomography for Shielded Casks (2020 – 2024)
Since its launch in September 2020, the international MuTomCa consortium [49], led by INFN Padua together with Forschungszentrum Jülich, EURATOM and BGZ, has been developing a muon detector based on drift tubes to non-invasively map the contents of CASTOR® dry storage casks. This method focuses on proliferation (safeguards). During a field campaign in 2023 at BGZ's interim storage facility in Grafenrheinfeld, muon imaging data was successfully recorded on two CASTOR® V/19 casks with different loads, demonstrating that fuel-free dummy fuel elements can be recognised. The project was funded by the BMUKN (FKZ 02W6279) and EURATOM Safeguards programmes, among others. The results are documented in several publications [50] and the project reports [51]. BGZ uses the findings to further develop its monitoring and safety concepts in the VisCas project.

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List of abbreviations

Abbreviation	**Explanation**
ADR	European Agreement concerning the International Carriage of Dangerous Goods by Road
AVR Jülich	Jülich Experimental Reactor Consortium
BAM	Federal Institute for Materials Research and Testing
BAM-GGR	Dangerous goods regulations issued by the Federal Institute for Materials Research and Testing
BASE	German Federal Office for the Safety of Nuclear Waste Management
BGE	Bundesgesellschaft für Endlagerung mbH
BWR	Boiling water reactor
BWR-FA	Boiling water reactor fuel assembly
BZA	Ahaus Interim Storage Facility for SNF
DIN	Deutsches Institut für Normung e. V. (German Standardisation Organisation)
DPC	Dual-purpose cask for the transport and storage of HAW
EPRI	Electric Power Research Institute
ESC	Energy supply company
ESCP	Extended Storage Collaboration Program
ESK	Nuclear Waste Management Commission
ESTRAL	Replacement transport cask storage facility
EURATOM	European Atomic Energy Community
EWN	Entsorgungswerk für Nuklearanlagen GmbH
FA	Fuel assembly
GDR	German Democratic Republic
GGV-SEB	German Ordinance on the Transport of Dangerous Goods by Road, Rail and Inland Waterways
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
IAEO	International Atomic Energy Agency
IAM	Institute for Applied Materials – Materials Science
INE	Institute for Nuclear Waste Disposal
ISO	International Organization for Standardization
JEN	Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH
K	Kelvin (absolute temperature, 273.15 K = 0 °C)
KIT	Karlsruhe Institute of Technology
KTA	Nuclear Safety Standards Commission (Kerntechnischer Ausschuss)
KTE	Kerntechnische Entsorgung Karlsruhe GmbH
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
MW	Megawatt (unit of measurement for heat generation, 1 MW = 1,000 kW)
NEA	Nuclear Energy Agency
Orano NPS	Orano Nuclear Packages and Services
PSI	Paul Scherrer Institute
PWR	Pressurised water reactor
PWR-FA	Pressurised water reactor fuel assembly

RID	Regulation concerning the International Carriage of Dangerous Goods by Rail
RR-FA	Research reactor fuel assembly
SINQ	Swiss Spallation Neutron Source
SKB	Svensk Kärnbränslehantering AB
SLS	Swiss Light Source
SZL	On-site interim storage facility
THTR	Thorium high-temperature reactor
THTR-FA	Thorium high-temperature reactor fuel assembly
TUM	Technical University of Munich
UF & HLW	Used Fuel & High Level Waste
VDI	Verein Deutscher Ingenieure e. V. (Association of German Engineers)
Weight %	Weight percent (percentage by mass in the mixture of substances)
WTI	Wissenschaftlich-Technische Ingenieurberatung GmbH
ZLN	Interim Storage Facility North (Zwischenlager Nord)
ZWILAG	Zwischenlager Würenlingen AG

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DOI: 10.69152/BGZRESEARCHPROGRAMME/2025

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Year of publication: 2025