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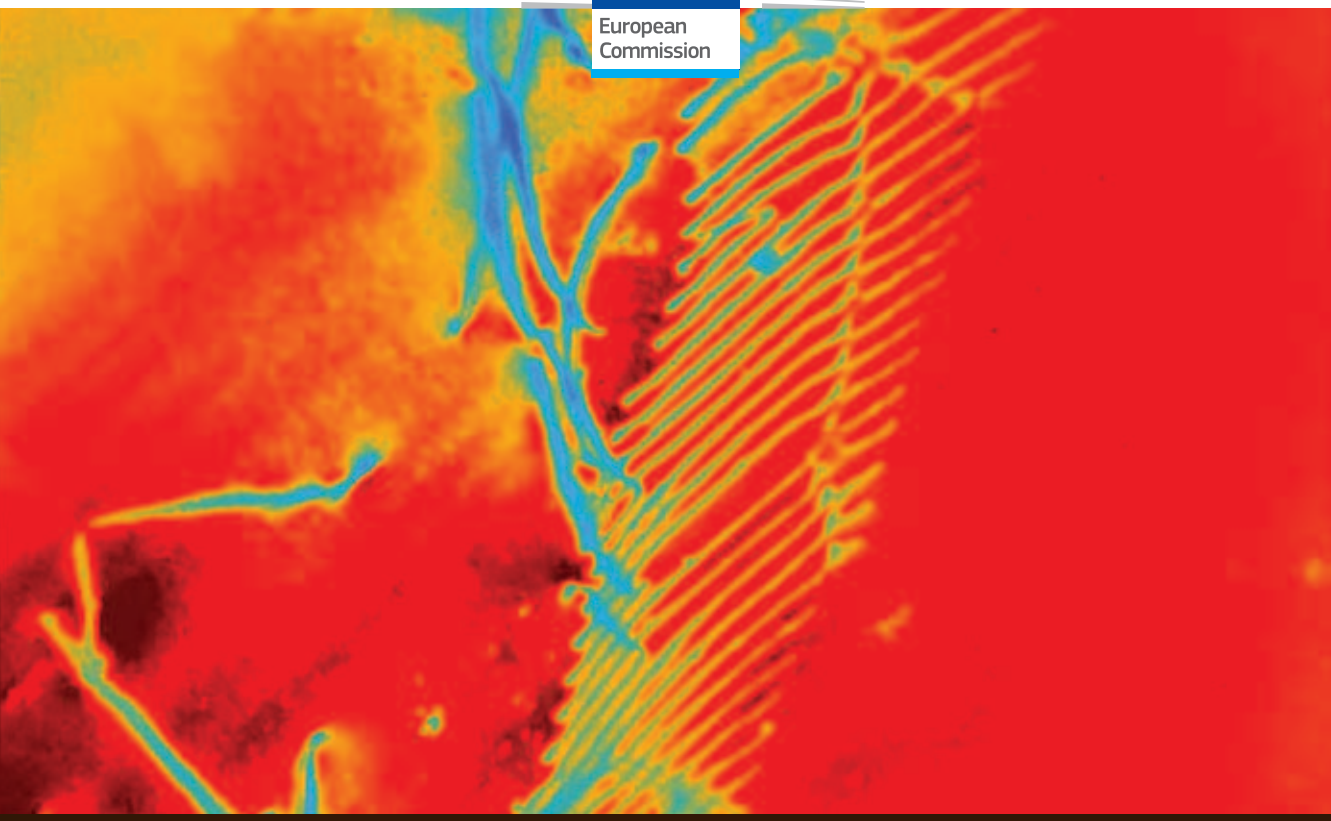

PROGRAMME & ABSTRACTS

EURADWASTE '13

8th EC Conference on the Management of Radioactive Waste Community Policy and Research on Disposal

14-17 October 2013
Vilnius, Lithuania

Co-organised by the European Commission and the Lithuanian EU Presidency 2013

Research and Innovation

EUROPEAN COMMISSION

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EUROPEAN COMMISSION

Directorate K - Energy
Unit K.4 - Fission

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2013

Directorate-General for Research and Innovation
Nuclear Fission

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BACKGROUND

Since 1975 the European Commission (EC) has been implementing, under the provisions of the Euratom Treaty, European Union (EU) research and training programmes on radioactive waste management. A key principle of these programmes remains the support for EU collaborative activities in implementation of safe and publicly acceptable solutions in the processing and disposal of radioactive waste.

While near-surface disposal of short-lived low and intermediate level waste has reached industrial maturity, implementation of geological disposal of long-lived high-level waste and spent fuel is not yet as far. Nevertheless, there is world-wide scientific and technical consensus that deep geological disposal represents the safest and most sustainable solution as the end point of the management of high-level waste and spent fuel considered as waste. Furthermore, knowledge on fundamental processes of confinement and isolation of radioactive waste in suitable geological repository environments is now very mature. The 7th Euratom Framework Programme (FP7, 2007-13) has therefore focussed its support on all remaining key scientific and technological aspects required for actual implementation of such repositories. The objectives of the programme have been the continuing development of a common European understanding of the scientific/ technical issues, the establishment of common research priorities, and support for joint implementation of related research and coordination activities.

In all these areas, FP7 has seen major progress. One notable success has been the establishment of the Implementing Geological Disposal Technology Platform (IGD-TP), piloted by leading European radioactive waste management organisations. This has enabled FP7 to focus support for research, development and demonstration (RD&D) activities on the realisation of the current vision of the platform: “By 2025, the first geological disposal facilities for spent fuel, high-level waste and other long-lived radioactive waste will be operating safely in Europe.”

In addition, depending on the energy mix choice of EU Member States (MS) and in view of the potential future role of nuclear energy in the EU energy landscape, as presented in the EU 2020 Energy Strategy as well as the EU Energy Roadmap 2050 scenarios for reduction of greenhouse gas emissions by 80-95%, the successful implementation of geological disposal is an important prerequisite for socio-political acceptance and support.

In July 2011 a new legislation establishing a framework for the responsible and safe management of spent fuel and radioactive waste was adopted in the EU, namely the Council Directive 2011/70/Euratom. Each EU Member State remains free to define its fuel cycle policy – the spent fuel can be regarded either as a valuable resource that may be reprocessed or as radioactive waste that is destined for direct disposal. Whatever option is chosen, the disposal in geological repositories of high-level waste resulting from spent fuel reprocessing, or of spent fuel regarded as waste should be considered. The storage of radioactive waste, including long-term storage, is an interim solution, but not an alternative to disposal. As per the Directive, each EU Member State has to establish, maintain and implement national policy, framework and programme for the management of spent fuel and/or radioactive waste from generation to disposal. Strong provisions are foreseen for ensuring a high level of safety, including in the long term, avoiding undue burdens on future generations and enhancing transparency. Member States will invite international peer reviews with the aim of ensuring that high standards are achieved.

Objectives and Scope

EURADWASTE'13 will be organized in parallel with FISA 2013 under the auspices of the Lithuanian EU Council Presidency 2013. A common introduction and closure to the two conferences will be organised to set the scene at international and EU levels and to obtain a synthetic view of issues, policies and research and implementation programmes in both reactor systems/ safety and radioactive waste geological disposal.

A joint FISA and EURADWASTE poster session will be held in the middle of the two conferences. For the first time in the history of the Euratom programmes the two reactor and waste communities will be able to interact directly in presenting their respective activities and achievements in a large and joint event. The objective is to enhance the common understanding and synergies on common issues and prepare for future activities.

The objective of EURADWASTE'13 is to present an overview of recent EC activities in the field of radioactive waste management with focus on geological disposal (GD) as the main challenge, including RD&D, policy, strategic, legislative and socio-political aspects.

The specific EURADWASTE objectives are:

- to address the main aspects of the European policy for the responsible and safe management of spent fuel and radioactive waste and its implementation at both EU and national level;
- to discuss key challenges in geological disposal programmes;
- to present the achievements of more than 20 Euratom 7th Framework Programme research projects in science and technology for radioactive waste geological disposal;
- to achieve consensus views on the current state-of-the-art and to determine future priorities at European level;
- to enhance the transparency in the decision-making process regarding spent fuel and radioactive waste management.

Overview of sessions

The joint opening of EURADWASTE'13 and FISA 2013 conferences will be held on 14 October morning, see separate overview programme.

The objective of this session is to present an overview of the EU energy policy and the contribution of the EC nuclear fission research. The session will include two keynote presentations on international perspectives in the field of reactor systems/ safety and the management of spent fuel and radioactive waste.

EURADWASTE'13 will continue with five sessions. Sessions I & II are dedicated to policy aspects of the responsible and safe management of spent fuel and radioactive waste, followed by session III bridging policy to research and implementation of radioactive waste disposal. There will be further two sessions (IV & V) on the Euratom 7th Framework Programme research projects and results. Session IV will address the challenges in science for disposal whereas session V will focus on the challenges in repository technologies and construction.

Both conferences will be closed jointly on 16 October afternoon, including concluding FP7, outlining future priorities, opening up to the Horizon 2020 Framework Programme for research and innovation. FISA and EURADWASTE will be concluded by a panel and three keynotes to consider the role of research in the safety and radioactive waste frameworks seen from the point of view of the technology platforms and regulatory stand point, see separate overview programme.

A technical visit will take place on 17 October.

Monday, 14th October

Joint introduction – EURADWASTE '13 and FISA 2013

EURADWASTE PROGRAMME

Session I: Responsible and safe management of SF and RW
Session II: Decision making on SF and RW management

Tuesday, 15th October

Session III: Challenges in Geological Disposal: from policy to research and implementation
Session IV: EURATOM FP7 – Challenges in Science for Disposal: advances in phenomenology understanding of the source term & migration and performance assessment for the safety case

Tuesday, 15th October, evening

Poster session: Presentation of generic and detailed posters from the FISA reactor systems and safety and the EURADWASTE radioactive waste disposal communities.

Wednesday, 16th October

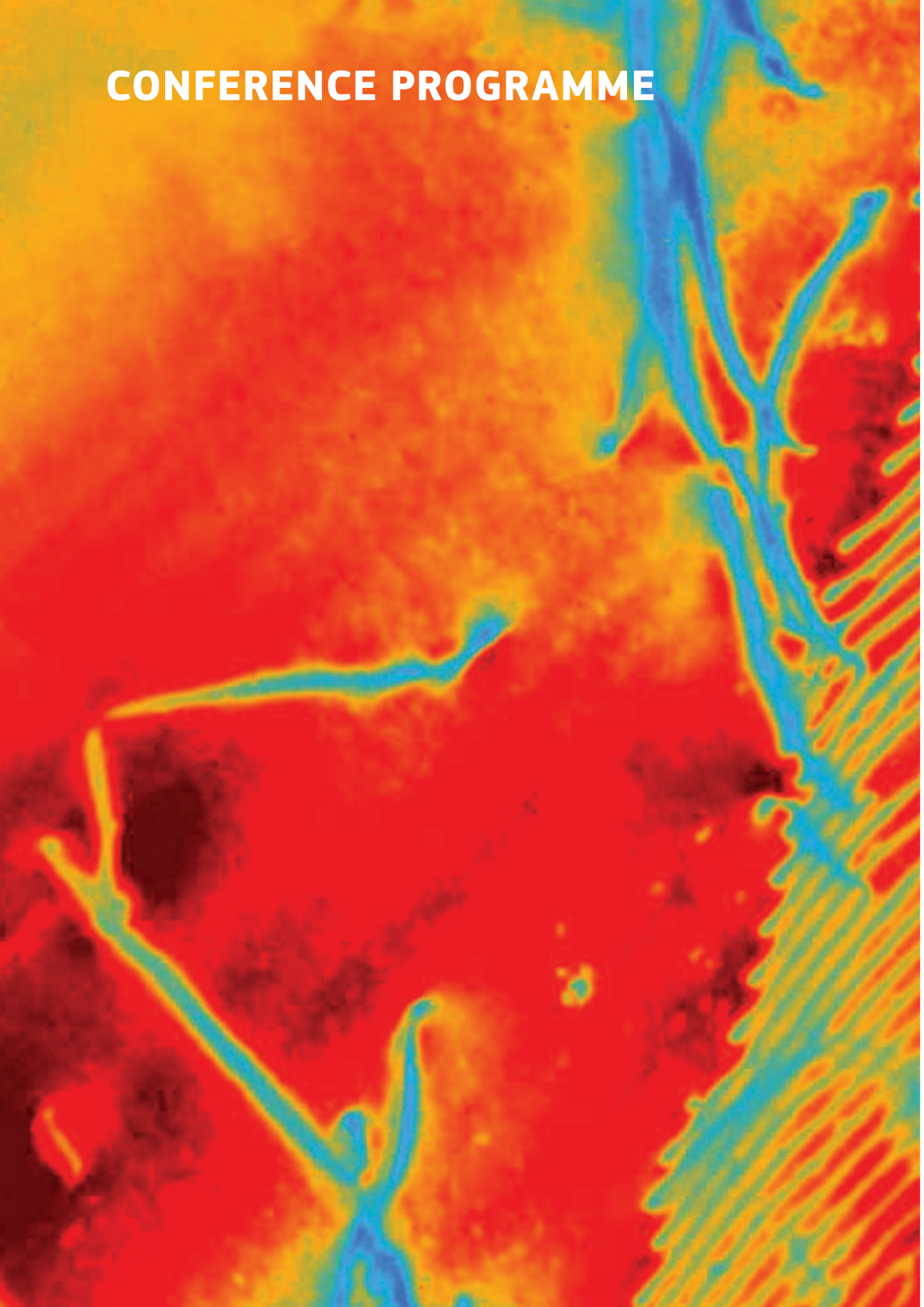
Session V: EURATOM FP7 – Challenges in repository technologies and construction: Engineering, Design, Demonstration and Monitoring

Joint conclusion – EURADWASTE '13 and FISA 2013.

Thursday, 17th October

Ignalina Nuclear Power Plant technical visit

Programme including a selection from the reactor hall, turbine hall, control room, spent fuel pools, spent nuclear fuel storage facility, free-release measurement facility and Solid Waste Management and Storage Facility.



Monday 14th October 2013 (AM)

Joint introduction – FISA/EURADWASTE
Chairperson: **Rudolf Strohmeier, Deputy Director-General**
DG Research & Innovation, European Commission

9:00 Chair introduction	
9:10	**Video-message from the Commissioner** Günther Oettinger, European Commissioner for Energy
9:20	**Message from the Minister** Dainius Pavalkis, Minister, Ministry of Education and Science of the Republic of Lithuania
9:40	**EU policy developments, achievements and perspectives in nuclear fission research and innovation** Rudolf Strohmeier, Deputy Director-General, DG Research & Innovation, European Commission
10:00	**Regulator's views on nuclear safety and radioactive waste management research and training – national and European perspective** Michail Demčenko, Head of State Nuclear Power Safety Inspectorate, Lithuania
10:15	**Public involvement and role of local and regional authorities** Arnoldas Abramavicius, Mayor of Zarasai municipality, Lithuania, Member of the EU Committee of the Regions
10:30 Coffee break	
11:00	**Keynote: Recent developments in research policy and implementation in reactor safety: How to ensure that knowledge derived from Euratom safety research is used to improve safety culture?** José I. Villadóniga (TECNATOM, S.A., ES)
11:30	**Keynote: International overview on the status of the long-term management of spent fuel and radioactive waste** Marie-Claude Dupuis (ANDRA, FR)
11:00	Press conference (tbc)
12:00 Lunch	

EURADWASTE PROGRAMME

Conference general rapporteur: Mr Alan Hooper, Visiting Professor of Repository Science and Engineering, Imperial College London, UK

Sessions on Policy Aspects of the Responsible and Safe Management of Spent Fuel and Radioactive Waste

SESSION I (13:20-15:30):
Responsible and Safe Management of Spent Fuel and Radioactive Waste

Chairperson: Mr Gábor Szendrö, Ministry of Environment, SE
Vice-Chair: Mr Maurizio Boella, European Commission (DG ENER)
Rapporteur: Mr Mariano Molina Martín, ENRESA, ES

13:20 Chair introduction	
13:30	**EU approach on responsible and safe management of spent fuel and radioactive waste: Council Directive 2011/70/Euratom** Mr Maurizio Boella (European Commission, DG ENER)
13:50-14:10 International developments	
13:50	**Address to the conference** Mr Pil-Soo Hahn (IAEA)
14:00	**Address to the conference** Mr Michael Sieman (NEA/OECD)
14:10	**ENSREG Work for Facilitating the Implementation of the Council Directive 2011/70/EURATOM of 19 July 2011 Establishing a Community Framework for the Responsible and Safe Management of Spent Fuel and Radioactive Waste** Mr. Gerald Hennenhöfer (Chair of ENSREG)
14:30	**ENEF work facilitating the implementation of the Waste Directive** Mr Hans Codée (COVRA, NL)
14:50	**The IAEA's Programme on the Safety of Radioactive Waste Management** Mr Kai Moeller (IAEA)
15:10	**The RWMC and concept development at NEA** Mr Jean-Paul Minon (ONDRAF/NIRAS, BE)
15:30-16:00 Coffee break	

Monday, 14th October 2013
Afternoon

SESSION II (16:00-18:10):
Decision Making on SF and RW Management

Chairperson: Mr Michael Sailer, Öko-Institut e.V., DE
Vice-Chair: Mr Michel Hugon, European Commission (DG RTD)
Rapporteur: Ms Christina Necheva, European Commission (DG ENER)

16:00 Chair introduction	
16:10	**Decision making on Spent Fuel and Radioactive Waste Management in France** Ms Marie Thomines (MEEDDM, FR)
16:30	**Long-term Planning of Spent Fuel and Radioactive Waste Management in Hungary** Mr Bálint Nős (PURAM, HU)
16:50	**Management of spent Nuclear Fuel and its waste,** Mr Manuel Martin-Ramos (EC, DG JRC)
17:10	**Challenges in Cost Estimation of Geological Disposal:** Mr Timo Äikäs (POSIVA, FI), et al.
17:30	**Public Involvement in the Decision-making Process** Mrs Saida Laârouchi Engström (SKB, SE)
17:50	**The UK's Graded Approach to Radioactive Waste Management** Mr Colin Mackie (DECC, UK)
Conference welcome cocktail (18:00-19:00) Welcome speech by the Lithuanian EU Presidency 2013	

Tuesday, 15th October 2013
Morning

SESSION III (08:50-12:00):
Challenges in Geological Disposal Programmes: from Policy to Research and Implementation

Objective

While geological disposal is the only accepted end-point for the responsible and safe management of high-level and long-lived radioactive waste and spent fuel, the status of policy, research and implementation varies widely among the EU and associated Member States (MS). Three countries (Finland, Sweden and France) have already or are about to submit construction licence applications. Other national programmes have advanced knowledge for the safety case but have either experienced setbacks in legislative or decision-making processes or have longer-term implementation horizons. Many other Member States are still working on establishing policies, strategies and time-frames for meeting the requirements of the EU "Waste Directive", 2011/70/Euratom.

Notwithstanding the requirements of the Directive and the need to perform scientific and technical research for the safety case, successful implementation of a repository programme over many decades requires a carefully planned research programme as well as sustained effort in other horizontal aspects.

Among these aspects, in the case of national programmes still at an early stage of development, useful experience can be drawn from more developed programmes on what it entails and the need to plan and organise a long-term R&D programme in building a country and site specific safety case. Also, the recent repository licence applications have highlighted the difficult role of regulatory functions in setting out sufficiently in advance regulatory expectations regarding the safety case. This requires interaction with the applicants but the nature and extent of this interaction remains a sensitive issue in order to respect the necessary independence of both sides. Furthermore, the successful implementation of a repository does not rest only on the scientific soundness of the safety case. The socio-political dimension plays an equally important role and several examples of set-backs in Europe in the last twenty years have demonstrated the need to involve non-technical stakeholders in the whole life cycle of a repository programme. Finally, the development and implementation of a repository programme from start to post closure may last one hundred years. Therefore, a well-conceived programme needs to plan and organise the development and maintenance of the necessary scientific / technical knowledge and competence over very long time periods. Competence will be needed anyhow to fulfil the need perform regular reviews of the safety case required by the Directive. This competence development and maintenance needs to be supported by adequate education and training activities.

All these horizontal aspects of a programme, seemingly not central, are in fact serious challenges that all national programmes are facing now or will have to face sooner or later. The aim of this session is therefore precisely to address these challenges in setting the scene regarding the safety case as starting point, in highlighting the need, experience and implications of setting-up and implementing a long-term national R&D programme in support of the Safety Case, the role of guidance and expectations of regulators in the review of the safety case, socio-political challenges, and training and education in developing and maintaining competence.

The chair is opening the session presenting key messages from the preceding Safety Case symposium, 7-9 October 2013. A keynote presentation is then given on an "experience of requirements and implications when setting-up and implementing a long-term national R&D programme". A report will be made of the work performed within the SITEX project on the development or regulatory expectations for the safety case. On governance and public involvement, a combined presentation of stakeholder involvement studies performed in several Euratom FP7 projects, mainly InSoTec and IPPA, will be made highlighting the evolution towards a more active involvement of stakeholders and social sciences in repository programmes which should not be limited to the question of acceptability. On competence maintenance and education and training, the strategy and activities being developed as part of the Implementing Geological Disposal – Technology Platform (IGD-TP) will be presented. This will be followed by a presentation on vocational training, accreditation, professional recognition and mobility based on the outcome of the PETRUS project. The session is finalized with a panel discussing the role of the R&D programme on a European level for countries with different levels of progress in implementation, stakeholder involvement and how to build and defend a Safety Case.

Chairperson: Mr Klaus-Jürgen Röhlig, Institute of Disposal Research, Clausthal University of Technology, DE
Vice-Chair: Mr Gunnar Buckau, European Commission (DG JRC)
Rapporteur: Mr Claudio Pescatore, NEA/OECD

08:50	**Chair introduction including key messages from the NEA Safety Case symposium,** 7-9 October 2013
09:10	**Keynote "Challenges in establishing a national RD+D programme"** Mr Piet Zuidema, Science Director, NAGRA, CH
09:40	**Regulatory Expectations Regarding the Safety Case: Moving from Conceptualization to Implementation** Mr Frank Lemy (FANC, BE), et al.
10:00 - 10:30 Coffee break	

Tuesday, 15th October 2013
Morning

Socio-political challenges - public involvement in geological disposal programmes		
10:30	**Opening up the technical: Involving stakeholders in developing repository programmes** Mrs Anne Bergmans (University of Antwerp, BE), et al.	
10:50 - 11:30 Competence Maintenance Education and Training (CMET) – challenges		
10:50	**Towards a European CMET Strategy: The CMET Working Group Actions within IGD-TP** Mrs Marjatta Palmu (POSIVA, FI), et al.	
11:10	**PETRUS initiative: overview of Seven years E&T experience in geological disposal** Mr Behrooz Bazargan-Sabet (Université de Lorraine-Ecole des Mines de Nancy, FR), et al.	
11:30	**Panel 1 – Role of R&D and other horizontal aspects on a European level in addressing the challenges facing national research and implementation programmes for Geological Disposal**	
	• What are key R&D and other horizontal activities (such as regulatory expectations and guidance, stakeholder involvement, ensuring sustained implementer and regulatory competence, ...) that can benefit from implementation on a European level? • What are the common R&D challenges and horizontal activities for advanced programmes? • What are the particular R&D challenges for programmes that are still in an early programmatic phase, including how can newcomers benefit from preceding activities and experience feedback on a European level? • What should the role be of stakeholder involvement in geological disposal programmes (at which phases, level of involvement in defining the implementation strategy/programme/technology, involvement in decision making, input to definition of R&D, ...)? • Is there an added value of developing safety issues, R&D and review methodologies of the Safety Case jointly on a European level for use in national programmes?	1. Panel Chair: Peter Wikberg (SKB, SE); 2. Kjell Andersson (Karita, SE), coord. IPPA project; 3. Thomas Fanghänel (EC-JRC); 4. Povilas POŠKAS (LEI, LT); 5. Christophe Serres (IRSN, FR), coord. SITEX project; 6. Cherry Tweed (NDA, UK).
12:00-13:20 Lunch		

Sessions on the seventh European Atomic Energy Community for Nuclear Research and Training Activities (EURATOM FP7, 2007-2013) in the Field of Management of Ultimate Radioactive Waste – Geological Disposal

SESSION IV (13:20-18:30):

Challenges in Science for Disposal: Advances in Phenomenology Understanding of the Source Term & Migration and Performance Assessment for the Safety Case

Objective

A Safety Case of a geological repository is a synthesis of evidence, analyses and arguments to quantify and substantiate that a repository will be safe during the time periods defined by the national regulations. Description and quantification of the various safety functions within the Safety Case builds on scientific knowledge for their conceptual understanding as well as for their quantification. The Safety Case thus is the basis for determining if a specific repository meets applicable safety requirements.

The Euratom Framework Programme Fission FP7 (2007-2013) in the area of management of radioactive waste, geological disposal, through implementation oriented R&D aimed at (i) establishing a sound scientific and technical basis for demonstrating the technologies and safety of disposal of spent fuel and long-lived radioactive waste in geological formations, and (ii) underpinning the development of a common European view on main issues related to the management and disposal of waste. A large number of cross-European consortia have been supported with projects on different topics, also providing for European added value.

In the absence of representative forums of the research community, the EC identified the topics to be included in the first calls for proposals following stakeholder consultation at different levels. Later on the key messages delivered from the EURADWASTE 2008 conference introduced an additional set of topics towards the middle of the FP7 programme. In 2009, the Implementing Geological Disposal – Technology Platform (IGD-TP) was launched and it developed its Research Agenda and Deployment Plan. Through this evolving consultation process, ten scientific projects were supported and these are reported in the present session.

The main objective of this session is to establish the status, at the end of FP7, of the research in support of the disposal safety case. The aim of the keynote presentation is to “review the scientific issues and uncertainties in geological disposal that led to the FP7 research projects on source term, migration and their impact on safety assessment and/or the safety case” and presented in this session. Presentation on individual projects describe their rationale and achievements: (a) What was the problem that led to this research and its impact on the safety assessment (for example quantifying or defining missing data, uncertainties and conservativeness), (b) what was the impact on the Safety Case (such as which processes are involved, and the ability to demonstrate that they are understood), and (c) at the end of the project, what is the differential progress on all these aspects. The task of the rapporteur is to round up the session by producing a synthesis of the key issues, outcomes of the projects and the outcome of the panel discussions, including open issues identified.

