safe solutions for radioactive waste

Implementing Geological Disposal for Radioactive Waste - Technology Platform

IGD-TP Exchange Forum No. 7

25-26 October 2016

Cordoba, Spain

Book of Abstracts

Foreword

The IGD-TP Exchange Forum (EF) gathers the geological disposal of radioactive waste community in order to i) informally exchange information on common interests in research, development and demonstration (RD&D), ii) highlight IGD-TP ongoing activities and EC projects, iii) initiate or deepen collaborations with other organisations, iv) explore new ideas that could complement the IGD-TP Strategic Research Agenda and v) prepare future activities and projects that may be developed in the framework of the EC Euratom work programme.

The 7th Exchange Forum (EF6) was held on 25-26 October 2016 in Cordoba, Spain and gathered about 100 participants from 19 different countries. It was themed around increasing the maturity of technology. Specific key-notes were given in plenary sessions on:

- National policy in Spain
- R&D activities in Ukraine
- Updates from the European Commission (EC) on the Joint Programme structure, Strategic Research agenda, schedule, mechanisms and funding.
- Adaption of mature disposal concepts to less advanced programmes

For this 7th edition, it was decided to review EC projects that were completed in 2016. These were the BELBaR, LOCOEX and DOPAS projects. The aim was to present the outcomes and assess the achievements of these projects and present how they have helped to reach the IGD-TP's Vision.

In addition to the plenary sessions, four parallel technical sessions were organised around four topics with the objective to discuss topics that could be submitted as EC project in the forthcoming call:

- WG1 – Industrialisation and optimisation
- WG2 – Canister Design
- WG3 – High temperature clay interactions
- WG4 – Spent fuel characterization

This document presents the abstracts of the presentations made during EF7.

All presentations are available on the IGD-TP website www.igdtp.eu.

Role of the IGD-TP

- Share information, promote debate and understanding between member waste management organisations (WMOs) in the field of geological disposal research, development and demonstration (RD&D);
- Define RD&D priorities that could lead to project s of European added value, potentially with the support of the European Commission;
- Identify projects of European added value which may be undertaken as joint collaborative projects for the benefit of all programmes.

IGD-TP Exchange Forum

- The Exchange Forum is the regular meeting of the IGD-TP, gathering participants and stakeholders that are committed to our Vision;
- The forum is intended to give an overview of on-going activities and to promote discussions on how to prioritize different activities to reach the Vision;
- It may act as an incubator for new projects and collaborations in the field of geological disposal.

Domain of Activities

- The RD&D activities oriented towards geological disposal for high-level and long-lived waste are the driver of the work of the IGD-TP;
- The Strategic Research Agenda offers the possibility to enlarge the scope of the work toward radioactive management issues such as the waste forms and their behaviour;
- The IGD-TP platform acts as an information exchange platform with other initiatives and fora such as SNETP, IGSC-NEA, MELODI, ENTRA.

Some numbers

11	Waste Management Organisations
124	participating organisations committed to the project
6	ongoing EC funded projects
6	proposals under discussion

IGD-TP 7th Exchange Forum - Highlights

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Plenary Session 1

National policy in Spain

Pablo Zuloaga, Enresa, Spain

(No Abstract)

R&D activities in Ukraine towards geological disposal of radioactive waste

Iurii Shybetskyi, Radioenvironmental Centre of National Academy of Sciences, Ukraine

The R&D activities, which has been undertaking in Ukraine over the last twenty years, generally correspond to the needs of the current stage of the Ukrainian program on geological repository creation. This is the early stages of site selection and preconceptual study of the disposal system.

The main focus of R&D activities in Ukraine remains the work on site selection and development of methodology for such a selection. Three promising areas for waste geological disposal were determined within Chernobyl Exclusion Zone and in adjacent territories.

Relatively less attention was paid to developing the concepts of geological repository and its engineered barriers (containers, buffer and backfill materials etc.). Therefore, the assessments of the prospects of various regions and areas were mainly carried out on the basis of geological data.

In recent years Ukraine has made significant progress in developing advanced safety case methodology for radioactive waste geological disposal. Results of preliminary safety assessments demonstrate suitability of crystalline rocks of the Chernobyl Exclusion Zone for waste geological disposal, as well as urgent necessity of its field investigations (including drilling) within the promising areas.

Update from the European Commission

Christophe Davies, European Commission, Belgium

(No Abstract)

Plenary Session 2

Focus on less advanced programmes

Jiri Slovak, SÚRAO / Radioactive Waste Repository Authority, Czech Republic

(No Abstract)

BELBaR end of project overview

Patrik Sellin, SKB, Sweden

(No Abstract)

LUCOEX end of project overview

Erik Thurner, SKB International, Sweden

(No Abstract)

DOPAS end of project overview

Johanna Hansen, Posiva Oy, Finland

(No Abstract)

Working Group 1 – Industrialisation and Optimisation

Outcomes

Chair: Johan Andersson

Rapporteur: Johanna Hansen

List of presentations

- **Session introduction – WG aims,** Johan Andersson, SKB and Johanna Hansen, Posiva
- **Industrialisation and optimisation of the KBS-3 repositories,** Johan Andersson, SKB and Tiina Jalonen, Posiva
- **Development of optimisation methodology and methodology illustration,** Daniel Galson, Galson Sciences Ltd.
- **International cooperation and consensus in geological disposal,** Andrejs Dreimanis, Radiation safety centre of the State Environmental Service
- **Reducing the cost of waste containers for decommissioning and geological disposal,** Jack Hardy, Nuclear Advanced Manufacturing Research Centre
- **Pyro(hydro)lysis of Spent Ion Exchange Resins for the Disposal of Organic-Free Radioactive Waste,** Jung, NUKEM
- **Looking for rational environmentally friendly approach for development of concept for safe nuclear waste repository in preparing for future projects,** Ivan Ivanov, TUS

Abstracts

Industrialisation and optimisation of the KBS-3 repositories

Johan Andersson, SKB, Sweden, and Tiina Jalonen, Posiva Oy, Finland

Nuclear waste repositories can only be developed in a staged iterative fashion. At early stages, e.g. prior to site selection, the aim of the design work is typically to develop concepts that have the potential to result in a safe repository. During a siting process the design work would need to show how a suitable design can be adapted to the available siting environments and potential host rocks as well as to provide guidance for the characterisation of the specific sites to be explored. When applying for a construction license the designs need to be further developed such that it can be demonstrated that the designs as adapted to the site will lead to safety and that the design can be realised in practice. Both in Finland and Sweden the development of the KBS-3 system for final disposal of spent nuclear fuel has reached this latter stage. Posiva in Finland has a construction license and SKB in Sweden is in the licensing process of potentially obtaining a constriction license. However, while a technically feasible reference design and layout has been developed at this stage, detailed designs adapted to an industrialized process designed to fulfilling specific requirements on quality, cost and efficiency need still be developed. Also, the repository layout may need to be adapted to the local conditions found when constructing the repository at depth. Furthermore, since repositories typically will operate over several decades further optimisation of the design and operational procedures can be envisaged.

