Institute for Energy

PROCEEDINGS

WORKSHOP ON

Design and Assessment of Radioactive Waste Packages

November 21st - 22nd 2006
Petten, The Netherlands

Organised by
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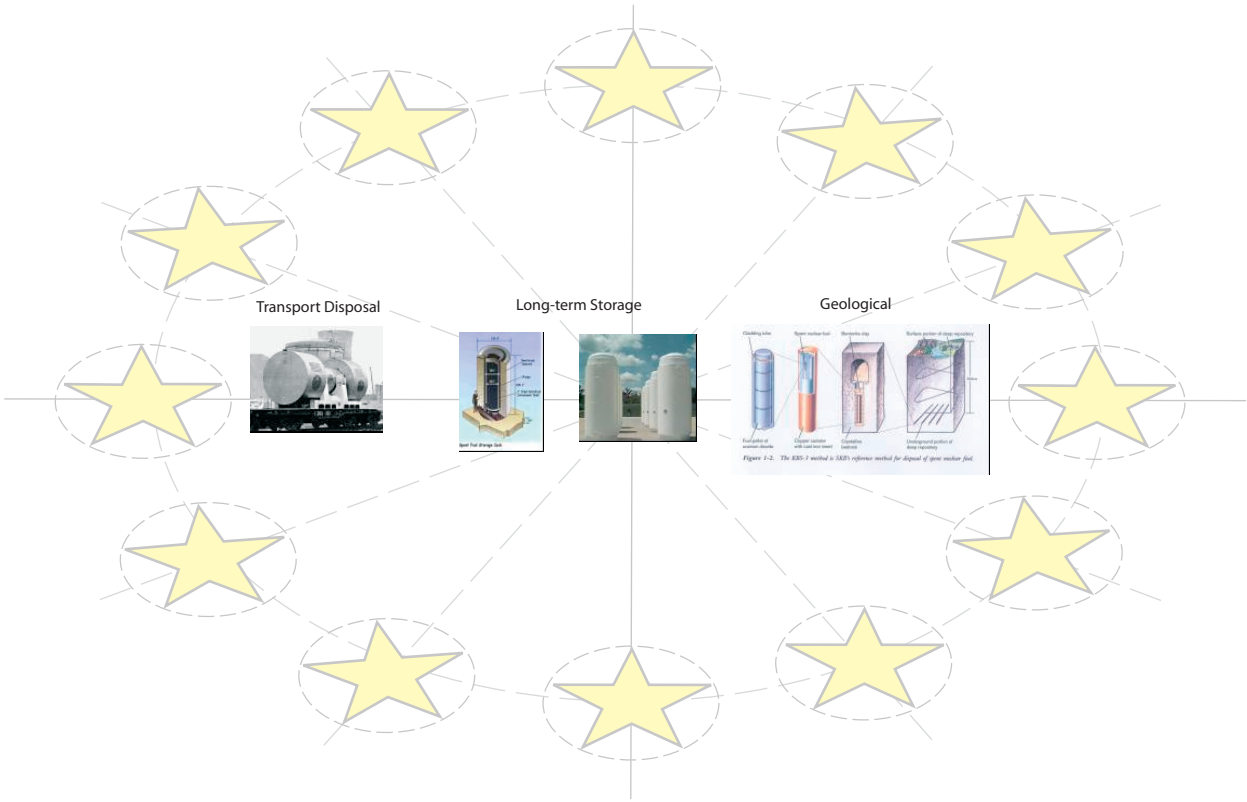

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2006 Workshop on Design and Assessment of Packages for Radioactive Waste

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Contents

1 INTRODUCTION

Effective waste management is recognised as a determining factor for the continued use of nuclear energy and the waste package itself is central to the process for transport, storage as well as for geological disposal. Different designs and concepts have been and are being developed at national level in many countries. However as with many nuclear technologies, acceptance can be enhanced by developing expert consensus on generic issues.

With this aim in mind, the Joint Research Centre decided to organise a dedicated workshop on waste package design in 2005. Its aim was to provide an overview of ongoing R&D and best practices for design and assessment of waste packages for handling, storage, transport and disposal of radioactive waste. This 2006 Workshop was organized to follow-up the successful 2005 Workshop. The technical presentations covered the following topics.

- Design premises and guidelines for waste package design,
- Non-destructive testing and qualification of components,
- Container degradation and integrity,
- Drop test of transport containers,
- International collaboration.

During the 2005 Workshop there was a strong interest to explore the possibilities for international collaboration on design and assessment of packages for radioactive waste and spent nuclear fuel. To bring this forward the Workshop participants were split into three groups to identify areas for which there would be a strong added value for international collaboration:

- Long-term behaviour and degradation of waste packages;
- Manufacturing, inspection and qualification of waste packages;
- Drop tests to ensure structural integrity for dynamic impact.

This proceedings volume provides a overall synthesis of the workshop, together with the slides used in the presentations themselves.

FINAL PROGRAMME, Day 1

Time	Presentation
09:00-09:20	Welcome and Introduction K-F Nilsson, JRC Petten
09:20-10:00	Tae Ahn, NRC, USA, *"Issues in the licensing review of the potential Yucca Mountain repository: container, cladding and waste form"*
10:00-10:30	Hidekazu Asano, RWMC, Japan, "Study on Evaluation Method of the Long-term Integrity of Waste Package Final Closure*"*
10:30-11:00	Hannu Hänninen, Helsinki University, Finland, *"The time effect on the integrity for canisters for geological disposal"*
11-11:20	Coffee break
11:20-11:50	Neil Carr, NIREX, UK, "*Shielding Integrity Testing of Transport Containers*"
11:50-12:20	Ian Norris, The Welding Institute, UK, *"NDT techniques and structural integrity evaluation for nuclear waste containment applications"*
12:20-13:30	Lunch
13:30-14:00	Bernhard Droste, Uwe Zencker, Holger Völzke and Frank Koch, BAM, Germany, *"BAM Guidelines for the Safety Assessment of Transport and Storage Casks for Radioactive Materials"*
14:00 – 14:40	Pitkänen, J.*, Leskinen, N.**, Rydén H.** & Emilsson G.*** , "Defect types, NDT and preliminary acceptance criteria approach based on the damage tolerance analysis and international standards in the manufacturing of nodular cast iron insert". * Posiva Oy,** SKB,*** Bodycote
14:40-15:10	Chris De Bock, ONDRAF/NIRAS, Belgium"The ONDRAF/NIRAS Waste Overpack and Supercontainer Design"
15:10-15:30	Coffee break
15:30-16:00	Miroslav Picek and Petr Kotnour, KOD A JS a.s., Czech Republic, *"Storage and transportation system – ŚKODA VPVR/M casks system"*
16:00 – 16:30	P. Poskas, A. Smaizys, E. Narkunas, Lithuanian Energy Institute, Nuclear Engineering Laboratory, Kaunas, Lithuania, *"Modelling of RBMK-1500 Spent Nuclear Fuel Characteristics and Comparison with Available Data"*
16:30-17:00	LPM Van Velzen, NRG The Netherlands, *"Proficiency Test for Non-Destructive Assay of 220 ltr Waste Drums by gamma systems: Feasibility Study & Design if First Proficiency Test"*
17:00-17:30	Dong-Hak Kim, Kyoung-O Nam, Ju-Chan Lee, Kyung-Sil Bang and Ki-Seog Seo, Korea Atomic Energy Research Institute, "Conceptual Structural Assessment of Polymer Concrete HIC for NPP Waste"

Day 2

Time	Presentation
08:30-09:00	R. Hüggenberg, GNS, Germany, *"Comparison of the Behaviour of a finned and smooth Surface of a Type B(U)-Cask made of DCI during a pinion Drop Test"*
09:00-09:30	Pierre Van Isegem, SCK-CEN, Belgium , *"ENTRAP - the European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages. Objectives, organisation, main output"*
09:30-10:00	Bernhard Droste and Volcker Noack, BAM Germany, "*German support for utilization of nuclear submarines in Russia"*
10:00-10:20	Coffee break
10:20 – 11:00	Short presentations and ideas for future collaboration – Tae Ahn, NRC – H. Asano, RWMC – K-F Nilsson, JRC
11:00-12:30	Group 1: Long term behaviour of Waste Packages Group 2: Manufacturing, Quality Control and NDT Group 3: Impact Accidents
12:30-13:30	Lunch
13:30 – 14:00	Summary of Working Group Discussions
14:00 – 15:30	Discussions on forms for future collaboration and administrative issues for Forum and conclusions of Workshop
15:30	End of Workshop

2 SUMMARY OF TECHNICAL PRESENTATIONS

2.1 General Design and Guidelines

Tae Ahn from NRC made an overview of the licensing issues for the Yucca Mountain project as regards containers, spent fuel cladding and waste form [1]. In the US a canister system for Transport, Ageing and Disposal (TAD) is now being considered. The spent fuel will be packaged into the TAD containers at the nuclear sites and transported to Yucca Mountain for final disposal. The TAD container brings several advantages, in particular it minimizes the radiological risks at the Yucca Mountain site. Defect criteria and structural integrity for dynamic impact will be important requirements.

H. Hänninnen made an overview of the cyclic climate changes in Scandinavia and where the inevitable ice-ages need to be considered [3]. Major ice-ages occur with intervals of about 100,000 years. The consequences of future ice-ages constitute the most important uncertainty factor in the long-term performance assessment of Scandinavian repositories. The Swedish and Finnish waste management organizations have therefore dedicated much effort to this. The mechanical loads from ice-age and rock shear movements that may follow after ice ages is one important item. A recent large study by SKB and partners showed that the probabilities for mechanical failure remain extremely small. Saline groundwater and oxygen-containing glacial melt water may infiltrate as a consequence of the land movement from ice-age. This clearly complicates the prediction of the very long-term corrosion behaviour of copper canisters, but generally safety margins should be quite large.

N. Carr [4] made a brief overview of the UK concept for disposal and storage of spent fuel and high level nuclear waste where the Swedish KBS-3 concept has been identified as viable reference concept. There is no specific geographic area identified in the UK but it is preferable if different kinds of waste (low and intermediate level, HLW/SNF) can be disposed in the same location (co-location concept). Specific waste package specifications are now being defined.

Bundesantalt für Materialprüfung in Germany provides Guidance to support the assessment procedures and design for safe transport and storage casks. B. Droste [6] presented three different Guidelines on a) Application of Ductile cast Iron, b) Numerical analysis procedures and c) calculation and assessment of lid and trunnion systems. These Guidelines are based on application of state-of-the art methodologies in finite elements and fracture mechanics.

Skoda is a major manufacturer and developer of canisters for radioactive waste and spent nuclear fuel. M Picek and P. Kotnour [9] made an overview of the Skoda's product family which includes CASTOR and CONSTOR casks designed under license from GNS as well as their own developed casks for research reactor fuel.