Tuesday, 15th October 2013
Afternoon

Chairperson: Mr Bernd Grambow, SUBATECH, FR
Vice-Chair: Mr Roger Garbil, European Commission (DG RTD)
Rapporteur: Mr Pierre Toulhoat, INERIS, FR

13:20	**Introduction by the European Commission on Euratom research programme on radioactive waste management – outline & state of play** Mr Christophe Davies (European Commission, DG RTD)
13:35	**Keynote: Review of the Scientific Issues and Uncertainties in Geological Disposal as Addressed in the Euratom FP7 Research Projects on Source Term, Migration and their Impact on Safety Assessment and/or the Safety Case** Mr Bernd Grambow, Director, SUBATECH, FR.
14:00	**First-Nuclides:** **"Investigation of Fast/Instant radionuclide release from high burn-up Spent Nuclear Fuel and its impact on the performance of geological repositories"** Mr Bernhard Kienzler (KIT-INE, DE), et al.
14:20	**CARBOWASTE:** **"Assessment/development of technologies and management options for irradiated-graphite and carbonaceous waste"** Mr Werner von Lensa (FZJ, DE), et al.
14:40	**ReCoSy:** **"Understanding of redox phenomena controlling the long-term release/retention of radionuclides in nuclear waste disposal"** Mr Marcus Altmaier (KIT-INE, DE), et al.
15:00	**SKIN:** **"Investigation of slow processes in close-to-equilibrium conditions in water/solid systems and their the impact on the mobility of radionuclides from radioactive waste geological repositories"** Ms Tomo Suzuki (ARMINES/Subatech, FR), et al.
15:20–15:50 **Coffee break**	
15:50	**REDUPP:** **"How the surface properties of spent nuclear fuel materials influence dissolution: Results of the REDUPP project"** Mrs Lena Zetterstrom Evins (SKB, SE), et al.
16:10	**FORGE:** **"Investigation of gas generation and migration processes for improving performance assessment of repositories"** Mr Richard Shaw (BGS, UK)
16:30	**Catclay:** **"Studies for improving the understanding of cation migration processes in Clayrocks"** Mr Sebastien Savoye (CEA, FR), et al.
16:50	**PEBS:** **"Improvement of the long-term performance prediction of clay-based engineered barriers systems"** Mr Lawrence Johnson (NAGRA, CH), et al.
17:10	**BELBaR:** **"Investigation of erosion processes in bentonite engineered barriers systems of a repository in crystalline rock and their impact on the long-term performance of the repository"** Mr Patrick Sellin (SKB, SE), et al.
17:30	**CROCK:** **"Investigations of uncertainties in radionuclide transport processes in the far-field of a repository in crystalline rock"** Mr Thomas Rabung (KIT-INE, DE), et al.
17:50	**Panel 2 – Role and achievements of European R&D in science for geological disposal and scientific challenges ahead**
	• Have the Euratom FP7 projects results been of international standard and so far contributed to improve the scientific and conceptual understanding of processes and thus improved confidence in the Safety Case, and which priority should this have in future activities? • To which extent have the scientific issues addressed in the different Euratom FP7 projects provided data and tools for reducing uncertainties in the Safety Assessment? • What is the significance of uncertainties in predictions for hundreds of thousands of years where all data indicate negligible risk for the next tens of thousands of years? • Is there evidence that R&D contributes to stakeholder confidence at large? • Have theEuratom FP7 projects established a common European view on the scientific issues addressed? • Are there additional topics emerging from for example source-term determination, waste forms, near- and far-fields, as well as advanced and innovative fuel cycle and waste management developments that could benefit from R&D on a European level?	1. Panel Chair: Bernd Grambow, (SUBATECH, FR); 2. Lawrence Johnson (NAGRA, CH); 3. Bernhard Kienzler (KIT-INE, DE); 4. Patrick Landais (BRGM, FR); 5. Barbara Pastina (Saanio & Riekkola Oy, FI); 6. Bo Strömberg (SSM, SE). Panel rapporteur: Pierre Toulhoat (INERIS, FR)
18:30–21:00 **Poster session and conference reception**		

Wednesday, 16th October 2013
Morning

SESSION V (08:45-11:45):
Challenges in Repository Technologies and Construction: Engineering, Design, Demonstration and Monitoring

Objective

Advanced EU national programmes on geological disposal can be expected to start operation within one or two decades. Therefore, in these advanced programmes, construction and engineering solutions as well as showing that the facility behaves as predicted play a central role. Engineering amongst others includes waste emplacement technology, but also construction of waste packages and the engineered near-field barrier system as well as engineering solutions used during closure and back-filling of shafts and galleries. Advanced national programmes thus face a transformation where emphasis and resources gradually move from scientific demonstration of long-term safety to industrial application.

The long-term safety of a geological repository is built on several redundant, robust and passive safety functions. However, monitoring is also considered in several countries as an essential component in demonstrating that the repository is performing as predicted and according to requirements. Consequently, in due time prior to operation it needs to become clear what the purpose of monitoring is (why), systems and processes to be monitored and their indicators (what) and the technology to be used or developed (how) and for how long. There does not seem to be a unique approach on this aspect. In addition, monitoring may be implemented to provide trust for non-technical stakeholders.

The objectives of this session are to present the status and challenges in repository technologies and construction of near-field engineering solutions and sealing of shafts and galleries as well as monitoring. For this purpose, a keynote gives a "review of the technical challenges in demonstrating feasibility and constructing a geological repository according to the technical design and quality performance requirements from commencement through to operation and closure". This is followed by presentations of three projects on technologies for realization of the engineered barrier systems, on full-scale in situ demonstration of five full-scale seals and plugs in shafts and tunnels and on monitoring. The three main host rocks formations in Europe are covered i.e. crystalline, clay and salt. The investigations include engineering and in situ testing of excavation methods of tunnels and cells, and manufacturing and emplacement techniques of buffer around waste canisters, as well as developing new technology for plug and seal materials, assembly and construction of plug and seal systems.

The project presentations will highlight what are the key challenges facing the actual implementation of the repository concept concerned (from drawings to construction), i.e. demonstrating the technical feasibility and ability to construct the necessary repository components in full respect of the long-term safety requirements, as well as to present the technical achievements and problems to solve. With respect to monitoring, the objectives include presenting the status of the various or possible strategies as to why, what, where, how and how long could monitoring of a repository project be decided or required. Against this, what are the current available/possible technologies, methods, strategies and organisations to implement this and the limitations (what is feasible and what is not feasible), in the different stages of the disposal process/programme (from surface exploration to post closure phase). The session ends with a panel on advances in these topics and considerations for future activities.

Chairperson: Mr Juhani Vira, POSIVA, FI
Vice-Chair: Mr Bruno Schmitz, European Commission (DG RTD)
Rapporteur: Mr Juan Carlos Mayor, ENRESA, ES

08:45 Chair introduction	
8:50	**Keynote: "Challenges in implementation of geological disposal: Licensing processes started for high-level nuclear waste repositories"** Mr Juhani Vira (POSIVA, FI)
9:15	**LUCOEX:** **"State-of-Art Demonstrations of Geological Disposal for High Level Waste"** Mr Jan Gugala (SKB, SE), et al.
9:45	**DOPAS:** **"Full-scale in situ demonstration of tunnel plugs and shaft seal in clay, crystalline and salt repository host-rock formations"** Ms Johanna Hansen (POSIVA, FI)
10:15-10:45 Coffee break	
10:45	**MoDeRn:** **"Overview of the MoDeRn project: A reference framework for developing a monitoring programme"** Mr Nicolas Solente (ANDRA, FR), et al.

11:15	**General discussion – Advances in EBS and Repository Technologies**	
	Themes and questions proposed for discussion by the audience. **Theme - Repository technologies and components, construction, testing and demonstration: From conceptual to industrial phase:** • What is sufficient for a positive licensing decision to operate the repository? • What is the difference, if any, between testing and demonstration? • What kind of residual uncertainties are acceptable/ tolerable? • Are the repository technical developments carried out in the LUCOEX and DOPAS Euratom projects adequately fulfilling the testing and demonstration requirements? • Are the concepts of robustness in repository design and Best Available Technique (BAT) complementary or antagonistic? • Is there a need and is it feasible to implement confirmation programmes of repository designs and performance? **Theme – Repository monitoring:** • What is the purpose of monitoring in a geological repository facility and programme? and what should it consist in? • Are the technical options, methods and developments in MoDeRn timely, adequate and sufficient? and what future promise could these deliver? • Should the qualification of the installation process, the repository system components (canister, buffer, backfill etc) and/or the repository operation depend on direct or indirect observations? • And should then monitoring continue after repository closure and if so for how long? • Should the public play a role in repository monitoring development and implementation? And if so in what form? • Will monitoring be another helpful step in the stepwise implementation of geological disposal, or will it just add to the burden of future generations? • If the monitoring of a closed repository is so very difficult, will it mean that the closure of repositories will be delayed (to make it possible to continue monitoring)?	Session Chair: Mr Juhani Vira and a panel of invited experts

11:45-13:30
Lunch

13:30-16:10
Change over to the joint FISA/EURADWASTE conclusion

Joint conclusion FISA/EURADWASTE

Chairperson: Director D, DG Energy, European Commission (tbc)
Vice-Chair: Bruno Schmitz, Head of Unit, DG Research & Innovation, European Commission
Rapporteur: Hans Riotte, Former Head of Radiation Protection and Radioactive Waste Management Division, OECD/NEA

13:30 Chair introduction	
13:40	**New instruments for Euratom research and innovation under the Horizon 2020 Framework Programme for Research and Innovation** András Siegler, Director, DG Research & Innovation, European Commission
14:10	**EURADWASTE General rapporteur report** Alan Hooper, Visiting Professor of Repository Science and Engineering, Imperial College London (UK)
14:30	**FISA General rapporteur report** Claes Nordborg, Former Head of Nuclear Science Division (OECD/NEA)
14:50	**Keynotes and General discussion**
14:50	**Safety Framework: Potential contribution of Euratom Research & Training** František Pazdera, Chair of SNE-TP
15:05	**Waste Framework: Potential contribution of Euratom Research & Training** Philippe Lalieux, Chair of IGD-TP
15:20	**Regulatory Requirements: Regulatory requests for EU-wide research and training in the areas of reactor safety and waste management** Johan Anderberg (SSM, SE)
15:35	**General discussion** Session chair, General rapporteurs (2), Keynote speakers (3), Director RTD K
15:50	**Closing remarks by the Commission** András Siegler, Director, DG Research & Innovation, European Commission
16:00	**Closing remarks by the Lithuanian Presidency** Representative of Lithuanian Presidency
16:10 Closure	

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JOINT INTRODUCTION

Public Involvement and Role of Local and Regional Authorities

Arnoldas Abramavicius
Mayor of Zarasai Municipality, Lithuania

Twenty-five per cent of the total energy produced in Europe has nuclear origin. That implies the presence of this kind of facilities in a significant part of the territory. They had been set in rural areas, near hydraulic resources, far away from populated or potential developing areas. As of today, the use of nuclear energy remains controversial due, in part, to security concerns and the long term management of radioactive waste. Our communities are engaged in a permanent round of debates in order to achieve consensus in the processes of democratic decision making, this is being emphasized, if possible, in the emerging model of knowledge society. Public participation and transparency are key elements, especially when these processes affect the environment or the future of our citizens. The activities of the nuclear facilities, including the ones related to waste disposal, have been carried out according to the 89/618 Euratom's Directive "Information to the population on the risks of radioactive facilities". This Directive has been applied and developed in different ways by each one of the Member States. In Spring of 1993 municipalities located near nuclear facilities started to coordinate in order to look at, and increase the weight of our municipalities in, the decision making processes of the European Union that affect our territories and citizens. Europe wide municipalities are organized in the association of Local and Regional authorities GMF (Group of Municipalities with Nuclear Facilities) which set common main targets: Information, especially to the population as a basic element for improving their capacity to get knowledge on nuclear issues; The future of municipalities after a nuclear plant shutdown (LT case); Getting enough economical capacity for municipalities in order to promote some kind of alternative economical development; Management of radioactive waste in a safe manner, in particular, their future and their definitive storage. Last but not least, increasing participation of our municipalities in the decision making process as a guarantee of respect to the principles of local democracy.

Local and Regional authorities' commitments generally are: Defending and ensuring that European nuclear municipalities take part in existing discussion forums and in decision-making processes that are being carried out in the European Union; Having information exchanged among its members about the nuclear reality in every European country and about municipal experiences in relation to territorial arrangement, economic development and civil protection; Carrying out future projects with the participation of all or part of its members in order to favour their integration in the European Union; Organising annual encounters so that European municipal authorities can gather and exchange information.

Construction of the only nuclear power plant in the Baltic States – the Ignalina Nuclear Power Plant – began in 1978. There were originally RBMK 1500 (Chernobyl type) four reactors planned in the power plant project. However, only two were ultimately built and launched (in 1983 and 1987). Precondition of the Lithuania EU accession – to close RBMK 1500 type reactors with EC financial support up to EUR 1 billion in 2004-2013. Decommissioning strategy were chosen: dismantling of the units, work done by Ignalina NPP work staff; on-site repositories/storages (brown field). To mitigate social and economical impact for local communities special target region were set in 2002. In anticipation of the job losses resulting from the Decommissioning of the Ignalina Nuclear Power Station in Visaginas, funding was launched from 2002 in order to prepare legal activities for a Regional Development Agency for the INPP Region which was seen

as a mechanism for coordinating the development actions of both the state and civil society in the region of the three Municipalities of Ignalina, Visaginas and Zarasai (total population of 75 000). Objectives for the Regional Authorities were set: to mitigate the negative social and economic consequences arising from the decommissioning of INPP on the inhabitants and business of the INPP Region; to diminish the immense difference of economic, cultural, educational and living conditions between the INPP Region and other regions; to provide assistance to enterprises when arranging the preparation and implementation of social and economic projects.

31 December 2004, the first unit (1500 MW) of Ignalina NPP was stopped. 31 December 2009, the second unit was stopped (1500 MW) with the capacity producing 9.8 TWh a year – approx. total LT demand. Having EU request to decommission both units, nuclear wastes disposal site were selected in the proximity of the NPP (onsite). Local municipalities has agreed with that solution as further financial support were promised from the LT Government and Ministry of Energy. Reduced funding from National Decommissioning Fund has to be substituted with the State budget and other sources. Happily by the local public and authorities were supported Plans to build a new 1 350 MW nuclear power plant at Visaginas by 2020–2022 were subsequently developed by the Lithuanian government together with Latvia, Estonia and the Japanese company Hitachi. Unlikely in 2012 Lithuanian Parliament made a decision to hold an advisory referendum on the construction of the new Visaginas NPP. According the vote NPP project were halted. Local communities left concerned with the uncertain future.

Recent developments in research policy and implementation in reactor safety: how to ensure that knowledge derived from Euratom safety research is used to improve safety culture?

José I. Villadóniga
TECNATOM, S.A., Spain

This paper aims to address two broad objectives.

The first objective is to summarize progresses made and new challenges identified, since the last FISA Conference that took place in the Czech Republic in 2009, regarding EURATOM research programmes in reactor systems with emphasis on safety. The topics covered are: safety of existing nuclear installations, severe accidents prevention and mitigation including emergency management, advanced nuclear systems for increased safety and sustainability, research infrastructures, education & training, and other cross-cutting actions.

The second objective is to place the focus on one specific area: Safety Leadership and Culture, which is widely accepted as an essential ingredient to achieve high safety and performance levels.

The EURATOM FPs, in addition to tackling very complex technological safety issues, have been responsive for many years to the importance of management aspects for Safety. Precursor projects in previous FPs, like ORFA and LEARNSAFE, helped to develop a better understanding of how organizational aspects have an influence on safety.

The Fukushima accident has reminded all of us that Safety Culture plays an essential role in the safe operation of nuclear installations and in the response to emergencies resulting from accidents. The EURATOM FP7 presently has at least five projects that directly support the enhancement of safety culture for obvious actors (utilities, regulators, health physics practitioners, etc) but recently also for actors not covered before (governmental organizations, opinion leaders, researchers, suppliers, contractors, etc). Therefore an important development was the understanding that all organizations that have an influence in the safety of nuclear installations need to cultivate a strong Safety Culture. The Projects with the most significant role in that direction are TRASNUSAFE, CORONA and NUSHARE.

The EURATOM FPs are contributing importantly to stimulate and develop new technical knowledge important for Safety. A sign of a strong Safety Culture in the users of that knowledge is precisely the willingness to use it effectively to improve the Safety of installations and activities. Several characteristics of a Strong Safety Culture are fully aligned with a proper use of knowledge developed by research: Safety is a clearly recognized value; Safety is learning-driven; Leadership and Accountability for Safety are clear and Safety is integrated in all activities.

Safety culture is driven by Leadership, therefore the role of leaders – of all organizations and at all levels - is essential in developing a strong Safety Culture that generates excellent levels of safety and performance. Leadership approaches should also be based on sound research.

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International Overview of the Status of the Long-term Management of Spent Fuel and Radioactive Waste

Marie-Claude Dupuis
ANDRA, CEO

All countries which have electronuclear production are concerned by the final disposal of their radioactive wastes, including spent fuels and/or vitrified residues in the case of reprocessing. This issue has been raised from the time of commissioning the first reactors. However, along with complex scientific and technical questioning, there were also social and political challenges.

Despite the fact that all citizens benefit from electricity generated by nuclear production and that as a consequence they must bear the burden of the associated wastes, the perspective of having a geological disposal for these highly dangerous wastes is difficult to accept. There is a strong need for information, for education, but also for sharing the decision-making processes with all the stakeholders, including the local public who have to live in the neighborhood of these facilities. Stimulating the acceptance via support to the local development is one way to show national solidarity to the communities which have accepted the implementation of disposal sites.

The challenges are the same everywhere, and the siting of these facilities have confronted similar difficulties in most of the countries. Therefore, new processes have had to be conceived in each case, with options matched to respective cultural values.

It is in this context that from very early on, all concerned countries began to share their respective viewpoints, especially through international organizations. Positions were taken respectively by IAEA and by NEA for the geological repository to be the reference solution for the disposal of spent fuels and high-level activity wastes, as well as intermediate-level long-lived wastes.

Finally, in July 2011, the 2011/70/Euratom European Directive also decided that geological repository would be the reference solution.

Three countries are at present well advanced: Finland, France and Sweden. Finland decided to dig the access to the repository level in a crystalline rock at the beginning of the previous decade. They reached it in 2011 and then began a set of experiments, including demonstration programs. POSIVA, the Finnish radioactive waste management organization, filed its license application at the end of 2012, with the perspective of commissioning the geological repository by 2022. After a 15-year research program, France has developed its project in a clay site, in an argillite formation at 500m of depth. ANDRA, the French National Radioactive Waste Management Agency, has moved to an industrial phase and is now preparing its license application, to be submitted in 2015. The last step before this milestone was the Public Debate which was held over a 4-month period in 2013 on the Cigéo project, as the French geological repository is known. In Sweden, in crystalline rock, as in Finland, the license application was submitted in 2011. SKB, the Swedish radioactive waste management organization, is also moving to an industrial phase, with a target date of commissioning the geological repository in 2025, as in France.

There are therefore several countries which started stepwise siting processes in the early 2000s. Canada, Switzerland, the United Kingdom are typical examples. We can as well include Belgium which is at a stage of political definition of the options. Lastly, the USA and Germany have joined this group by re-setting the scene. The US has adopted the options proposed by the Blue Ribbon Commission, with a geological repository to be made available in 2048. Germany, has defined a new Federal scheme towards a new siting process.

The last group includes Russia, China, India and several other countries, which have projects for commissioning a repository around 2050. Their schedules are still under definition. Japan, after the Fukushima event, is still redefining its approach. Finally, there is also the case of small nuclear countries, mostly interested in shared or regional solutions for disposing of their spent fuels.

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SESSION I

EU Approach for Responsible and Safe Management of Spent Fuel and Radioactive Waste: Council Directive 2011/70/Euratom

Ute Blohm-Hieber
Head of Unit "Nuclear energy technology, nuclear waste and decommissioning", DG Energy, EC

In July 2011 a new legislation establishing a framework for responsible and safe management of spent fuel and radioactive waste was adopted in the EU. It is anchored in internationally endorsed principles and requirements of the IAEA Safety Standards and the Joint Convention. All 28 EU Member States have to bring into force the laws, regulations and administrative provisions necessary to comply with this legislation by August 2013.

The EU approach aims at ensuring responsible and safe management of spent fuel and radioactive waste to avoid undue burdens on future generations. It reaffirms the ultimate responsibility of the Member States for the management of spent fuel and radioactive waste, including to establish and maintain national policies and frameworks, and to assure the needed resources and transparency.

Each Member State remains free to define its nuclear fuel cycle policy. Spent fuel can be regarded either as a valuable resource that may be reprocessed or as radioactive waste destined for direct disposal. Whatever option is chosen, the disposal of high-level waste, separated at reprocessing, or of spent fuel regarded as waste has to be considered. The storage of radioactive waste, including long-term storage, is an interim solution, but not an alternative to disposal. To this end, Member States are obliged to establish, implement, review and update national programmes for management of spent fuel and radioactive waste from generation to disposal, including significant milestones and clear timeframes.

Safety is absolute priority. Strong provisions are foreseen for demonstrating safety of spent fuel and radioactive waste management, including in the long term ("safety case" concept introduced). Prime responsibility of the licence holder for the safety of spent fuel and radioactive waste management under the supervision of its national competent regulatory safety authority is also reaffirmed. The role of the national regulators is reinforced and their independence strengthened.

Member States are obliged to make arrangements for education and training for their staff, as well as research and development activities to cover the needs of their national programmes in order to obtain, maintain and to further develop necessary expertise and skills. They have to ensure that adequate financial resources are available when needed for the implementation of national programmes, taking due account of the primary responsibility of spent fuel and radioactive waste generators for the long-term management of these materials. Member States are also obliged to ensure that the public is given the necessary opportunities to participate effectively in the decision-making process regarding spent fuel and radioactive waste management. They will invite international peer reviews to exchange experience and ensure that high safety standards are achieved.

The EU is the first major regional actor to provide a binding legal framework on nuclear safety (2009) and on responsible and safe management of spent fuel and radioactive waste.

ENSREG Work for Facilitating the Implementation of the Council Directive 2011/70/EURATOM of 19 July 2011 Establishing a Community Framework for the Responsible and Safe Management of Spent Fuel and Radioactive Waste

Gerald Hennenhoefer
Chair of ENSREG

On the 19th of July 2011 the council of the European Union has passed the Directive 2011/70/EURATOM establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste. The transposition of the Council Directive in national laws, regulations and administrative provisions had to be finalized on 23 August 2013.

While a lot of work has to be done in the respective member states, ENSREG will contribute and work to facilitate the implementations of the Council Directive.

Next to a national programme which has to be drawn up by each Member State by August 2015, first reports are due in August 2015. ENSREG decided to impose guidelines for the three-year reporting to the Commission (Guideline regarding Member States' Reports as required under Article 14.1 of the Council Directive 2011/70/EURATOM).

The purpose of these guidelines is to provide guidance to Member States regarding information and material that should be included in the reports and to establish a unified structure for reporting. Topics covered in the guidelines are general principles, national framework, competent regulatory authority, license holders, expertise and skills, financial resources, transparency, and national programmes.

ENSREG is positive that these guidelines will help to simplify the reporting for the Member States. It can therefore be regarded as a valuable tool for facilitating the implementation of the Council Directive.

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ENEF Work Facilitating the Implementation of the Waste Directive

Hans Codée
COVRA N.V., The Netherlands

Introduction

"Atoms for Peace", now 60 years ago, started the wide-spread use of nuclear energy and nuclear technology. The down side of its beneficial use is the production of waste that has to be taken care of. The safe and responsible management of radioactive waste is positively influenced by the facts that radiation can easily be measured at very low levels and that the volume of waste is small and hence manageable. Societal concern is a factor that can influence negatively the practical implementation of waste management solutions, but otherwise assures that a high level of safety is aimed at.

Low-level waste forms the largest volume of the radioactive waste. In Europe practical solutions for this waste have been found that are in operation in most Member States. Solutions for high-level waste will become operational between 2025 and 2030 for the countries with the largest volumes. In order to harmonise radioactive waste management in the European Union several attempts have been undertaken, ultimately resulting in the "Waste Directive" that has to be implemented by the Member States in 2013.

Waste Directive

The Council Directive 2011/70/EURATOM ("Waste Directive"), entered into force on 22 August 2011. As a key obligation under this Directive, Member States shall ensure the establishment, maintenance and implementation of a National Programme covering all aspects of spent fuel and radioactive waste management from generation to disposal.

ENEF

ENEF, the European Nuclear Energy Forum, was founded in 2007 as an initiative of the EC and endorsed by 27 Member States. ENEF brings together relevant stakeholders across the nuclear field. ENEF has three Working Groups: Risk, Opportunities and Transparency. Under each of these Working Groups Sub-Working Groups have been formed to address specific issues. A sub-working group on National Programmes was formed in order to assist in the establishment, documentation, communication and notification of National Programmes.

A Guide

The main function of a National Programme, the key obligation under the Waste Directive, is to set out how the national policy is transposed into practical solutions. While having to be notified to the Commission, the main role of a National Programme is to serve as the key tool and basic reference for the respective national actors dealing with the practical implementation of national spent fuel and radioactive waste management policies. Here a Guide was thought to be useful. The objective of the Guide is to provide a descriptive, but non-exhaustive tool to Member States to assist in the establishment, documentation, communication and notification of National Programmes, which satisfy both the needs of national actors and the requirements of the Waste Directive. It also assists the Commission in the review of National Programmes.

The IAEA's Programme on the Safety of Radioactive Waste Management

Magnus Vesterlind
Waste and Environmental Safety Section, IAEA, Austria

The IAEA has responsibilities for the safety of spent fuel and radioactive waste management, and implements a wide range of safety demonstration projects related to:

- Predisposal management and disposal of radioactive waste and spent fuel;
- Decommissioning of facilities such as nuclear power reactors, research reactors and fuel cycle facilities;
- Remediation of sites that have been radioactively contaminated;
- Control and assessment of radioactive releases to the environment with emphasis on the protection of the public and the environment.

The foundation for these projects is the safety standards which the IAEA is authorized, by its Statute, to establish for protection of health and minimization of danger to life and property. The safety standards are regularly reviewed and revised in close cooperation with the Member States.

The safety standards and the supporting projects together aim at assisting Member States to achieve and maintain a high level of safety in all activities and facilities where radioactive waste is generated or managed.

The IAEA also arranges international peer reviews of spent fuel and radioactive waste management, decommissioning and environmental remediation at the request of Member States. The reviews are based on the IAEA safety standards and on proven technical practice from around the world. Reviews can range from "high-level" reviews of the organisation, national plans/programmes and infrastructure to detailed assessment and technical advice on implementation of specific activities.

The present paper describes recent developments in regard to the IAEA's waste safety standards, and some of the ways the IAEA makes provision for their application.