Development of optimisation methodology and methodology illustration

Daniel Galson, Galson Sciences Ltd, UK

The requirement to demonstrate optimisation becomes increasingly important as implementation programmes for radioactive waste disposal facilities progress through the lifecycle of generic studies, site-specific studies, construction approval and operation approval. Thereafter there is an ongoing requirement to ensure disposal practices continue to be assessed and that optimisation considerations are kept live, particularly as closure approaches. There is no commonly accepted approach to demonstrate optimisation for radioactive waste disposal facilities and requirements and expectations may differ between countries. This presentation will elucidate some of the main expectations on optimisation in advancing proposals for radioactive waste disposal facilities. The presentation will call for two tasks to be worked jointly at European level through provision of examples and starting points:

- **Methodology elaboration** – what do we mean by optimisation in terms of geological disposal, and to what extent can we agree at European level a generic "workflow" or approach to optimisation during the lifecycle of the implementation process, identifying how requirements, constraints and level of detail will change as a project progresses? Can optimisation / risk reduction be linked to a holistic and balanced analysis of safety functions for the operational and post-closure periods? Are there generic steps that waste management organisations should follow to demonstrate progression in technology readiness level, for example associated with full-scale testing and commissioning of facilities ahead of receiving

a licence to operate? The Design Basis Workflow elaborated in the recently completed EC DOPAS project (Demonstration of Plugs and Seals) provides a framework for an optimisation process that must be undertaken to move conceptual design to detailed design implementation. It could serve as one possible starting point for considering the feasibility of developing a generic optimisation workflow.

- **Methodology illustration** – via contribution of worked examples from national programmes at different stages of advancement. There are good examples from several national programmes, for example the successive development of deposition tunnel plug design by SKB and Posiva via the Stripa Shaft and Tunnel Sealing Tests, the Äspö Backfill and Plug Test, the Äspö Prototype Repository, and the Äspö Dome Plug (DOMPLU) Experiment. There are also good worked examples in other countries (e.g. the US Waste Isolation Pilot Plant) illustrating how optimisation approaches and requirements evolve through the repository development process. In the UK, work on geological disposal facilities is still at the generic stage, but the new LLW disposal facility brought into operation earlier this year at Dounreay illustrates how the concept of optimisation has been extended to cover each aspect of the development and design of a disposal facility, including site selection and characterisation, design and operation, and backfilling and closure – and also how these analyses have been periodically updated over time to demonstrate the progression of thinking as the project advances.

International cooperation and consensus in geological disposal

Andrejs Dreimanis, Radiation safety centre of the State Environmental Service, Latvia

We consider an approach how to promote development of multinational deep geologic repositories - providing disposal of high-level radioactive waste (RW) and spent fuel, at the same time essentially contributing to physical protection of nuclear material in the currently elevated global nuclear threat level.

Recognizing the choice of the host country as a prior problem in siting such facility we propose an interdisciplinary approach to multilevel stakeholder involvement and consensus building for siting of shared multinational RW deep repositories. The approach is based on a novel model of societal optimization of RW management in an extended environment being a multitude of physical, ecological, economic, socio-cultural and psychological factors by using theself-organization (SO), chaos and fuzziness concepts as well as the principle of requisite variety.

In the problem of a multi-national repository siting there appears an essential novel component of stakeholder consensus building: to reach consent – political, social, economic, ecological – among international partners, in addition to the intra-national consensus building tasks. An entire partnering country is considered as a national stakeholder, represented by the national government, being faced to simultaneous seeking an upward (international) and a downward (intra-national) consensus in a stressed environment, having possibly diverse political, economic and social interests.

Following theses about building of multilevel consensus are developed:

1) owing to stakeholder informational SO via their cooperation and competition there are formed knowledge-creating stakeholder communities,
2) building of international stakeholder consensus could be promoted by activating and diversifying multilateral interactions between intra- and international stakeholders, including web networks and various institutional networks for international cooperation,
3) development of partnership between inter-national and intra-national stakeholders - a key towards democratic dialogue, with the aim to observe distinguishing interests and to reach a shared understanding of disputable issues,
4) emerged controversies are resolvable using synergetic approaches of conflict resolution: a) moderate chaos (mutual flexibility) succeeds to non-rigid step-by-step approach to the choice of the host country, b) fuzziness in the siting strategy could promote societal SO by reducing mutual misunderstanding in decision-making,
5) social learning, cross-cultural and integrative thinking and knowledge - basic prerequisites for developing participative consciousness and co-awareness of existence of shared goals and favouring adequate risk perception.

The proposed approach for reaching multinational facility siting consensus can be extended to solving similar problems for arrangement of nuclear power plants and research units. One can recommend to develop - in the frame of international cooperation - further systemic interdisciplinary studies being goal-oriented towards implementing actual shared projects.

Reducing the cost of waste containers for decommissioning and geological disposal

Jack Hardy, Nuclear Advanced Manufacturing Research Centre, UK

An overview of the Waste Container Integrated Project Team (IPT) - including its background, programme of activity and opportunity for international collaboration - will be shared as part of this presentation. Technical details and preliminary results of a case study regarding a 3m³ box manufacturing improvements programme will also be shared and discussed.

Waste Container IPT

The waste containers used for disposal of nuclear waste are expensive, with current projections suggesting that over the lifetime of the Sellafield site in the UK, up to £4 billion will be spent on waste containers alone. The current cost estimates to procure individual waste containers (typically $3m^3$ and $4m^3$ boxes) can be in excess of £50,000 each. This high cost is driven by the constraints of Sellafield Ltd (SL) and Radioactive Waste Management Ltd (RWM) for interim storage and final disposal of waste packages in the geological disposal facility (GDF) respectively.

Aligned with SL's Technical Innovation Plan, the current challenge for waste containers is to:

"Provide an improved range of cost-effective waste container options that have been designed to facilitate decommissioning and, where appropriate, to meet the criteria for final disposal, reducing the overall cost of waste containers to Sellafield by 50% from the current baseline."

To achieve this cost reduction, three areas of opportunity are envisaged:

1. A review of waste containers currently utilised across the UK and international nuclear organisations.
2. Utilisation of advanced manufacturing techniques or materials that have potential to provide more cost-effective waste container solutions than are currently available.
3. Feasible alternatives to the current waste package constraints.

An IPT consisting, in the first instance, of the Nuclear Advanced Manufacturing Research Centre (Nuclear AMRC), SL, National Nuclear Laboratory (NNL) and RWM was launched in July 2016 to address this challenge. It is expected that the IPT will change and grow over time as solutions are developed.

Case Study

Ongoing research & development (R&D) is underway to reduce the cost of waste containers both in the UK and internationally. This R&D will be captured and incorporated into the Waste Container IPT. The Nuclear AMRC have reviewed the manufacturing methods of numerous waste containers and transport flasks and are beginning to challenge both design and manufacturing baselines. A current project between Nuclear AMRC and SL aims to reduce the cost of manufacture of a $3m^3$ box designed for Intermediate Level Waste (ILW). This will be achieved by:

- Identifying and automating cost-effective welding techniques.
- Casting of large, complex components as opposed to fabrication.

Preliminary results suggest that there are a number of automated welding techniques that have the potential to provide cost savings during bulk manufacture. It has also been proved feasible to cast the top flange of the box as a one-off component, which presents significant cost savings during manufacture.