C. de Bock from the Belgian waste management organization ONDRAF/NIRAS presented the development of their container for geological disposal in Boom clay [8]. The previous concept, SAFIR-2, was modified after various reviews, including a NEA review. There were some doubts regarding the reliability of the system functioning as

well as for the barriers. For the container pitting corrosion was a major concern. A new container, referred to as ”Supercontainer” has been designed for vitrified high level waste (v-HLW), and UOX and MOX spent fuel. The Supercontainer has a 30 mm thick carbon steel overpack, surrounded by a buffer made of Ordinary Portland Cement (OPC), and a stainless steel liner. There is cementitious backfill plus concrete wedge blocks between the Boom Blay and the Supercontainer. The corrosion protection is provided by the OPC buffer’s alkaline environment. The carbon steel overpack should isolate the radionuclides in a watertight container during the thermal phase in the host rock (minimum 200 years for spent fuel and 500 years for the vitrified waste). The quality of the lid weld is important to guarantee the long-term integrity. Different welding techniques (electron beam, friction stir, arc, and laser beam) have been compared with respect to cost and weld quality.

P. Poskas from Lithuanian Energy Institute made an overview of the situation in Lithuania and presented development of codes to determine the characteristics of RMBK spent nuclear fuel [10]. Lithuania plans to store its spent nuclear fuel in GNS dual purpose casks for a period of 50 – 100 years at the Ignalina site. There is no decision whether geological disposal is the end-point. Lithuania is considering different disposal options. The emphasis has been for crystalline rock and clay, but salt domes could also be an alternative. An accurate characterization of the spent nuclear fuel is very important for the safety analysis of storage as well as geological disposal systems. P. Poskas presented the basic assumptions and the capabilities of the code Scale-5, which has been specifically for characterization of spent nuclear fuel of different types. The code computes nuclide content, activities, neutron and gamma source spectrums, and radiation dose for a cask. The analyses can be done for different temperatures, geometry, material compositions, enrichments and burn-ups. There was general very good agreement between predictions and experimental data for different spent fuel characteristics such as mass fraction of different radionuclides or activity versus burn-up.

2.2 Waste Package Integrity and Long-term Degradation Mechanisms

H. Asano from RWMC, presented ongoing research to evaluate the long-term integrity of the welds in the carbon-steel overpack. The integrity evaluation incorporates experimental as well as analytical programmes. It considers uniform corrosion and stress corrosion cracking (SSC); neutron radiation embrittlement; and fracture mechanics analysis of weld defects. Uniform corrosion takes preferentially place in the weld material, whereas SSC is more likely in the base material. Neutron irradiation is not expected to affect the long-term integrity. Critical defect lengths have been determined from stress analysis combined with fracture mechanics. The computed critical lengths are quite large and the probability of detection for these defect is close to 1. Although the analysis indicate very large safety margins against failure, further research should still be pursued to improve analyses and better understanding of basic degradation mechanisms.

Corrosion is generally the long-term degradation mechanism which is most critical for disposal canister integrity. The allowance of corrosion and probability of canister failure depends on the disposal concept. The isolation of the radionuclide by the canisters is the basis for both the Scandinavian KBS-3 concept as well as for the Yucca Mountain concept. The canisters are therefore designed to be very resistant or

even immune to corrosion. There is convincing evidence that the pure copper used for the KBS-3 canisters is virtually immune against corrosion in the reduced conditions at the repository. According too H. Hänninen [3], the corrosion conditions may, however, change after glaciation due to infiltration of deep saline groundwater and oxygen-containing melting water. Although the likelihood for corrosion induced failure remains extremely low, the issue may need to be looked into further. The corrosion problem at Yucca Mountain is quite different since there will be oxygen present and temperatures will be relatively high. The corrosion resistance for the Alloy 22 relies on a passive film. Uncertainty effects from long-term chemical or structural changes in passive film stability warrant additional consideration for uniform corrosion [1]. Stress corrosion cracking and localized corrosion are deemed as very unlikely

2.3 Non-destructive Testing and Qualification

Non-destructive testing and qualification of waste packages to ensure that acceptance criteria are fulfilled is a very important aspect to ensure that waste package integrity. Inspection of waste packages is quite challenging for various reasons as discussed by I. Norris from TWI [5]:

- a large number of containers and large areas need to be inspected;
- the geometry of the components could be quite complex;
- the environment such as temperature and possibility for vibrations;
- what to inspect (corrosion, crack-like flaws, material microstructure);
- acceptance criteria may require that quite small defects or variations in material structure or geometry need to be detected;
- radiological protection may require remote access.

TWI has explored different techniques such as acoustic emission, penetrant, eddy currents, phased array ultrasound, radiography, thermography and shearogrpahy. Their conclusion is that there is no universal technique. The techniques are complementary and which one to use depends on the time requirements, geometry, size and type of defects to detect. Several examples and list of the pros and cons of the different techniques are given in [5].

N. Carr from Nirex presented work performed in the UK do develop requirements for standards and guidance documents for detection of different kinds of flaws and how they may affect dose rated [4]. The methods include gamma and x-ray radiography, ultrasonics.

Sweden and Finland have essentially the same geological disposal concept (copper/cast iron canisters in crystalline rock) and plan to have repositories in operation between 2017 and 2020. SKB submitted their license application for building an encapsulation plant in November 2006. The qualification of canisters and the associated non-destructive techniques are key areas and much effort is dedicated to it. SKB works now in close collaboration with the Finnish waste management organization POSIVA to develop the specific requirements and design of non-destructive testing and qualification. J. Pitkänen presented an overview of this work

[7]. SKB/Posiva are in the process of developing preliminary acceptance criteria. The canister is then split into different zones. The inspection requirements for the zones differ because the loading differ as well as the possibility to perform non-destructive inspections. Linear phased array ultrasound is the main technique for inspection of the inserts, but how the inspections are performed varies for the different zones. The suitability of the proposed methods will be evaluated in the coming years and probability of detection will be established. In parallel to the technical development, there will also be supporting activities to ensure that the work is done in accordance with national requirements and international standards.

2.4 Drop Tests and Analysis

Germany has a very active research programme on dynamic impact for transport casks. An important reason to this is probably that GNS, who produces the CASTOR and CONSTOR casks is one of the leading manufacturers. B. Droste made a brief overview on numerical simulations of drop tests and aircraft crash into spent fuel storage casks [6]. BAM has also build a large-scale drop test facility outside Berlin Examples of large-scale drop tests were presented.

R. Hüggenberg from GNS [11] made a presentation on a recent test programme where one cask with cooling fins were subjected to a 1 metre drop test on a pin. The purpose of the test was to assess how cooling fins affect impact response. The fins were locally machined away at the impact location in one of the tests whereas the fins were left intact for the second test. The most important observation was that the impact load and peak strains were lowered by the fins. For the strain the reduction was about 20% and is caused by the additional energy needed to plastically deform the fins. The experiment is now being analysed by dynamic finite element codes by JRC-Petten as well as by GNS.

Korea develops a polymer concrete High Integrity Container (HIC) for transport and storage of radioactive waste. D-H Kim presented ongoing numerical simulation work to determine the integrity and safety margins for some different design concepts under different dynamic loading conditions [12]. The basic design which uses only polymer concrete failed for the drop tests. Different reinforcement concepts with steel plated and steel structures have also been analysed. The concept with only steel plate could also fail for drop tests whereas the concept with steel structure did not fail.

2.5 Networking and international collaboration

The management of the radioactive waste from nuclear submarines in Northwest Russia is a safety issue for Europe. Consequently there has been a large number of joint activities between Russia and different EU countries such as Sweden, UK and Germany to ensure to support safe management. B. Droste [13] presented the results of a joint project between the German and Russian governments to design and construct an onshore long-term storage facility for nuclear submarine reactor compartments in Sayda Bay near Murmansk. The construction started in 2004 and the first section was commissioned in 2006. The project is expected to continue until at

least 2013. After that the site will gradually become larger and it will be possible to store different kind of radioactive components.

EN-TRAP is a European Network that started in 1992 [14] for regulatory quality checking of radioactive waste packages. It consists of twelve full members and associated laboratories. The work done in EN-TRAP include round-robin exercises which has lead to harmonisation of procedures. A number R&D project proposals have been developed by the Network and specific bi- and multi-lateral collaborations have been generated. An example of a recent EN-TRAP activity was presented by L. van Velzen [15] on Proficiency Test for Non-Destructive Assay of waste drums by Gamma Assay Systems. The idea is to organize on a regular basis non-destructive assay systems for radiological waste characterization. The first step, which is the specification of a European reference drum was presented.

3 BRAINSTORMING DISCUSSION ON FUTURE COLLABORATION

To promote the discussion three presentations were first made. T. Ahn made an overview of collaboration models and topics for which the NRC would be very interested in working with European partners [16]. NRC is interested in general information exchange and can provide information from NRC guides as well as procedures and Guides from ASME and ASTM. Specific technical items of interest include: long-term behaviour of fuel cladding (in particular hydride cracking). Another area is drop test testing and analysis where NRC already has a collaboration with BAM, which potentially could be extended to other partners. Testing of fuel properties (dissolution, solubility and sorption) is also of high interest.

H. Asano made a presentation for the basis for a technical project on the long-term integrity of welds in waste packages [17]. The project should answer the following two basic questions: how to guarantee the long term integrity and how to develop methodologies to understand the ageing. Such a project would be multi-disciplinary and involve NDT, basic material science and material characterization, fracture mechanics, corrosion, technical welding aspects and treatment of uncertainty.

K-F Nilsson made a presentation with the objective to stimulate a discussion on future collaboration [18]. There is a tradition in nuclear safety, which is perhaps even stronger for waste management, of working together. This collaboration is driven by the mutual interest to minimize risks by consensus on best-practice procedures. Nevertheless the specific requirements depend on waste form and technical solutions as well as on public acceptance and national legislation. Hence collaboration means agreeing on basic criteria but not on imposing specific solutions. A number of trends and aspects call for a closer collaboration in the field of design and assessment of packages for radioactive waste.

- Regional or shared repositories and encapsulation plants are discussed in Europe. This would require harmonization of waste package design and requirements.
- Even if there will be a national repository and storage facility in each nuclear country it is likely that the concepts that are now close to implementation (such as the Swedish KBS-3 concept) will serve as reference for systems in other countries through technology transfer.
- The concept of very long-term storage and increasing burn-up need to be addressed.
- Increase of transport of radioactive waste requires public acceptance, which can be enhanced by international consensus on safety criteria and best-practice procedures.

Different specific topics suitable as technical projects had been proposed by Workshop participants prior to the meeting including: manufacturing, quality control and operation; long-term degradation; drop testing with supporting dynamic analysis; integrity and behaviour of spent fuel under long-term storage, and; development of standard casks.