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The RWMC and Concept Development at NEA

Jean-Paul Minon
ONDRAF/NIRAS, Belgium

The OECD Nuclear Energy Agency (NEA) has been at the forefront in sharing experience and providing practical guidance in addressing radioactive waste management issues for over 35 years. The NEA countries represent democracies with the most advanced waste disposal programme worldwide. The main body advising the NEA member countries in this area is the Radioactive Waste Management Committee (RWMC), composed of senior policy-makers, regulators, implementers and R&D specialists. The mission of RWMC is to assist NEA member countries in the development of safe, sustainable and broadly acceptable strategies for the long term management of radioactive waste, particularly long-lived waste, and spent fuel considered as waste. Its mandate is:

- To provide an international forum to senior representatives from national waste management organisations, regulatory authorities, policy-making bodies and research institutions to exchange experience and information in order to advance both the technical and societal aspects in managing radioactive waste.

- To consolidate and disseminate existing knowledge on radioactive waste management through joint co-operative research activities and projects, publications and specialist meetings. Also to promote initiatives to maintain relevant competencies and knowledge.

- To derive common bases for developing national regulatory frameworks and to promote adoption of common approaches in managing radioactive waste.

- To advance best practice by supporting international peer reviews.

The main objectives of the RWMC work programme are:

- To evaluate and develop waste disposal topics (disposal optimisation, long-term safety and safety assessment methods, industrial implementation of repositories, operational phase,…).

- To carry out studies and workshops to further enhance societal confidence and to explore the roles and requirements of regulatory authorities.

- To develop decommissioning topics (management of large components, cost control of decommissioning projects and standardisation of decommissioning cost estimates,…).

- To provide the framework for peer reviews (waste management, decommissioning studies).

- To consolidate and transfer knowledge to policy makers and non-specialist audiences.

The presentation gives an overview of the recent accomplishments and initiatives of the RWMC.

SESSION II

Decision making on Spent Fuel and Radioactive Waste Management in France

Marie Thomines
Ministry of Ecology, Sustainable Development and Energy, France

Radioactive materials and waste must be sustainably managed for the protection of human health, safety and the environment. The French policy regarding radioactive materials and waste is based on three pillars, specified in the French law: elaboration of a national plan as asked today by the 2011 Waste Directive, transparency and association of civil society representatives to the policy-making process and securing of long-term nuclear costs to limit undue burdens to be supported by future generations.

These pillars, as well as the main objectives and key milestones set by France for the waste management, are inscribed in a program act adopted in 2006. The French national plan for the management of radioactive materials and waste is a privileged tool for the implementation of the framework described in this Act. It presents a global vision of the management of radioactive materials and waste with the two-fold goal of ensuring the existence of management routes adapted to each category of radioactive substances over the short as well as the long-term and improving the consistency between the management routes. It defines a strategic roadmap for the three next years for the overall management of the radioactive material and waste.

Transparency, dialogue and consultation, notably with civil society representatives, must be at the core of the elaboration of public policies, especially for the management of radioactive materials and waste. The French National Plan falls within this framework: it results from exchanges in a pluralistic working group led by the Ministry of Ecology, Sustainable Development and Energy and the Nuclear Safety Authority. This working group is composed of environment protection associations, control and evaluation authorities, ministry representatives, radioactive waste producers and the national radioactive waste management agency (Andra).

This working group is in charge of the elaboration of the national plan, based on the descriptive and prospective data given by the national inventory of radioactive materials and waste, the results of the studies led by the waste producers and the feedback of the existing waste management routes. In accordance with the demands of the national plan, the waste producers shall report and provide the results of their studies to the Government, which are then evaluated by the Nuclear Safety Authority. The pluralistic working group is also ensuring a close follow up of the actions asked in the plan.

The current edition of the French national plan is the third one. Every three years, when elaborated, the national plan is transmitted by the Government to the Parliament for evaluation. The Government issues a ministerial decree that sums up the main recommendations of the national plan and gives them a legal basis.

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Long-term Planning of Spent Fuel and Radioactive Waste Management in Hungary

Bálint Nős
Public Limited Company for Radioactive Waste Management, Hungary

In Hungary, pursuant to the Act on Atomic Energy (Act), the work to be carried out for the final disposal of radioactive wastes, for the interim storage of spent nuclear fuel, for the back-end of the nuclear fuel cycle and for the decommissioning of nuclear facilities shall be the responsibility of an organisation appointed by the Government. The Government assigned the Hungarian Atomic Energy Authority to establish the Public Limited Company for Radioactive Waste Management (PURAM) to cover the above mentioned tasks.

The Act also created the Central Nuclear Financial Fund (Fund), from which these activities are financed. In order to accumulate enough money in the Fund, PURAM is responsible to compile the Mid- and Long-term Plan for the activities to be financed from the Fund. This plan contains the cost calculation of the activities. On the basis of a net present value calculation, the fee to be paid by Paks NPP (the only NPP in Hungary) into the Fund can be derived for each year of operation.

The basis of the Mid- and Long-term Plan is the inventory of radioactive waste and spent nuclear fuel, which contains the current amounts and the prognosis for the future arising, including the decommissioning of nuclear facilities. For each stream of waste and spent fuel, a management route with milestones and necessary technical solutions is defined. In line with Directive 2011/70/Euratom on the responsible and safe management of spent fuel and radioactive waste, the endpoint of each management route is a final solution: disposal.

In spite of the fact that there is no approved policy for the back-end of the nuclear fuel cycle in Hungary, a reference scenario (direct disposal) is defined for the management of spent nuclear fuel. This serves as a basis for cost calculations. The end location point is clearly defined: final disposal in a domestic deep geological repository.

In the presentation, the Hungarian practice for planning the long-term management of spent nuclear fuel and radioactive waste will be introduced including the main milestones, the technical solutions, the necessary facilities and the corresponding financing system.

MANAGEMENT OF SPENT NUCLEAR FUEL AND ITS WASTE

Manuel Martin-Ramos
European Commission, Joint Research Centre

A report has been prepared jointly by the European Commission's Joint Research Centre (JRC) and the European Academies Science Advisory Council (EASAC), which aims to inform policy makers on important issues to take into consideration in developing national programmes for the future management of spent fuel and the waste generated by fuel treatment.

The report describes the options for spent fuel management, their present state of development, and the consequences of choices between them, as well as important issues for national programmes to take into consideration in order to implement Directive 2011/70/EURATOM.

The report discusses in a synthetic way:

- the need for a national policy;
- the fuel cycles to consider;
- the decision factors in fuel cycle choice;
- experience with the involvement of stakeholders in decision-making; and
- the key decisions to be taken and their consequences.

To inform preparation of the report, a seminar was held in February 2013 to get the views of an international group of experts on the challenges associated with different strategies to manage spent nuclear fuel, in respect of both open cycles and various steps towards closing the nuclear fuel cycle. The report integrates the conclusions of the seminar, which considered issues of sustainability, non-proliferation, safety, organisational and economic factors, and public involvement.

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Challenges in Cost Estimation of Geological Disposal: Optimisation as a Goal

Timo Äikäs, Jussi Palmu
Posiva Oy, Finland

The program for direct geological disposal of spent fuel in Finnish bedrock was started in the early 1980s as a response to the Government decision in 1983 on the principles and time schedule for nuclear waste management. Since a suitable concept for disposal was already there (KBS-3), the focus of the program was on site selection. In 2001 the Finnish Parliament approved the so-called Decision-in-Principle (DiP) on siting a KBS-3 type repository for spent fuel in an area near the Olkiluoto nuclear power plant. After the decision Posiva Oy started the detailed design and construction of an underground rock characterisation facility named as "ONKALO". In December 2012 Posiva submitted an application for the construction license of a disposal facility for spent fuel (SF) consisting of an above-the-ground encapsulation facility and an underground repository. The time schedule presented in the application aims at submitting the operational license application in 2020 and commissioning the operation in 2022.

During its course the programme has experienced significant changes since 1983 in terms of the inventory of spent fuel. This is, of course, linked to the plans of Posiva's owners to use the existing power plants and plan for the new reactors. In the early 1980s the amount of SF to be disposed was 1200 tU accumulating from two reactor units planned to operate for 30 years. Today Posiva applies the construction license for 9000 tU predicted from six NPP units, four in operation, one under construction and one in planning phase. The planned operation lifetime of reactors is typically 60 years. The changes in the nuclear programme have set challenges both on the concept development for geologic disposal and as well on the cost estimation.

According to the Finnish nuclear legislation Posiva's owners as the waste producers are responsible for all activities related to waste management including financial provisions. The provisions are collected in an external fund managed by the government. These provisions collected in the fund cover in principle future liabilities for the waste generated so far in the operating NPP units. The total cost estimate for spent fuel management of Olkiluoto and Loviisa NPP's is estimated for the whole expected lifetime of the reactors and for the required duration of the disposal. Prediction of future costs over a time span of one hundred years is very challenging while bearing in mind that the technology is continuously developing. Posiva's owners have adopted the International Financial Reporting Standard (IFRS) as basis for their financial reporting. IFRS reporting is based on the total cost estimate and requires that the waste management liabilities are accounted for in the balance sheet.

Based directly on realistic predictions of SF inventory, waste management plans and technical designs Posiva has produced cost estimates for its owners for the spent fuel disposal. The total cost estimate for the construction, operation and closure of the spent fuel disposal facility presented is EUR 3 300 million (year 2009) for current four operating reactors and one under construction. Can the total cost be optimised, is a topic for discussion. In parallel with the concept development and the design the timing of disposal during the more than 100 years seems to be important element in the efforts for optimisation. The tentative results show that there is potential for optimisation.

The UK's Graded Approach to Radioactive Waste Management

René McTaggart
Department of Energy & Climate Change, United Kingdom

In the UK we have diverse and well established uses of radioactive materials. From nuclear materials to spent fuel to reprocessing materials to hospital wastes. All need to be dealt with in a coherent and responsible manner.

The UK has a long standing approach that seeks to ensure that safety remains of paramount importance when managing radioactive substances. However, it is clear that not all radioactive substances are equal and therefore each can and should be dealt with differently according to the level of risk they pose.

Proportionality in dealing with the risk is delivered in the UK via a graded approach – this is key to ensuring safety while allowing the benefits of the uses of radioactive substances to be realised.

The UK's rationale and approach to the application of the graded approach will be presented including how the UK uses measures, such as licensing and the safety case regime, to ensure the risks are controlled.

Additionally, how the roles and relationships between the UK Government (as the policy makers) and stakeholders, including the regulators and operators, work in practice will also be explored.

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SESSION III

Challenges in establishing a national RD+D programme

Piet Zuidema
Nagra, Switzerland

Each geological disposal facility is to some extent “a first of its kind” due to the unique geological environment at each site. Therefore, it will never be possible to implement a repository without any research and development work. Therefore, research, development and demonstration (RD+D) is essential for each programme that has a mandate to develop geological disposal facilities for long-lived wastes.

Strategic planning of RD+D requires consideration of the programme needs over the sequence of steps and milestones with clearly defined time frames until repository construction and start of repository operation. It is also important to define the level of detail of the work needed for each phase of repository development.

RD+D (normally mainly under the responsibility of the implementer) does not work in isolation. Its work programme is informed by policy and regulatory guidance and provides input to and receives feedback e.g. from repository design and development of the safety case. Considering the national context, it is also important to give sufficient thought to the role of RD+D independent of the implementer (e.g. by the regulator, by technical support organizations, or through other independent channels). Thus, at the national level the organizational framework for RD+D should be well thought out and clear rules should be established for decision-making in RD+D; including, e.g., the role of a periodically updated and externally reviewed RD+D plan. On the one hand RD+D should focus on the needs of the respective programme and should not be dominated by pure research interests, but on the other hand the scientific needs of a programme should seriously be considered and recognized. To summarize, although many RD+D results are available from “elsewhere”, each national programme has to have an RD+D programme tailored to its own specific needs.

The RD+D programme will evolve with time. In the early phases, fundamental issues are often addressed, then the focus typically moves towards implementation feasibility and finally, RD+D often has to look at ”industrialization” of repository implementation and to investigate how to optimize the repository implementation (construction, operation, closure).

The team in charge of RD+D has to have a good network and a broad overview and has to understand the context of its work. This allows the team to decide if work is needed within its own infrastructure (e.g. if there is a need for a dedicated URL or lab) or if the work can be done somewhere else (e.g. in a lab or URL of a partner organization). The team should also be able to recognize if the information needed is available from somewhere else and no substantial work is needed. Then the team should be able to see its work in the framework of the worldwide scientific community to ensure that it is state-of-the-art and in line with the current scientific thinking.

Thus, when establishing an RD+D programme, sufficient thought has to be given to the infrastructure necessary to do the work; i.e. is there a need to have one’s own infrastructure or is access to infrastructure somewhere else possible and sufficient? In this context, the EU programmes offer several platforms for co-operation in RD+D that can be an efficient way to perform the necessary RD+D work. Specifically, the IGD-TP provides further opportunities for RD+D collaboration in a flexible manner, which permits a range of possible collaboration modes.

Regulatory Expectations Regarding the Safety Case: Moving from Conceptualization to Implementation

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Regulatory bodies are responsible for the establishment of requirements and conditions for the development, operation and closure of disposal facilities. Performing an independent verification of compliance with these requirements and conditions involves different types of activities such as reviews, on-site inspections and R&D activities. In particular, the regulatory body has a continuing role to review the safety case which has to be regularly updated to remain an adequate basis for making decisions throughout the repository lifecycle. The evolving nature of the safety case is largely acknowledged and addressed by on-going international initiatives bringing together regulators and technical safety organizations such as the EC FP7 SITEX Project, the development of Safety Reference Levels for radioactive waste disposal undertaken by WENRA and, particularly, the European Pilot Study on the Regulatory Review of a Safety Case for Geological Disposal of Radioactive Waste.

Diverse types of decisions are expected to be made owing to the stepwise nature of a disposal programme. Well-defined decision points are associated with each programme phase such as host rock and/or site selection, decisions to grant a license to construct, to operate or to close the facility. Hence, the safety case needs to be updated when moving from one phase to the next in order to provide the information supporting the next decision while ensuring traceability of previous decisions and giving reasonable assurance that subsequent stages will allow meeting the safety objective. In particular, the safety case should show that an adequate safety strategy has been developed and implemented and that choices on the disposal system derive from a process involving optimisation of radiological protection. The safety case forms also the basis for assessing the safety implications of any change to the disposal system such as modifications of the design, operational procedures or waste inventory as a result of experience from practice or technological and scientific developments. Therefore, regular updates of the safety case are also needed to maintain an adequate basis for the decisions regarding these changes.

Regulatory decisions are made in presence of both irreducible and reducible uncertainties considering that the safety case will mature as new information such as “as-built” properties, monitoring data, characteristics of the emplaced waste or results from research become available. The safety case should describe the approach that is followed to manage these uncertainties throughout the different phases of the disposal programme and show that existing levels of uncertainty are acceptable given the decisions to be taken. Several choices made on the basis of limited information in early phases may have to be confirmed before or during the construction and operation of the facility. Information on how significant uncertainties will be managed in subsequent phases is thus also essential. Furthermore, assurance has to be given that the concept will be or has been properly developed and implemented by means of a management system ensuring the quality of deliverables and fostering a strong safety culture.

ABSTRACTS
Invited Lectures

Opening up the technical: Involving stakeholders in developing repository programmes

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In this paper, we will present findings from InSOTEC (International Socio-Technical Challenges for implementing geological disposal – GA 269909), IPPA (Implementing Public Participation Approaches in Radioactive Waste Disposal – GA 269849), and MoDeRn (Monitoring Developments for safe Repository operation and stage closure – GA 232598). All three projects have in common that they examine the potential for and consequences of broadening involvement in radioactive waste management (RWM) and geological disposal (GD) beyond the technical community.

It is clear that the concept of GD has developed, and will continue to do so, not only because of developments in scientific knowledge, but also as a consequence of debates on how this technology relates to society. A clear example of this, is the introduction, by legal obligation, of the seemingly contradictory notion of retrievability into the concept of GD in Switzerland[1] and that of reversibility in France[2]. The adapted concept of GD that is being developed in these and other countries today (see for example the NEA's R&R project: NEA, 2011; NEA 2012) still has to prove its capacity for resolving competing values with regard to the safe disposal of radioactive waste. But it does show that we need to think about GD - or more generally any technique for the long term management of high level waste (HLW) or spent fuel (SNF) - not simply as a technology designed by scientists and experts, but as a socio-technical concept the meaning and characteristics of which are negotiated and value laden.

InSOTEC focusses on this entanglement between the social and the technical in the development, siting and implementation of RWM options, and GD in particular. Work so far has shown that the active engagement of potential or actual host communities changes the geographies of HLW/SNF disposal, as voluntaristic siting regimes or local demands placed on the more traditional siting programmes tend to reconfigure the relationships between GD and society. Furthermore, the estimated operational lifetime of a facility seems to be changing in many cases, from a few decades to at least a century. Today even the strongest proponents in geological disposal have come to recognise that the state of passive safety that is fundamental to the concept will not be achieved instantly. The end of operations of a disposal facility will therefore be just as much a social and political decision as it will be a technical one, making the long-term governance of the physical site within a host community one of the key challenges to consider.

Monitoring, the focus of the MoDeRn project, illustrates the necessity of a sociotechnical understanding of geological disposal. Monitoring, broadly understood as any data gathering relating to the behaviour of a repository and its natural and social environment and potentially spanning a period from site investigation to post-closure, is considered by both experts and citizen stakeholders interviewed in the course of the MoDeRn project as having the potential to contribute to building and maintaining confidence, among experts, regulators, decision makers and citizens, in a repository system. However, from exploratory engagement activity organised within MoDeRn we identified an important tension between the perspective of technical experts and that of potentially affected citizens on the purpose of monitoring. Where the experts tended to view monitoring in terms of 'performance confirmation', as a tool for validating the repository design concept and its construction, citizens tended to see monitoring in terms of the 'critical assessment' of safety, acknowledging uncertainties and serving as a form of surveillance that would detect unanticipated problems in a repository. This is more than merely a semantic difference and needs to be addressed and negotiated if a mutual understanding is to be developed and the question of long-term safety is to be treated meaningfully for all involved.

The IPPA project is focused on enhancing the quality of decision-making processes in nuclear waste management through clarity, awareness, fairness and trust. A key principle is the implementation of participative processes and transparency, and the involvement of stakeholders in "safe spaces". The RISCOM model and other approaches to public involvement are implemented in five radioactive waste management programmes in central and eastern European countries (Czech Republic, Poland, Romania, Slovenia and Slovakia). The intention is to find ways to 'vaccinate' decision-making processes against unnecessary narrow framing and fragmentation that may later become serious obstacles to decision-making. Experience has shown that restrictive framing is often the result of technological experts failing to acknowledge broader societal perspectives, but it could also come about in part due to social sciences neglecting 'hard facts'.

An important step has been taken in building a knowledge base for participation processes. In order to assess the usefulness of various methods, it is necessary to describe them in terms of representative properties and in IPPA properties have been divided into three main groups: instrumental properties (enhancing the quality of decision-making), procedural properties (conduct of the process) and constitutive properties (benefits implied by participation). Experiences so far bring about some observations to consider for future activities. There are indications that some stakeholders hesitate to take part in participative activities. Not all stakeholders have the same interests and most often consensus between them cannot be expected, although some of them may agree to cooperate to reach common goals. Any public participation processes must take this into account to be trustworthy and meaningful. The IPPA project also considers how negotiations on compensation and added value can be implemented at the local level. A report has been produced which gives readers a holistic picture of the different kinds of arrangements for adding value to benefits and of their potential drawbacks, and to raise consciousness about the different approaches.

[1] Kernergiegesetz, vom 21. März 2003 (*Nuclear Energy Act* - 21 March 2003).
[2] Loi n°2006-739 du 28 juin 2006 de programme relative à la gestion durable de matières et déchets radioactifs (*Radioactive Materials and Waste Planning Act* - 28 June 2006).

ABSTRACTS
Invited Lectures

Towards a European CMET Strategy: The CMET Working Group Actions within IGD-TP

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The Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) community according to its vision (Vision 2025) aims to proceed to obtaining licenses to construct and to operate deep geological repositories for spent fuel, high-level waste, and other long-lived radioactive waste in their respective Member States. Their commitment to the Vision 2025 includes developing joint means to facilitate access to expertise and technology and maintain competences in the field of geological disposal for the benefit of the European Member States. In the Strategic Research Agenda (SRA 2011) a need for the Cross-cutting Activity Competence Maintenance, Education and Training (CMET) was identified. Preliminary work towards setting up a permanent Organisational Working Group to address this area has begun. In 2013, the CMET Working Group convened for its first meeting with the support of the IGD-TP Secretariat (FP7 project: SecIGD2).

The CMET Working Group has four main objectives: 1. To carry out transfer of the state of the art of strategies and activities for Competence Maintenance, Education and Training related to the implementation of Vision 2025. This requires identifying what the specific CMET needs are for implementing the SRA 2011 and the IGD-TP's first Deployment Plan (DP 2012) until 2016. 2. To develop quality assurance of training aimed at new and experienced professionals in the field of geological disposal by developing quality assurance procedures and criteria for the voluntary accreditation of training (and education) for the sector. The work starts with carrying out a feasibility study for an accreditation scheme for informal learning will be undertaken that can also be applied to the formal setting. The background of the scheme is derived from the ECVET (European Credit system for Vocational Education and Training) approach. 3. To compile the content of training i.e. a type of "curriculum or curricula" for professionals in geological disposal for pooling joint training efforts or alternatively engaging educators and trainers to address the IGD-TP's RD&D work's E&T needs. The identification of current state of curricula already developed for geological disposal and mapping them in relation to the generic stages of repository development identified in the SRA 2011 are the starting points of the CMET work towards this objective. 4. To ensure indirectly the sustainability of providers and the necessary infrastructures/facilities for CMET and new personnel and their development in the future. The first three objectives and the voluntary pooling of resources for the development and implementation of CMET action plan are foreseen to strengthen the sustainability of supply of expertise in various forms. The CMET strategy and action plan are in the formulation process by the CMET Working Group volunteers from 12 European Member States.

PETRUS initiative: overview of Seven years E&T experience in geological disposal

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Radioactive Waste management organisations (WMOs) need new type of approaches to meet their human resources demands and to maintain the competence and availability of skilled personnel over the life time of a repository operation that will exceed a century. Considering the small size of the waste management community, collaborations and partnerships at the European level are key actions. Pooling all available resources for creating a common strategy with long-term sustainability objective is the only way towards assembling the necessary resources for avoiding the emergence of the generation gap and assuring the transfer and preservation of knowledge with fewer risk and unintended consequences.

In line with the Lisbon strategy and the 2020 perspective, "PETRUS" initiative coordinates since 2005 universities, WMOs, training organisations and research institutes efforts to develop cooperative approach to education and training (E&T) in the geological disposal. The objective is to ensure the continuation, renewal and improvement of the professional skills by filling the gap between growing demand for structured education and training and the offering that is fairly limited. By transposing the principles and the concepts of the "systematic approach for training" (SAT) into a rotary dynamic system encompassing analysis and planning, development and implementation, and evaluation and validation processes, PETRUS proposes an innovative strategy for sharing resources from both academia and industries. Launched as a part of ENEN II project under FP6 and later granted in the frame of FP7 EURATOM EFTS, PETRUS programme has already i) established and analyzed the end-users' needs and the existing resources in E&T, ii) successfully implemented in several partner universities 130 hours of specialised common courses within the Bologna process that address Master's students in Geosciences, iii) developed the framework for qualification oriented modular training programmes for professionals, compatible with the "European Credit system for Vocational Education and Training" (ECVET), iv) set up synchronous e.learning methodology allowing broadcasting live lectures into multiple distance sites and v) initiated the PETRUS end-user council for long-term collaboration between end-users and E&T providers. Efforts made by PETRUS partners will be consolidated in the frame of a new project (PETRIS III) as the continuation of the European Cooperation targeting i) practical implementation of PETRUS training programme following ECVET principles, ii) elaboration of multidisciplinary training and research framework for PhD students and iii) development of strategies and frameworks for maintaining PETRUS initiative over the long-term through collaboration with the **Implementing Geological Disposal - Technology Platform** (IGD-TP) and the ENEN association for the overall management of the radioactive waste disposal E&T activities.

The paper provides an overview of the current development of PETRUS programme, highlights results and experience already acquired, identifies difficulties to overcome and discusses processes and pathways that shape further achievements.

ABSTRACTS
Invited Lectures

SESSION IV

Review of the scientific issues and uncertainties in geological disposal as addressed in the Euratom FP7 research projects on source term, migration and their impact on safety assessment and/or the safety case

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It is inconceivable that we leave to our children our highly toxic radioactive wastes in today's interim surface storage sites without investing large efforts in implementing geological disposal to reduce multi-generational risks. Large scientific and technological advances in many European projects have increased our understanding on how to construct, exploit, and close a geological repository and how to reduce uncertainties in demonstrating its long term safety. The 7th FWP was in this sense a large step forward, characterized in particular by the establishment of IGD-TP and its exchange forum, gathering the major technical stakeholders willing to contribute to the key topics of the strategic research agenda, interested in pooling of resources, coordinating efforts and managing knowledge and dialogue.

Heading for the implementation of repository projects, new issues arise and old questions are to be seen in a new light, asking for in-depth research efforts to assess the impact of repository construction and operation on safety functions or to capture the evolution of the various thermo-hydro-mechanical-chemical (THMC) transients from the time of emplacements of the waste leaving many open void spaces to the more compacted conditions in the long term. Research efforts are addressing more and more realistic questions like how to simulate the evolution of the THMC behavior and long term integrity and sealing properties of engineered barriers materials (studied in the project PEBS), providing a more convincing connection between the initial transient state and its long-term state at thermal and hydraulic equilibrium, considering as well materials interaction or fluid transport (gaz, water) with time along gaps between materials and disturbed zones (EDZ). Similarly, since at the beginning of the 7th FWP, gas transport was still poorly understood,the project FORGE addresses the fate and impact on sealing functions of gas generation and migration in the repository. Certain waste emplacement cavities may remain open for tens of thousands of years. Source term, mechanical evolution, fluid transport and chemical interactions under such non-saturated transient conditions still deserve further research attention.

Repository sites will remain a rare resource in the management of the nuclear fuel cycle whose use needs to be optimized as compact as possible. Certain wastes, like graphite, may need treatment, to dispose only the most radioactive concentrates, a challenge studied in the Carbowaste project, but large efforts are still necessary for industrial solutions. Transport of radionuclides (C14) by gaseous pathways may be important under certain conditions of gas generation or if large void spaces remain available.