Pyro(hydro)lysis of Spent Ion Exchange Resins for the Disposal of Organic-Free Radioactive Waste

Dr. Hagen Gunther

As a nuclear engineering company NUKEM Technologies can support facilitating radioactive waste disposal in underground repositories by providing technical systems and relevant consulting. Actual fields to which NUKEM Technologies can contribute include design and construction of above-ground facilities, waste treatment/conditioning, waste package handling, safety assess-ments, radiation protection/monitoring etc.

To demonstrate our relevant abilities one of our most-progressive waste conditioning technologies will be elucidated – Pyro(hydro)lysis:

Internationally agreed, up-to-date waste acceptance criteria (WAC) require radioactive waste, which should be disposed of underground, to be organic-free. To obtain organic-free waste generally advisable is thermal treatment and combustion appears to be the technology of choice here. However, as combustion is usually based on oxidative incineration, high temperatures (up to 1250° C) are achieved and radionuclides like Cs-137 can become volatile. By consequence, as any filtering, afterburning or washing of the flue gas has no effect here, emission of Cs-137 would be possible.

By contrast, Pyro(hydro)lysis – adapted to nuclear technology originally by NUKEM with improvement by NGK – employs lower temperatures (max. 550° C) and thus, achieves reductive decomposition of organic compounds without release of Cs-137. "Heart" of this technology is a pyrolysis pebble-bed reactor, where eligible raw waste (e.g. spent ion exchange resins) is completely decomposed supported by dosing of steam. Applications to other waste materials are also possible.

The resulting ash is high-force compacted in case of low-activity or in-drum cemented in case of intermediate activity and ready for underground disposal as the treated waste became free from any organic compound.

Looking for rational environmentally friendly approach for development of concept for safe nuclear waste repository in preparing for future projects

Ivan Ivanov, Technical University of Sofia (TUS), Bulgaria

A sketch, preconditions and considerations for development and application of rational environmentally friendly approach addressed to development of concepts for safe nuclear waste repository will be presented. The approach is built on the basis of overview of the safety requirements of the IAEA, European and national legislation and regulations concerning spent fuel and radioactive waste (SF&RAW) management and experience gained from previous projects in the nuclear energy field with participation of Safety and Environmental Engineering Laboratory (SE&EL) of Technical University of Sofia (TUS).

Briefly some activities for SF&RAW research and management in Bulgaria also will be present as background for the considered approach with defined main steps in behalf of joint preparing for future projects and joint activities within IGD-TP.

Working Group 2 – Canister Design

Outcomes

List of presentations

- **Technical Requirements for High Level Waste Disposal Containers with regard to their Retrievability during Operation and Recovery after Repository Closure,** Teresa Orellana Perez, BAM
- **Candidate Material Solutions for the Design of Nuclear Waste Storage Canisters,** Stuart Holdsworth, Empa
- **Czech Disposal Canister Programme,** Ilona Pospiskova, SURAO
- **Experience in manufacturing of nuclear storage containers via casting, forging and machining techniques,** Michael Blackmore, SFIL
- **Implications of canister design and materials on closure welding for deep geological disposal canisters for high level nuclear waste and spent fuel,** Chris Punshon, TWI Ltd
- **Developments of the Canadian Copper Coated Used Fuel Container,** Peter Keech, NWMO

Abstracts

Technical Requirements for High Level Waste Disposal Containers with regard to their Retrievability during Operation and Recovery after Repository Closure

Teresa Orellana Perez, BAM, Germany

(No Abstract)

Candidate Material Solutions for the Design of Nuclear Waste Storage Canisters

Stuart Holdsworth, Empa, Switzerland

Consideration of candidate material solutions for the containment and storage of nuclear waste has been the topic of research on several occasions by various national authorities during the past 40-50 years. The options range from carbon steel with an adequate corrosion allowance, or outer corrosion resistant barrier, to passive metallics and ultimately to ceramics. A recent study in Switzerland has considered a range of options in terms of a number of risk assessment criteria, including: mechanical integrity, environmental damage, impact on geological barrier, robustness of lifetime prediction, fabrication and costs, and the outcome is summarised.

The advantage of such an approach is that it provides the opportunity to assess the risks associated with different design requirements relating, for example, to the type and configuration of the waste to be stored (e.g. vitrified high level waste, spent fuel rods, …), to canister fabrication constraints, to the type and character of the geological barrier to be adopted for emplacement (e.g. granite, clay, ….), and to the emplacement arrangement.

While the mechanical and very-long time corrosion properties of candidate materials are key selection criteria, verifying the feasibility and effectiveness of canister closure and inspection solutions in a hot cell environment is equally important. The remote handling and final sealing of spent fuel rod containment canisters, which are typically over 5m in length and can be in excess of 1m in diameter, is challenging.

Perhaps unsurprisingly, cost is also an important consideration; not so much the unit cost, but the prior research and development costs. For example, the unit cost of a large ceramic canister would be relatively small, but the R&D costs could be very large (>1 M€), with the need to develop: i) a solution with acceptable strength and fracture toughness properties, ii) the manufacturing capability to produce the required size of container when the overall market requirement for such large parts in this class of material is insufficient to encourage industrial investment, and iii) reliable sealing/closure and associated inspection procedures. The associated R&D costs for other strong candidate canister solutions are also high, but different, with a main difficulty being funding, in particular by industry, for such a niche (albeit strategically important) market. The only realistic solution is probably well-co-ordinated internationally funded R&D.

Carbon steel with an outer corrosion resistant barrier involving a passive metallic or possibly even a ceramic coating provides an increasingly attractive solution, with technology advances and experience feedback from other industry sectors. Some of the possibilities and their practical implications are reviewed.

Czech Disposal Canister Programme

Ilona Pospiskova, SURAO, Czech Republic

Disposal canisters fulfill one of the principal safety functions in DGR, i.e. close/contain radionuclides in the canister until a substantial portion of fission products is transformed into stable nuclides.

Following the detailed review of general research, four candidate materials were identified: carbon steel, stainless steel, Copper and a titanium alloy. Two design concepts were proposed: the first was assigned as “corrosion acceptable”, using carbon steel as an outer shell canister material and stainless steel as that of the inner shell. The second concept was termed “corrosion resistant” using copper or titanium alloy as the outer shell material and stainless steel as that of the inner shell. The canister design would be then based on material corrosion resistance and mechanical properties.

The design and the various corrosion assumptions will be confirmed in the context of the ongoing experimental programme, leading by SURAO, and by means of detailed safety assessment. The experimental programme is focused on long term material behavior under DGR conditions; both in laboratory and in-situ (underground laboratories).

Further research should extend the background knowledge, gained in activities, mentioned above, including also consideration of coating material for canister development.

Experience in manufacturing of nuclear storage containers via casting, forging and machining techniques

Michael Blackmore, Sheffield Forgemasters International Limited, UK

Sheffield Forgemasters International (SFIL) has decades of manufacturing experience in the nuclear industry delivery safety critical components. SFIL’s forged and cast product portfolio spans the operational lifetime of the nuclear fuel cycle, including primary nuclear island and reactor pressure vessels through to transportation and end of life waste disposal canisters.