After the three presentations the Workshop participants were split into three "Breakout session groups" to deal with the following topics:

- Long-term behaviour of waste packages,
- Manufacturing and inspetion issues, and
- Dynamic impact.

Each group was asked to address the following issues:

1) What do you see as the most important challenges in the coming 2-10 years in terms of R&D and practical implementation?
2) Which of these areas do you consider suitable for international collaboration?

Each group summarized its findings which are given in the appendices [19],[20],[21]. The following main conclusions were thereby drawn.

- The participating organizations expressed a strong support for international collaboration. The support is particularly strong among countries with small nuclear programmes. There is some interest for standard casks that are accepted in different countries.
- There are different forms of collaboration. A first step is to compile available information in a form so that different designs can be compared and the philosophy behind design concepts and material selections are understood. This could for instance be as a state-of-the-art report.
- There is a clear interest to compare manufacturing standards for waste packages. This aspect is expected to become more important in the future as technology transfer and possibly shared facilities for storage and disposal are likely scenarios for management of spent nuclear fuel and radioactive waste in Europe.
- The life extension of storage facilities and the long-term storage, particularly in combination with increased fuel burn-up will raise a number of issues related to container and cladding integrity.
- External funding, through for instance the European Commission's Framework Programmes, would clearly increase the possibilities to conduct R&D projects on specific topics.

The following topics were identified as interesting to pursue as research projects.

- Long-term behaviour of container and cladding materials. Methods to predict long-term degradation and methods for accelerated testing need to be further developed.
- Application and development of existing non-destructive testing methods. Some kind of round-robin exercise would be a suitable project form.
- Application of fracture mechanics and development of analysis methodologies for transport packages under severe dynamic loads as well as disposal and storage canisters under handling and operation.
- Acceptance criteria for ductile cast iron containers. Ductile cast iron is used in canisters for transport, storage and disposal. Harmonization of methods for NDT, material characterization and analysis procedures would be of interest.

- Long-term performance of welded structures. Welded structures are used for HLW and SNF canisters and the weld is often the weakest part. Accelerated test methods and micro-mechanical modelling are two research areas that need to be developed to understand the long-term degradation of welds.
- Dynamic structural integrity of waste packages is a very complex area for which more research is needed. One obvious reason is that drop tests are very expensive and it makes sense to share the costs. A major issue is transferability of results between different scales, i.e. how to ensure the relevance of small component and specimen test data when applied to large components. The research needs to include development of dynamic material characterization tests (in particular fracture toughness), round robin tests to validate computational methods and modelling of radioactive release from damaged packages.

Following the successful 2006 Workshop JRC has drafted a proposal for an "International Forum on Design and Assessment of Packages for Radioactive Waste and Spent Nuclear Fuel", [22]. There was a strong support for such an activity among the Workshop participants and JRC was asked to take this further on.

Acknowledgements

The JRC Institute for Energy wishes to thank the presenters and their co-authors for their excellent contributions to the Workshop. The support of the JRC's Enlargement and Integration Programme, which provided financial support for participants from the new EU member states and from the Candidate Countries, is also gratefully acknowledged.

4 REFERENCES/PRESENTATIONS

[1] Tae Ahn, NRC, USA, *"Issues in the licensing review of the potential Yucca Mountain repository: container, cladding and waste form"*

[2] Hidekazu Asano, RWMC, Japan, "Study on Evaluation Method of the Long-term Integrity of Waste Package Final Closure*"*

[3] Hannu Hänninen, Helsinki University, Finland, *"The time effect on the integrity for canisters for geological disposal"*

[4] Neil Carr, NIREX, UK, "*Shielding Integrity Testing of Transport Containers*"

[5] Ian Norris, The Welding Institute, UK, *"NDT techniques and structural integrity evaluation for nuclear waste containment applications"*

[6] Bernhard Droste, Uwe Zencker, Holger Völzke and Frank Koch, BAM, Germany, *"BAM Guidelines for the Safety Assessment of Transport and Storage Casks for Radioactive Materials"*

[7] Pitkänen, J.*, Leskinen, N.**, Rydén H.** & Emilsson G.*** , "Defect types, NDT and preliminary acceptance criteria approach based on the damage tolerance analysis and international standards in the manufacturing of nodular cast iron insert". * Posiva Oy,** SKB,*** Bodycote

[8] Chris De Bock, ONDRAF/NIRAS, Belgium"The ONDRAF/NIRAS Waste Overpack and Supercontainer Design"

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[10] P. Poskas, A. Smaizys, E. Narkunas, Lithuanian Energy Institute, Nuclear Engineering Laboratory, Kaunas, Lithuania, *"Modelling of RBMK-1500 Spent Nuclear Fuel Characteristics and Comparison with Available Data"*

[11] R. Hüggenberg, GNS, Germany, *"Comparison of the Behaviour of a finned and smooth Surface of a Type B(U)-Cask made of DCI during a pinion Drop Test"*

[12] Dong-Hak Kim, Kyoung-O Nam, Ju-Chan Lee, Kyung-Sil Bang and Ki-Seog Seo, Korea Atomic Energy Research Institute, "Conceptual Structural Assessment of Polymer Concrete HIC for NPP Waste"

[13] Bernhard Droste and Volcker Noack, BAM Germany, "*German support for utilization of nuclear submarines in Russia"*

[14] Pierre Van Isegem, SCK-CEN, Belgium , *"ENTRAP - the European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages. Objectives, organisation, main output"*

[15] LPM Van Velzen, NRG The Netherlands, *"Proficiency Test for Non-Destructive Assay of 220 ltr Waste Drums by gamma systems: Feasibility Study & Design if First Proficiency Test"*

[16] *Ideas on future collaboration T. Ahn, NRC*

[17] *Ideas on collaboration, H. Asano, RWMC*

[18] *Ideas on future collaboration, K-F Nilsson, JRC*

[19] *Conclusions Working Group Long-term degradation and performance of waste package*

[20] *Conclusions Working Group Manufacturing, Quality Control and NDT*

[21] *Conclusions Working Group Drop Tests*

[22] *Proposal for International Forum on Design and Assessment of Packages for Radioactive Waste and Spent Nuclear Fuel*

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Abstract
The Workshop was organized during the 21st and 22nd of November 2006 on Design and Assessment of Packages for Radioactive Waste. The proceedings contain a summary of 15 technical presentations on: General Design and Guidelines; Waste Package Integrity and Long-Term Degradation Mechanisms; Non- Destructive Testing and Qualification; Drop Tests, and; Networking and International collaboration. The Workshop partners were split into three working groups dealing with Long-term behaviour of Waste Packages; Manufacturing and Inspection Issues, and; Dynamic impact and drop tests. Each of the groups were asked to identify areas where challenging research and development is needed in the next 5-10 years and based on this propose topics for international collaboration.

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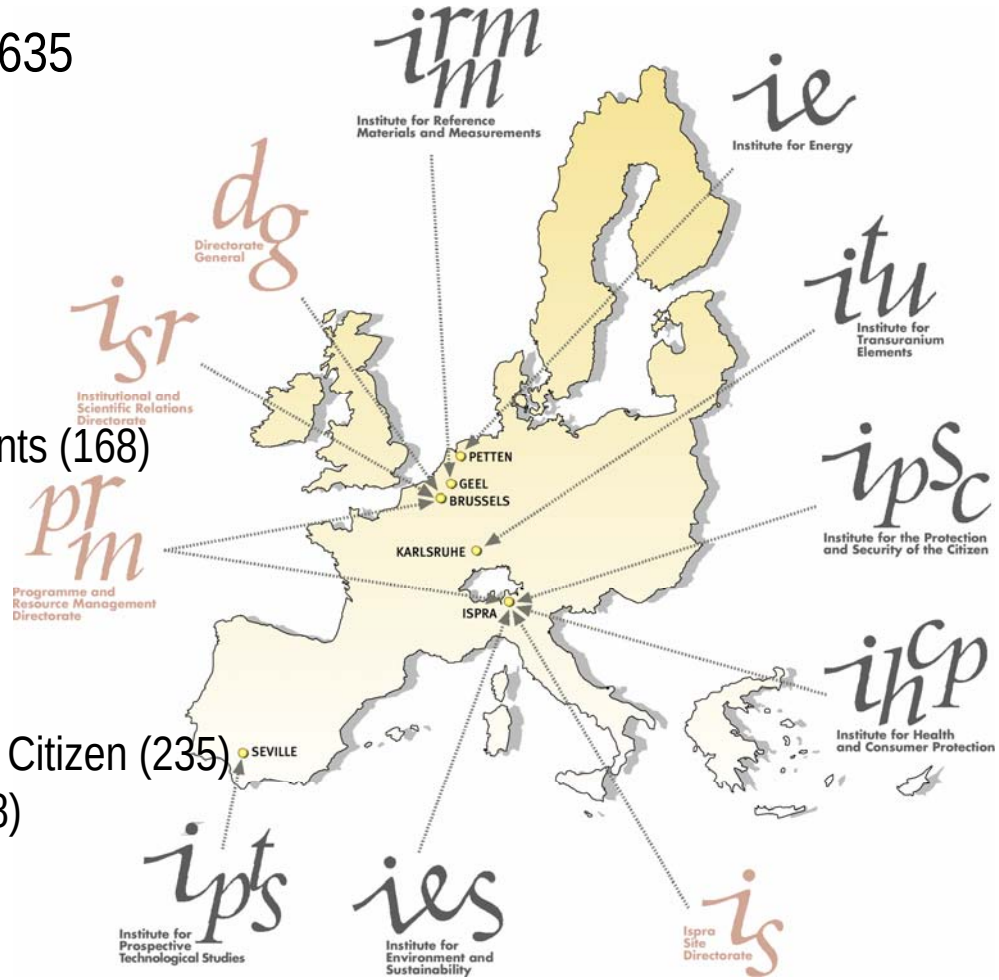

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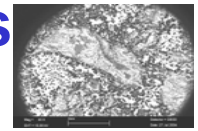
Radioactive Waste research at JRC

- ITU : Research on P&T, spent fuel behaviour and characterization
- IE: SAFECASK (6th FP)/SAFEWASTE (7th FP). Below is a summary of SAFECASK achievements and SAFEWASTE plans

Summary FP6 Achievements (1)

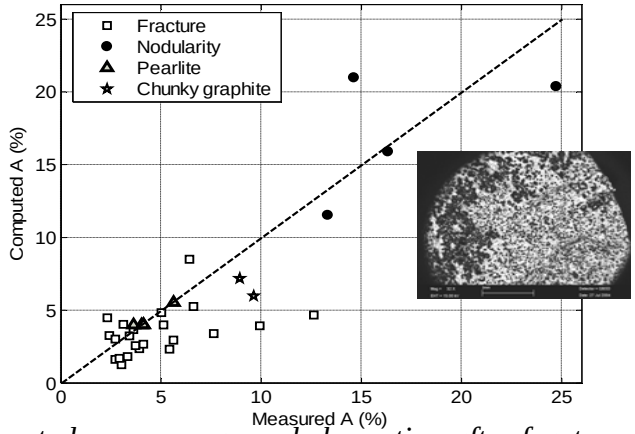
Completion KBS-3 canisters under ice-age loads