More than 30 years of European research on source terms for nuclear waste glass and spent fuel have resulted in structural, kinetic and thermodynamic multi-element understanding, coupled in overall mechanism and mathematical models, which describe the contribution of the waste form to radionuclide retention, which capture the evolution of the near field conditions (redox, temperature, and interface to near field materials…) in the various phases of disposal, as studied for example in NFPRO (6th FP) or Recosy. Uncertainties remain for example in the impacts of long term evolution of spent fuel (radiation field, surface maturation) and more knowledge is needed to keep up with the tendencies to use nuclear fuel to higher average burn-up. Certain aspects have been addressed in the 6th FP using simulated alpha doped UO_2, while the role of water chemistry, fuel surface maturation and high energy sites of the corresponding fluorite structures are addressed in the ongoing REDUPP project, the very slow kinetic, thermodynamic and structural transformations at nuclear fuel surfaces close to equilibrium are studied in the SKIN project and radionuclide release from grain-boundaries and from high-burnup spent fuel in the FIRST-Nuclides project. Alteration products of glass and spent fuel deserve still more research attention: for glass the thermodynamics and structure of incorporation and local distribution pattern of sparingly soluble radionuclides in the amorphous gel and the underlying atomistic binding modes are still uncertain and difficult to predict. Similarly, surface alteration products of spent fuel formed under reducing conditions have not yet been identified.

Studying the processes which govern the migration of radionuclides in porous or fractured repository rock, the FUNMIG (6th FWP) project has already addressed the challenge to bridge the gap in sorption/diffusion description form atomistic to meter scale, both in experiments, analytics and models. In the ongoing CLAYCAT project, this approach is extended to strongly sorbing nuclides. However, we still do not fully understand the relationship between microscopic heterogeneous dynamics (diffusion in heterogeneous media) and macroscopic heterogeneity and mechanical and chemical and transport properties. Incorporation of radionuclides in solids by diffusion and solid solution formation provide additional irreversible entrapment, which needs to be quantified in future (first steps in SKIN…). Our understanding allows to predict the mass transfer in space and time of the most mobile radionuclides (those which may contribute to dose to future humans) even though uncertainties needs to be reduced in particular in crystalline rock considering barrier effects and preferential path ways. Two projects address this issue: CROCK and BELBAR, the latter to study the impact of bentonite erosion on colloid facilitated radionuclide transport. We know bounding and realistic values for retention and solubility of the most toxic sparingly soluble radionuclides, sufficient to show negligible release to the biosphere, but we do not know exactly which fraction of the inventory of these nuclides will remain for how long time bound to the reaction products of an altered waste matrix, which fraction at a given time becomes sorbed on metal containers corrosion products or which fraction has migrated after tens of thousands of years for 1, 10, 100 cm etc. This knowledge is not important for safety analyses since the nuclides will not lead to dose contributions in typical exposure scenarios, but it is important for feeling safe.

RD&D on geological disposal is driven by the need to demonstrate long term safety and to optimize the safety architecture of the repository. How can we upscale results of short-term studies and models derived from such data to long-term projections? How can we be sure we're extrapolating results correctly to project to hundreds of thousands of years? Our models are based on fundamental scientific understanding, derived from experimental and field studies and from underground laboratories, using the very best analytical tools available and dedicated experimental setups to work with radioactive matter available only in few countries, to characterize multiple coupled complex processes and structural evolutions from molecular to meter scale and to develop adequate models. THMC model couplings need to grasp all scales in time and space. But at the same time as our scientific understanding rises, so does the complexity of the problem to describe. The difficulty is to avoid that information on key safety functions becomes buried in a large quantity of less important information.

Safety assessment is today a routine exercise used at various stages in the development of repository projects, in the triangle between evolving regulatory imperatives, research orientations and repository design optimization. Compared to PAGIS, 30 years ago, our analyses today take much more detailed process understanding into account. At the end of the 6th FWP, the PAMINA project has reviewed safety

assessments and have shown that the associated explicit or bounding case uncertainties need to be handled in different ways, depending on probability of events, on whether they impact an important safety function, whether one can take benefit from additional barriers to make some uncertainties less important and the safety case more robust. The uncertainty on solubility values of tetravalent actinides by many orders of magnitudes (studied in SKIN) has for example little impact on radiological risk if even the highest reasonable solubility value assures fixation. However in other fields, for example of crystalline rock far field migration processes (CROCK project) uncertainty needs to be reduced due to avoid too much over-conservatism. Today almost any EC project studies also the impact of its scientific results and process models on safety assessment, often providing feedback to repository design optimization like the FORGE project on gas generation or the PEBS project which deals with the evolution of the engineered barrier system over time. But we need to keep in mind that the safety case is both a scientific and a social concept. A more active involvement of social sciences in risk assessment is necessary which shall not be limited to the question of acceptability.

FIRST-Nuclides: "Investigation of Fast/Instant radionuclide release from high burn-up Spent Nuclear Fuel and its impact on the performance of geological repositories"

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The CP FIRST-Nuclides deals with understanding the fast / instant release of safety relevant radionuclides from high burn-up uranium oxide (UO_2) spent nuclear fuels in deep geological repositories. Investigations on the rapid release have been performed in previous European projects as well as in several studies mainly within French research programs. However, important issues are still open and consequently, the CP FIRST-Nuclides aims on covering this deficiency of knowledge: Fuel elements from pressurized water and boiling water reactors, with different enrichments, burn-ups and average power rates need to be disposed of in Europe. Safety analyses require the quantification of the rapid release of radionuclides directly after loss of integrity of a disposed fuel element canister. The fission products iodine, caesium, technetium and selenium but also chlorine and carbon belong to the fast/instant release fraction (IRF). The relevance of the IRF is obvious, as some of these elements form negatively charged species which will be mobile in the engineered barriers as well as in many of the repository host rocks selected in Europe.

The project is implemented by a consortium of ten beneficiaries. Eleven Associated Groups (AG) contribute to the project without receiving funding by the CP. These groups have particular interest in the exchange of information. A group of six implementation and regulatory oriented organizations participate as an "End-User Group". This group ensures that interests of the waste management and regulator organisations are reflected sufficiently.

The CP is organized in six work packages (WP) including the selection, characterization and preparation of the spent nuclear fuel to be used for the experiments and modelling studies (WP 1). WP 2 covers "Gas release and rim and grain boundary diffusion experiments" and WP 3 addresses "Dissolution based release studies". This includes determining the chemical form of released radionuclides. WP 4 "Modelling" deals with modelling of release processes of fission products with respect to the spent fuel structure. Special attention is attributed to fission product migration along the grain boundaries, the effects of fractures in the pellets and of holes/fractures in the cladding. The modelling work within FIRST-Nuclides will help to clarify which geometric scales dominate the fast/instant release. WP 5 "Knowledge, reporting and training" is responsible for the knowledge management, the state-of-the-art report, and organizing training measures. WP 6 covers the management of the Collaborative Project.

CP FIRST-Nuclides started in January 1, 2012 and extends over 3 years. The achievements of the first project year are documented and published in the Annual Workshop Proceedings (KIT Scientific Report KIT-SR 7639, 2013). This volume includes summaries of the achievements within the WPs, in total 20 scientific/technical contributions by the project partners as well as abstracts of the presentations of AGs and external experts. The 2nd Annual Workshop will be held November 5-7, 2013 in Antwerp, Belgium. The workshop proceedings and further information is available from the project's web page: www.firstnuclides.eu

ABSTRACTS
Invited Lectures

CARBOWASTE: "Assessment/development of technologies and management options for irradiated-graphite and carbonaceous waste"

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The history of irradiated-graphite (i-graphite) began on December 2, 1942, with the very first self-sustained nuclear fission reaction in the Chicago Pile-1 facility. Graphite served as moderator for the fast fission neutrons and allowed the use of natural uranium as nuclear fuel, due to the extremely low cross section for parasitical neutron capture. This concept of nuclear reactors was then applied for military plutonium production and experimental piles, across the World. From the mid-1950s, the commissioning of the graphite-moderated and gas-cooled 'Calder Hall' and the 'Chinon A1' plants marked the beginning of the commercial use of nuclear energy, for Europe in the electricity market. Graphite has been used in over 100 reactors (MAGNOX, UNGG, AGR, HTR, RBMK) as well as in many research reactor types accumulating more than 250 000 tons of irradiated graphite, to date. The first generation of graphite-moderated nuclear facilities are now moving into the decommissioning phase. The sheer volume and diversity of upcoming i-graphite waste requires the identification and development of i-graphite waste management options, reflecting the varying contents of radionuclides like ^{3}H, ^{14}C, ^{36}Cl, ^{60}Co etc., due to neutron activation of natural isotopes (e.g. ^{13}C) and of impurities (e.g. ^{6}Li, ^{35}Cl, ^{59}Co) within the graphite or in the coolant gas (e.g. nitrogen).

The European Project on 'Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste (CARBOWASTE)' was launched in 2008 under the 7th EURATOM Framework Programme (FP7-211333) and terminated in March 2013. The 30-partners consortium addressed the retrieval, characterization, treatment, reuse and disposal of irradiated graphite including other carbonaceous waste such as non-graphitized carbon materials or pyrocarbon with the following main results:

- I-graphite waste features significantly depend on the specific manufacture process, ingredients (filler & binder) and residual impurities
- The nuclear reactor operational conditions (neutron dose, atmosphere, temperature etc.) strongly determine the i-graphite waste characteristics
- Radiolytic oxidation of graphite core internals leads to partial releases of activation products during reactor operation
- The neutron activation process generates significant recoil energies breaking pre-existing chemical bonds resulting in dislocations of activation products and new chemical compounds
- Most activation products exist in different chemical forms and at different locations
- I-graphite can be partly purified by thermal and chemical treatment processes leaving more stable waste products
- Leach tests and preliminary performance analyses show that i-graphite can be safely disposed in a wide range of disposal systems, after appropriate treatment / conditioning

The 'CARBOWASTE Toolbox' that is underpinned by Multi-Criteria-Decision Analysis (MCDA) methodology allows the assessment of different choices for retrieval, treatment, storage, conditioning and disposal of i-graphite with regards to ecologic, economic and social issues. The results will be used within the CARBOWASTE partner countries and beyond in collaboration with IAEA to optimize the national i-graphite management strategies and extend related international cooperations.

A 'programme-related' logic should be established for common future R&D under 'Horizon 2020' supporting the national programmes on i-graphite management and moving the new treatment and conditioning options from laboratory to pilot or industrial scale (**CarboSOLUTIONS**).

ABSTRACTS
Invited Lectures

ReCoSy: "Understanding of redox phenomena controlling the long-term release/retention of radionuclides in nuclear waste disposal"

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The EURATOM FP7 Collaborative Project *REdox phenomena COntrolling SYstems (RECOSY)* started in April 2008 and extended over 4 years. Main objectives of RECOSY were related to the improved understanding of redox phenomena controlling the long-term release and retention of radionuclides in nuclear waste disposal and providing tools to apply the results to Performance Assessment and the Safety Case. The RECOSY consortium had 32 Beneficiaries, 6 Associated Groups to RECOSY and support of an End-User Consultancy Group. The work program had six RTD workpackages (WP1-6) covering near-field and far-field aspects as well as all relevant host-rocks considered in Europe. Workpackages included "Development of redox determination methods", "Redox response of defined and near-natural system", "Redox reactions of radionuclides", "Redox processes in radionuclide transport" and "Redox reactions affecting the spent fuel source-term".

The four Annual Project Workshops were key-activities of the project where 67 poster presentations and 129 oral contributions were presented. Workshop Proceedings, published as KIT Scientific Reports, include 71 reviewed S&T papers. The RECOSY Intercomparison Exercise produced a report on "Intercomparison of Redox Determination Methods on Designed and Near-Natural Aqueous Systems". Work performed within RECOSY has been presented in 144 contributions to 77 conferences, workshops and seminars. More than 46 papers have been published in peer-reviewed journals with 10 more papers submitted or under review as of spring 2012. Another 28 publications to peer-reviewed journals are under preparation and 19 publications contributed to internal reports or PhD thesis. 36 students and 19 Post-Doc researchers were involved in RECOSY, some partially supported by the RECOSY Mobility Measures tool. Women represented 40-45% of these young researchers involved in RECOSY. RECOSY has set up a project webpage at http://www.recosy.eu where further information is made available.

The RECOSY project has provided a significantly improved understanding of Redox Controlling Systems with implications for Performance Assessment and Safety Case. Within this contribution to EURADWASTE, we present an overview of the RECOSY project including a discussion of main project features and structure, discuss important S&T results and summarize general conclusions from RECOSY.

SKIN: "Investigation of slow processes in close-to-equilibrium conditions in water/solid systems and their the impact on the mobility of radionuclides from radioactive waste geological repositories"

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Solid/liquid chemical equilibrium hypotheses (sorption, solubility, solid-solution formation) are key concepts in the assessment of nuclear waste safety. For radionuclides at trace concentrations this corresponds to constant solution concentrations, or solid/liquid distribution ratios, if environmental conditions remain constant. However, these concepts do not account for irreversible incorporation of radionuclides in the solid phases. Indeed, there is often a gradual and very slow transition from simple adsorption processes to incorporation of trace elements in the surface structure of solid phases. For certain tetravalent actinides, apparent solubility equilibrium applies to only the surface without bulk phase equilibrium. This can dramatically increase uncertainties in solubility values and derived thermodynamic constants. Equilibrium concepts are characterized by a dynamic state of equal forward and backward reaction rates, under conditions where phase compositions remain constant. Most of the problems arise from a lack of understanding of the dynamics of slow processes close to equilibrium, specifically in the coupling of sorption with other surface reactions such as dissolution/precipitation, recrystallisation, isotopic exchange in relation with the bulk phase equilibrium.

The project SKIN aims to study slow processes influencing radionuclide mobility in close-to-equilibrium conditions in a detailed and systematic manner in relation to surface properties, surface site detachment/attachment kinetics, irreversible sorption and surface incorporation, for cases relevant to the assessment of radionuclide mobility (Ra, Tc, U, Sr, Se…) in nuclear waste repository sites. The development of methods using adsorption from isotopically spiked solution at equilibrium to assess reversibility of solubility equilibria and to quantify the number of surface sites participating in the establishment of equilibrium is important in this project. Models are developed to describe this behaviour.

As an example of the major results of the project, radium released from the nuclear waste may react with barium and sulfate from the repository to form barite $Ba(SO_4)$. The ^{226}Ra uptake by barite at ambient conditions and 90°C was studied in batch experiments with a special focus on the radio-barite recrystallization mechanism, the solid-solution composition, and the Ra uptake rates. The recrystallization experiments showed a very efficient uptake of Ra by barite. Microscopic observation of samples taken at various times during the experiments indicates a significant impact of the presence of Ra on the recrystallization of barite. In addition, TOF-SIMS analyses of the final radiobarite powders indicate a full recrystallization leading to a homogenous distribution of Ra within the particles. The experimental data can be fitted with a solubility product of the $RaSO_4$ end-member in supporting thermodynamic modelling calculations using the GEM-Selektor code (http://gems.web.psi.ch). These studies demonstrate the importance of accounting for the formation of solid solutions, and in a more general way, the relevance of local/partial equilibrium concepts in describing solid-solution solubility with implications for the safety analyses.

ABSTRACTS
Invited Lectures

How the surface properties of spent nuclear fuel materials influence dissolution: Results of the REDUPP project

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The safety case of a spent nuclear fuel repository relies on assessments of the features and processes which are to provide containment and retardation of radionuclides in the repository and the geosphere. One process which plays a significant role in these assessments is spent fuel dissolution, which begins when water intrudes into a breached canister. The REDUPP project presented here aims to reduce some uncertainties that are still left regarding the dissolution processes. The interactions between the solid surface and the fluid during the dissolution process are complex. A number of remaining questions relate to basic scientific research, however, they are also in direct connection to specific questions relating to the safety assessment of the repository. In the REDUPP project a combination of dissolution testing, surface characterisation and theoretical modelling of the dissolution process is enhancing our understanding of the long-term changes in dissolution rate of spent nuclear fuel.

The processes occurring at the solid-fluid interface of the spent nuclear fuel are affected by the compositional and structural complexity of the material as well as the radioactivity. Some remaining question marks relate to how results of laboratory experiments can be translated to the real repository environment. The REDUPP project addresses two aspects of this question: 1) how do we extrapolate the results of laboratory dissolution experiments to the very long times the process will be active? 2) will the process be affected by the complex composition of the real ground water? By answering these questions, we have taken another step towards a general model to be used in further nuclear fuel dissolution research.

The main component of nuclear fuel is uranium dioxide (UO_2), which has the fluorite crystal structure. The REDUPP research is focused on a succession of materials with the fluorite structure: CaF_2, CeO_2, ThO_2 and UO_2. Laboratory dissolution experiments are mainly performed using fragmented or powdered samples that contain sharp edges and defects. During dissolution, material is more rapidly dissolved from these surface sites than from smoother areas, so that the fragments become more rounded as the experiment proceeds. This 'ageing' of the sample surface is likely affecting the measured dissolution rate. The integration of laboratory results with first principles calculations aims to provide a model of how the surface evolves during dissolution. In addition, the effect of trace elements in the solution is studied using real groundwater in dissolution experiments with UO_2. Previous experiments involved solutions which did not contain all of the chemical elements that occur in natural ground waters. The effect of the chemical complexity of natural waters on the dissolution rate is a non-trivial uncertainty that needs to be reduced. These studies combined will advance the understanding required for predicting long-term dissolution rates of spent nuclear fuel, a key parameter in performance prediction of a final repository.

FORGE: "Investigation of gas generation and migration processes for improving performance assessment of repositories"

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Understanding the behaviour of gases in the context of radioactive waste disposal was the focus of the FORGE Project. Of particular importance in the disposal of radioactive waste are the long-term performance of bentonite buffers, plastic clays, indurated mudrocks and crystalline formations. FORGE has provided experimental data to reduce uncertainty relating to the quantitative treatment of gas in performance assessment. This has been achieved through a series of laboratory, field-scale experiments (performed at a number of underground research laboratories throughout Europe) and modelling, including the development of new methods for up-scaling allowing the optimisation of concepts through detailed scenario analysis. It is important to understand a system to an adequate level of detail to allow confidence in the assessment of site performance, recognising that a robust treatment of uncertainty is desirable.

The results of a series of long-term laboratory experiments to examine the mechanisms controlling gas flow and pathway sealing in the Callovo-Oxfordian Claystone (COx), the proposed host rock for the French repository, demonstrate that advective gas flow is accompanied by dilation of the samples (i.e. the formation of pressure induced micro-fissures) at gas pressures significantly below that of the minimum principal stress. Flow appears to occur through a local network of inherently unstable pathways, whose properties vary temporarily and spatially within the claystone. The coupling of parameters results in the development of significant time-dependent effects, impacting many aspects of COx behaviour, from gas breakthrough time, to the control of deformation processes. Variations in gas entry, breakthrough and steady-state pressures are indicative of microstructural heterogeneity which may exert an important control on the movement of gas. With the availability of enhanced data sets from FORGE understanding of these processes improves and a new conceptual model for advective gas flow in COx has emerged. This is one in which the onset of gas flow and the hydromechanical response of the material are integrally linked.

Data collected during the study of gas migration in bentonite clearly demonstrate a strong coupling between total stress, pore-pressure and applied gas pressure. In tests so far completed, the evidence is for gas migration through saturated bentonite by way of dilational pathways. This provides more evidence that, in some circumstances, gas flow through clay rich materials is, at least partially, through dilatant pathways.

Experimental validation of critical stress theory applied to repository concepts has greatly increased our understanding and database of fracture flow properties. Experiments have highlighted the importance of stress-history on the flow properties of fractures evidenced by the considerable hysteresis observed during testing. Shearing has been shown to be a very effective self-sealing mechanism. Repeat gas injection testing has shown repeatability in "gas entry" values, but considerable differences have been seen in gas peak pressures. The effect of healing of the interfaces between manufactured bentonite blocks has been demonstrated by measuring the shear strength properties of the healed interface. The observation of significant cohesion confirms the "real" healing of the interface.

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CATCLAY: "Studies for improving the understanding of cation migration processes in Clayrocks"

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Clayrock formations are under consideration as host formations and geological barriers for deep geological repositories for radioactive wastes because of their very low permeability and capacity to retain most radionuclides by adsorption on their clay minerals. The dominant transport mode in these materials is diffusive and depends mainly on parameters such as the mobility of the species in water, the accessible porosity, the pore space geometry and retardation, if any. While the scientific basis for explaining and modelling the diffusion-driven transfer of non-sorbing anionic species can be considered as solid for all spatial scales considered in safety cases, the picture is not so clear for cationic radionuclides, especially for strongly-sorbing ones. Indeed, results obtained in the framework of the FP6 'Funmig' Project indicated that these species, known to form highly stable surface complexes with sites on mineral surfaces, would migrate more deeply into clay rock than calculated by models based on coupling bulk solution diffusion characteristics and sorption behavior measured on dispersed material.

The CATCLAY project has the overall objective of providing the experimental evidence and phenomenological understanding needed to justify the parameter values used in safety cases for representing diffusion-driven transfer of actinides and other strongly-sorbing cationic radionuclides in clay rocks under consideration for hosting radioactive waste disposal facilities (Opalinus Clay in Switzerland, Callovo-Oxfordian claystones in France, Boom Clay in Belgium). The scientific approach is of the 'bottom-up' type, in which simpler, analogous systems (here a compacted clay, 'pure' illite) are experimentally studied and modelled, and then the transferability of these results to more complex materials (the three above-mentioned clay rocks) is verified. Since the June 2010 start-up of this 4-year-long project, much effort has been devoted to developing and applying specific experimental methods needed for acquiring the high precision, reliable data needed to test the alternative hypotheses represented by different conceptual-numerical models. The enhanced-diffusion of sorbing cations such as Zn(II) and Co(II) has been confirmed and the robustness of the proposed models capable of reproducing their diffusive behaviour under multiple experimental conditions is being evaluated.

PEBS: "Improvement of the long-term performance prediction of clay-based engineered barriers systems"

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The main aim of the project PEBS (Long-term Performance of the Engineered Barrier System) is to evaluate the sealing and barrier performance of a clay-based EBS with time, through development of a comprehensive approach involving experiments, model development and consideration of the potential impacts on long-term safety functions. The experiments and models cover the full range of conditions from initial emplacement of wastes (high heat generation and EBS resaturation) through to later stage establishment of near steady-state conditions, i.e. full resaturation and thermal equilibrium with the host rock.

The consortium involves 15 European organizations as well as the Beijing Research Institute for Uranium Geology.

The work performed within the project builds on existing knowledge and experience generated during recent years and supported by on-going national and prior EC research programmes. In particular, the project takes advantage of a broad spectrum of large-scale laboratory and in situ experiments, some of which have been in progress for many years (e.g. the FEBEX full-scale laboratory mock-up and the associated in situ experiment at Grimsel, as well as the EB experiment at Mont Terri) in addition to new in situ experiments that were developed specifically for the project (e.g. the half-scale heater experiment with clay-based barrier at Mont Terri).

Results of experimental and laboratory studies are integrated into coupled THMC (thermal, hydraulic, mechanical, and chemical), HM and THM models of the processes in the engineered barrier system and near-field rock. In order to integrate the information into the context of performance of engineered barrier systems in actual planned configurations, a number of specific cases have been defined that represent a combination of a configuration (the defined EBS with its initial conditions) and the description of an evolution of the EBS reflecting an identified uncertainty of process-based transient repository evolution. These are Case 1 – Uncertainty in water uptake in the buffer (T<100°C), Case 2 – Uncertainty in T evolution in the buffer (T>100°C), Case 3 – Uncertainty in HM evolution and Case 4 – Uncertainty in geochemical evolution.

For each case the importance of uncertainties arising from disagreement between models and experiments and implications for extrapolation of results over the transient phase are being assessed and the possible impact on safety functions over the short and long term are being evaluated. In order to come to a comprehensive outcome, this requires structured discussions between experimentalists, modellers and safety assessors and engagement from everyone to place specific knowledge in a wider context.

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BELBaR: "Investigation of erosion processes in bentonite engineered barriers systems of a repository in crystalline rock and their impact on the long-term performance of the repository"

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BELBaR is a Collaborative Project within the Seventh Framework Programme of the European Atomic Energy Community (Euratom) for nuclear research and training activities. The main aim of BELBaR is to increase knowledge of the processes that control clay colloid stability, generation and its ability to transport radionuclides. The overall purpose of the project is to come up with a new way of treating issues in long-term safety/performance assessment for radioactive waste repositories. The project started March 1, 2012 and has a duration of 48 months. The project has 14 partners from seven European countries.

The main aim of BELBaR is to reduce the uncertainties in the description of the effect of clay colloids on the long term performance of the engineered barrier and on radionuclide transport. This is done by:

- Improving the understanding on when bentonite colloids are unstable.
- Improving the quantitative models for erosion on the bentonite barrier for the cases when the colloids are stable.
- Improving the understanding of how radionuclides attach to clay colloids. This information will be used to formulate improved transport models for the assessment of radionuclide transport in the geosphere.

To meet the main aim a number of experimental and modelling activities will be undertaken within the project. BELBaR consists of six RTD (research and technical development) work packages and one project management work package.

WP1 will have the responsibility to ensure that that the type and values of the parameters selected for experimental and modelling work are those that represent as much as possible the full range of conditions and situations that can be expected in a repository. Drawing on the work undertaken in WP 2 to 5, the general objective of this work package is to consider how colloids and related phenomena can be considered in the long-term safety case, to 1 million years following the closure of a geological repository, and to make recommendations on the quantitative and qualitative approaches that a safety case could pursue to adequately address this issue.

The main objective of WP2 will be to understand the main mechanisms of erosion of clay particles from the bentonite surface and to quantify the (maximum) extent of the possible erosion under different physico-chemical conditions. Additionally, these studies will point out under what conditions compacted bentonite is able to produce colloidal particles, free to move into the contacting aqueous phase and to determine the bentonite colloids "source-term". Potentially relevant physico-chemical factors to the bentonite erosion processes will be studied in an extensive experimental programme.

Clay colloids potentially generated in the nuclear waste repository near-field from the bentonite-buffer/backfill material might be stable under the geochemical conditions of the fractured rock far-field and could be a carrier of radionuclides. WP3 will focus on the process understanding of colloid mobility controlling processes and their appropriate description, the reversibility of the sorption of radionuclides onto colloids and the kinetics of the sorption/desorption process as well as the identification of additional retention processes. The potential release of divalent cations (Ca^{2+}, Mg^{2+}) via matrix diffusion is expected to increase the colloid attachment probability and reduce the colloid mobility.

WP4 focuses on clay colloid stability studies under different geochemical conditions with respect to ionic strength and pH. The colloids formed at the near/far field interface would be stable only if favourable conditions exist and therefore their relevance for radionuclide transport will be strongly site-specific. The experimental work will be concentrated on effects of removing colloidal particles from the liquid phase (such as reaching critical coagulation concentration, the effect of surfactants, coagulation) and understanding the influence of complexing agents (organic/humic substances) on clay colloid stability to reduce the uncertainty of naturally occurring organic matter being present.