Many variations in high level waste (HLW) disposal canister materials and designs are currently being explored by multiple organisations. Typically these canisters are cylindrical in shape and approximately 5000mm long with an aspect ratio in the region 5:1. This high aspect ratio poses many manufacturing challenges for designers to guarantee the fabricated canisters integrity for their extremely long design lives.

SFIL has performed manufacturing feasibility studies on cast, forged and machined canister and closure head designs in a variety of steel and iron grades. Technical limitations during machining of forged canister preforms with very high aspect ratio (>20) blind holes to depths of 4500mm have been shown and achievable manufacturing tolerances defined. Cast canister design simulations have been performed on two different iron based alloy grades. Multiple design iterations utilising two different spent fuel configurations and two different casting core materials have also been explored as a manufacturing option.

Once the spend HLW has been loaded into the canister it is sealed be welding on a closure head. Various flat, shaped and domed head designs have been assessed by SFIL and the pros, cons and a cost analysis has been carried out. This study concludes it is invaluable for the canister design organisation to identify and engage its supply chain sufficiently early, so that manufacturing limitations can be established so that a cost effective, reliable and repeatable product can be delivered to meet the strict requirements of HWL disposal canisters.

Implications of canister design and materials on closure welding for deep geological disposal canisters for high level nuclear waste and spent fuel

Chris Punshon, TWI Ltd, UK

The drivers for long term integrity of nuclear waste canisters for long term deep geological disposal are quite different to those for most engineering structures and pressure vessels. The potential failure modes and uncertainty in environmental conditions lead to consideration of different constraints being applied to the canister closure method and joint performance over time. This presentation will address the issues concerning choice of multi-barrier canister materials, canister geometry and environmental conditions on the requirements for a reliable closure welding process. Welding induced residual stresses, stress relief, weld properties and corrosion mechanisms will be addressed including suggestions for minimising risk of loss of hermeticity over the geological time periods for which the design requires that the closure weld remains intact. The presentation will be supported by a number of design case studies in which TWI has provided practical closure welding experimental evidence.

Developments of the Canadian Copper Coated Used Fuel Container

Peter Keech, NWMO, Canada

In 2011, NWMO initiated a design optimization for its Deep Geological Repository solution of nuclear waste for Canada. Efforts have been made to improve emplacement strategies, bentonite manufacturing and most noticeably, the used fuel container (UFC). The result of the UFC optimization can be seen in Figure 1.

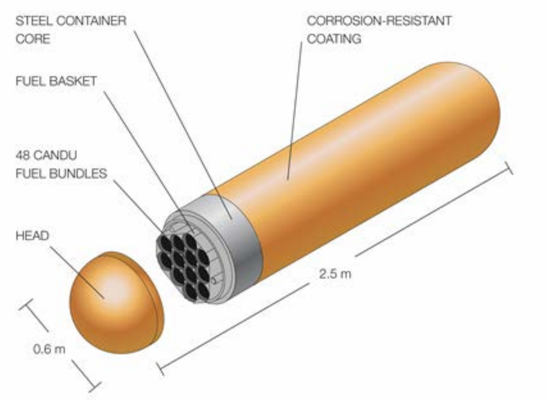

Figure 1: The Canadian Used Fuel Container

Design features include a steel body and head, fabricated from standard sized nuclear grade pipe and plate to give the container strength; an internal basket for loading the modular CANDU fuel; a hybrid laser seal weld; and a copper coating, integrally bonded to the steel container, to prevent corrosion during its long service life (i.e. > 100,000 a).

Copper coating is produced via two manufacturing strategies: electroplating of the container body and head and cold spray, following weld closure over the uncoated region visible in Figure 1. By using a copper coating vs. a freestanding copper shell, the thickness of copper can be limited to a scale similar to the conservatively calculated corrosion losses which are typically on the order of a millimetre; the NWMO reference copper coating thickness is currently 3 mm. It can also be applied directly over the welded steel, and inspected using non-destructive methods to ensure complete coverage of the vessel, prior to its entrance into service life.

During the development of copper coatings, other waste organizations such as Nagra and RWM have participated through co-funding; in the case of the former, this collaboration has included other substrate materials that may be applicable to larger or alternative canister designs (i.e. cast iron). In addition to summarizing the NWMO results, this paper will explore some strategies to enhance participation by other organizations in copper coating and welding activities that Canada has been exploring.

Electroplated copper and alternative design possibilities

Andrew McClusky, BEP Surface Technologies Ltd, UK

Copper electroplating offers some interesting design options in the design of high level waste nuclear waste canisters. These options to be discussed would include

- free standing copper canisters of literally any thickness
- coating of canisters/canister inserts to different thicknesses
- possibilities of alternative shaped canisters
- Canister sealing

BEP has been electroplating copper since 1970 and therefore has 45+ years of lessons learnt to help control the copper chemical composition and other physical properties eg grain size, hardness etc to be able to optimise the copper required for your canister. We are also able to demonstrate a highly efficient, stable, automated proven process where we have successfully manufactured over 250 similar canisters in the last 15 years.

We will explain the electroplating process and how this method of manufacture is suitable for nuclear canisters

We will discuss what an electroplating plant might look like, given the different canister designs.

We will discuss how known engineering difficulties such as “the gap” and how they may be overcome

Overview of Cold Spray Coating Technology

Heidi Lovelock, TWI Ltd, UK

Cold spray is a solid state deposition process that can deposit thick (several mm), dense, oxide-free layers of materials that are of interest to a number of nuclear waste storage strategies; for example, Cu, Al-B4C, Fe, Ni, Ti, Ta, amorphous materials, etc. The presentation will review the advantages and limitations of cold spray deposition, with specific reference to its potential use in nuclear waste containers.

Mechanical-Corrosion Effects on the Durability of High Level Waste and Spent Fuel Disposal Containers

Cristiano Padovani, Radioactive Waste Management, UK

The design of waste containers for the disposal of High Level Waste (HLW) and spent fuel needs to take into account the chemical and mechanical stressors likely to be present in a geological disposal facility (GDF). In some circumstances, the interaction of chemical and mechanical processes is expected to be limited, so that waste containers can be designed to ensure they are able to withstand mechanical loads taking into account a simple corrosion allowance. In other circumstances, however, coupling between mechanical and corrosion processes is expected (particularly in the case of effects associated with hydrogen embrittlement) and the evaluation of the integrity of a waste container can become more complex. A study of the degree of coupling expected in a variety of conceptual container designs featuring a single-shell carbon steel container with and without a copper or titanium coating/cladding was carried out making use of Failure Assessment Diagrams (FAD). The study considered a variety of disposal scenarios (different types of host rock and buffer materials), thus providing a broad analysis. The results of this study can be used to evaluate the nature and expected importance of corrosion/mechanical interactions in a variety of situations. This work is therefore of relevance to a variety of national disposal programmes, particularly those currently at a conceptual design stage, as it helps the development of more detailed/mature container designs.