- Objective: to determine failure probabilities and acceptance criteria for KBS-3 canister under ice-age load

- JRC contribution
  - Leading role Planning project
  - Perform tensile, fracture & tensile tests
  - Performed microstructural analysis
  - Developed probabilistic fracture model
  - Organized large mock-up test
  - Main contributor final reports

- Results
  - Demonstrated very large safety margins, identified main defects, proposal for improved design
  - Basis for license application of encapsulation plant
  - A number of peer reviewed papers published or in press

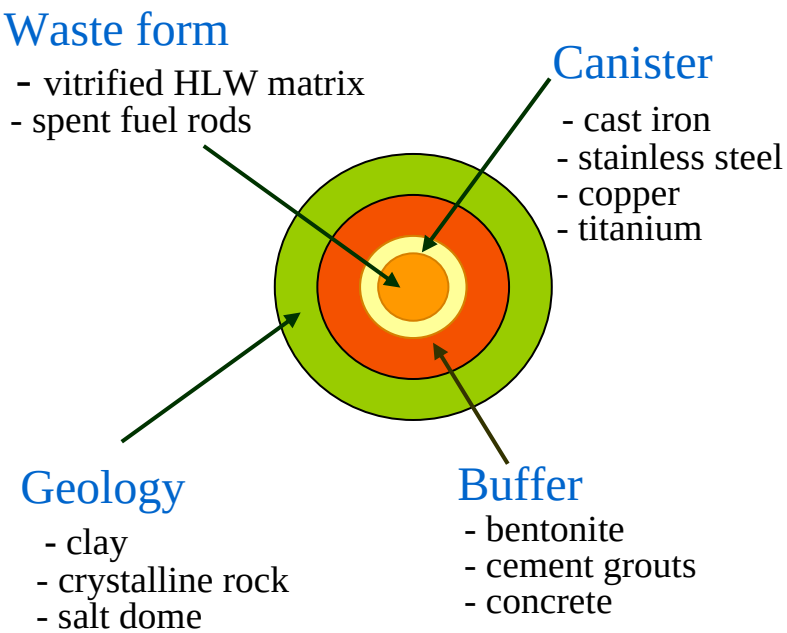


Computed versus measured elongation after fracture for cast iron with associated failure mode

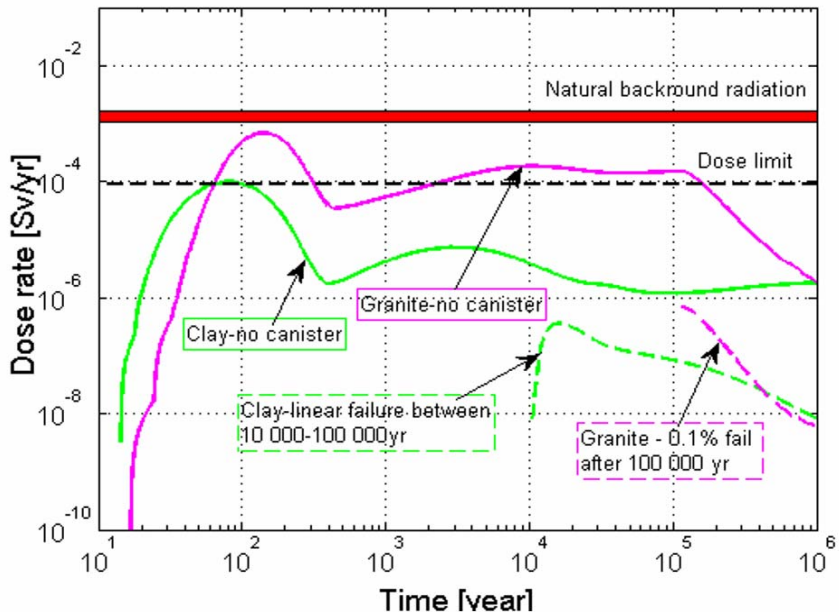
Mock-up after final collapse139 MPa

Summary FP6 Achievements (2)
Start-up of PA analysis for Geological Disposal

- Performance assessment of repositories in clay and crystalline rock for different waste package failure scenarios



Multi Barrier concept



Calculated surface Dose rate from clay and crystalline rock repository systems ("1000 canisters").

- Crystalline rock: canister main barrier
- Clay: geology main barrier
- Overall safety is similar!

Scientific and Technical Description of the Work FP-7
Three main areas:

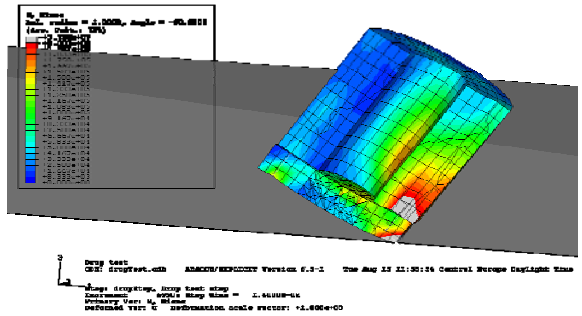
a) Design and assessment of waste packages

b) Performance of systems for long-term storage and disposal of HLW and SNF

c) European Centre for dissemination and harmonization

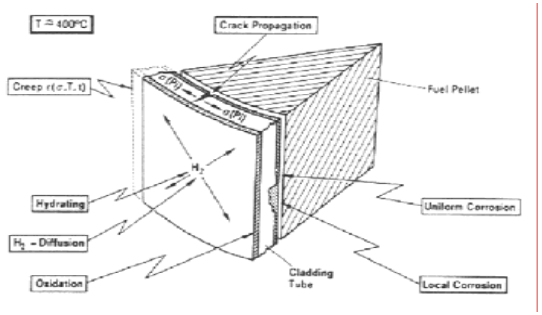
a) Design and assessment of waste packages

- Probabilistic Methodology for Defect Tolerance
  - Further development of methods to relate material properties <=> defects/microstructure,
  - Assessment of welding defects
  - Development/application of NDT methods to detect and size defects.



Simulation of drop test with computed stresses

- Dynamic Impact on Packages for Transport & Storage
  - Numerical modelling of waste packages under severe impact,
  - Characterization & modelling of dynamic material properties,
  - modelling release of RN from damaged waste packages.



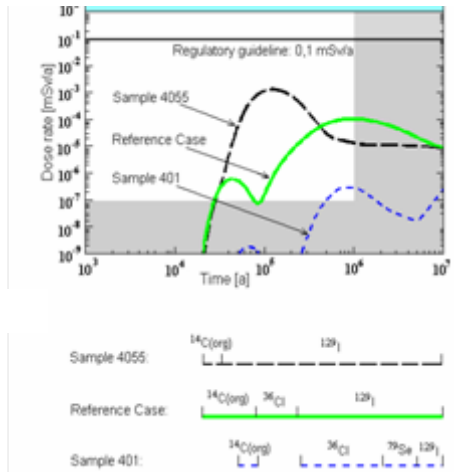
Mechanisms that may effect the cladding under long term storage

- Integrity of SF Cladding under Long-term Storage
  - Delayed hydride cracking, creep, corrosion…….

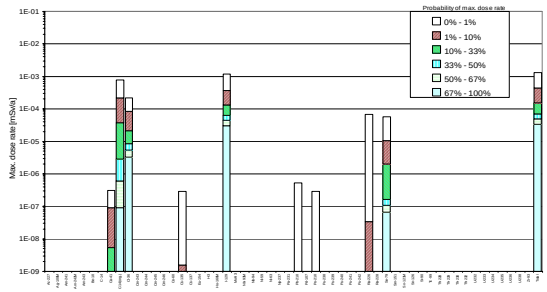
b) Performance of systems for long-term storage and disposal of HLW and SNF

- Development of probabilistic methods for Performance Assessment of geological disposal systems
  - 2007-2008 French clay repository. Collaboration ANDRA (F) in IP PAMINA
  - Assessment of data, Monte Carlo Simulations, sensitivity studies……

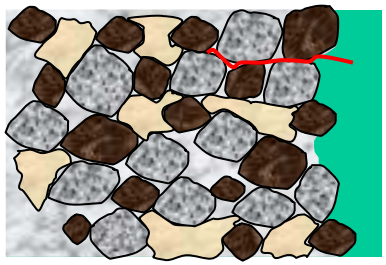
Deterministic



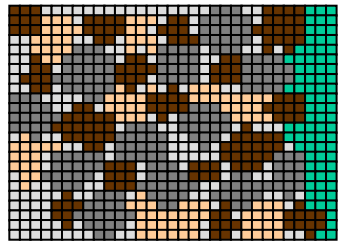
Probabilistic



- Development of methods for specific processes governing migration of radionuclides
  - Diffusion in micro-cracked rock using random walk technique
  - Coupling corrosion RN transport in saturation phase bentonite/canister



Mineral



Grid representation

c) European Centre for dissemination and harmonization

- Manage “Forum” for Design & Assessment for Packages for Radioactive Waste and SNP
- Actively participate international activities
- Establish knowledge system to serve policy DG’s

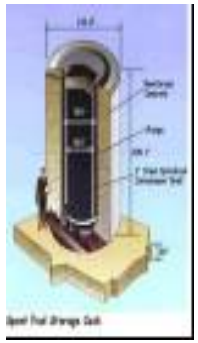
Workshop on Design and Assessment of Radioactive Waste Packages

21th-22nd November 2006 at Hotel Marijke,

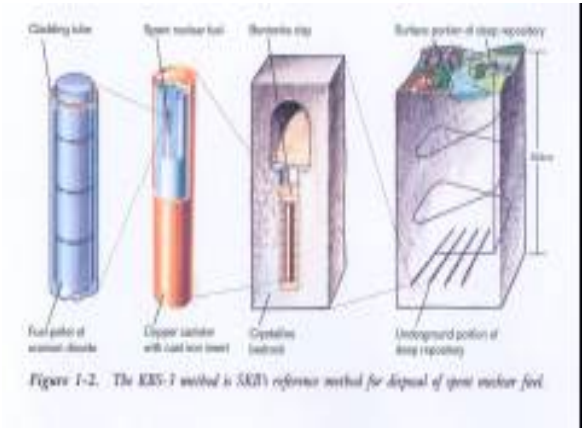
Organized by Institute for Energy, Petten, The Netherlands

Transport

Storage



Geological disposal



Objectives of the Workshop

- Share research R&D results and data for design and assessment of radioactive waste packages (transport/storage/geological disposal);
- Assess the interest for future collaboration in this field as outlined in the scoping document;
  - Identify R&D/technical areas where there is an added value in international lcollaboration;
  - Discuss forms for collaboration
- The Workshop is funded by the EU programme for “Enlargement and Integration”. Special attention should be given to the needs of new member states and candidate countries.