The objective of WP5 is to validate and advance the conceptual and mathematical models used to predict mass loss of clay in dilute waters and clay colloid generation as well as clay-colloid-facilitated radionuclide transport relevant to geological disposal of high level nuclear waste. Validation of the current conceptual as well as mathematical models is pending. The target is to obtain validated advanced models for the purposes of geological disposal of high level nuclear waste.

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CROCK: "Investigations of uncertainties in radionuclide transport processes in the far-field of a repository in crystalline rock"

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The EURATOM FP7 Collaborative Project "Crystalline ROCK retention processes" (CP CROCK, grant agreement No. 269658) is established with the overall objective to develop a methodology for decreasing the uncertainty in the long-term prediction of the radionuclide migration in the crystalline rock far-field. The project is launched in response to the need identified in conjunction with selection of retention data for the forthcoming crystalline host-rock HLW disposal Safety Case. The process of selecting a set of data for this purpose showed that the spread in data is broad and cannot presently be related to material properties or processes. Consequently, very conservative numbers need to be used in order to be defendable within the Safety Case. This does not lead to unacceptable dose predictions, but remains highly unsatisfactorily.

The project makes use of the broad set of existing analytical approaches, methodologies, and general knowledge from decades of past investigations. It builds on the output and main conclusions of the 6th FP IP FUNMIG project and the Swedish site selection program. The experimental program reaches from the nano-resolution to the Performance Assessment (PA) relevant real site scale, delineating physical and chemical retention processes. Existing and new analytical information provided within the project is used to set up step-wise methodologies for up-scaling of processes from the nano-scale through to the PA relevant km-scale. Modelling includes testing up-scaling process and parameters for the application to PA and in particular, the reduction of uncertainty.

The scientific-technical work program of the project is structured along six RTD work packages (WP). WP1 started at the very beginning of the project providing new drill core fracture samples and characterizing the experimental materials. WP2 focuses on radionuclide transport and sorption studies. WP3 deals with matrix diffusion and natural chemical homologue analysis. The general objective of WP4 is to conceptualize and model radionuclide transport processes on systems at different scales. In WP5 is described how the outcome of the other WPs can contribute to decrease the uncertainty in PA related with transport treatment. WP6 is a cornerstone of the project, since its first objective is to establish a state-of-the-art of the current knowledge on retention processes in crystalline rocks, then to continuously collect the results obtained in the other WPs, and finally to deliver a report summarizing the major advances which will have been accomplished at the end of the project. There is also one WP on knowledge management, dissemination and training (WP7). The last WP is on administrative and financial project management (WP8).

The project started on 1st January 2011 and will last until June 2013. CP CROCK is implemented by a consortium with 10 Beneficiaries consisting of large European Research Institutions, Universities and SME's from countries with dedicated crystalline host-rock disposal programs and particular competence in this field. National Waste Management organizations participate as associated groups, contributing with co-funding to beneficiaries, infrastructure, knowledge and information. They also contribute together with national regulators to guidance with respect to application of the project to the disposal Safety Case and scientific-technical review. The project is open for additional organizations entering into formal cooperation via Associated Group agreement.

More information about the project and the scientific output can be found on the project webpage: http://www.crockproject.eu/

SESSION V

Challenges in implementation of geological disposal: Licensing processes started for high-level nuclear waste repositories

Juhani Vira
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The systematic program for direct geological disposal of spent fuel in Finnish bedrock was started in the early 1980s as a response to the Government decision in 1983 on the principles and time schedule for nuclear waste management of the Finnish nuclear power plants. Since a suitable concept for disposal was already there (KBS-3), the focus of the program was on site selection. This also led to a success, since in 2001 the Finnish Parliament approved the so-called Decision-in-Principle (DiP) on siting a KBS-3 type repository for spent fuel in an area near the Olkiluoto nuclear power plant.

On the basis of the 2001 decision Posiva Oy could start the detailed design and construction of an underground rock characterisation facility ("ONKALO"), which, according to Finnish regulations, is needed before submission of the application for the construction license of the repository. The ONKALO, consisting of a tunnel and three shafts reaching to the planned disposal depth of 420 m, has now been used for various rock characterisation activities and technical tests for more than eight years. In December 2012 Posiva submitted an application for the construction license of a disposal facility consisting of an above-the-ground encapsulation facility and an underground repository. According to the application ONKALO would be used as an access way to the repository.

The application is now being reviewed by the licensing authorities. The first comments from the regulator, STUK, on the formal completeness and needs for supplementary documentation are already expected in early spring 2013 and more detailed comments, with possible requests for additional information, in Summer 2013. According to STUK, if Posiva is able to provide the documents and information requested by the end of the year 2013, they will be able to finish the review by mid-2014 and the license could be granted towards the end of 2014.

Meanwhile, Posiva is preparing detailed plans for the construction of the disposal facility. A summary of these plans is included in the 3-year nuclear waste management program "YJH-2012" published in September 2012 (in Finnish, the English translation forthcoming). Assuming that no major issues arise in the review of the application for construction license, the current plans envisage the possibility of submission of the operating license application by 2020.

In general, the main challenges for the implementation of the disposal facility relate to the fact that the Olkiluoto repository may become the first nuclear high-level waste repository in the world. From the implementer's point of view the potential issues relate to the definition of all technical requirements of the disposal facility in a way that allows the demonstration of their compliance through inspections based on observations and measurements at manufacturing and installation and the subsequent modelling predictions of the future evolution of the repository system. From the regulator's point of view the issue is the level of proof desired. The successful licensing process requires that a reasonable resolution is reached on the acceptable level of residual uncertainty. Compared with the licensing of nuclear reactors, a complicating fact is that the back-fitting of improvements to the repository system after the disposal tunnels have been closed may be difficult without entailing major costs.

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LUCOEX: "State-of-Art Demonstrations of Geological Disposal for High Level Waste Large Underground Concept Experiments"

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The overall objective of the four year LUCOEX project is to demonstrate the technical feasibility *in situ* for safe and reliable construction, manufacturing, disposal and sealing of repositories for long-lived high-level nuclear waste. The demonstration activities in the project take place in four different underground research laboratories (URL) in Europe, which have been constructed for the specific purpose of developing repository technology under repository-like conditions. The demonstrations include four different geological deposition concepts, which have been developed individually to the present state-of-the-art in national programmes. We are now combining knowledge and practical experiences to develop the demonstration installations and these concepts further in cooperation.

The four addressed repository concepts are:

- Horizontal disposal of waste packages in Opalinus Clay formation
- Horizontal disposal of waste packages in Callovo-Oxfordian clay formation
- Horizontal disposal of waste packages in crystalline hard rock
- Vertical disposal of waste packages in crystalline hard rock

In order to establish confidence in the technical feasibility of constructing and operating the four repository concepts the following key technical issues are addressed in LUCOEX:

- Gallery construction
- Manufacturing and emplacement of buffer around waste canisters
- Emplacement of waste packages
- Backfilling and sealing of galleries

The quality and use of obtained technical achievements are enhanced by engaging many more persons in the addressed issues than the ones taking part in the actual project. This is achieved by distributing out results and findings through workshops, conferences, our webpage, scholarships for participating in the project and most importantly through a direct dialogue with interested parties. Because of this we have ensured that all four participating URLs are open for domestic and foreign engineers, decision makers and the general public interested in getting information at the actual sites of the demonstrations.

At the current state of progress of the project (LUCOEX will be completed 2015), gallery construction and instrumentation of drifts is mostly finished. We are now focusing on the manufacturing of buffer material and development of the necessary support equipment to be able to handle the emplacement, backfilling and sealing of the galleries during 2013. Further information regarding our key technical issues will be presented at the EURADWASTE 2013 conference and on our webpage www.lucoex.eu

DOPAS: "Full-scale in situ demonstration of tunnel plugs and shaft seal in clay, crystalline and salt repository host-rock formations"

Johanna Hansen
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Fourteen nuclear waste management organisations and research institutes from eight European countries are participating in a technology development project for assessing tunnel plugging and sealing systems in geological disposal facilities for radioactive waste - the DOPAS project ("Full-Scale **Demonstration Of Plugs And Seals**"). The project is built around a set of full-scale demonstrations, laboratory experiments, and performance assessment studies and is jointly funded by the Euratom's Seventh Framework Programme and European nuclear waste management organisations. The project is running from September 2012 to August 2016, and is being coordinated by Posiva Oy, the nuclear waste management company in Finland.

DOPAS project aims to improve the adequacy and consistency regarding industrial feasibility of plugs and seals to be used in different geological environments. The main challenges are related to: 1) site selection and construction technologies, 2) new material development, 3) in situ instrumentation and performance assessment, and 4) quality and safety assurance.

Selection the location for hosting the plug and its excavation, which will be an ordinary industrial process in future disposal facilities, has been tested in full-scale only to a limited extent. During the planning phase for plug location siting and excavation, best practices are established in the project related to occupational and operational safety, selection of excavation methods, and design of unique tunnel reinforcement to preserve the initial geological conditions. One of the experimental sites will host an actual future disposal facility for spent fuel, thus setting stricter requirements on documentation, quality management procedures and selection of materials.

The material development includes advances in cement-based and bentonite-based components used for plugs and seals. A comprehensive laboratory programme is required in order to ensure the specifications to be used in full-scale experiments. Low-pH cementitious materials have been developed in several projects for more than a decade, but still their use at full-scale requires modifications in mixes and laboratory verification of their properties before field use at a decametric scale. Laboratory analyses for mechanical and chemical evolution of the materials' performance, such as pH verification, takes several months. The laboratory information is also needed, as source data for the assessment of behaviour over time in repository conditions, in the early stage of the project. The production of bentonite components for large-scale tests requires understanding of manufacturing and emplacement processes, including quality assurance, storage and transport of the materials, ensuring achievement of planned design and accounting for the interactions between cementitious- and bentonite-based components in field conditions.

Instrumentation and monitoring of full-scale experiments is required to gain information on plug feasibility, but also for assessment of the plug and seal behaviour during accelerated testing conditions. Great emphasis is put on planning monitoring needs and techniques, to ensure optimal use of results for performance and safety assessments. During the project it will be beneficial to evaluate the level of uncertainty arising in the demonstrated concepts. Possible disturbances by monitoring are mitigated by testing new techniques, such as wireless sensors.

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The DOPAS project with full-scale experiments provides the basis for future needs related to plug and seal technologies for nuclear waste management. Outcomes from the DOPAS project will be disseminated via a web site, public reports, papers and presentations at various conferences. Knowledge transfer will benefit from planned workshops and seminars as well as staff secondment.

Overview of the MoDeRn project: "A reference framework for developing a monitoring programme"

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This paper provides an overview of the MoDeRn projects main goals and results. The main goal of this collaborative, European Commission 7th Framework project is to take the state-of-the-art of broadly accepted, main monitoring objectives and to develop these to a level of description that is closer to the actual implementation of monitoring during the staged approach of the disposal process. It should be noted that the MoDeRn project recognizes the diversity of monitoring activities that will be required in a repository, in particular related to operational safety and environmental impact assessment. The project's emphasis, however, is on verifying – with the aim of confirming and possibly enhancing the prior license basis for safety and pre-closure management options – expected repository system evolutions (i.e. natural environment and engineered system evolutions) during a progressive construction, operation and closure phase that may last on the order of a century. To achieve this goal, 18 partners representing 12 countries and including 8 Waste Management Organizations joined their efforts since 2009, and have developed a "roadmap to repository monitoring".

MoDeRn has progressed on both the associated Process issues – why to monitor, how to develop a program, and how to use monitoring results – and Technology issues – technical requirements and constraints, technology state-of-the-art, and focused R&D and in situ demonstrators. To achieve progress on process issues, the basis for a structured development of monitoring programmes was established, focused on justifying and proposing key objectives; how to attain them; and how to use monitoring results to assist decisions of disposal process management. Their application was tested through developments of case studies in various host rocks. Furthermore, the results of a focused sociological study provide a basis for associated stakeholder engagement activities as well as a better understanding of whether and how monitoring will contribute to enhancing confidence in and acceptance of the disposal process.

To achieve progress on the technology issues, an overview of typical technical requirements and constraints was developed. Further on, a technical workshop involving other monitoring Research and Technology Development (RTD) projects was hosted to identify RTD techniques that enhance our ability to monitor deep geological repositories. An overview of monitoring technology state-of-the-art was established and several focused R&D and in situ monitoring experiments were launched.

The research leading to these results has received funding from the European Atomic Energy Community's Seventh Framework Programme (FP7/2007-2011) under grant agreement n° 232598.

JOINT CONCLUSION

Safety Framework: Potential contribution of Euratom Research & Training

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The FISA-2009 formulated a number of conclusions regarding current status, main achievements and key remaining research issues in the two areas of interest (Safety and Competitiveness of Nuclear Installations, and Education and Training and Large Infrastructures). It also appreciated the new SNETP (Sustainable Nuclear Energy Technology Platform), step towards strengthening the cooperation and integration across the EU, and its "Strategic Research Agenda" (SRA).

Since, SNETP has prepared several additional strategic documents. However, the main SNETP activity was structured around three main pillars, and resulted in:

- NUGENIA, launched in March 2012, an international association mandated by SNETP. Its main role is to develop R&D supporting safe, reliable, and competitive GEN II and GEN III systems;

- The European Sustainable Nuclear Industrial Initiative (ESNII) launched in November 2010, promotes advanced fast reactors, with the demonstration plans defined in its Concept Paper;

- The Nuclear Cogeneration Industrial Initiative (NC2I) aims at demonstrating an innovative and competitive energy solution for the low-carbon cogeneration.

The Fukushima accident, on March 11th, 2011, confirmed the importance and value of past research enabling, among other, to keep the leading position of Europe in reaction to it (Stress Tests) and was followed by many others countries. The GB of the SNETP decided to establish a Task Group with the main objective to assess implications of the accident on the medium and long term research and development Platform's programme. Results are summarised in its report "Identification of Research Areas in Response to the Fukushima Accident".

The importance of the role of nuclear energy is acknowledged also in the EC Communication – "Energy roadmap 2050". Electricity is expected to play a greater role in the future energy supply, and it also calls for a new electricity market model. Also the EU economical and financial crises, security of supply challenged in winter 2011/12 and electricity grids stability are new challenges for energy R&D as whole.

All this lead in this year to several actions among other:

- Interdisciplinary Study. Benefits and limitations of nuclear fission for a low-carbon economy. Defining priorities for Euratom fission research & training, followed by February Symposium in Brussels;

- New updated SNETP "Strategic Research and Innovation Agenda", issued in February 2013;

- Contribution to the preparation of an *Integrated Roadmap* under the guidance of the SET Plan.

The FISA 2013 reviewed the past four year's activities and made main recommendations for the following period. It underlines the importance of safety aspects, which must be integral part of all R&D activities in nuclear fission as well as focused on the societal challenges of the society and citizen needs. It gives important recommendations for the new Work Programmes under the Research and Training Programme of the European Atomic Energy Community (2014-2018) complementing the Horizon 2020.

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Waste Directive: Potential Contribution of Euratom Research & Training and of the Implementing Geological Disposal Technology Platform (IGD-TP)

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Despite the fact that the Council Directive 2011/70/Euratom covers all steps of radioactive waste management, geological disposal as a safe long-term management solution for spent fuel, high-level waste and other long-lived radioactive waste remains the key societal and technological challenge. The potential contribution of the Euratom Research & Training programme to the implementation of the Directive 2011/70/Euratom should therefore continue to be focussed on geological disposal and also consider a series of recent evolutions at national and European levels in the development of this solution.

Recent evolutions in the development of geological disposal and of supporting RD&D

Amongst the recent evolutions in the development of geological disposal and of supporting R&D, the following ones could influence the future roles of Euratom Research &Training programme:

- Several European Member states are approaching implementation of geological repositories and therefore need to shift part of their R&D towards demonstration (i.e. Research, Development and Demonstration – RD&D), operational issues, industrialisation, optimisation...;

- At the same time, numerous newcomer Member states are at the very early stages of the development of their disposal programmes and need to build their own expertise while maximising the transfer of knowledge from more advanced programmes;

- There is a need to establish and disseminate a shared scientific and technical base for geological disposal;

- There is an increased acknowledgement of the need and interest to involve the public at large throughout the development and implementation of disposal solutions, including at the RD&D stage; as a consequence, the public's concerns should be considered as one of the bases in the definition of RD&D activities;

- There has been a shift in R&D multilateral cooperation mode between Member states, from national based projects to shared priorities and pooling of resources. This shift has notably resulted in the creation of the Implementing Geological Disposal of Radioactive Waste – Technology Platform (IGD–TP) which helps tackle the remaining RD&D challenges with a view of fostering the implementation of national geological disposal programmes;

- There is an increasing awareness of the need to continue RD&D well after the completion of a safety case and/or after a licence for constructing and operating a geological repository has been granted;

- The risk of a shortage, at European level and at short to medium terms, of skilled human resource is high; such shortage will impact countries that will implement geological disposal as well as countries that start developing such a management solution.

Potential roles of the Euratom Research & Training programme

The Euratom Research & Training programme could pursue its evolution towards (without being comprehensive):

- *The establishment of a common, recognised knowledge base for geological disposal*: this should notably include the consideration of interdependencies with the management phases prior to disposal, the consideration of public's concerns, a systematic scientific review process and the integration of JRC activities.

- *The dissemination of the knowledge base:* only if the dissemination goes beyond the classical radwaste community, will it help foster transparency and appropriation by the various stakeholders.

- *Fostering the integration of the concerns of all stakeholders (including the public at large) into RD&D priorities:* a bottom-up approach may raise new concerns or force the radwaste community to have a fresh look at some established knowledge. The EC, with its various established stakeholders' groups or representatives, could help bridge the gap between the public and the scientists.

- *Fostering training and maintenance of expertise over long period of time:* such goal can only be achieved if a coherent approach at Euratom/JRC level is established (rather than an effort at individual project level). It will also be helped by the recognition by the broader scientific community of the up-front quality of the work carried out in disposal programmes.

- *Transfer of technologies:* going beyond document sharing and experts' training represents a huge challenge for both Member countries and the EC. Indeed, the primary objective of advanced programmes is, and will remain, the implementation of their respective national geological repository. Such focus will become even more vivid if one considers the forecast shortage in skilled human resources as mentioned previously. Therefore, there is an opportunity for the Euratom Research and Training programme, in combination with other European initiatives, to set up a specific and innovative approach towards technology transfer, including helping to develop shared repositories.

Potential roles of the *Implementing Geological Disposal – Technology Platform (IGD-TP)*

The IGD-TP Strategic Research Agenda, that defines shared RD&D priorities with an important cooperative added value, should continue to be used as a basis for the Euratom programme as it provides a vehicle to emphasise RD&D and networking activities that are important for establishing safety cases and fostering disposal implementation, the latter being the ultimate goal of the Directive. As the IGD-TP brings together the national organisations which are legally entrusted to industrially implement geological disposal and science providers, its SRA also ensures a balance between fundamental science, implementation-driven RD&D and technological demonstration.

The need for disposal related R&D will persist for a long period of time, to ensure appropriate conception and assessment of future repositories, establishment and evaluation of related safety cases, as well as adaptative implementation and optimisation of the operation of repositories. The IGD–TP should therefore continue to contribute to a sound, broadly shared, societally accepted and transparent scientific and technological basis for geological disposal.

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POSTERS ON COORDINATION ACTION AND EDUCATION AND TRAINING AND GOVERNANCE PROJECTS

Secretariat of the Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) - SecIGD2 project

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The secretariat of the Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) - Phase 2 project (SecIGD2) is a European project co-financed by the FP7 EURATOM programme (coordination and support action). It aims to support the secretariat activities of the IGD-TP over the period 2013-2015.

The IGD-TP secretariat is part of the overall concept of implementing the IGD-TP and aims to support the IGD-TP vision and mission in its practical administrative and operational tasks so that IGD-TP mission and objectives are achieved and the benefits from the work carried out are widely spread among its participants and in Europe.

The SecIGD2 project consists of three main support action work packages (WP1-WP3) which have been set up in order to:

- Facilitate the smooth management of the IGD-TP by coordinating work and activities, organising the IGD-TP meetings and conferences. In addition, the secretariat acts as an information centre, carrying out dissemination activities, publishing and providing open access to the produced documents and maintaining a public website (WP1, led by Andra).
- Network, structure and develop Research Development and Demonstration (RD&D) competences in countries with less advanced geological disposal programmes. The main objective of this activity is to foster the transfer of strategic knowledge towards members who have less advanced programmes. A specific Working Group has been set up, SecIGD2 will support its actions, provide management guidelines and help it in the organisation of two international conferences for disseminating the public scientific and technical information and results derived from the IGD-TP activities and from other RD&D efforts (WP2, led by NDA).
- Support the development and implementation of end-user needs-based on competence maintenance, education and training activities in the field of radioactive waste management and disposal. More specifically, SecIGD2 will assist the IGD-TP Working Group "Competence Maintenance, Education and Training" (CMET) in achieving its goals, which are to identify and share the needs in knowledge, skills and competences and to discuss and study approaches for mutual recognition of learning outcomes (WP3, led by Posiva Oy).

The secretariat also ensures that synergies with existing projects (such as Sitex or InSOTEC) or platforms (such as SNETP) are generated at a European level. An additional work package (WP4) is dedicated to the project management (reporting, deliverables, financial management). The duration of the SecIGD2 project is 36 months (2013-2015). The role of secretariat general will be filled by Andra in 2013-2014 and by NDA in 2015-2016.

Sustainable Network of Independent Technical Expertise for Radioactive Waste Disposal

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SITEX is a 24 months FP7 project led by IRSN and bringing together 15 organisations representing technical safety organisations (TSOs) and safety authorities, as well as civil society outreach specialists involved in the "regulatory" review process of geological disposals for radioactive waste. SITEX aims at establishing the conditions required for developing a sustainable network of experts from various horizons (authorities, TSOs, academic organisations, civil society...) capable of developing and coordinating the technical expertise that is required from the stakeholders in charge of delivering opinion, independently from the waste management organisations (WMOs), on the safety of geological disposals.

The SITEX program of work is split into a set of workpackages that address technical and organizational issues allowing in fine to propose a structure of the missions and operating mode of the future network. A number of issues relate to the potential for sharing and developing technical expertise practices amongst stakeholders: the needs of National Safety Authorities and of TSOs for dialogue and/or guidance development and harmonization were identified, a harmonized safety review method is proposed and the practical ways to engage with civil society in expertise function on a durable process are studied. In addition, the ability to implement coordinated R&D programs run by TSOs in order to develop the scientific knowledge necessary to perform independent technical assessments constitutes an important challenge for SITEX. A set of R&D activities was structured into main key safety issues related to the safety functions and components of a deep geological disposal. This will contribute to engaging in scientific dialogue with IGD-TP in the context of the new EC H2020 program.

Two major perspectives are identified for the future of the SITEX network: its ability to foster cooperation between regulatory bodies, TSOs, implementers and civil society with the view to enhancing common understanding of key safety issues and challenges and to identifying possible harmonization of practices; the constitution of a scientific task force (mainly driven by TSOs) for research definition and implementation at European level allowing to improve the coordination of scientific programmes between TSOs and developing its own skills and analytical tools, independently of the WMOs.

From ACTINET to TALISMAN

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Understanding and being able to predict the actinides behaviour is a key issue to ensure the highest safety level all along the nuclear fuel cycle. It is therefore a key issue to increase our knowledge on their very specific chemical and physical properties and therefore to be able to assess their behaviour all along the fuel cycle. It is thus of cardinal importance to keep and increase our knowledge on actinide sciences. In a context where the number of skilled scientists in this field is decreasing due to retirement, nuclear countries have also to educate the next generation of scientists and engineers who will contribute to develop safe actinide management strategies.

However, actinides studies require specific hot laboratories and characterization tools that are only available in a limited number of places worldwide. Actually, only a few academic and research organisations have the capabilities to work on these elements under safe conditions in Europe. To ensure a sustainable development of actinides sciences, it is therefore considered as a strategic issue to coordinate the existing actinide infrastructures available in Europe, and to strengthen the community of European scientists working on actinides.

Since 2004, Europe supports the strengthening of the European actinides sciences scientific community through the funding of dedicated networks: from 2004 to 2008, the ACTINET6 network of excellence (6th Framework Programme) gathered major laboratories involved in nuclear research and a wide range of academic research organisations and universities with the specific aims of funding and implementing joint research projects to be performed within the network of pooled facilities; from 2010 to 2013, the ACTINET-I3 integrated infrastructure initiative (I3) has supported the cost of access of any academics in the pooled EU hot laboratories. In this continuation, TALISMAN (Transnational Access to Large Infrastructures for a Safe Management of ActiNides) gathers now the main European hot laboratories in actinides sciences in order to promote their opening to academics and universities and strengthen the EU-skills in actinides sciences. Furthermore, a specific focus is set on the development of advanced cutting edge experimental and spectroscopic capabilities, the combination of state-of-the art experimental with theoretical first-principle methods on a quantum mechanical level and to benefit from the synergy between the different scientific and technical communities.

The paper comes back on more than 10 years of successful collaboration in this field in Europe and presents TALISMAN in more detail.

Engineered Barrier Systems for Radioactive Waste Disposal (EBSSYN)

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The EBSSYN project has developed a Synthesis Report on a multi-year European Commission (EC) and Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA)-sponsored project on the engineered barrier system (EBS) used in geological disposal of long-lived radioactive wastes. The EBS Project examined how to design, characterise, model and assess the performance of the EBS, and how to integrate EBS issues in the safety case for disposal.

EBSSYN has been conducted by TerraSalus Limited (the beneficiary) in collaboration with an EC/NEA-coordinated Task Group comprising representatives from the EC and the NEA, as well from national radioactive waste management and disposal organisations in Canada, France, Finland, Germany, Japan, Spain, Sweden, Switzerland, the UK and the US.

TerraSalus Limited provides high-quality, professional consultancy advice on a range of environmental issues, including radioactive waste management.

In addition to producing the EBS Synthesis Report, one of the most important benefits of the project has been its role in facilitating high-level consideration, and international discussion, of how the various necessary studies on detailed aspects of the EBS should be integrated within the safety case.

The EBS has a central role in the safety case for disposal. EBS design and optimisation requires a significant programme of work, typically lasting from several years to several tens of years. During such a programme, it is essential to maintain good links all the way from the fundamental understanding of the processes and phenomena that may affect the behaviour of the wastes, the EBS materials and the host rock, to their representation in safety assessment.