Sharing information on container materials for High Level Waste and Spent Nuclear Fuel (IGD-TP activity JA11a)

Cristiano Padovani, Radioactive Waste Management, UK

(No Abstract)

Working Group 3 – High temperature clay interactions

Progress in understanding up to 100°C

Effects of elevated temperatures on the EBS in clay formations

Thorsten Schäfer, KIT-INE, Germany

The temperature upon which thermodynamic data used for performance assessment purposes is usually 25°C. However, in the near field environment of a spent nuclear fuel or vitrified high level waste repository the temperature will remain elevated for a long time period. Thermal calculations (Johnson et al., 2002) have shown that the temperature in the bentonite backfill of a repository in a clay formation may attain 70–90°C under saturated conditions for 1000 years after repository closure. According to these calculations temperature then decreases to 40– 50°C after about 10000 years, which is the expected life span for the canisters. For the Swedish KBS-3 concept temperatures for the bentonite backfill are calculated to reach up to 80°C at the inner boundary (Ageskok and Janson, 1999). After 1000 years temperatures are assumed to decrease to 40–55°C.

Safety relevant effects of a thermal pulse at the canister / bentonite interface, in bentonite backfilling and in argillaceous host rock are mainly related to radionuclide retention, evolution of hydrogen production and fracturing, as well as reactions of organic materials and pore water sulfate at elevated temperatures. Since clay mineral interactions with ferrous iron are expected at the canister / bentonite interface, radionuclide retention to altered clay minerals and ferrous secondary phases for elevated temperatures have to be considered, in particular for early canister failure scenarios. Cronstedtite and odinite are the two Fe-rich 1:1 phyllosilicates expected to form at the canister / bentonite interface. No parental link has been evidenced between the two minerals despite their contrasting crystal morphologies and thermodynamic predictions (Lanson et. al., 2012). Moreover, a thermal pulse may influence the coupled evolution of abiotic / biotic degradation and maturation of organic matter, sulfate reduction and canister corrosion. Hydrogen production and thermal expansion, as well as the secondary mineral phase formation/precipitation in the pore space may cause a build-up of pore pressure and fracturing in the argillaceous barriers. With respect to high temperatures, there is a lack of knowledge on radionuclide retention in argillaceous systems of elevated ionic strength (> 1 mol/L), as in the Lower Cretaceous clays in Northwest Europe and in shales of the Canadian Shield.

Ageskok, L. Janson, P. 1999. Heat propagation in and around the deep repository. Thermal calculations applied to tree hypothetical sites: Aberg, Bberg and Cberg. Tr-99-02, SKB Technical Report, Stockholm, Sweden.

Johnson, J.W., Niemeyer, M., Klubergans, G., Siegel, P., Gribi, P., 2002. Calculations of the temperature evolution of a repository for spent fuel, vitrified high-level waste and intermediate level waste in Opalinus clay. NTB 01–04. Nagra, Wettingen.

Lanson B., Lantenois S., van Aken P., Bauer A., Plançon A. , 2012. Experimental investigation of smectite interaction with metal iron at 80 °C: Structural characterization of newly formed Fe-rich phyllosilicates. American Mineralogist, Volume 97, pages 864–871, 2012

Clay interactions at high temperature by molecular dynamics, thermodynamic modelling and lab analysis

Markus Olin, VTT, Finland

Clays have been intensively investigated during last decades for many application and therefore, many models, analysis methods and experimental setups of scales varying from lab to full scale are available. However, it has been not so common to carry out these studies at other than room temperature, while it is well known that temperature changes both the equilibrium and non-equilibrium thermodynamics (kinetics).

However, it is assumed to be beneficial, if higher (common upper limit 100 oC) temperatures are allowed for buffer clay materials in high level waste repositories. Therefore, it is essential to study clay materials at these higher temperatures, and if the conservative assumptions behind the application of lower temperature can be released, it is possible to obtain those, mainly economic, benefits in high level repositories.

We have studied clays as a function of temperature both experimentally and by modelling.

The structural changes of water saturated bentonite or montmorillonite at high temperature can be monitored by x-ray scattering methods and nuclear magnetic resonance. SAXS (or WAXS or XRD) allows monitoring the basal spacing as the function of the temperature (figure) and NMR can show changes in the behaviour of pore water in connection with the structural changes (the relaxation signal is depending on water mobility and pore sizes).

Atomic level modelling has been demonstrated to give reliable estimates of smectite swelling pressures for various compositions at standard temperature and external pressure.[1] Swelling pressure and structural changes at elevated temperatures are proposed to be studied with molecular dynamics methods. The outcome of the simulations will be useful in the evaluation of clay interactions and pressure behaviour at high temperature.

We propose studies of clay as function of temperature consisting of

1) Molecular level modelling

a) Swelling (distance between layers)

b) Swelling pressure

2) Structural experiments

a) Distance between layers by SAXS

b) Fraction of bound water by NMR

3) Thermodynamic parametrisation

a) Model parameter dependence

The main impacts from these studies much better understanding of the clay behaviour as a function of temperature, and the results combine swelling, structural, chemical and mechanical behaviour. Together with other studies long term behaviour of clays at high temperature may be better evaluated in final disposal conditions.

[1] Sun, L et al. J.Phys.Chem C 119 (2015) 19863.

Progress in understanding above 100°C

Research topics from the state of the art on THMCB aspects of thermal compatibility of clays

Artur Meleshyn, GRS, Germany

A literature survey of research on thermally-induced changes on clay properties reveals that temperatures up to 150°C can be expected not to compromise the integrity of the geotechnical or geological clay barriers, but rather to contribute to their improved performance. This is because of – amongst other positive effects – (i) the increase of the consolidation of the clay and (ii) the sterilization of the repository's near-field and thus inhibition of detrimental microbial activity. Still, quantitative characteristics of clay performance at temperatures >100°C and especially ≥150°C, which are required for performance assessment modeling and understanding of the disposal system evolution, are very scarce. We discuss on the following topics that in our view have high priority with respect to the needed experimental research: (i) dependence of solution and gas permeabilities, gas entry pressure, swelling pressure, sorption capacity and gas production on the temperature and duration of thermal treatment for different bentonites and crushed clays, (ii) determination of the temperature of expansion-contraction threshold in compacted clay materials (iii) changes in the osmotic swellability of bentonite due to interaction with water vapor at 100-180°C, (iv) temporal and spatial quantification of the thermally induced drainage and of the influence of inhomogeneities at in-situ scale as supported by THM modeling, (v) quantification of thermally induced decline in detrimental microbial effects.

Coupled THMC interactions at temperature beyond 130 °C : experimental investigations and modeling of materials alteration, swelling processes and overpressure build-up

Francis Claret, BRGM, France

In their critical review dealing with the performance of bentonite barrier as a function of temperatures, Wersin et al. (2007) point out the lack of reliable information at temperatures beyond 130°C regarding hydraulic, mechanical and mineralogical changes. While the thermal perturbation will have a limited impact on clay mineralogy (illitisation, chloritisation, cementation effects, et cetera), our prediction of its impact on swelling pressure, clay dehydration and hydraulic conductivity still lacks a comprehensive and consistent understanding of the processes at stack (Couture, 1985; Villar et al., 2010). Recent advances in the development of oedometer cell coupled to microstructure visualization (Gaboreau et al., 2016; Massat et al., 2016) can be used to give a better insight on bentonite swelling capacity as a function of temperature as well as on the overpressure development and its mechanical impact on the bentonite macroscopic behaviour. Regarding overpressure, the contribution of thermo-osmotic processes is also foreseen (Tremosa et al., 2012), the lack of accurate calibration data currently prevents any definitive conclusion on the possible contribution of this process to overpressure during the transient state associated with temperature decrease after reaching the “peak” of temperature. New experiments coupled to innovative modelling approaches could be proposed to circumvent this uncertainty. At last but not least in some concept, the canister is not surrounded by bentonite but by cementitious materials in order to reduce the corrosion rate. Hydraulic and mineralogical changes of these cementitious materials at elevated temperatures could be detrimental to the safety of radioactive waste disposal, and they certainly need to be investigated.