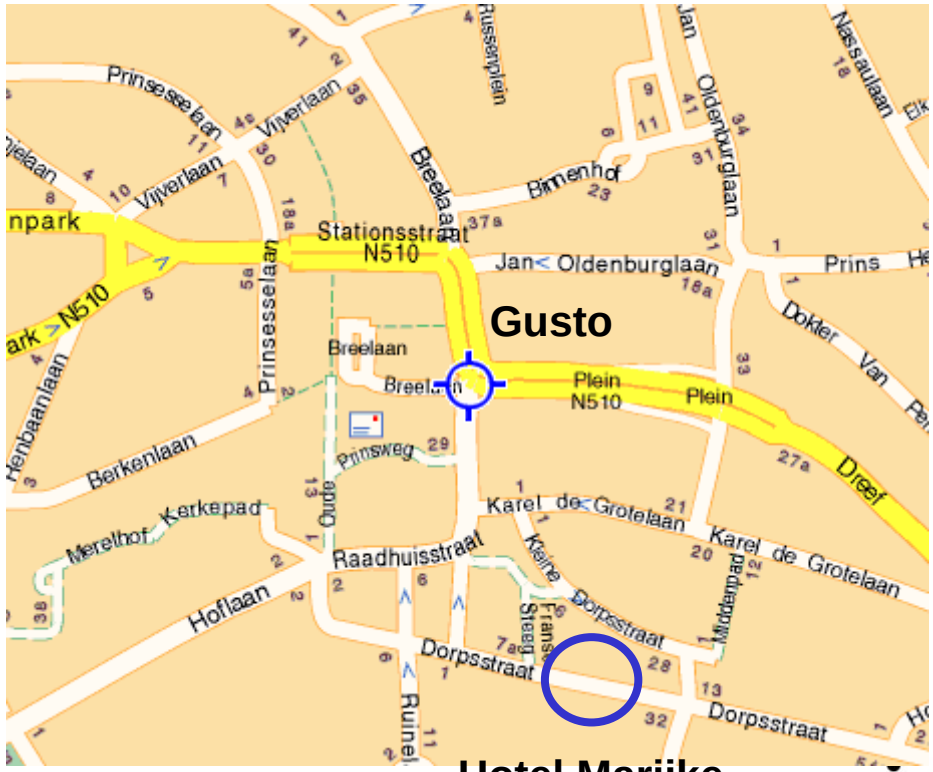
Practical information

- Internet/emails: There are two lap-tops with internet connection in the meeting room. You can check emails, read newspapers etc…..
- Speakers: Put all presentations on the lap-top
- Reimbursements: Bring signed reimbursement request and originals of travel documents to Ms Sylwia Zamana. You need to send the boarding card for return trip by mail.
- Proceeding: All presentations will be compiled in a "Workshop Proceedings" together with general conclusions. Target time for publication: end of the year

Workshop Dinner

You are all invited to a Workshop dinner at 19.30 at Gusto's restaurant, Breelaan 2, Bergen.

Gusto is a couple of minutes walking distance from Hotel Marijke. I will pick you up at 19.30



Issues in the Licensing Review of the Potential Yucca Mountain Repository: Container, Cladding and Waste Form

Tae M. Ahn

Division of High-Level Waste Repository Safety

U.S. Nuclear Regulatory Commission

Washington, D. C., U. S. A.

Workshop on Design and Assessment of Packages for Radioactive Waste, Bergen, The Netherlands

November 21 – 22, 2006

Objectives

- Review potential technical issues concerning a waste package (WP - container, cladding and waste form) associated with the U.S. Department of Energy's potential license application for disposal of high-level waste at the proposed Yucca Mountain repository

  - operational safety during the preclosure period and

  - waste isolation during the postclosure period

- Present potential collaboration work with European Commission (EC)

Outline

- Process for Preclosure Safety Analysis
- Processes for Total System Performance Assessment
- Repository Temperature and Waste Package
- Operational Safety (Preclosure)
- Waste Isolation (Postclosure)

Process for Preclosure Safety Analysis

- Areas of Conseideration:
  Engineered Barriers, Waste Package Emplacement, Ventilation,
  Waste Package Transportation, Waste Handling,
  North Portal Surface Facilities, Performance Confirmation
- Event Sequence Analysis
- Normal and Accident Conditions
- Worker Dose
- Public Dose
- Preclosure Safety Analysis (PCSA)

Processes for Total System Performance Assessment

- Normal Processes:

  Climate, Unsaturated Zone Flow, Effects of Decay Heat on Water Movement, In-Drift Physical and Chemical Environment, Water Diversion Performance of the Engineered Barriers, Waste Package and Drip Shield Degradation, Waste Form Degradation and Radionuclide Release, Engineered Barrier System Transport, Unsaturated Zone Transport, Saturated Zone Transport, Biosphere

- Feature, Event and Process (FEP) Analysis

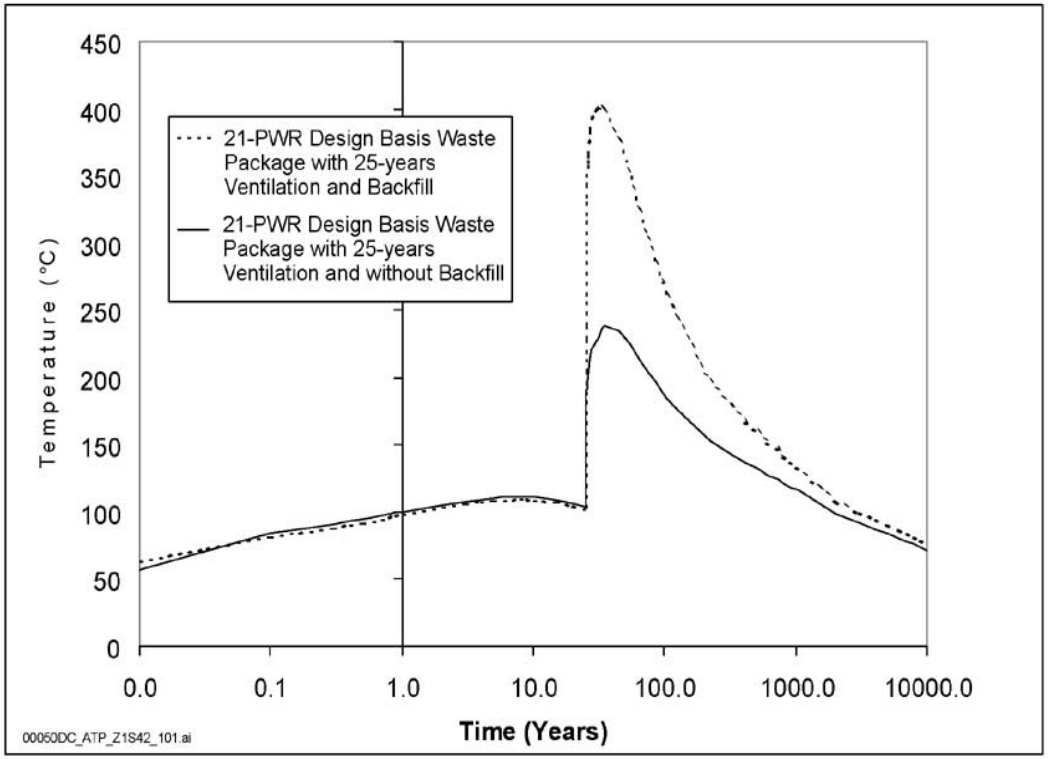
- Nominal and Disruptive Scenarios

- Public Dose: individual, groundwater, human intrusion

- Total-system Performance Assessment (TPA)

Repository Temperature and Waste Package

DOE (2002)



100, 200, 300, 400 °C [212, 393, 572, 752 °F]

- Alloy 22 (w/o):
  Cr (20.0 – 22.5),
  Mo (12.5 – 14.5),
  Co (2.50 max),
  W (2.5 – 3.5),
  V (0.35 max),
  Fe (2.0 – 6.0), Ni (Bal.)
- Ti – 7 (w/o):
  Pd (0.12 – 0.25), Ti (Bal.)
- Ti – 24 (w/o):
  Al (6.0), V (4.0),
  Pd (0.04 – 0.08)

Operational Safety (Preclosure)

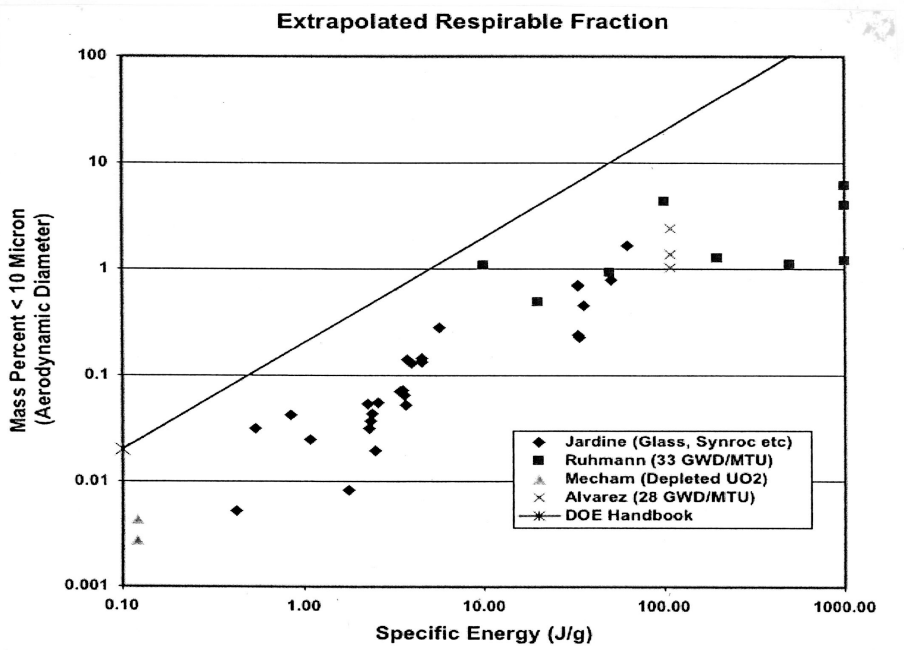
Canister Drop

- Transport, Aging and Disposal (TAD) canister is being considered
- Canister inside canister handling building may need to be sufficiently robust to withstand drop
- Drop height, fabrication defects, canister and internal materials may be important parameters
- For Part 63 (disposal) safety analysis, canister may be partly credited in the PCSA compliance assessment
- Safety of cask (with canister) during transportation and interim storage is reviewed under 10 CFR Part 71 and 72, respectively

Operational Safety (Preclosure)

Source Term

Sanders et al. (1992)



Comparison of the DOE Handbook Respirable Fraction Equation to Experimental Values of the Specific Energy Input into the Brittle Material