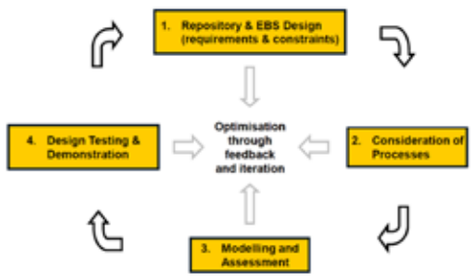
The rationale for the project was to enhance understanding of how safety cases for disposal can be used to integrate results from various activities. These activities include:

- Defining the requirements of the disposal system and the EBS.
- Understanding the materials of the EBS components and the processes that may affect them.
- Modelling the behaviour and performance of the EBS components.
- Demonstrating that the EBS can be manufactured, constructed and installed.
- Providing reasonable assurance that the disposal system will provide acceptable levels of safety.

The results of the project will be disseminated as a public-domain EUR-report and will be made available for download from the EC website and the NEA website. The final report will also be widely disseminated by the EC and the NEA to stakeholders worldwide.

The key result from EBSSYN is a highly visible international publication that addresses directly many issues associated with performance and safety assessment for geological repositories, as well as giving high-level guidance on approaches to repository design and implementation.

The project has led to a better understanding of the EBS design process. EBS design and optimisation is necessarily an iterative process that follows an initial step of defining the safety strategy for disposal. The optimisation process then involves a range of studies as illustrated in Figure 1.



The project has illustrated aspects of each of these steps and their combination by examining a series of examples and trends in radioactive waste disposal.

The EBS Synthesis Report should assist radioactive waste disposal programmes to proceed in a better informed, well-integrated and planned manner and, therefore, have an increased chance of success.

Sharing of knowledge and best practices between different disposal programmes should increase stakeholder confidence, help to ensure the safety of waste disposal, and reduce the costs and emissions associated with necessary research and development works.

The results from the project should be of interest to waste disposal organisations, regulatory authorities and stakeholders in radioactive waste generation and disposal.

CINCH–II:
Next Step in the Coordination of Education in Nuclear- and Radiochemistry in Europe

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Any of the potential options for the nuclear power – both the renaissance, if any, or the phase out – will require significant numbers of the respective specialists, amongst others the nuclear and/or radiochemists. In parallel, a significant demand exists for these specialists in non energy fields, such as environmental protection, radiopharmacy, nuclear medicine, biology, authorities, etc. Since the numbers of staff in teaching and the number of universities with facilities licensed for the work with open sources of ionizing radiation has decreased to or sometimes even below the critical level, coordination and collaboration are required to maintain the necessary teaching and training capabilities.

The CINCH-II project, aiming at the Coordination of education and training In Nuclear CHemistry in Europe, is a direct continuation of the CINCH-I project which, among others, identified the EuroMaster in Nuclear Chemistry quality label recognized and guaranteed by the European Chemistry Thematic Network Association as an optimum common mutual recognition system in the field of education in Nuclear Chemistry in Europe, surveyed the status of Nuclear Chemistry education and training in European universities and industries including the needs of the end-users, developed an efficient system of education/training compact modular courses, or developed and tested two electronic tools – NukWik (an open platform for collaboration and sharing teaching materials in nuclear- and radiochemistry based on a wiki engine) and CINCH Moodle (an e-learning platform) – as a basis of a future efficient distance learning system.

The expected outcomes of the follow-on CINCH-II project will be described in detail. The CINCH-II project is built around three pillars - Education, Vocational Education and Training (VET), and Distance Learning - supported by two cross-cutting activities – Vision, Sustainability and Nuclear Awareness that includes also dissemination, and Management. Its main objectives, expected to have the broadest impact to the target groups, are further development and implementation of the EuroMaster in Nuclear Chemistry, completion of a pan-European offer of modular training courses for the customers from the end users, development of a Training Passport in Nuclear Chemistry and preparing the grounds for the European Credit system for Vocational Education and Training (ECVET) application in nuclear chemistry, implementation of modern e-learning tools developed in CINCH-I and further development of new tools for the distance learning, laying the foundations of a Nuclear Chemistry Education and Training Platform as a future sustainable Euratom Fission Training Scheme (EFTS) in Nuclear Chemistry, development of a Sustainable Systems for Mobility within the Nuclear Chemistry Network, or development of methods of raising awareness of the possible options for nuclear chemistry in potential students, academia and industry. The CINCH-II project will mobilize the identified existing fragmented capabilities to form the critical mass required to implement the courses and meet the nuclear chemistry postgraduate education and training needs, including the high-level training of research workers, of the European Union. Networking on the national level and with existing European as well as international platforms will be an important feature of the project.

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Research and Development Programmes on Geological Disposal as Networks for the Co-Production of Knowledge

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This paper aims at a better understanding of the networks around research and development (R&D) on geological disposal at the national level. In particular, we want to know whether national R&D programmes aim at joint knowledge production between different disciplines and stakeholders and to what extent this enables socio-technical knowledge to be addressed. This paper is conducted as part of the InSOTEC project, a three-year (2011-2014) collaborative research project funded under the 7th Euratom Framework Programme (Grant Agreement nr. 269906).

Work so far has identified common socio-technical challenges to the implementation of geological disposal of radioactive waste in fourteen countries. Science and technology studies provide the conceptual underpinnings for this endeavour. The results of this analysis suggest that in some countries, socio-technical challenges are only discussed in techno-scientific circles.

In this paper, we explore and map out R&D networks around geological disposal and the extent to which socio-technical research is being explicitly recognised and addressed. We look at the contributions and limitations of technically driven programmes compared to more comprehensive R&D programmes which address social and technical challenges in a more integrated way. The absence of stakeholder involvement in different phases of the R&D process might lead to the separation of social and technical aspects in R&D. Even in some cases, societal R&D might be understood as being part of public relations or public information campaigns of the implementer. Countries with more advanced geological disposal programmes, then again, seem to show a greater awareness of the socio-technical character of research. Social science research is integrated as part of a wider structured multidisciplinary research programme and stakeholder involvement takes place in one phase or another of the research cycle (e.g. agenda-setting, defining priorities, review, etc).

The mapping of these R&D networks will be based on desk top analysis and structured interviews. In a first stage we explore the different R&D programmes based on existing documents. Secondly, representatives from the 14 countries are interviewed to characterize the interactions that take place and the integration of multidisciplinarity within these programmes.

Technical Training for Local Community Administrations as a Tool of Implementation of Public Participation Approaches in Radioactive Waste Disposal

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The implementation of nuclear power in Poland forces a new approach to the current procedures and systems related to the management of radioactive waste and spent nuclear fuel. Due to the fulfillment of disposal volume in the National Radioactive Waste Repository in Różan it will be necessary to build a new repository for low and intermediate radioactive waste with a capacity of about 170 000 m^3. Poland as one of the Arhus Convention Signatories is force to follow the European recommendation on public participation in decision-making in environmental matters. The Recommendations are the tool to improve the implementation of the Arhus Convention's provisions on public participation in decision-making.

The new approach to public participation in environmental decision-making methods was the reason for Poland's participation in IPPA FP 7 project where the methods of communication with society and transparency of decision-making actions are developed. The main goal of IPPA project in Poland was to build the safe base and space for dialogue between interested parties involved in decision process related to radioactive waste management in Poland. Model RISCOM which is supposed the core method to be used in Poland starts from INFORMATION level but the BASIC EDUCATON level is not stressed.

In order to fill this gap it was organized special technical training at National Centre for Nuclear Research Świerk and Radioactive Waste Management Plant Różan for selected group represented the local community administrations which is elected community representation gifted by highest confidence of the habitants. The preliminary opinions and basic knowledge about radioactive waste were tested and the new approach after training was compared. The improvement of the knowledge does not transfer to increase of acceptance but the way of training helps to polarize the opinion of people previously undecided.

The organized training gives the new suggestions for future action.

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POSTERS BASED ON THE RECOSY PROJECT

RECOSY Intercomparison Exercise on Redox Determination Methods

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Redox processes play an important role in defining the aqueous chemistry of redox sensitive radionuclides (i.e. Se, Tc, U, Np, Pu, etc). In order to assess the chemical behaviour of such elements in the context of nuclear waste disposal it is necessary to predict and quantify the impact of redox conditions on solution chemistry. The reliable measurement and assessment of redox potentials in solution therefore is a matter of highest importance for many applications related to long-lived radionuclides and geochemistry in general.

In order to analyse how to perform reliable measurements of the redox potential in aqueous media, an Intercomparison Exercise (ICE) on redox determination methods was conducted within the EURATOM FP7 Collaborative Project RECOSY. ICE was hosted in 2009 by the Institute for Nuclear Waste Disposal at the Karlsruhe Institute of Technology (KIT-INE). More than 40 scientists working on different topics related to redox chemistry from 20 RECOSY partner organisations and associated groups contributed to RECOSY ICE, thus providing a broad scientific basis for ICE. The objectives of the RECOSY ICE were to compare different redox determination methods in order to (i) identify critical redox determination issues, (ii) provide the basis for more confidence in redox determinations for the individual groups, and (iii) identify future activities that could contribute to gain confidence in determination of the redox state of nuclear waste disposal Safety Case relevant systems and conditions.

The intercomparison was based upon different redox determination methods, i.e. static electrodes (platinum, gold glassy carbon, single/combined electrodes), dynamic electrochemical measurements, amperometric measurements, optodes (optical fibres with oxygen sensitive tips) and thermodynamic calculations based on chemical composition and physicochemical properties (such as pH, ionic strength and temperature). A wide set of samples was used with three different types of origin and properties, namely (i) simple samples with well defined composition, (ii) natural samples kept under near-natural conditions, and (iii) samples with microbial cultures.

The main conclusion of RECOSY ICE is that the redox state of an aqueous system can be determined by the existing experimental techniques, although the degree of confidence strongly depends on the kind of aqueous system investigated and the degree of optimisation of the experimental equipment and handling protocols. It is strongly recommended to use a combination of several experimental approaches to identify and assess systematic errors as there is no single “best method” to determine the redox state of a given system. This is especially true for the analysis of (intrinsically highly complex) real systems. Based upon the outcome of RECOSY ICE and the joint data evaluation and interpretation, several recommendations and a list of open topics were defined as input for future research activities, both of which are detailed in the Final RECOSY ICE report.

The final report on RECOSY ICE has been published as *KIT Scientific Report7572*. The printed report is available from the authors. Electronic pdf copies are available via http://www.recosy.eu or as free pdf download from KIT-Scientific Publishing at http://www.ksp.kit.edu

RECOSY - Microbial mediated reduction of Pu(IV) and Tc(VII)

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Understanding the migration behaviour of radioactive elements in geological environments is essential for the long-term safety assessment of nuclear waste repositories and in nuclear waste management. The potential of migration of radionuclides at contaminated sites and also from the waste repository depends on their environmental behaviour determined by its chemical speciation. The physicochemical characteristics of the environment and microbial activity play an important role for radionuclide behaviour in various environmental systems. Under appropriate conditions, the activity of microorganisms could affect the chemical nature of the radionuclide due to changing redox conditions and biotransformation of their oxidation states.

Within RECOSY, investigations relevant for the assessment of the effect of microbial processes on mobility of redox sensitive radionuclide (Pu, Tc) have been done. The study of Pu oxidation states after Pu(IV) interaction with microorganisms isolated from samples collected in the low-level radioactive waste repository in the Ignalina NPP was performed. Dominance of Gram-positive and spore-forming bacteria in the radioactive dust shows their resistance to radiation and ability to survive under the nutrient limited conditions. The highest effect in a change of Pu(IV) to Pu(III) at low pH under aerobic conditions was demonstrated by Gram-positive bacterium Bacillus mycoides (strain DPKI-01) and Gram-negative bacterium Serratia marcescens (strain DPKI-06). The amount of reduced plutonium (III) in brine solution came up to 15%. Tested fungi Absidia spinosa var. spinosa (strain DPKI-25) and Paecilomyces lilacinus (strain DPKI-12) showed no pronounced influence on the redox process in the used experimental system under aerobic conditions. The presence of a significant amount of Pu(IV) in biomass suggests that the process of bioaccumulation is dominant. Both fungi and bacteria showed their good abilities to accumulate Pu from a solution at low pH.

A set of experiments to determine the factors (pH, incubation time) that influenced TcO_4^- reduction and sorption onto iron oxides and to explore the effect of microbial activity on the migration behaviour of technetium was performed. Gradual sorption of technetium, added as TcO_4^-, in aquatic solution onto FeO/Fe_3O_4 mineral under aerobic conditions was observed. Under alkaline and acidic conditions, no pronounced effect of sorption of technetium onto hematite (Fe_2O_3) was determined. The mineral hematite at neutral or slightly alkaline pH under oxic conditions is attributed to minerals which do not immobilize Tc(VII). Adsorption of Tc (VII) onto hematite was achieved because of the presence of specific microorganisms.

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RECOSY - Microbial mediated immobilization of radionuclide in a future crystalline nuclear-waste repository

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The world's first permanent nuclear-waste repository for high-level waste (HLW) will be the underground rock characterization facility tunnel ONKALO, situated near the nuclear power plants of Olkiluoto, approximately 300 km northwest of Helsinki (Finland). Geological mapping showed that the bedrock of the tunnel is mainly composed of veined gneisses with a migmatic appearance, which are interspersed by numerous fractures. Massive 5–10-mm thick biofilms (microbial communities) are observed attached to the tunnel walls where groundwater is seeping from bedrock fractures feeding the biofilm. Groundwater and biofilm samples were taken during the EURATOM collaborative project RECOSY for laboratory experiments in flow cells, where uranium(VI) was added to the circulating groundwater with a final uranium concentration (4.25×10^{-5} M), which is a concentration observed near uranium-contaminated sites. The aim of our studies was to simulate the fate and behaviour of uranium(VI) in the presence of microorganisms using different methods to find evidence of the possibility of uranium retention.

In our experiments, aqueous uranium percolated the biofilm and induced interaction between biofilm microorganisms and uranium(VI), forming an intracellular U-phosphate mineral similar to autunite ($Ca[UO_2]_2[PO_4]_2 \cdot 2\text{-}6H_2O$ or meta-autunite $Ca[UO_2]_2[PO_4]_2 \cdot 10\text{-}12H_2O$) as determined by means of EF-TEM/EELS. The potential toxicity of uranyl was consequently avoided by the uranium forming these strong complexes with phosphate in the cytoplasm of these microorganisms. Inorganic phosphate ($H_2PO_4^-$) was released from the cellular polyphosphate as a cells' response to the heavy metal stress. In contrast, aqueous uranium carbonate species, likely calcium uranyl carbonates species ($Ca_2UO_2[CO_3]_3$), were formed using the large available amount of carbonate in the uranium-contaminated circulating groundwater. The results of the laser-induced fluorescence spectroscopy studies performed at 283 K are in excellent agreement with the thermodynamic calculations of the theoretical predominance fields of the uranium species formed in the uranium-contaminated circulating groundwater. These complexes consequently influence the speciation of uranium, contributing to the transport and migration of uranium(VI).

Our studies aimed to improve our understanding of the mechanisms by which biofilms respond to the exposure to radionuclide with respect to safety assessments in the far-field of nuclear-waste repositories. In this environment, microorganisms must be considered, along with minerals, as an important factor influencing radionuclide transport.

Redox processes in the safety case of deep geological repositories of radioactive wastes: contribution of the European RECOSY Collaborative Project

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Redox processes influence key geochemical characteristics controlling radionuclide behavior in the near and far field of a nuclear waste repository. A sound understanding of redox related processes is therefore of highest importance for developing a Safety Case. The EURATOM 7th EC Framework Programme Collaborative Project REdox phenomena COntrolling SYstems (RECOSY) started in April 2008 and extended over four years until March 2012. Main objectives of RECOSY were related to the improved understanding of redox phenomena controlling the long-term release or retention of radionuclides in nuclear waste disposal and providing tools to apply the results to Performance Assessment and the Safety Case. The RECOSY consortium had 32 Beneficiaries and 6 Associated Groups to RECOSY. The work performed within RECOSY was critically reflected by an End-User Consultancy Group throughout the project. The work program had six RTD work packages covering near-field and far-field aspects as well as all relevant host-rocks considered in Europe. Work packages were focussing on specific relevant aspects, for example the development of redox determination methods or the redox response of defined and near-natural system. Additional research activities were centred on investing the redox reactions of radionuclides and also addressed open questions related to redox processes in radionuclide transport. Focussing particularly on high level waste material, redox reactions affecting the spent fuel source-term were analysed under several different conditions.

It is the intention of this paper to specially highlight in which way the scientific and technical results obtained from RECOSY can feed the process understanding needed for the stepwise development of the Safety Case associated with deep geological disposal.

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RECOSY- Kinetics of Pu(IV), Pu(V) and Am(III) Sorption to Natural Clay

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Natural clays of complex composition are reactive minerals which can retard migration of radionuclides from a repository of radioactive wastes as components of the engineered and natural barriers. An important characteristic of the retardation is the reaction kinetics of radionuclides with clays. Sorption of Pu(IV), Pu(V) and Am(III) to clays and time changes of bonding of both the elements in the clays were studied with the aim of better understanding their sorption mechanism to materials selected as engineered barrier for the cement (concrete) based near surface low and intermediate level radioactive waste repository. Clay samples with different amounts and compositions of iron oxides in their coatings as well as synthetic magnetite, goethite and hematite were used in equilibrium and kinetic sorption experiments. In order to obtain information about Pu and Am bonding to clays the sequential extraction (SE) was carried out as a function of time (361 days, 5 – 520000 min). The composition of clay mineral coatings and their leaching during the SE procedures were investigated by means of the Mössbauer spectroscopy and X-ray diffraction. Sorption was studied using natural groundwater. Pu(IV) and Pu(V) – a mixture of 238Pu, 239Pu, 240Pu and 241Pu or 241Am spikes were used to achieve initial concentrations of the elements 1·10-10 and 3·10-11 mol/L, respectively. After desired sorption time the solids were separated from the liquid phase by centrifugation at 10,000 G. Pu and Am in the solution and in the solid phase after radiochemical separation using the UTEVA and TRU resins (Eichrom Industries) were determined by alphaspectrometry. 242Pu and 243Am were used as tracers in the separation procedures. The experimental data were modeled using ion exchange and surface complexation models. Although the highest Pu *Kd* value was found for clay with the largest content of crystalline Fe oxides and mineral hematite, Pu was sorbed to various sorption sites on the clay surface, whereas Am sorption was restricted to the carbonate sorption sites. Disagreement between the SE and modelling results was explained by a particular role of ferrihydrite in the sorption process. The mineral siderite (Fe(II)CO3) found in the clay coatings played an important role in the catalytic transformation of ferrihydrite to crystalline Fe oxide species. A possible mechanism was a hydratation of mineral surfaces and the reductive dissolution of ferric minerals via electron transfer from ferrous-iron-bearing minerals, with subsequent oxidation of ferrous ions in the liquid phase and formation of secondary minerals containing ferric iron. Recrystallization of ferrihydrite to lepidocrocite, goethite and hematite induced by the trace concentration of Fe(II) was found to be important in Pu incorporation into clay mineral coatings.

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POSTERS BASED ON THE CROCK PROJECT

Np(V) migration in a single fracture from Äspö, Sweden: Experiments and reactive transport modeling

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In the context of nuclear waste disposal reactive transport modelling represents an important tool for long-term predictions of radionuclide migration. Within the CROCK project, the Lagrangian-based framework FASTREACT (FrAmework for STochastic REACive Transport) (Trinchero et al., 2013) coupled to the geochemical code PhreeqC is applied which is able to solve for multicomponent reactive transport. This newly developed approach is tested to model experimental data on Np(V) migration through a natural single fracture in Äspö diorite.

Migration experiments have been conducted on a single fractured drill core (length 13.5cm; diameter 5.05cm) from Äspö, Sweden using a radionuclide cocktail on basis of natural Grimsel ground water to mimic the influence of glacial melt water intrusion into a repository. Here, only the results of the conservative tracer (Tritium (HTO)) and of $^{237}Np(V)$ ($[Np]_{TOT}$ = ~1.25x10^{-6} M) are presented.

To model the Np reactive transport we tested different approaches with increasing complexity ranging from simplified K_d-type sorption models (1st order reversible sorption/desorption kinetics) to more complex, mechanistically based surface complexation models for sorption of NpO_2^+ onto one ($Fe_2O_3x2H_2O$ (HFO)), two (HFO and biotite (Bt)) and three (HFO, Bt and kaolinite) different mineral phases.

The experimental Np breakthrough curve (BTC) shows a retardation (retardation factor R_f = 1.8) compared to the conservative tracer (HTO) BTC and a pronounced tailing indicative of an interaction with the fracture surfaces. Np recovery is ~76% at the end of the experiment, but the tailing seems to be still ongoing thus a quantitative Np recovery is reasonable to expect. The high Np recovery is in line with measured Eh values (~ +200 mV) and thermodynamic modeling yielding no Np reduction and NpO_2^+ as dominating species, which is known to sorb weakly.

The experimental HTO BTC is captured well by FASTREACT verifying the correctness of the coupling of PhreeqC and FASTREACT. Using a single Kd derived from the migration experiments (K_d = 0.18) in the different SCM models, it is not at all possible to adequately describe the Np BTC. While the peak position is captured relatively well, the model completely fails both in describing the Np BTC peak concentration and tailing. In case of the 1st order kinetic model, the model fit describes the ascending part of the curve quite well, and also shows a pronounced tailing, but the overall fit is not satisfactory. Adjusting the (unknown total reactive) mineral surface area using the parameter estimation code PEST improved all model fits, but still considerable deviations to the experimental BTC are observed. By coupling of FASTREACT to the reactive transport code HBGC123D the SCM applied can be modified to include a kinetic description of the sorption and desorption processes rather than an equilibrium SCM sorption/desorption approach as used so far. The results of these simulations, which are still on going, are expected to reproduce fairly well both the peak position and concentration and the curve tailing.

To sum up, the FASTREACT approach coupled to PhreeqC has been shown to represent an effective computational framework for the intepretation of laboratory experiments of radionuclide transport.

ABSTRACTS
Posters

Extending geochemical modelling with automatic stochastic simulations

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MCPhreeqc is an application to apply a stochastic analysis to geochemical models implemented in Phreeqc. When modelling geochemical processes in Phreeqc, the parameter uncertainty is normally not considered. The degree of this uncertainty can be studied using MCPhreeqc. The application allows the user to specify a probability distribution for each parameter. It will generate random values from these distributions and automatically run Phreeqc using these combinations as input. To improve run times these Phreeqc simulations are run in parallel using all the available processor cores. The results are summarized in histograms and scatterplots. The cross-platform application MCPhreeqc, which has been developed in the framework of the FP7 CROCK project, is written in Python and comes with a graphical user interface. It is released as open-source (LGPL) and can be downloaded from http://www.amphos21.com/software

The numerical tool has been applied to study the uncertainty related to key radionuclide retention processes in a typical Fenoscandian site. More specifically, probability density functions of fracture filling minerals (i.e. clay minerals) inferred by in situ mineralogical analyses have been used to simulate the (uncertain) amount of Cation Exchange Cacity (CEC) of the medium. The results (i.e. Ka values) have been interpreted in probabilistic terms in order to provide realistic ranges of Ka values.

POSTERS BASED ON THE CATCLAY PROJECT

Influence of the Na+ counter-ion concentration on the diffusion in clay rocks of three sorbing cations: Sr^{2+}, Zn(II) and Eu(III)

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It is now well established that the transport properties of clayrocks, i.e. parameter values for Fick's law, are governed mainly by the fact that these rocks contain a high proportion of negatively charged clay minerals. While a good understanding of the diffusive behaviour of non-reactive anionic and neutral species is now achieved, much effort has been placed in the framework of the EC FP7 CatClay project on improving understanding of coupled sorption–diffusion phenomena for sorbing cations in clayrocks, since several cations known to form highly stable surface complexes with sites on mineral surfaces, migrated more deeply into clayrock than expected.

This paper aimed at investigating the influence of Na^+ counter-ion concentration on the diffusion of three cations in the Callovo-Oxfordian claystones. The cations of interest have been chosen for covering a representative range of cation families taken under consideration in the performance assessment, from a moderately sorbing cation, the strontium, to strongly sorbing cations, one being divalent, the zinc and the other being trivalent, the europium. For that purpose, three diffusion cells with Na^+ counter-ion concentration varying from 51 mmol/L (corresponding to the equilibrated pore-water) to 546 mmol/L were setup for carrying out successively the through-diffusion method for strontium, and in-diffusion method for zinc and europium. In a first step, diffusion of tritiated water (HTO) was studied for comparing the diffusive behaviour of the three cations of interest with a water tracer.

First of all, the diffusion experiments performed with the equilibrated pore-water, led to values of effective diffusion coefficient (D_e) revealing the enhanced-diffusion of these three sorbing cations when compared to that of the tritiated water (D_e(Zn) = 8 10^{-11} m^2/s > D_e(Sr) = 5 10^{-11} m^2/s > D_e(Eu) = 4 10^{-11} m^2/s > D_e(HTO) = 3.3 10^{-11} m^2/s). Moreover, the impact of the Na^+ counter-ion concentration on diffusion was clearly evidenced. While the D_e values of tritiated water remained stable, the three cations of interest showed a gradual decrease of their D_e values, by a factor of up to 4 for strontium, 10 for europium and 16 for zinc. In parallel, the extent of the sorption also displayed a sharp decrease with the Na^+ counter-ion concentration change. Their distribution coefficient (K_D) values varied from 5, 2700 and 5000 L/kg for strontium, zinc and europium, respectively to 0.75, 115 and 3000 L/kg. The evolution of all these parameters was finally discussed regarding the diffusion surface process.

Sorption and diffusion of Zn onto Na-illite under a wide variety of conditions

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In the framework of the FP7 EC project CATCLAY and the Belgian research program on radionuclide migration coordinated by ONDRAF/NIRAS, the sorption of Zn onto illite, a main constituent of argillaceous host rocks, is investigated by batch type experiments under a wide variety of conditions (pH, ionic strength, CO_2 pressure, presence of organic matter).With the acquired knowledge, diffusion of Zn in compacted Na-illite is investigated in order to demonstrate the upscaling from batch to migration experiments.

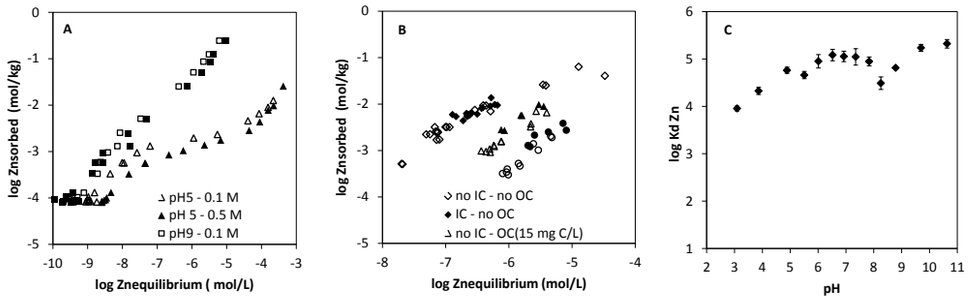

Figure: Zn sorption isotherms and edge on conditioned Na-illite: A. isotherms at pH 5 and 9 at two different ionic strengths (0.1 and 0.5M $NaClO_4$); B. istotherms at pH 8.4 at different dissolved inorganic and organic carbon concentrations (0.1M $NaClO_4$); C. sorption edge (0.1M $NaClO_4$) – distribution coefficient in function of pH.