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Tremosa, J., Goncalves, J., Matray, J.-M., 2012. Natural conditions for more limited osmotic abnormal fluid pressures in sedimentary basins. Water Resour Res 48.

Villar, M.V., Gómez-Espina, R., Lloret, A., 2010. Experimental investigation into temperature effect on hydro-mechanical behaviours of bentonite. Journal of Rock Mechanics and Geotechnical Engineering 2, 71-78.

Wersin, P., Johnson, L.H., McKinley, I.G., 2007. Performance of the bentonite barrier at temperatures beyond 100 °C: A critical review. Physics and Chemistry of the Earth, Parts A/B/C 32, 780-788.

Contribution to numerical modelling of high temperature clay interactions

Javier Samper, Universidad de a Coruña, Spain

The maximum temperature in the buffer of HLW/SF repositories has been often limited to 100 ºC to minimize potentially harmful chemical processes such as illitization and cementation. The assessment of the impact of increased temperatures on the buffer performance and properties requires the scientific understanding of the underlying processes at higher temperatures and the use of well-established and verified THM and THC numerical models and codes.

The UDC group has expertise on the scientific knowledge and the development of coupled hydrodynamic (H), thermal (T) and chemical (C) numerical models and codes for the engineered barrier system such as INVERSE-FADES-CORE. These models have been applied and tested within the framework of the FEBEX, NFPRO, PEBS, and Ventilation Experiment Projects. The contributions of our group to the Working Group 1 include: 1) The use of the coupled THCM models for the design stage of the HotBENT in situ test (dimensions, location of sensors), 2) The interpretation and modeling of the experimental data; and 3) The benchmarking of coupled THCM codes. These contributions will require the implementation of some extensions on the existing THCM capabilities of INVERSE-FADES-CORE to deal with high temperatures and very large pore water chemical concentrations.

Assessing THM consequences in clay host rocks for high T

Thermo-Hydro-Mechanical Effects of a Geological Repository at Macro-Scale – A Novel Modeling Approach Adapted from Plate Tectonics

Thomas Kämpfer, AF-Consult Switzerland, Switzerland

Highly radioactive waste in a geological repository emits heat that may affect the near and far field of the repository. Besides the pure thermal impact on e.g., material properties, heat diffusion into the host rock leads to thermal expansion of the solid skeleton and porous water. In host rocks with very low permeability and given the higher thermal expansion of water than of the solid skeleton, this implies a pressure build-up in the pore water that affects the effective stresses acting on the skeleton. Depending on the boundary conditions and in particular because of the free surface at the atmosphere, parts of the skeleton will be under traction and a risk of fracturing may result. On the one hand, such thermo-hydro-mechanical (THM) effects and optimal modeling approaches have been studied for a repository context quite extensively, namely within the European DECOVALEX project series. Nevertheless, successful modeling across the widespread relevant temporal and spatial scales remains a formidable challenge. On the other hand, similar THM modeling challenges appear when studying the dynamics of water- or melt-bearing rocks in natural geological processes as e.g., in subduction zones. In this plate tectonic setting, water- and melt-induced THM phenomena are, in particular, responsible for localized shearing of rocks, hydrofracturing and seismicity triggering. Currently, advanced multi-scale THM modeling rapidly evolves in basic Earth sciences (such as computational geodynamics and plate tectonics), which creates favorable conditions for inter-disciplinary exchange with applied THM modeling fields. In the course of a small project within the Marie Curie Innovative Training Network, we successfully adapted a novel, massively parallel staggered finite-difference, marker in cell modeling approach from plate tectonics simulations to the THM physics of a geological repository in clay host rock. The approach allows for the consideration of an appropriate viscous-elastic-brittle/plastic rock rheology, considers compressibility of both solid and liquid, and is particularly innovative with respect to the fully implicit numerical solution scheme allowing to resolve both short-term and long-term deformation. The achieved 2D proof of concept (Figure 1) indicates that the relevant phenomena are appropriately modeled over the desired scales in space and time and this within very reasonable computation times. The next step will consist in extending the approach to 3D and its quantitative validation through benchmarking tests and comparison with laboratory and field experiments.

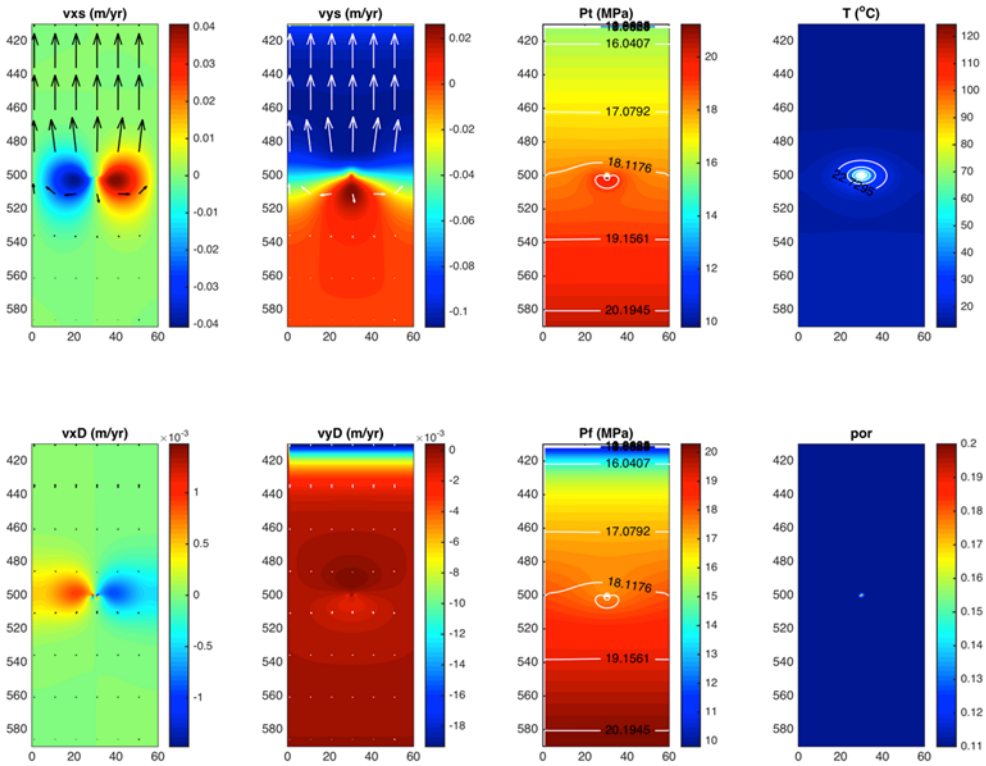

Figure 2: Qualitative results showing solid and liquid velocities, pressures, temperature and porosity near a generic emplacement cell.