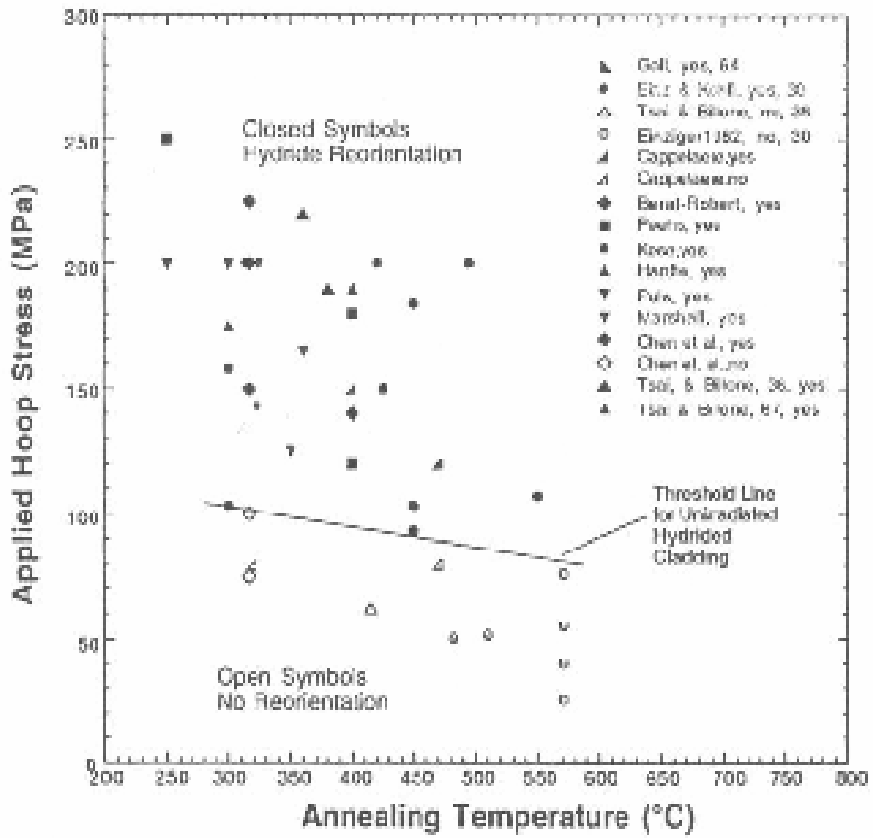
- Release Fraction - impact energy, oxidation from UO_2 to U_3O_8, high-burnup
- Leak Path Factor – HEPA efficiency, stack height, building leakage

10 μm [3.9×10^{-4} in]
1 J/g [1.1×10^{2} cal/lb]

Operational Safety (Preclosure)

Cladding Integrity

Chung (2004, copyright by the American Nuclear Society, La Grange Park, Illinois)

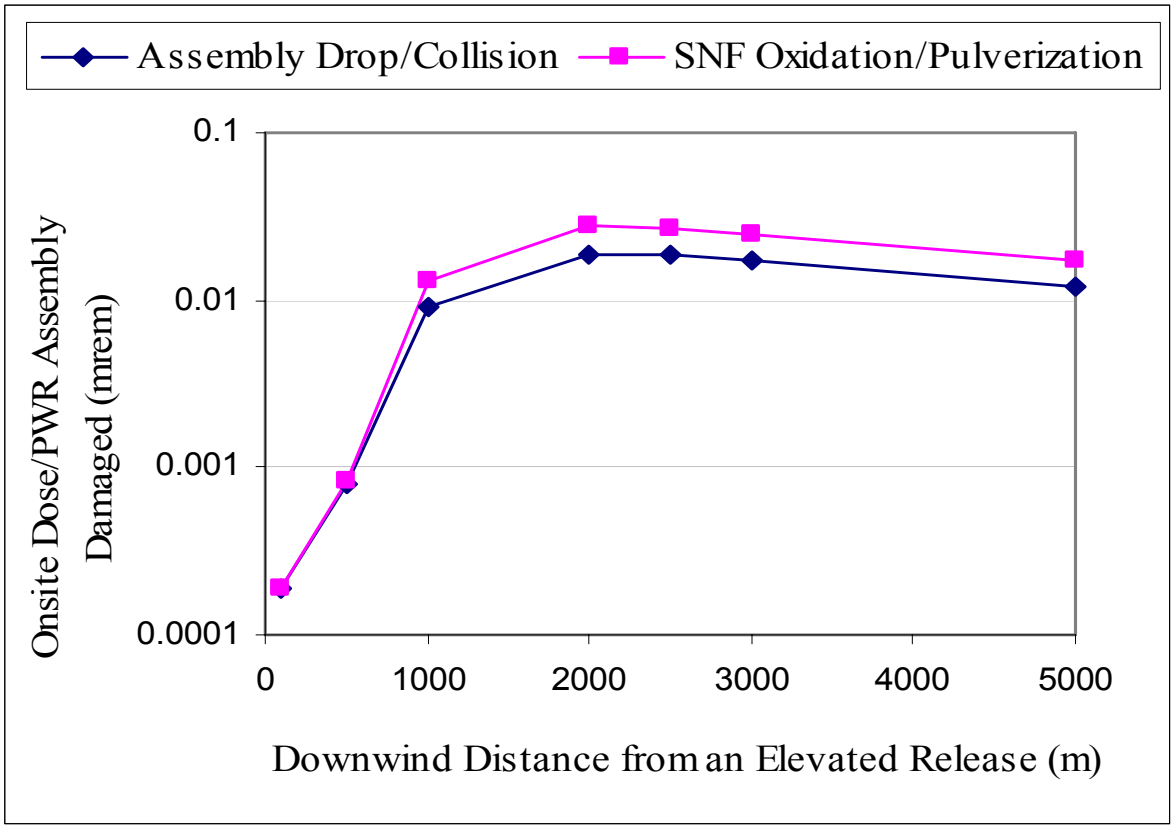


- Detection of pinholes and hairline cracks
- Mechanical vibration - transportation
- Hydride effects – reorientation and delayed hydride cracking (DHC): localized stress by temperature gradient, swelling and Zr/UO_2 interaction

100 MPa [14.5 KSI]

Operational Safety (Preclosure)

PCSA Exercise Results



1000 m [3281 ft]; 1 mrem [10 µSv]

Waste Isolation (Postclosure)

Dust Deliquescence Corrosion of Waste Package

Environments:

- Temperature: 130 to 220 °C [234 to 428 °F] for mixed salts of NaCl, $NaNO_3$ and $NaNO_3$
- Representative drift conditions: ambient pressure, no deaeration

Corrosion Test Results (Yang et al., 2006):

- General corrosion was the major mode of attack for Alloy 22
- General corrosion rate was from 1 to 10 μm/yr [0.39 – 3.9 x 10^{-4} in/yr] at 150 to 180 °C [270 to 356 °F]
- Uncertainties exist in susceptibility to localized corrosion

Waste Isolation (Postclosure)

Persistency in Passive Film

- Accelerated corrosion by passivity breakdown related to localized corrosion, assurance of extremely low general corrosion rates

- Structural change – micro-structure (crystalline or amorphous), various defects, compact/porous, void

- Change of chemical compositions

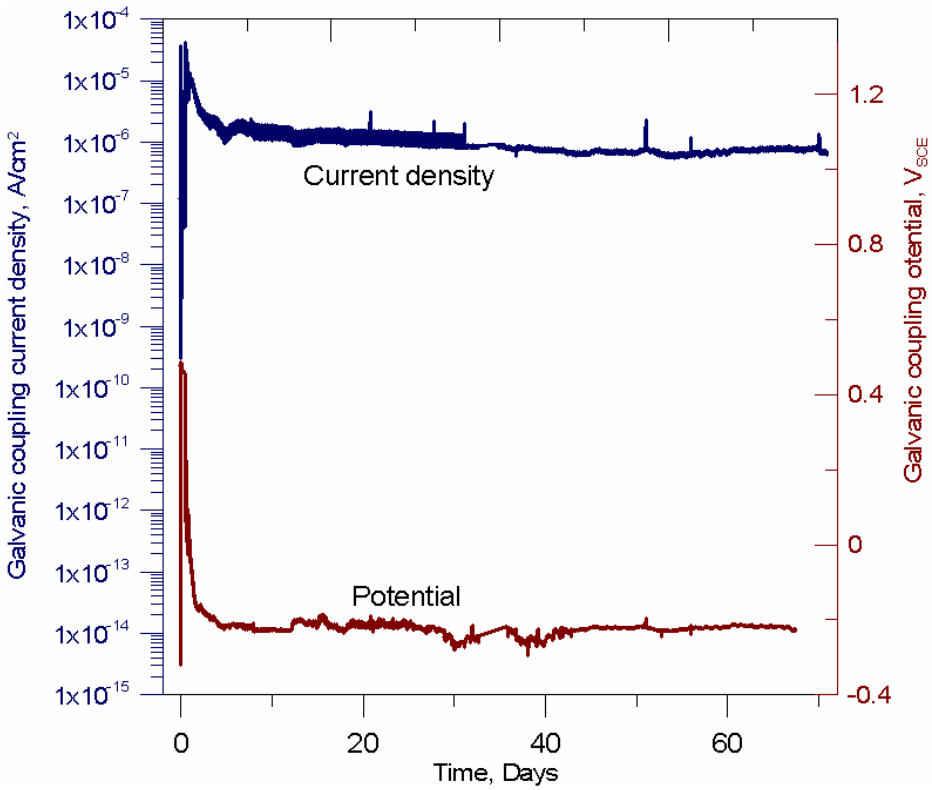
- Thickness change with time

- Examples:
  (i) transpassive dissolution
  (ii) anodic sulfur segregation
  (iii) development of porous structure
  (iv) mechanical spallation by void formation at film interface
  (v) development of large cathodic surface area
  (vi) anion selective sorption

Waste Isolation (Postclosure)

Localized Corrosion

Ahn, Pan and others (2006)



1 A/cm^2 [0.5 A/in^2]

- Initiation: corrosion potential > repassivation potential
- Propagation: decrease with time
- Stifling or Arrest
- Restricted Opening Area

Waste Isolation (Postclosure)