The presence of carbonates (10 times atmospheric partial pressure of CO_2) has no observable effect on the sorptionat pH 8.4 (figure B). A strong effect is observed, however, for organic carbon. In presence of dissolved organic carbon (humic acid) the uptake (as K_d) of Zn by illite decreased significantly by about one to two orders of magnitude (depending on Zn and DOC concentration) (figure B).

The sorption edge at the ionic strength of 0.1 M ($NaClO_4$) (figure C) shows that the sorption varies only one order of magnitude over the entire pH range from 3 to 11. This sorption behaviour on the Na-illite differs significantly with the sorption of Zn on Na-montmorillonite [1], which varies 4 orders of magnitude (log K_d ~2 to 6) in the same pH range.

Besides the batch sorption experiments, the focus in the Catclay project is mainly on diffusion experiments in order to investigate if Zn is subject to the so called 'surface diffusion' [2]. Currently, in-diffusion experiments are ongoing. The K_d values obtained from the batch sorption experiments can be compared with the capacity factor α (= $\varepsilon + \rho_b K_d$) obtained from the diffusion experiments.

In-diffusion experiments were performed at pH 5 (experiments at pH 7 and 8.5 planned). The clay plug with a density of 1.7 g cm^{-3} and a length of 15 mm was in contact with $NaClO_4$ 0.1M buffered with MES 10^{-3} M at pH 5, spiked with ^{65}Zn at the inlet side. Concentration profiles in the inlet solution and the trace profile in the clay are fitted with COMSOL Multiphysics.

Although Zn is strongly retained on the compacted clay, the observed clay profiles showed that Zn diffuses faster than expected. Modelling indeed resulted in a relatively high effective diffusion coefficient, D_{eff}.

This result therefore would indicate, for the first time, the presence of surface diffusion phenomena for transition metals in compacted illite.

ACKNOWLEDGEMENT

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ABSTRACTS
Posters

Eu^{3+}/Na-illite interaction: batch sorption and diffusion studies

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Due to their favourable properties (high surface area and sorption capacity, low permeability) sedimentary clay rock formations are regarded as potential host rock formations for a high-level nuclear waste repository in geological formations [1]. As illite is an important component in such rock types, data on radionuclide retention and transfer properties are mandatory for performance assessment. In the present work conducted within the CP CatClay project, retention and transport of Eu is studied by batch sorption and diffusion experiments using the Na-illite (Illite de Puy).

Eu sorption isotherm studies on Na-illite were performed with variable Eu concentrations (2E-9 and 2E-4 M) under various experimental conditions (pH 5.5, 7.0 and 8.3; I = 0.1 and 0.5 M), while for the sorption edge experiments the metal ion concentration was fixed at trace levels (Eu_{tot} = 2E-9 M) and the ionic strength was varied between 0.1 - 4 M NaCl. The experimental setup for the Eu in-diffusion comprised of a compacted Na-illite sample (1700 kg/m^3) in a PEEK diffusion cell connected to two reservoirs, one containing inactive Eu (1E-6 M). After 152 days running time, the solutions were exchanged stepwise by methanol/water, methanol and epoxy resin. After drying, the sample was cut and investigated by confocal microscopy investigations using an optical microscope/ND-YAG laser system (λ_{ex} = 395 nm). The laser light was focused onto the sample surface and the emitted fluorescence light was recorded by a CCD camera. In addition, fluorescence decay was followed under variable delay times.

The batch sorption results at trace metal ion concentrations show an uptake by cation exchange at low pH (< 5) and low ionic strength (0.1 M NaCl). With increasing pH, the sorption increases with an almost quantitative uptake at pH > 5. In this pH region, comparable K_d values are detected for I ≤ 2 M NaCl, pointing to a sorption mechanism by surface complexation. In addition, an increasing ionic strength leads to an decreased Eu uptake at low pH values. Near unity slope of the sorption isotherm up to Eu = 1E-8 M for all experimental conditions indicates a linear sorption behaviour, while at higher Eu equilibrium concentrations the slope decreases (Langmuir- and/or Freundlich type). The present (surface normalized) data are in good agreement with former results under similar conditions [2] with the exception at 1E-8 M < $Eu_{equilibrium}$ < 1E-6 M, where an increased Eu uptake is detected in the present system. The confocal microscopy results show a fluorescence zone at (d = 0 – 50 nm) and in the vicinity (d > 50 µm) of the sample edge, indicating the zone on the clay surface being in contact with the Eu reservoir. Similar fluorescence decay curves could be detected originating from different regions and showing a bi-exponential decay law with two Eu fluorescence lifetimes (τ_1 = 46 µs; τ_2 = 200 µs). Based on [3], the higher fluorescence lifetime can be assigned to 5 H_2O/OH^- entities in the first Eu coordination sphere, while the shorter lifetime points to fluorescence quenching effects, e.g. Eu-Eu or Eu-Fe interactions. Based on the pH of 5.3-6.4 in the diffusion experiment, $\equiv S-Eu(H_2O)_5$ and $\equiv S-Eu(OH)(H_2O)_4$ surface complexes can be deduced existing on the compacted Na-illite surface.

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Diffuse transport in clay media: µm to nm scale characterization of pore space and mineral spatial organization

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In clay materials, the dominant transport mode is diffusive and depends mainly on various parameters such as the mobility of the species in water, the accessible porosity, the pore space geometry and the retardation as a result of reactions such as sorption or ion exchange. In this way, the CATCLAY project was proposed to describe the cation migration processes in natural clayrocks. The project combines modelling and experimental studies from a simpler, analogous system (monophasic compacted clay system) to clayrocks (Callovo-Oxfordian argillites, Opalinus Clay and Boom Clay). Part of these experimental studies focuses on small scale structure (µm – nm) property of rocks in order to determine how the spatial distribution of mineral and pores at small scales can influence diffusion driven transport of sorbing cations. The present study focuses on compacted illite properties (simpler analogous system) in hopes to extent this study to the natural clayrock formation. Compacted illite material represents thus an analogy with the clay matrix constituting clay-rocks. Our approach is mainly based on imaging the small scale structural organization of compacted illite material and analyzing the obtained images in order to extract information on pore space and mineral spatial distribution.

Techniques for imaging the texture of illite material, in compacted state, were first developed. The small scale structure of impregnated sample was imaged in 2D using Transmission Electron Microscopy (TEM) and in 3D using Focused Ion Beam coupled to Scanning Electron Microscopy (FIB/SEM). Both methodologies (FIB-tomography and TEM techniques) are complementary methods for the up-scaling characterization of the structural organization of compacted clayey materials. TEM images analysis allow to scale down the resolution size since only a part of the pore space could thus be imaged with FIB/SEM method.

Viewing and performing a qualitative description of images constitute a major issue and can help us to better understand how the transfer pathways and retention sites are organized in the porous media. Thanks to image analysis method, pores and minerals can be thresholded from grey level TEM and FIB/SEM images. Quantitative parameters can be then computed based from segmented images. For that purpose, much efforts are currently given for determining the size and the morphology of pores, the main geometrical features of clay particles (number of layers, size, shape...), the spatial distribution of clay particles (individual/aggregates, type of contact between the clay particles, orientation...) and the pores connectivity. Finally, all these quantitative parameters are expected to be used in various transfer modelling approaches.

ABSTRACTS
Posters

POSTERS BASED ON THE REDUPP PROJECT

The disposal of spent nuclear fuel: The effect of high energy surface sites on dissolution rate

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High energy surface sites, including grain boundaries, step edges and naturally occurring surface defects are expected to play a role in the dissolution of mineral surfaces. These microstructural features are dependent on both surface chemistry and crystallography. Spent nuclear fuel, composed primarily of UO_2, is a ceramic material annealed at high temperatures, which leads to the formation of grain boundaries and oxygen vacancy defects. In the safety case for the geological disposal of nuclear waste, the release of radioactivity from the repository is controlled by the dissolution of the spent fuel in groundwater, therefore the dissolution characteristics arising from these features must be determined.

In the REDUPP project, we demonstrate the effect of high energy surface sites, grain boundaries and oxygen vacancies on the dissolution rate of CeO_2, a spent nuclear fuel analogue which approximates as closely as possible the mineral structure characteristics of fuel-grade UO_2. Due to the highly refractory nature of CeO_2, dissolution was conducted at high temperature and low pH (90 °C, pH 2 with nitric acid). CeO_2 was powdered to several different size fractions and subject to dissolution conditions. Contrary to expectations, it was found that larger size fractions exhibited a greater dissolution rate than smaller size fractions (0.45×10^{-4} g m^{-2} d^{-1} for 25 -50 µm compared with 0.81×10^{-4} g m^{-2} d^{-1} for 75 – 150 µm). Annealing the powders at high temperature removed some the high energy surface sites (confirmed by SEM and BET analyses) and resulted in a less rapid dissolution rate than for non-annealed powder (0.21×10^{-4} g m^{-2} d^{-1}). Monolith samples of CeO_2 were annealed to develop grain boundaries, dissolved and analysed periodically by AFM and ICP-MS. Grain boundary dissolution was rapid during the initial dissolution period (460 nm increase in grain boundary depth over 7 days), concurrent with a rapid dissolution rate (0.57 ppm d^{-1}). When grain boundary dissolution slowed, the dissolution rate also decreased (0.15 ppm d^{-1}). CeO_2 powders were annealed under reducing conditions (H_2/N_2), forming non-stoichiometric $CeO_{1.93}$, concurrent with the formation of oxygen vacancy defects. The dissolution rate of these powders was three orders of magnitude greater than for air-annealed powders. Coupled electron backscatter diffraction and Raman spectroscopy of a $CeO_{1.93}$ monolith sample showed that grains with different crystallographic orientation displayed differing stoichiometries, suggesting dissolution rates may differ between individual grains. These results demonstrate that a variety of high energy surface sites affect the dissolution spent nuclear fuel analogues and provide quantitative information to reduce uncertainty in the safety case for spent nuclear fuel disposal.

Immobilisation of technetium-99 on backfill cement: sorption under static and saturated flow conditions

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Technetium-99, a β-emitting radioactive fission product of ^{235}U, formed in nuclear reactors, presents a major challenge to nuclear waste disposal strategies. Its long half-life (2.1×10^5 years) and high solubility under oxic conditions as the pertechnetate anion [Tc(VII)O_4] is particularly problematic for long-term disposal of radioactive waste in geological repositories.

In this study, we investigate the effectiveness of the backfill cement, Nirex Reference Vault Backfill (NRVB) and other cements commonly used in nuclear waste disposal scenarios (crushed Ordinary Portland Cement (OPC) and OPC combined with Blast Furnace Slag (BFS) or Pulversised Fly Ash (PFA)) for their effectiveness towards immobilisation of Tc(VII). Sorption in batch experiments was shown to be dependent on cement type; NRVB, OPC and OPC/PFA weakly sorbed pertechnetate, while the BFS-containing OPC cement sorbed ~50% of the injected Tc(VII). Oxidation state µ-XRF mapping, combined with µ-XANES performed on a BFS-containing cement reacted with Tc(VII) showed that immobilisation in this cement was due to a rapid reductive-precipitation mechanism, with Tc(IV) precipitates localised on the surface of BFS particles. The cements that displayed poor sorption of technetium were found to contain only Tc(VII). Time-lapse, non-invasive, quantitative radiographic imaging of a ^{99m}Tc radiotracer through the different cement compositions was performed to investigate Tc(VII) immobilisation under dynamic conditions. A standard medical gamma camera was used to monitor pulse-inputs of ~15MBq ^{99m}Tc under saturated conditions and at a constant flow of 0.33ml/min. Dynamic gamma imaging was conducted every 30s for 2 hours. Spatial moments analysis of the resulting ^{99m}Tc plume provided information about the relative changes in mass distribution of the radionuclide in the various cement materials. ^{99m}Tc advected through NRVB demonstrated typical conservative transport behaviour, while OPC and OPC/PFA produced a slight reduction in ^{99m}Tc centre of mass transport velocity over time. BFS-containing cement was shown to be most effective at immobilising ^{99m}Tc under dynamic, rapid-flow conditions, with up to 50% of the injected activity retained irreversibly by the cement, indicating that the determined sorption mechanism has a significant effect on the transport of technetium.

ABSTRACTS
Posters

Surface Reactivity and Dissolution of Spent Nuclear Fuel:
The Value of First-principles Computational Modelling

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Dissolution of spent nuclear fuel such as uranium dioxide (UO_2) is both difficult and extremely costly to be studied under laboratory conditions. The difficulty of studying spent nuclear fuel stems from the restrictions in handling the radioactive material. Therefore, only a few highly specialized laboratories can perform such investigations, giving raise to high costs. There exists however, an alternative route to obtain detailed insight in dissolution processes, namely, that of first-principles computational modeling [1]. Chemical processes down to the atomic level can nowadays be ideally and very accurately simulated using modern computational modeling techniques, based on electronic structure calculations. Moreover, computational modeling tools have been perfected in the last decade, and have reached an absolute, predictive level. Therefore, first-principles materials' modeling is a low-cost alternative to obtain precise information on the dissolution.

We apply here first-principles materials modeling to simulate the dissolution processes of nuclear fuel materials (UO_2, PuO_2), as well as of CaF_2 and CeO_2, which are used as (non-radioactive) surrogates to UO_2 in dissolution experiments.

Utilizing first-principle simulations [based on density functional theory (DFT) corrected for on-site Coulomb interactions (DFT+U)] we have explained [2] the experimental stability in solution of materials like CaF_2 and CeO_2 that have the same crystal structure. We could show that the stability of a particular surface is mainly dependent on its atomic structure and the presence of sites where atoms are deficiently bonded. Using as reference planes the surfaces with low surface formation energies, viz. (111), (100) and (110), our results reveal a relation between the surface energy of any Miller-indexed plane and the surface energy of those reference planes, being dependent on the fluorite surface structure only. As a consequence, they follow the same trend for CaF_2 and CeO_2. Comparison with experimental results shows a direct correlation between the trends of dry surface energies and surface stabilities during dissolution of both CaF_2 and CeO_2.

Further calculations have been performed for ThO_2 and UO_2 surfaces to verify the predicted results for actinide surfaces.

To study the interaction of water with nuclear fuel, we have computed the dissociation and the coverage-dependent adsorption energies of water on CeO_2 and UO_2 surfaces, using *ab initio* molecular dynamics simulations. Our results show that water molecules adsorb and dissociate primarily at step edges, where they induce dissolution processes and release of hydrogen.

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POSTERS BASED ON THE FORGE PROJECT

FORGE – Fate of Repository Gases
Generation and Migration of Hydrogen in Deep Geological Repositories

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In deep geological repositories of radioactive waste gas is generated, either from degradation of waste, metal waste canisters, other metal engineered barriers or radiolysis of water. The pressure of generated gas can affect the functional barrier properties and consequently the time and the rate of penetration of radionuclides from the repositories to the environment. Although research of gas generation and migration was conducted in a large number of research works to show that generated gases will not affect adversely safety of radioactive waste repositories, there are still a lot of uncertainties concerning hydrogen generation rate in repository conditions. The problem is that hydrogen generation is affected by a variety of factors such as pH, temperature, composition of water, radiation or the amount of oxygen in water, which is in turn affected by all other mentioned factors. Reported values of hydrogen evolution rate therefore ranges from less than one tenth of moles per square meter and year to the values exceeding several tens of moles of generated hydrogen per square meter and year.

The Czech concept of deep geological repository is based on the steel canister surrounded by Czech bentonite in granitic host rock. Therefore the main aim of research work of ÚJV in the framework of 7th Euratom Framework Programme, EC project FORGE was focused on hydrogen generation from carbon steel corrosion and migration of hydrogen through compacted bentonite. Measurements of hydrogen generation from carbon steel corrosion in synthetic bentonite pore water were performed under anaerobic conditions (concentration of oxygen less then 0.1 ppm). The volume of hydrogen was measured by special device developed in ÚJV, which enable measuring volume under constant pressure. Short term and long term corrosion experiments were conducted at high temperature for determine hydrogen evolution rate.

Two types of device were used for measurement of hydrogen migration through saturated compacted Czech bentonite Rokle B75. The first device is based on pressure increases due to accumulation of hydrogen from reaction of iron powder with water (hydrogen pressure reactor) and the second one where constant hydrogen source was used. A large number of experiments have been conducted with various compacted densities of bentonite. Different results were obtained in various experiments conducted otherwise under the same conditions. This suggests that the initial heterogeneities in bentonite samples can affect the migration of hydrogen through compacted bentonite.

ABSTRACTS
Posters

Gas Induced Radionuclide Transport in Disturbed and Undisturbed Boom Clay and Boom Clay – Bentonite interfaces

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In deep geological repositories for radioactive waste, gas can be generated by different mechanisms like anaerobic corrosion of metals, radiolysis of water and organic materials, and microbial degradation of various organic wastes. Gas generated in the near field of a geological repository in clay can dissolve in the ground water and can be transported away from the repository by diffusion as dissolved species. However if the gas generation rate is larger than the capacity for diffusive transport of dissolved gas, the pore water will get oversaturated and a free gas phase will be formed, leading to a gas pressure build-up [1], eventually leading to gas breakthrough events. During gas breakthrough, some water could be expelled by the gas phase. Depending on the timing of gas breakthrough, dissolved radionuclides (RN) and contaminants could be driven out of the undisturbed clay faster than the normally expected diffusive transport. However, to what extent a gas pressure build-up may enhance radionuclide and contaminants transport in a clay host rock (undisturbed and disturbed) and at interfaces with engineered barriers remains subject to large uncertainties [2]. Therefore, the investigation of the gas-driven migration of aqueous born contaminants through the multi-barrier system is becoming an increasingly important element of safety cases for geological repositories in clay hosts. This research, performed in the framework of the FP7 project FORGE tried to answer these questions through the performance of a dedicated experimental programme [1,2].

To test the potential for gas-driven tracer transport, we designed a column experiment in which a water saturated clay core (h~4 cm, Ø = 3.8 cm) is put directly on top of a thin BC core (h~1 cm) which has been previously saturated with a tracer solution. To mimic the presence of radionuclides in the pore water, a non radioactive NaI solution of 0.01 mol/l was used as tracer solution. A He gas pressure (P~0.5-5 MPa) is applied at the bottom end of the iodide saturated plug while at the top end of the column, a known volume of natural pore water is put in contact with the plug. The gas pressure is stepwise increased until gas breakthrough occurs. Upon gas breakthrough, the water on top of the column is expelled and analysed for its iodide content. The measured concentration I^- is linked to the amount of NaI saturated pore water that was displaced [2]. Experiments have been performed with both undisturbed and disturbed (artificially fissured) Boom Clay cores and with combined Boom Clay – bentonite cores.

Based on the obtained results, we can state that the transport of radionuclides and contaminants due to a gas breakthrough is indeed possible but very limited. The amount of I^- that was transported and consequently the fraction of the volume pore water of the NaI saturated core transported during gas breakthrough was very low (< 0.5%). The effect of different parameters (orientation of the sample with respect to bedding plane, time of sealing of the fractures and porous medium type) on the fraction transported pore water was investigated [2].

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FORGE - Water retention and gas relative permeability of partially saturated bentonite/sand plugs

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In France, the deep-seated nuclear waste repository consists of a natural barrier located at 500m depth (in an argillaceous rock named argillite), associated to artificial barriers, including plugs of swelling clay (bentonite) for tunnel sealing purposes. For sealing a repository gallery, the clay barrier is formed by blocks of compacted bentonite arranged on vertical slices, which are put in place with initial construction gaps (Villar and Lloret, 2007). While fully water-saturated bentonite provides swelling capacity and low permeability. Bentonite/sand mixtures are usually compacted at an intermediate water content (w ≈ 10-15%), and they are progressively wetted by water coming from the host rock formation (Wang et al., 2011; King et al., 2010). It is expected that, during the process of saturation, a significant water saturation gradient will be present between the core and the external surface, itself in contact with the host rock and the underground site water (Villar et al., 2005; Villar and Lloret., 2007) (in the case of the FEBEX in situ experiment). Besides, owing to high water content, the external part of the massive barrier swells in contact with an extremely stiff host rock, so that it applies a confining pressure to the partially water-saturated core. In this context, it is essential to research sealing ability of the central part of the bentonite/sand barrier under confinement. To investigate this question, bentonite/sand plugs are uniformly partially water-saturated under free displacement conditions (which allows swelling/shrinkage), and then, tested for effective gas permeability at different confining pressure levels, usually up to 7-8MPa (which is the expected swelling pressure of fully water-saturated plugs).

Experimental results show that when subjected to relative humidity of 75% and below, bentonite/sand plugs shrink and lose mass, whereas these swell and gain mass for RH>75%. At RH=98%, swelling is as huge as 20% of the plug initial volume and water saturation reaches 87-92%. For gas permeability test, at a confinement equivalent to the expected full swelling pressure of 7MPa, gas permeability is lower than $10^{-20}\,m^2$ for samples initially saturated up to 86-91% only. It means that tightness to gas is achieved for partially-saturated bentonite/sand plugs under confinement.

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ABSTRACTS
Posters

POSTERS BASED ON THE FIRST-Nuclides PROJECT

FIRST-Nuclides - Numerical modelling of Spent Nuclear Fuel saturation with water under repository conditions: Implications for radionuclide instant release fraction under repository conditions

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Geological disposal of Spent Nuclear Fuel (SNF) in an underground repository requires quantitative Performance Assessment (PA) calculations. A factor that can significantly impact PA results is the Instant/Fast Release Fraction (IRF) – a fraction of the radionuclide inventory that can be mobilised "instantly" upon exposure to the intruding groundwater. Wetting of the fuel surfaces may be an important factor limiting the release rate for some radionuclides. Quantification of SNF saturation with water could therefore help better constrain the IRF and reduce PA uncertainty.

As a first approach, the problem was reduced to the saturation of a single SNF pellet through a network of discrete cracks. To this end a reference pellet representative of high linear power and burn-up conditions was conceptualised to be composed of two families of cracks with distinct aperture statistics. Saturation of the cracks was represented using a generalised Darcy equation for water flow under variably saturated conditions (the Richards equation). The model was applied to flow through cracks and implemented in the FEM code Comsol Multiphysics.

First results indicate that saturation with water of the crack network should proceed in two largely independent steps. First, the larger cracks are rapidly saturated (hours). This is followed by a much slower filling up of the smaller cracks, which process can take several months. However, uncertainty regarding some model parameters still exists. Experimental data on the hydraulic properties of the pellet are needed to increase the reliability of the model predictions.

Physical characterisation of spent nuclear fuel. First step to further Instant Release Fractions investigations

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The EURATOM FP7 Collaborative Project "Fast/Instant Release of Safety Relevant Radionuclides from Spent Nuclear Fuel (FIRST-Nuclides)" is established with the overall objective to provide for improved understanding of the fast / instantly released radionuclides from disposed high burn-up spent nuclear fuel. Many of the published results on the fast/instant release fraction (IRF) have been determined with spent fuel irradiated either with relatively low or extremely high neutron fluxes which are not typical of present spent fuel waiting for disposal. Procedures for selection, characterization and preparation of spent nuclear fuel (SNF) samples, which are relevant to SNF disposal, are established in workpackage #1 "Samples and tools" of FIRST-Nuclides. All experimentally working project partners (KIT, JRC-ITU, JÜLICH, PSI, SCK·CEN, CNRS, CTM, EK and STUDSVIK) contribute to this work package. Most partners study high burn-up spent nuclear fuels irradiated in commercial nuclear power plants, while the JÜLICH group studies so-called TRISO fuel irradiated in a research reactor at the Petten EC Joint Research Centre. CNRS investigates unirradiated TRISO particles, which will be used in successive corrosion experiments under alpha irradiation. EK studies damaged and leaking VVER fuel rods with a burn-up below 27 GWd (t HM)$^{-1}$, which were stored in water for several years after an incident at the Paks-2 reactor.

One of the first activities in work package #1 covers the compilation of fuel characterisation data from the partners. The IRF depends on critical characteristics of the SNF such as manufacturing process, burn-up history and fuel temperature history, ramping processes and storage time. Therefore, characteristic data of various high burn-up spent nuclear fuel rods are collected, dealing with the type of nuclear reactor, its electrical power, types of fuel assemblies, fuel manufacturing information and the discharge date of the fuel to be investigated. With respect to the cladding, the characterization covers the composition, cladding diameter, thickness and the initial radial gap width between pellet and gap. The information regarding the pellet comprises the initial enrichment, geometry, grain size, density and specifics of the production process. The irradiation history covers the burn-up, the irradiation time and the number of cycles as well as the maximum and average linear power rate. Critical parameters of the selected fuel samples are compared to those of high burn-up fuels which need to be disposed of in Europe, to assure the relevance of the samples for the Safety Case.

ABSTRACTS
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POSTERS BASED ON THE DOPAS PROJECT

DOPAS Experiment: Experimental Pressure and Sealing Plug

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DOPAS is a European project (FP7) involving cooperation between fourteen institutions from eight European countries. The aim of the project is the technology development for assessing tunnel plugging and sealing systems in geological disposal facilities for radioactive waste and the project is being coordinated by one of the Finnish partners – Posiva Oy. As part of this project one of the full scale experiments is the Experimental Pressure and Sealing Plug (EPSP), which will be constructed in a blind niche at the Josef Underground Laboratory in granitic rock by a consortium of three institutions from Czech Republic - RAWRA, the Faculty of Civil Engineering CTU and Řež NRI.

The aim of the poster is to provide information on the proposed design of the plug, the planned monitoring system and the technological centre which will allow the loading of the EPSP with various media (air, liquids and a bentonite suspension).

POSTERS BASED ON THE BELBAR PROJECT

Bentonite erosion: effects on the long term performance of the engineered barrier and radionuclide transport – the BELBaR project

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BELBaR is a Collaborative Project within the Seventh Framework Programme of the European Atomic Energy Community (Euratom) for nuclear research and training activities. The main aim of BELBaR is to increase knowledge of the processes that control clay colloid stability, generation and its ability to transport radionuclides. The overall purpose of the project is to come up with a new way of treating issues in long-term safety/performance assessment for radioactive waste repositories. The project started March 1, 2012 and has a duration of 48 months. The project has 14 partners from seven European countries.