In situ experiments regarding high T

Hot Mock-ups at Joseph URL

Jiří Svoboda, CTU, Czech Republic

Although most of the repository concepts envisages temperature in buffer bellow 100°C temperatures above this limit could bring some benefits namely higher capacity of deposition place which brings higher cost efficiency.

Going above 100°C however brings some challenges such as possible buffer degradation, water boiling/steam generation and sauna effect. These effects have been studied to some extend in the laboratory but not very much in-situ.

The CTU in Prague in partnership with SURAO are therefore in process of commissioning of two new in-situ hot mock-ups with target temperature 200°C. Two vertical mock-ups of deposition place in DGR are being built in Josef URL and they will be used to study the effect of elevated temperature and associated processes in repository like conditions on both - MX-80 (Na type) and Czech bentonite (Ca-Mg type).

The proposed presentation introduces these two mock-ups and associated scientific program.

HotBENT - High Temperature Bentonite Project Ideas

Florian Kober, NAGRA, Switzerland

A number of large in-situ heating experiments have been run for testing the performance of concepts for geological disposal of nuclear waste that consider the use of bentonite as buffer material between waste and hostrock, commonly exposing the bentonite to early disposal temperatures of ~100-130°C. Those experiments provide valuable information on the performance of various bentonites. Some repository concepts under development are considering exposure of the bentonite to temperatures of up to 150°C and beyond. Based on different studies the safety relevant properties of bentonite such as the swelling pressure or the hydraulic conductivity and its overall performance are well known for a temperature range up to 130°C only. However, the performance and parameters for higher temperatures, such as the physico-chemical response of materials (changes in mineral chemistry, e.g. the transformation of smectite to illite and the related decreased swelling capability, changed mechanical properties, etc.) are only known from small-scale lab studies. Additionally, the impact on the performance of bentonite at higher temperatures with respect to complex moisture transport processes and bentonite-rock interactions affecting early disposal system behavior under natural conditions (pore pressure build-ups, two-phase flow, bentonite saturation etc.) are yet not well understood. Temperatures of 150°C or higher in the bentonite may cause thermal drying and cracking of the near field and the impact on the evolution and the properties of the excavation damaged zone (EDZ) are less known. Numerical models for thermal coupled processes in natural and engineered barriers are not tested for such conditions.

A large-scale long-term experiment focusing on the behavior of bentonite-based barriers under very high temperatures will enable to evaluate different “hot” designs and how they can be optimized in terms of efficient implementation, disposal footprint, while ensuring the long-term safety of the repository system. The Grimsel Test Site (GTS) is bringing interests together from those programmes considering high temperature concepts thereby enhancing the bentonite understanding towards higher temperatures for the whole community interested in using bentonite as an engineered barrier.

The group of interested organizations proposes a HotBENT experiment which aims at increasing the data base on buffer / host-rock performance under high temperatures (up to 200°C and realistic scales/conditions). Current design discussions consider various bentonites, heater-materials and concrete-liner parts, as well as natural and/or artificial (saline) saturation. Such an experiment should be accompanied by laboratory investigations, scaled mock-up tests and should integrate modelling activities. The experimental design should allow for a first set of data and samples within 5 years or so, while certain parts of the setup can also allow for a longer duration. A sufficient simple monitoring system would ensure that samples can be taken at positions with significant temperature and saturation gradients.

Working Group 4 – Spent fuel characterization

Presentation on behalf of SNETP Nuclear Sustainable Platform

David Hambley, National Nuclear Laboratory, UK

Developments in Fuel Technology

Current Reactor Fuels

Advanced Fuels

Other Fuels

Integrating the Back-end for Fuel

SNETP : IGD-TP Interface

Spent fuel characterization with emphasis on Russian type reactors

Jitka Miksova, CV REZ, Czech Republic

Miksova Jitka 1, Nachmilner Lumir 1, Lysychenko Georgii 2, Dudar Tamara 3, Sayenko Sergii 2, Krasnorutskii Vladimir 2, Baltrunas Dalis 4

In this contribution a joint effort of the Research Centre REZ, Czech Republic (1), Department of Nuclear Physics and Energy of National Academy of Sciences of Ukraine (namely National scientific center «Kharkov physics – technical institute»)(2) and Institute of Environmental Geochemistry(3), Center for Physical Sciences and Technology, Lithuania (4) will be presented.

Intended research focused on spent nuclear fuel (SNF) characterization could support geological repository development in EU Member States or other countries using Russian type of reactors, mainly VVER 1000.

The main goal of the proposed research is the validation of modelling results by measuring radiochemical and thermal characteristics of SNF from different Nuclear Power Plants (NPP), mainly VVER-1000 reactor type of III generation (depending of SF samples accessibility) so that to more preciously determine the inventory and decay power of materials to be disposed of. The idea is to benchmark and standardize SNF characterization methods using both mathematical modelling and experimental measurement of key parameters. It is also intended to assess performance of irradiated cladding materials under repository conditions, in particular corrosion of Zr-alloy cladding in contact with simulated deep repository aqueous environment under a range of conditions

The team has modelling capacities for calculating isotopic composition and thermal output of spent nuclear fuel based on its history (SCALE, MCNP, SERPENT) and hot cells and radiochemical laboratories for experiments with high activity materials (irradiated cladding, SNF). The characterization methods to be employed involve a new system for measuring neutrons and gammas (Atoll -1M), calorimetry, a dosimetric system, autoclave, in-situ electrochemistry, surface analysis and metallographic equipment, optical and transmission electron microscopy, etc.

The additional aim of this proposed joint effort is to involve new Member states with less developed disposal programmes into European RD&D activities and enable them to share scientific work/capacities with colleagues from advanced programmes. This approach could help to increase new MS RD&D capabilities.

The presented work was financially supported by the Ministry of Education, Youth and Sport Czech Republic Project LQ1603 (Research for SUSEN). This work has been realized within the SUSEN Project (established in the framework of the European Regional Development Fund (ERDF) in project CZ.1.05/2.1.00/03.0108).

Thermal modelling of high heat generating wastes in the UK programme

Robert Winsley, Radioactive Waste Management, UK

The design of a geological disposal facility (GDF) for High level Waste and spent fuel needs to take into account the heat generated by these materials at the time in which they will be emplaced in disposal tunnels. In disposal concepts developed for different types of host rock, temperature limits in the buffer and/or host rock are typically identified to ensure that thermal alteration processes do not negatively affect the long-term safety of a disposal facility. In disposal concepts featuring the use of bentonite clays, limits of the order of 100-150°C are typically envisaged for the buffer. In concepts making use of a cement buffers, thermal limits may also be present. These limits can present substantial challenges to the timescales over which wastes (particularly spent fuels) may be accepted into a disposal facility and, once the waste has sufficiently cooled down to be acceptable in a GDF, can determine the spacing of deposition tunnels, the timing of backfilling and closure operations, and the size of the overall GDF. The impact of heat on the disposal schedule and size of a GDF has major strategic and cost implications for a disposal programme, particularly when considering disposal of spent MOX fuels and of spent fuels from new power stations that may be built in the future (likely to be of higher enrichment and burn-up).