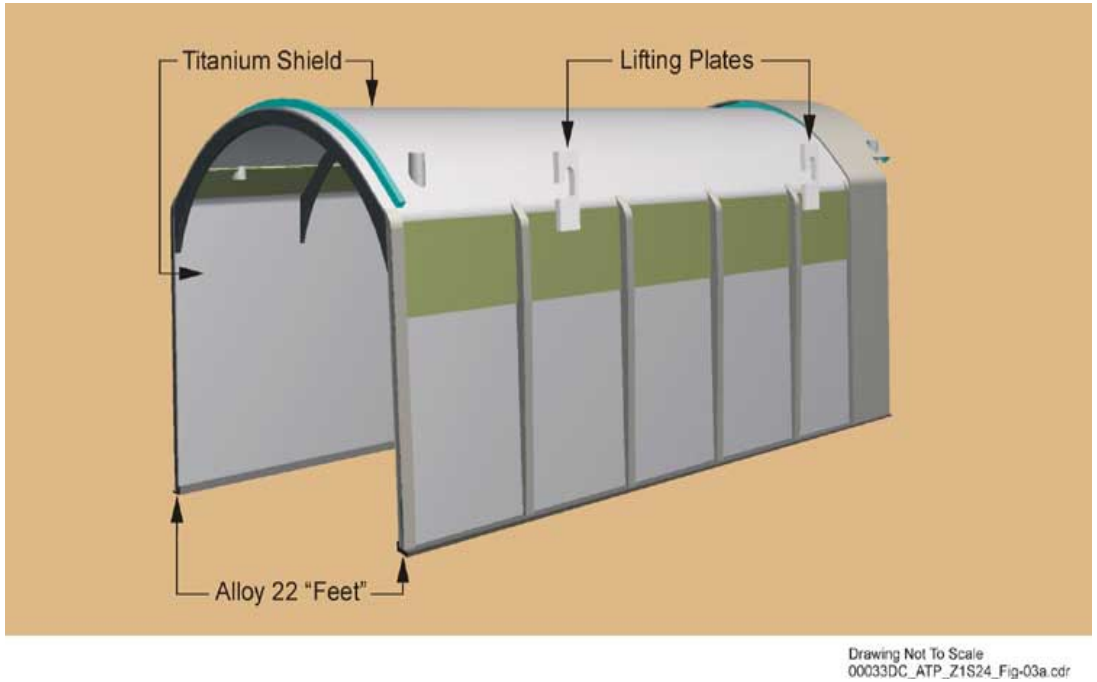
Stress Corrosion Cracking

- Unlikely to happen
- Long-term effects: threshold potential model or K_{ISCC} model
- Screening by crack plugging: corrosion products and mineral precipitation

Waste Isolation (Postclosure)

Titanium Drip Shield

DOE (2002)



- Uphill hydrogen effects in welds
- Mechanical buckling
- Low temperature creep: twinning and slip
- Dust deliquescence corrosion

Waste Isolation (Postclosure)

Spent Nuclear Fuel (SNF) Dissolution

- Major release modes of Tc-99 and I-129

- Factors affecting SNF Dissolution:

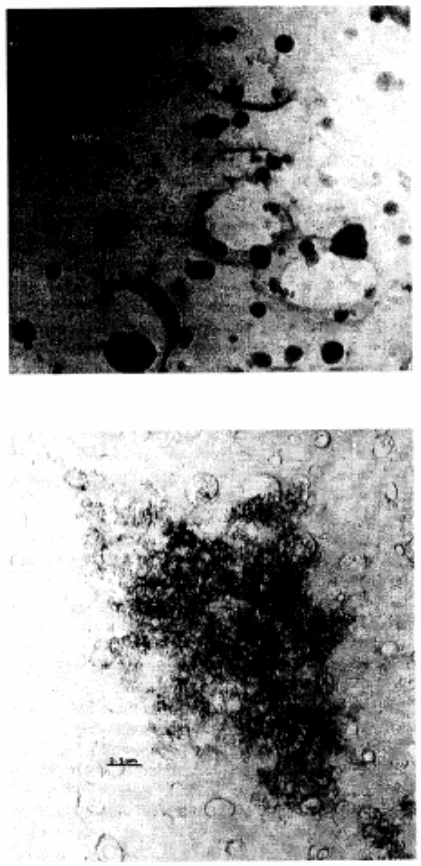
  pH, T, [CO_3/HCO_3], [O_2], [cation, Ca and Si], failed cladding, alteration during dry periods (e.g., hydration and oxidation)

- Important issue: radionuclide (RN) release measurements may include large amount of grain boundary inventory of RNs

Waste Isolation (Postclosure)

Colloids

Bates et al. (1992)



(a)

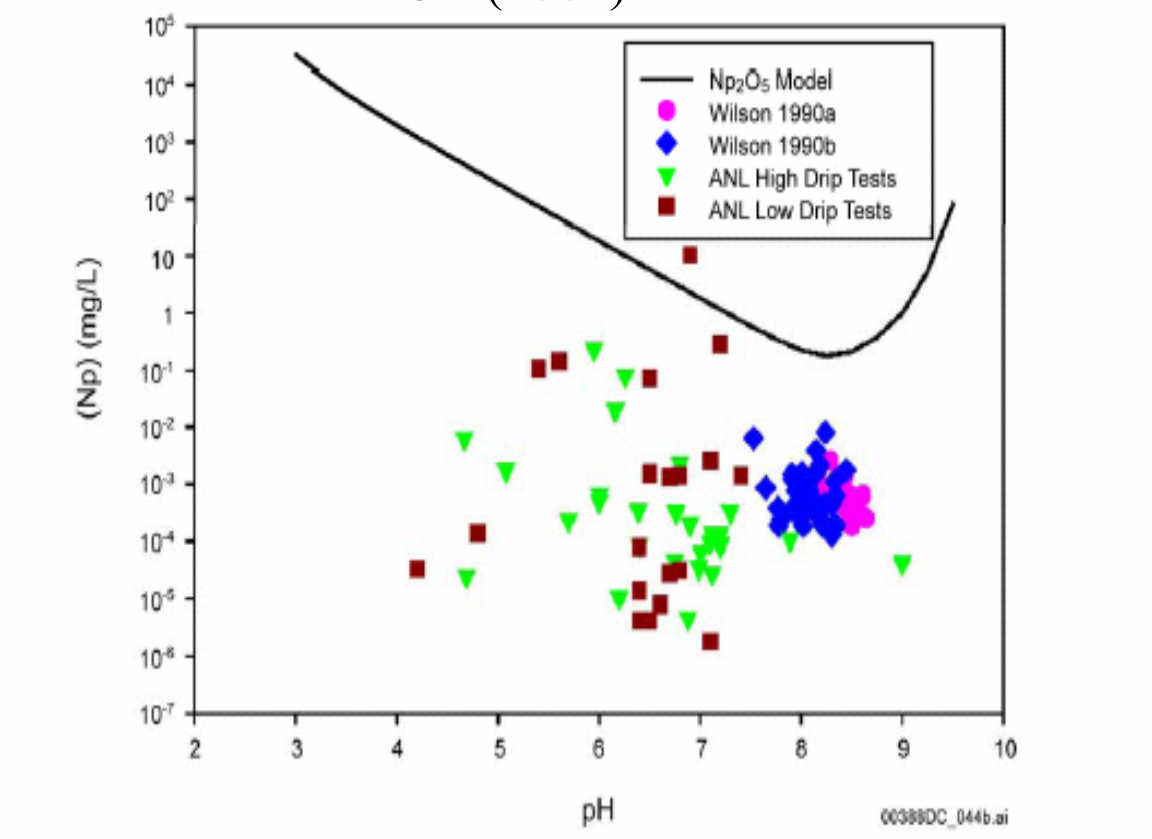
(b)

- Major carrier of Pu-(239 + 240)
- Types:
  - Waste form: radioactive alteration product colloids, true colloids
  - Pseudo-colloids: reversible and irreversible sorption on nonradioactive colloids
  - Nonradioactive colloids: groundwater colloids, iron colloids from WP internal corrosion

Waste Isolation (Postclosure)

Solubility Limits

DOE (2004)



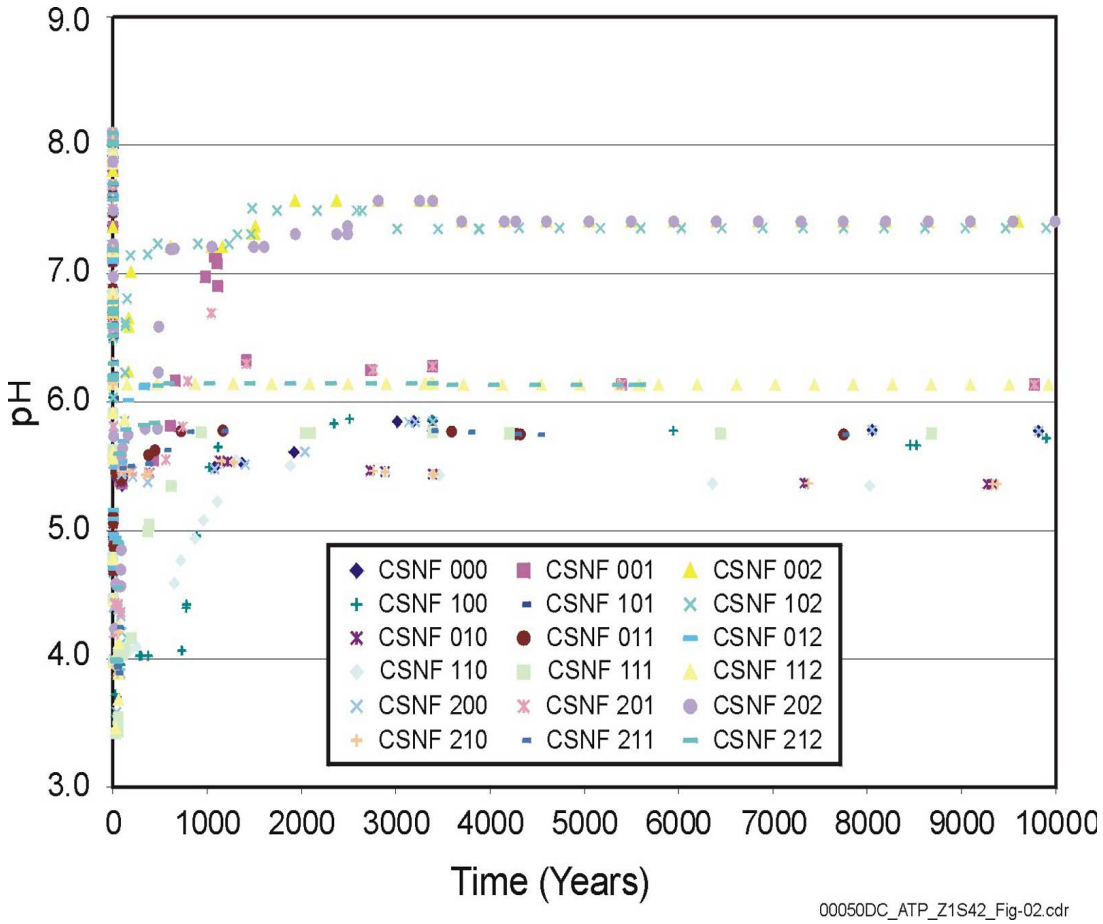
- Major release mode of Np-237
- Controlling factors: pH, T, secondary phase
  - Incorporation of Np-237 in Schoepite or Uranosilicates may substantially decrease the solubility limit