The main aim of BELBaR is to reduce the uncertainties in the description of the effect of clay colloids on the long term performance of the engineered barrier and on radionuclide transport. This is done by:

- Improving the understanding on when bentonite colloids are unstable.
- Improving the quantitative models for erosion on the bentonite barrier for the cases when the colloids are stable.
- Improving the understanding of how radionuclides attach to clay colloids. This information will be used to formulate improved transport models for the assessment of radionuclide transport in the geosphere.

To meet the main aim a number of experimental and modelling activities will be undertaken within the project. BELBaR consists of six RTD (research and technical development) work packages and one project management work package. The current presentation gives examples of activities within the different work packages:

WP1 has produced a state of the art report. In the report, each country represented in BELBaR Work Package 1 has produced a description of how it currently considers colloids in performance assessment – the current state-of-the-art. That information was subsequently used to identify current issues affecting the treatment of colloids in performance assessment in national programmes. These issues are identified in a series of tables. The issues are linked in these tables as appropriate to the Work Packages considered in the BELBaR project, clarifying how the issues raised by national programmes regarding the treatment of colloids in performance assessment will be progressed by the BELBaR project itself.

In WP2 the erosion of bentonite in artificial fractures has been studied. One observation provide evidence that, following erosive mass loss of montmorillonite through contact with dilute groundwater at a

transmissive fracture interface, accessory phases (from within bentonite) remain behind and form layers at the solid/solution interface. No apparent attenuation of the erosion of montmorillonite was observed in the tests with added accessory materials relative to montmorillonite alone in a 1 mm aperture fracture.

One aim of WP3 is to improve the understanding of colloid/rock interactions in a crystalline rock. Data on colloid filtration obtained at the macroscopic scale (by performing transport experiments) have to be related with data obtained at the micro-scale, where colloid/rock interactions take place. The effect of the presence of bentonite colloids is always clearly visible in the “Grimsel case” (favorable for colloid stability). However, the presence of colloids is not very significant for cesium transport because cesium is desorbing from colloids. In the “Äspö” case, the presence of colloids is almost insignificant on the overall transport. But, in both the cases, retained colloids possibly contribute to RN retardation.

Knowledge of rheological properties of bentonite is of importance for the modelling and the prediction of the behaviour of a bentonite buffer during the operational time of a repository. The viscosity of bentonite/montmorillonite suspensions is a quantity needed to predict the velocity field outside a deposition hole where the seeping groundwater comes in contact with the extruded bentonite. WP4 provides suggestions regarding what systems to study with regard to montmorillonite/bentonite volume fraction, cation composition and ionic strength of the water in equilibrium with the bentonite. Bentonite may be highly thixotropic and it is found that the handling and preparation influence the outcome from rheological measurement.

One objective of WP5 is to estimate the loss of buffer or filling material through contact with dilute groundwater at a transmissive fracture interface and to give scientific support for performance assessment scenarios and calculations together with confidence building. The current status is that typical results from the model calculations differ from experiments by higher erosion rate and a clear bentonite wake: an observation is that in the model calculations there is not so much bentonite upstream migration, while in downstream a clear wake has formed. This is in clear contrast to the almost circular montmorillonite extrusion in the experiment.

BELBAR – The Sol/Gel Interface in Bentonite Erosion

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One scenario of interest for repository safety assessment involves the erosive loss of bentonite buffer material in contact with dilute groundwater flowing through a transmissive fracture interface. Results from artificial fracture experiments simulating this process indicate that there is a sharp boundary between the extruding phase and the eroding phase (see Figure 1). Furthermore, when the fracture system is rotated vertically, the eroding material sediments with gravity whereas the extruded material shows complete static resistance to flow under gravitational load.

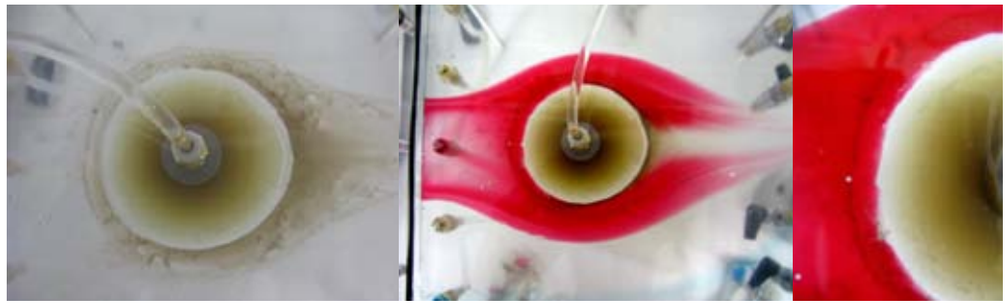

Figure 1. *Overhead photographic images of an artificial fracture test with sodium montmorillonite against deionized water in a 1 mm aperture fracture at 456 h (far left image) and after dye injection (middle and far right images).*

In order to further examine the sol/gel transition behavior observed in artificial fracture tests at the extrusion/erosion interface, a series of rheological experiments on sodium and mixed sodium/calcium montmorillonite suspensions covering a range of solids contents spanning from the “swelling paste phase” (15 vol-%) to the sol phase (0.1 vol-%) were initiated. The results clearly demonstrate the flocculated nature of sodium montmorillonite suspensions in the presence of electrolytes at the relatively low concentration of 17 mM NaCl. Even at very low solids contents (0.5 vol-%), an elastic response was observed. More surprising was the observation that the elastic component of the response for 50/50 calcium/sodium montmorillonite mixtures was much weaker than for sodium montmorillonite at the same salt concentrations. In fact, 50/50 calcium/sodium montmorillonite mixtures were less elastic in general than sodium montmorillonite suspensions, even when comparing the former in the presence of electrolytes to the later in deionized water.

Although these results indicate that 50/50 calcium/sodium montmorillonite might be more susceptible to erosion than sodium montmorillonite, artificial fracture experiments indicate that, at ionic strengths ≤ 1.1 mM, both materials erode equivalently. However, such comparisons are rather tenuous given that the rheological samples are prepared under batch conditions and the artificial fracture results are observed during dynamic (both swelling and erosion), flow-through conditions. Protocols for preparing rheological samples under more relevant conditions are being developed. The research has been performed as part of the (BELBAR, Bentonite Erosion: effects on the Long term performance of the engineered Barrier and Radionuclide Transport).

ABSTRACTS
Posters

Characterization of different bentonites to assess their erosion behaviour

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The quantification of colloid erosion from compacted bentonite barrier, in a high-level waste repository, is considered especially relevant because the eroded colloids may be stable and mobile, affecting radionuclide transport. Additionally, if particle erosion were continue the barrier integrity will be compromised.

At present, many experimental and modelling efforts are being devoted to elucidate the mechanisms, and the main chemical and physical factors affecting the formation of the bentonite gel, as forerunner of colloid formation.

In this study, colloid erosion behaviour of five bentonites form different origin (FEBEX (Spain), Mx-80 (USA), Mylos (Greece), B75 (Czech) and Khakassiya (Russia) was analysed.

Colloid erosion behaviour of the different bentonites was compare with an experimental set-up designed to study the colloid formation in a confined system to minimize the loss of density of the bulk bentonite during hydration. Experiments were carried out with raw bentonites, compacted at 1.65 and 1.8 g/cm^3 and in deionised water, to measure the maximum generation masses under most favourable conditions for colloid erosion.

A complete characterisation of the studied bentonites (smectite content; presence of certain accessory minerals, cations present in the interlayer; exchange capacity,...) was carried out to explain the erosion behaviour of the different bentonites.

In addition, the stability behaviour of colloid suspensions for the five studied bentonites was also performed as a function of the pH and ionic strength, by combining electrophoretic measurements, allowing determining the surface potential of the clay particles (Zetapotential) and Photon Correlation Spectrometry (PCS) that allows establishing the mean size of the particles under different conditions. The stability of bentonite colloid suspensions is critical to assess their role in radionuclide transport.

The research leading to these results has received funding from EU Seventh Framework Programme (FP7/2007-2011) under the grant agreement Nº 295487 (BELBAR, Bentonite Erosion: effects on the Long term performance of the engineered Barrier and Radionuclide Transport).

POSTERS BASED ON THE CARBOWASTE PROJECT

Characteristics of the Irradiated Graphite from the Triga Thermal Column

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Irradiated graphite in Romania arises only from the thermal column of the VVER and TRIGA research reactors. While the first is under decommissioning and requires a fast solution for its irradiated graphite management, the second one is still in operation.

Under CARBOWASTE project graphite from the TRIGA thermal column was investigated in order to define the changes occurring in microstructural characteristics as effect of irradiation, impurity content of the virgin graphite as well as the radionuclide content of an irradiated graphite brick.

Combining gamma scanning and beta spectroscopy, a large number of graphite samples were measured for the characterization of the radionuclide content. Techniques and methods available in IRN allowed the determination of Co-60, Cs-134 and Cs-137, Eu-152 and Eu-154, C-14 and H-3 activities.

Based on the impurity content as measured by X-Ray Fluorescence and on the irradiation history and real distribution of the neutron flux along the column, numerical simulations were performed to assess the main radionuclide accumulation such as C-14 and Cl-36. To overcome incomplete impurity content and the unknown position in the thermal column of brick used in experimental investigations, the Eu152/Eu154 ratio as function on the irradiation flux was used.

Numerical calculation of radioisotopes content allowed the estimation of C-14 and Cl-36 activity for the entire graphite of the thermal column. This made possible the performance assessment for its disposal in a near surface repository focused on the impact of these two radioisotopes, based on a conceptual dose assessment model aiming to serve as a template on which adjustments are still to be made as soon as new data become available.

Performance assessment was accomplished with GoldSim software for irradiated graphite from TRIGA reactor, following only C-14 (gas and water pathway) and Cl-36 (only water pathway) contribution. The GoldSim simulations predict that maximum individual dose due to C-14 inhalation as CO_2 released from repository is 0.013 mSv/year (lower than dose constrain of 0.3 mSv/year) while maximum individual dose for Cl-36 is 0.4 mSv/year, exceeding the dose constrain. According to these data near surface disposal could be accepted only if adequate treatment to reduce Cl-36 activity in the waste package will be applied or very long-lived engineered barriers are considered to reduce its release from the near field. Cementitous barriers would decrease the C-14 release and its contribution in the total dose due to carbonation reaction, but they can only delay the Cl-36 release and not reduce it. Further graphite characterization and leaching tests should be performed to confirm the predicted values and determine the C-14 forms and Cl-36 activity.

Removal of gamma emitting radionuclides such as Co60, Cs137, Eu152, Eu154 from irradiated graphite of TRIGA reactor was tested along the project in a large through a large number of experiments using different acid solutions. Highest efficiency (72%) in gamma emitters removal was obtained for a mixture of nitric and phosphoric acid. Chemical treatment with acid solutions can also lead to some beta emitters' removal. Decontamination efficiency with regards to gross beta activity is highest (71%) in case of sample treated in solution of hydrochloric and sulphuric acids in demineralized water.

ABSTRACTS
Posters

The Study of Relation between Treatment and Disposal on the Performance of RBMK-1500 Graphite Disposal in Crystalline Rock

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As the other countries having graphite moderator reactors, Lithuania is facing the open question on further graphite management and disposal technologies with the dismantling of RBMK-1500 type reactors at Ignalina NPP. In the Final Decommissioning Plan of Ignalina NPP it is foreseen to store irradiated graphite (reactor core blocks and sleeves) at interim storage for app. 50 years. However, the international consensus exists that interim storage itself is not a solution for any radioactive waste and it should be followed by its safe disposal. Different options of graphite disposal (near surface, geological disposal) are being analysed in different countries considering country specific aspects (the amount, radiological characteristics, availability of repositories, etc.).

In this paper the results on relation between treatment and disposal on the performance of RBMK-1500 graphite disposed of in crystalline rocks are presented. Modelling of ^{14}C migration in the near field and far field was performed with the source term based on LEI results on RBMK-1500 inventory, illustrative rates for representation of possible differences on non-treated/treated graphite and conceptual models developed. The importance of waste leaching rate was studied within the context of different performance of engineered barrier (in terms of sorption, limited solubility) and considering possible graphite encapsulation in cementitious material. It was obtained that the impact of the options (treatment vs. no treatment of graphite) on the ^{14}C flux to geosphere is not straightforward. While reasoning the option of treatment/not treatment the inventory, leaching rates, barrier performance and transport conditions need to be considered.

This research was performed within the framework of the European Project on 'Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste (CARBOWASTE)' under the 7th EURATOM Framework Programme (Grant agreement No 211333).

Modelling of ^{14}C Distribution in Ignalina NPP Unit 1 RBMK-1500 Reactor Graphite Rings

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Long-lived ^{14}C is one of the most important radionuclides in the irradiated RBMK reactor's graphite. The main precursors of this radionuclide are nuclides ^{13}C (raw material) and ^{14}N (initial impurity). The propagation of cooling helium-nitrogen gases mixture (which circulates around RBMK reactor stack) into the graphite pores may additionally increase the quantity of nitrogen impurity and consequently increase ^{14}C production in RBMK-1500 reactor's graphite. So a conservative estimation of possible nitrogen impurity content coming from the cooling gases of RBMK-1500 reactor was made.

Axial distribution of induced ^{14}C activity in the RBMK-1500 reactor's graphite rings was obtained by the help of neutron activation modelling. The neutron fluxes in the graphite rings were modelled firstly and then using these fluxes activation modelling was performed for two basic cases, i.e. using maximal and minimal concentrations of impurities in graphite.

Calibration of developed model against experimental activity measurement data (measured by the Institute of Physics of the Centre for Physical Sciences and Technology) of irradiated graphite ring specimens from RBMK-1500 reactor) was performed. It was done by the reverse activation modelling way, i.e. the concentration of nitrogen impurity, which activation lead to the generation of ^{14}C radionuclide, was altered until the modelled activity of ^{14}C matched best the measured one. In this way the "explanatory" initial concentration of nitrogen impurity was derived. As measurements for ^{14}C activity were made in three different positions along the reactor axis, the best match was obtained by the help of least squares method, i.e. the "explanatory" nitrogen concentration was the one that gave the minimal sum of the squares of the differences between measured and modelled activities in all these positions.

Comparison of the modelled specific activities of ^{14}C using derived "explanatory" concentration of nitrogen impurity in the graphite rings in the region of active core, top and bottom reflectors shows very good agreement with the measurements (modelled and measured specific activities of ^{14}C coincide taking into account measurement uncertainties). However, it should be noted that performed numerical modelling of induced ^{14}C activity in the graphite rings does not take into account two processes – the possible release of emerged ^{14}C from the graphite to the surrounding material and opposite process – uptake of ^{14}C radionuclide (or impurities which are activated later in the graphite) from the surrounding material.

This research was performed within the framework of the European Project on 'Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste (CARBOWASTE)' under the 7th EURATOM Framework Programme (Grant agreement No 211333).

ABSTRACTS
Posters

Investigation of irradiated graphite decontamination techniques by high temperature pyrolysis under the CARBOWASTE Programme

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The CARBOWASTE Progamme was initiated under the 7th Framework Programme in April 2008 as a consortium of 31 partners (co-ordinated by the Forschungs Zentrum, Jülich) from industry, university and research institutes of the EU, but also included PMBR and NECSA of South Africa. The total grant was EUR 6 million for a project value of EUR 11.55 million.

The objective was to examine the possibilities for decontamination, re-cycling and storage evaluation of irradiated graphite (i-graphite) coming from the moderators and other sources in the older test and commercial reactors in the EU, as well as an assessment of this in waste storage and identification of remaining uncertainties.

The Transuranium Institute (ITU) Karlsruhe contributed to the Work Package on the physical decontamination processes of irradiated graphite (WP4) and to monitor releases of the principal contaminants (C-14 and H-3) during high temperature pyrolysis by means of Knudsen Cell spectroscopy. This was done on three i-graphite types: moderator from the Merlin reactor in Jülich, i-graphite from the LATINA reactor in Italy and finally a sample cored from the dense, outer i-graphite layer of a HTR pebble fuel irradiated in the HFR, Petten. For the i-graphite from all 3 sources (Merlin and Latina reactors and from the HTR fuel) the loss of C-14 was too low to establish compared to background of contaminations at mass 14 (such as N-14).

Despite the failure of the Knudsen Cell technique to establish any results for C-14 or H-3, without much more controlled background signals, other testing has shown that these isotopes are released under various annealing conditions. Nevertheless, it was possible to demonstrate with the i-graphite of the HTR fuel outer layer that there was a loss of a volatile fission product – Cs-137 - from the irradiated graphite at low temperatures under vacuum in the Knudsen cell, and suggests that other volatile or gaseous radioactive fission products (not just C-14 and tritium) could also be released at such low temperatures. This in turn suggests that physical decontamination processes can be useful as a general decontamination process of i-graphite.

POSTERS BASED ON THE PEBS PROJECT

Generic poster on the PEBS project: PEBS- Long-term performance of Engineered Barrier Systems

PEBS: Modelling the interaction of corrosion products with compacted bentonite in heating and hydration laboratory tests and long-term conditions of a HLW repository

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The main aim of the project PEBS (Long-term Performance of the Engineered Barrier System) is to evaluate the sealing and barrier performance of the EBS with time, through development of a comprehensive approach involving experiments, model development and consideration of the potential impacts on long-term safety functions. The experiments and models cover the full range of conditions from initial emplacement of wastes. These aspects are integrated in a manner that will lead to a more convincing connection between the initial transient state of the EBS and its long-term state that provides the required isolation of the wastes. The existing coupled thermal (T), hydrodynamic (H), chemical (C), and mechanical (M) models have been improved to: (1) Account for different types of waters in clays; (2) Account for the porosity changes caused by swelling phenomena; (3) Improve the chemistry of reactive gases. Here we present coupled THCM models of the heating and hydration corrosion experiments performed at the Ciemat facilities to study the corrosion products generated at the canister/bentonite interface under repository conditions and analyse how the corrosion products affect the properties of the bentonite. The experiments were performed on samples of several lengths and temperatures. They were modeled with 1D finite element grids. The chemical system is defined in terms of the following primary species: H_2O, H^+, Ca^{2+}, Mg^{2+}, Na^+, K^+, Cl^-, SO_4^{2-}, HCO_3^-, O_2(aq), Fe^{2+} and SiO_2(aq) and accounts for homogeneous and heterogeneous reactions. Magnetite precipitation is controlled by a kinetic law. For the most part, simulations agree well with experimental data. The kinetic parameters of magnetite precipitation were calibrated by using Fe weight content data. For the most part, simulations agree well with experimental data. Model results indicate that: 1) The main properties of the bentonite remain unaltered; 2) Magnetite is the main corrosion product; 3) Fe^{2+} is sorbed by surface complexation; 4) Fe^{2+} cation exchange is less relevant than Fe^{2+} sorption; and 5) Corrosion products penetrate a few mm into the bentonite.

A reactive transport model of the long-term hydrochemical evolution of the bentonite porewater in the bentonite barrier of a spent-fuel carbon-steel canister repository in granite has been performed which improves the models performed within the NFPRO and PAMINA Projects. Long-term simulations of the evolution of the system have been performed for the reference scenario and for a set of variant scenarios of sensitivity runs. Model results indicate that canister corrosion causes a marked increase in pH and the concentration of dissolved Fe^{2+} and a decrease in Eh. Most of the released Fe^{2+} diffuses from the canister into the bentonite where it precipitates mainly as magnetite and to a lesser extent as siderite. Fe^{2+} sorbs by surface complexation on weak sorption sites and undergoes cation exchange. Sorption plays a relevant role in the geochemical evolution of bentonite. The competition of Fe^{2+} and H^+ for the sorption sites near the canister causes pH, Eh and sorption fronts. Model results indicate a significant reduction of bentonite porosity due to mineral precipitation near the canister/bentonite interface which could result in the clogging

of the bentonite pores. A detailed sensitivity analysis of the key geochemical variables has been performed to: 1) The corrosion rate; 2) The effective diffusion coefficient of solutes; 3) The water flow through the granite at the bentonite/granite interface; 4) The cation selectivities; and 5) The chemical compositions of the bentonite and granite porewater. Our results provide a more realistic description than those of previous models of the long term evolution of the geochemical conditions in the near field of a repository in granite.

POSTERS BASED ON THE SKIN PROJECT

SKIN - Thermodynamic and Structural Data for the Radium and Barium Sulphate System

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Radium chemistry accounts for a little bit more than one century and radium, being a naturally occurring element, was not discovered until 1898 in trace amounts by M. and P. Curie. Due to its radioactive nature few data and reliable experiments exist in the literature and the number of research groups at universities able to work with relevant amounts are few these days. Radium is one of the radioactive elements that dominate the naturally occurring radioactive material (NORM) and its technically enhanced equivalent (TNORM). The main anthropogenic sources of radium are oil extraction and natural gas production, manufacturing of phosphoric acid, uranium mining and the storage of spent nuclear fuel. When performing a safety analysis of a final repository for spent nuclear fuel of the underground type planned for example in Sweden, radium will dominate the dose to man in the long term perspective. In all the cases mentioned above modelling of the radium behaviour is one of the paths to understand and minimise the doses to the public either living in areas where radium contamination exists or working with radium containing material.

In this work the radium sulphate crystal structure has been investigated using XRD and, for the first time, EXAFS. The unit cell parameters, a = 9.07 Å, b = 5.52 Å, c = 7.28 Å and V = 364.48 Å^3 belonging to orthorhombic space group were determined and support the fact that radium sulphate is isostructural with barium-, strontium- and lead sulfates. The study of co-precipitation and recrystallization of the radium and barium sulphate systems has been done. The reactions of the radium, barium and strontium co-precipitation systems in sulfate media are showing an Arrhenius behavior, *i.e.* the relationship between ln*(k)* and 1/*T* is linear in the temperature range of 10 – 30 ºC. These findings show that it is possible for radium to co-precipitate with barium, strontium and lead in sulphate media to form a substitution solid solution. The activation energy of the pure radium, barium and strontium sulphates decreases in the order Sr>Ba>Ra, which may be correlated to their ionic size, since this affects the ability to lose their hydration water. The activation energy (E_a) is lower for the precipitation of pure radium than for radium in a mixture of radium and barium.

The study of the kinetics of ^{223}Ra/^{133}Ba recrystallization on the surface of synthesized barium sulfate crystals has been studied. The use of short-lived ^{223}Ra isotope (half-life 11.4 days) allows reaching concentrations below 10^{-13} M as found in nature and avoids generation of long-lived radioactive waste.

The research leading to these results has received funding from the European Union's European Atomic Energy Community's (Euratom) Seventh Framework Programme FP7/2007-2011 under grant agreement n° 269688 ("SKIN" project) and the Swedish radiation protection authority (SSM).

ABSTRACTS
Posters

Uptake of Radium during barite recrystallization

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In recent safety assessments for the direct disposal of spent nuclear fuel, Ra has been identified as a main contributor to dose after 100,000 years [1]. Due to the likely presence of $BaSO_4$ which is formed by a reaction of Ba present as a fission product and sulfate ubiquitous in ground waters, it is expected that Ra will form a $Ra_xBa_{1-x}SO_4$ solid solution via recrystallization, which would lower the Ra concentration in solution significantly. However, due to a lack of reliable data, this solid solution system is currently not considered in long term safety assessments for nuclear waste repositories. For instance, the solubility of the pure $RaSO_4$ end-member is not well defined. Furthermore, available literature data for the interaction parameter a_0, which describes the non-ideality of the solid solution, vary by about one order of magnitude [2,3]. The final Ra concentration in solution in this system is extremely sensitive to the amount of barite present, the difference in the solubility products of the end-member phases, and the degree of non-ideality of the solid solution phase.

The EU-funded SKIN project investigates slow processes in close-to-equilibrium conditions for radionuclides in water/solid systems of relevance to nuclear waste management. Within the SKIN project we have combined a macroscopic experimental approach with ToF-SIMS and thermodynamic modelling to study in detail how a Ra containing solution will equilibrate with solid $BaSO_4$ under repository relevant conditions. Batch experiments at close to equilibrium conditions show a final Ra concentration which is several orders of magnitude lower than the solubility of pure $RaSO_4$. For the first time, the spatial distribution of Ra and Ba within the recrystallized barite powders - analysed using an ION-TOF TOF-SIMS - confirms the uptake of Ra into the solid. Solid-solution aqueous-solution equilibrium calculations were carried out for the $BaSO_4$ – $RaSO_4$ – H_2O system with the GEMS – PSI code [4] in combination with the NAGRA – PSI thermodynamic database [5]. Thermodynamic modelling fits the experimental data best with an interaction parameter a_0 = 0.6 and a solubility product K_{RaSO4} = -10.41.

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SKIN - Slow processes in close-to-equilibrium conditions for radionuclides in water/solid systems of relevance to nuclear waste management

URANIUM/IRON(III) OXIDES

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SKIN project is a 36 months (2011-2013) Collaborative Project under the Seventh Framework Programme of the European Atomic Energy Community (EURATOM).

The project studies slow processes influencing radionuclide mobility in close-to-equilibrium scenarios in a detailed and systematic manner in relation to surface properties, surface site detachment/attachment kinetics, irreversible sorption and surface incorporation, for cases relevant to the assessment of radionuclide mobility in nuclear waste repository sites. WP3 of this project is focussed on (a) the assessment of the kinetics of dissolution of tetravalent oxides under near to-equilibrium conditions, and (b) the assessment of the impact of major systems present in the repository environment on the rate of dissolution of matrix-related material and retention/release of radionuclides. Specifically, WP3.1 deals with the interaction between uranium and Iron (III) oxides. The retention of radionuclides, U(VI) among them, by Fe(III) solid phases has been long studied by different authors, mainly from the perspective of sorption equilibrium. The long time frames of interest for the safety assessment, and the massive occurrence of iron compounds in the repository highlight the possibility of more intimate interactions between U(VI) and iron solid phases.

ABSTRACTS
Posters

POSTER FROM THE EUROPEAN COMMISSION JOINT RESEARCH CENTRE

EU Support on Disposal of High Level Radioactive Waste and Spent Fuel to neighbouring and other countries

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Recognising the importance of safety of radioactive waste management worldwide, the European Union is providing significant technical support to regulatory authorities and operating organisations dealing with radioactive waste management in neighbouring and other countries. This support has been implemented through European Commission mechanisms such as the Technical Assistance to the Commonwealth of Independent States Programme - TACIS, the Instrument for Nuclear Safety Cooperation - INSC and the Instrument for Pre-accession Assistance - IPA.

This paper summarises the main mechanisms and technical areas of assistance, as well as the main TACIS and INSC activities on improving safety measures for safe disposal of high level radioactive waste and spent nuclear fuel. It also outlines the way forward and the scope and objectives of the INSC follow-up instrument envisaged for the 2014-2020 period.

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