Currently, the heat generated by spent fuels is evaluated on the basis of radioactive decay and associated thermal models under specific assumption (particularly fuel burn-up). Such models, however, have not been validated experimentally, leading to uncertainty on their level of conservativism (or optimism). This paper will present the findings of a project recently completed in the UK programme (outputs of Integrated Project Team on High Heat Generating Wastes), highlighting the need of experimentally characterising the heat generated by a real population of spent fuels, which is a key factor required to underpin the engineering and future optimisation of a GDF. Considerations related to the need of ensuring both radiation shielding and a subcritical condition within disposal containers - characteristics which are strongly related to, and can be measured jointly with, the heat generated by spent

Development of depletion models for radionuclide inventory, decay heat and source term estimation in discharged fuel

Stefano Caruso, Nagra, Switzerland

The capability to reliably predict the spent nuclear fuel composition in terms of radionuclide inventory, elemental content, decay heat and radiation source term is relevant for both operational and long term safety assessment as well as for disposal cost key factors. The uncertainty of the fuel inventory can be dominated by several components: irradiation history of the fuel, exact composition of the fresh fuel (especially the level of impurities), and the large heterogeneities of the fuel design being used as well as code modeling limitations and reactor physics code characteristics themselves. However, the availability of experimental data allows to test and validate codes and models in order to be able to simulate the irradiation conditions quite close to the reality, or , at least, to assess a solid baseline as start point for further uncertainty assessments. If the baseline (i.e. code and fuel model) is known, being properly characterized by a well determined uncertainty, further assumptions can be analyzed and incorporated, such to encompass in a conservative direction all variables, but avoiding to bring the final estimates to a level of over-conservativism.

The methodology used at Nagra for fuel depletion and cladding activation calculations is here illustrated. A number of LWR fuel assembly designs, coming from the Swiss reactors, have been modeled employing the SCALE/Triton sequence and validated against available experimental data. Main results from Beznau, Gösgen (both PWR) and Leibstadt (BWR) reactors are presented here and discussed. Part of the work performed was used to successfully support the Swiss NPPs for their transport/storage cask license application.

Authors: S. Caruso, A. Shama, M. M. Gutierrez

Spent fuel online monitoring: opportunities for new technologies

Paola Finocchiaro, INFN, Italy

New technological developments in radiation sensors and electronics allow to set up arrays of low-cost detectors around the canisters for online continuous monitoring of radiation levels (gamma and neutrons, primarily). Following the full characterization of each canister these sensors placed around it would act as an active electronic seal, warranting the integrity of the canister itself and of its content. Any subsequent detected anomaly in the distribution of gamma and/or neutron counting rate would be an indication of a change in the status of the canister and/or of its content. In some respect this is similar to the CRC control code assuring the data integrity in IT systems.

Such a solution would also provide an important contribution in the direction of the social acceptability, as it allows a better transparency of data and procedures around radwaste and spent fuel management.

The proposed SPIRE Project: challenge, scope, impact

Peter Jansson, Uppsala University, Sweden

Time is progressing towards 2025 when the vision of IGD-TP is to have an operating geological disposal facility in Europe, only nine years remains. As mature national programs are developing details on what is needed regarding characterisation of spent nuclear fuel, i.e. determining source terms to decay heat, criticality or dose rates, it becomes more evident that the uncertainties associated with such parameters and the underlying knowledge of source terms have large impacts on economy and safety of interim storage and deep geological facilities. With the Spent fuel characterization Program for the Implementation of (geological) Repositories (SPIRE), we try to address remaining issues for spent fuel characterization in relation to its storage, both short term and long term. We attack the operators' challenge both from a measurement and simulation point of view, including development of both old a new measurement techniques, enhancing the precision and reliability with which parameters of the fuel can be calculated and perform validation and verification efforts in this context. We believe that spent fuel characterization is an important contribution to enabling the vision of the IGD-TP.

The proposed SPIRE Project: measurement techniques applied on SNF

Henrik Widestrand, Vattenfall AB, Sweden

Within the proposed SPIRE project both measurements and simulations will be performed for validation and verifications of the employed methods. A range of spent fuel are proposed to be measured in order to provide as broad a data set as possible for comparison with models and hence enable uncertainty quantification. Measurements will be made on LWR fuels in form of fuel assemblies and well-characterised fuel samples. The results will be used for validation of prediction of the decay heat, neutron and gamma-ray emission, inventory and reactivity. By reducing the uncertainties associated with determination of the mentioned parameters we believe that SPIRE can offer unique data and an important contribution for fuel characterization and thus improved possibilities for the European Community to reach the vision of an operating geological disposal facility in Europe by 2025. The presentation will give an overview of the spent fuel measurements in SPIRE.

SF inventory calculations in SPIRE project: current limits and expected improvements

Dimitri Rochman, Paul Scherrer Institute, Switzerland

The SPIRE project will address some issues related to the spent fuel characterization in the framework of implementations of repositories. Both measurements and simulations will be performed for validation and verifications of the employed methods. As an example of the existing challenges and limitations of the simulations capabilities, differences between measurements (E) and calculations (C) for specific fuel inventories will be presented with precise estimations of calculation uncertainties. These "C/E" values with uncertainties are related to penalty factors and cost increase, and should therefore be close to 1. Possible computational solutions proposed in SPIRE will be sketched in this presentation.

Safety of extended interim dry storage of spent nuclear fuel (SAEXFUEL)

Joaquin Cobos, CIEMAT, Spain

This project proposes to develop cooperative research on interim storage of spent fuel and other high level radioactive wastes to contribute to the development of common fundamental approaches for the safe management of the predisposal activities for the spent fuel, focusing on its storage during extended periods of several decades. A number of Member States are at present finalizing their preparation of the predisposal activities including the selection and detailed design of an interim storage system that should be able to safely preserve the spent fuel for extended time interval (more than 100 years), until a suitable final disposal facility is available or there is a decision to recycle these materials in new reactors. Different conditions define the technological choices favoured by each Member State; several of these spent fuel management technologies require R&D to optimize solutions and demonstrate that they are safe, all the way from the transport from the reactor till the recuperation and repackaging in the way to their final destination, and that the design of the storage systems allow to perform safely all the required operations.

The technological choices and conditions are different in different member States, but in all cases the optimization and safety validation raises a number of challenges that require significant R&D effort, particularly on the behaviour of spent fuel stored under dry conditions for extended period of times. This project intends to develop coordinated research and share efforts from organizations around Europe, taking into account the different reactor technologies, fuel cycles and stages of implementation of predisposal actions implemented in different Members States. The goals include integrating this R&D with the plans of the nuclear waste management organizations and the corresponding European Technological Platforms. In particular, the project is designed to be complementary to the ongoing research on deep geological disposal that is and has been the subject of research projects included in national and the EURATOM Framework Programmes.

Appendix A Useful Links

Websites:

IGD-TP www.igdtp.eu

JOPRAD www.joprad.eu

SITEX www.sitexproject.eu

SNETP www.snetp.eu

Cebama www.cebama.eu

DOPAS www.posiva.fi/dopas

FIRST-Nuclide www.firstnuclides.eu

MIND www.mind15.eu

Modern2020 www.modern2020.eu

PEBS www.pebs-eu.de

IGSC - NEA www.oecd-nea.org/rwm/igsc/

MELODI http://melodi-online.eu/

EC DG Research & Innovation: http://ec.europa.eu/research/

Appendix B List of participants

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