1 mg/L [3.6×10^{-5} lb/in^3]

Waste Isolation (Postclosure)

In-Package Chemistry

DOE (2002)



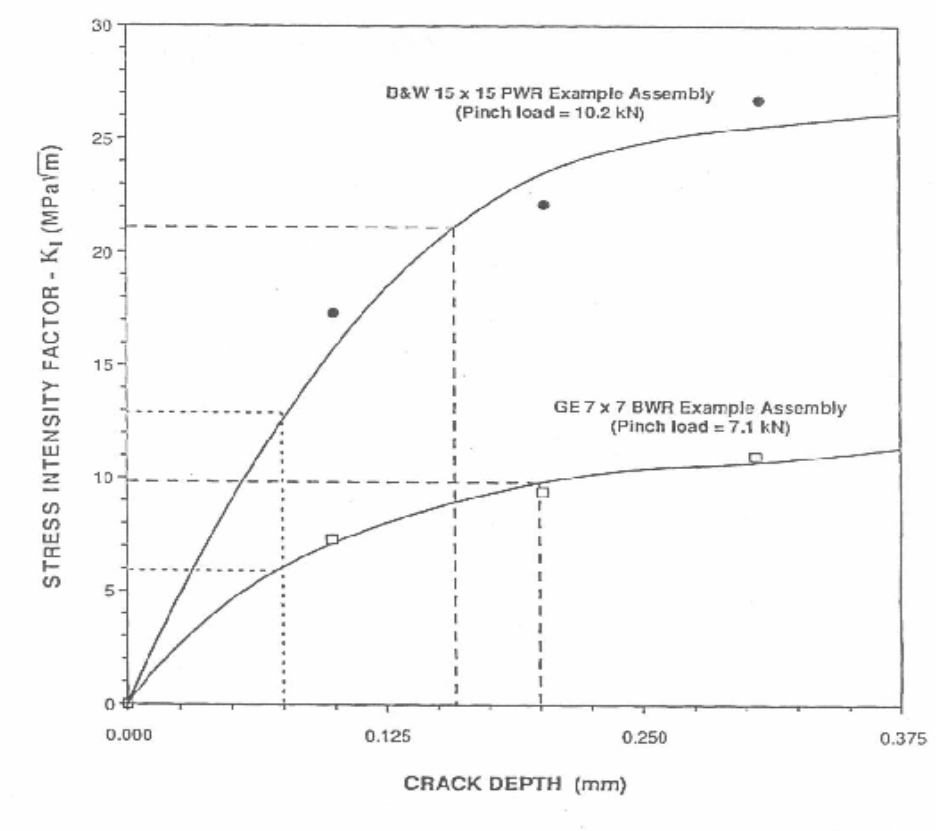
00050DC_ATP_Z1S42_Fig-02.cdr

- Affects the dissolution rates of the SNF matrix
- Affects colloid stability
- Affects the solubility limits of radionuclides
- Controlling components: waste form, basket structure, TAD canister, neutron poison

Waste Isolation (Postclosure)

Cladding Performance

Sanders, Seager et al. (1992)

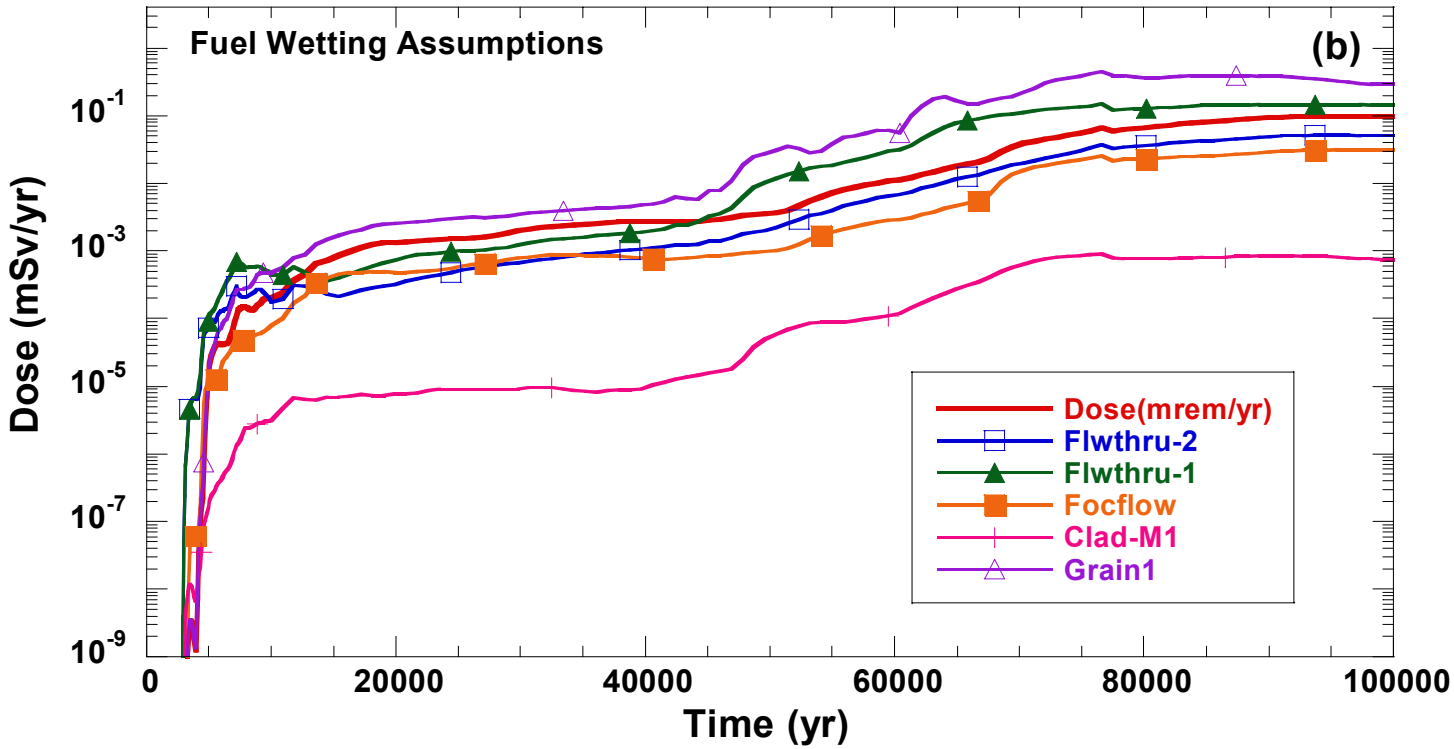


- Hydride effects
  - Cooling rates during postclosure period are much slower than preclosure operation: more susceptible to hydride reorientation
  - Repository may be subject to seismic shaking: more susceptible to delayed hydride cracking

Waste Isolation (Postclosure)

TPA Exercise Results

(Mohanty et al., 2002)



Groundwater Dose from the Basecase and the Fuel-Wetting Alternative Conceptual Models for 100,000 Years, Using the Mean Value Data Set

References

T. Ahn, Y. Pan and others, Summary of NRC Work and Waste Package Corrosion Risk Insights, NWTRB Workshop on Localized Corrosion of Alloy 22 in Yucca Mt. Environments, Las Vegas, NV, 2006

J. Bates et al., ANL Technical Support Program for DOE Environmental Restoration and Waste Management: Annual Report – October 1990 – September 1991, ANL-92/9, Argonne National Laboratory, 1992, Argonne, IL

H. Chung, Understanding Hydride- and Hydrogen-Related Processes in High-Burnup Cladding in Spent-Fuel-Storage and Accident Situations, Proc. 2004 Int. Mtg. LWR Fuel Performance, Orlando, FL, p. 470, American Nuclear Society, 2004

S. Mohanty, R. Codell and others., System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code, CNWRA 2002-05, Revision 1, Center for Nuclear Waste Regulatory Analyses, San Antonio, TX, 2002

T. Sanders, K. Seager and others, A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements, SAND90-2406, Sandia National Laboratories, Albuquerque, NM, 1992

U. S. Department of Energy, Technical Basis Document No. 7: In-Package Environment and Waste Form Degradation and Solubility, Revision, Bechtel SAIC, Las Vegas, NV, 2004

U. S. Department of Energy, Yucca Mountain Science and Engineering Report, DOE/RW-0539, Revision 1, Office of Civilian Radioactive Waste Management, Las Vegas, NV, 2002

L. Yang and others, Corrosion of Alloy 22 in Salt Environments at Elevated Temperatures, NWTRB Workshop on Localized Corrosion of Alloy 22 in Yucca Mt. Environments, Las Vegas, NV, 2006

Disclaimer

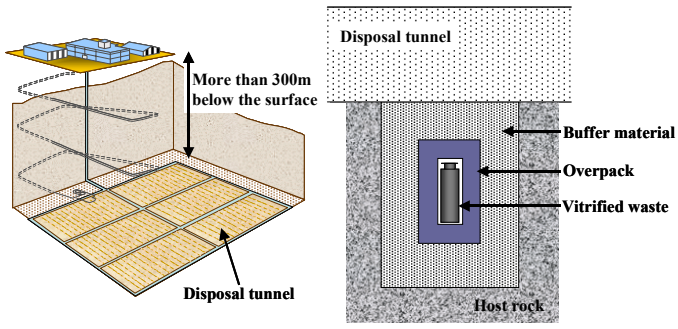
The NRC staff views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed or of the acceptability of a license application for a geological repository at Yucca Mountain.

Study on Evaluation Method of the Long-Term Integrity of Waste Package Final Closure

H.Asano, RWMC

Workshop on Design and Assessment of Radioactive Waste Packages
November 21- 22, 2006
Petten, The Netherlands

Long-Term Integrity of Overpack Final Closure



(a) Surface and underground facility (b) Engineered barrier system

Carbon steel overpack (JNC's H12 Report)

Wall thickness

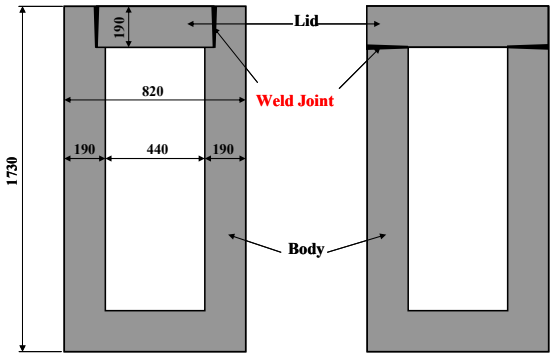
Corrosion allowance : 40 mm

Pressure resistance : 110 mm

Radiation shielding : 150 mm

190 mm

1,000 years containment



(a) Drop-in Lid Type (b) Flat Lid Type

Two(2) different types of lid structure which are modeled for this study

Back to the last Workshop in Petten, October, 2006

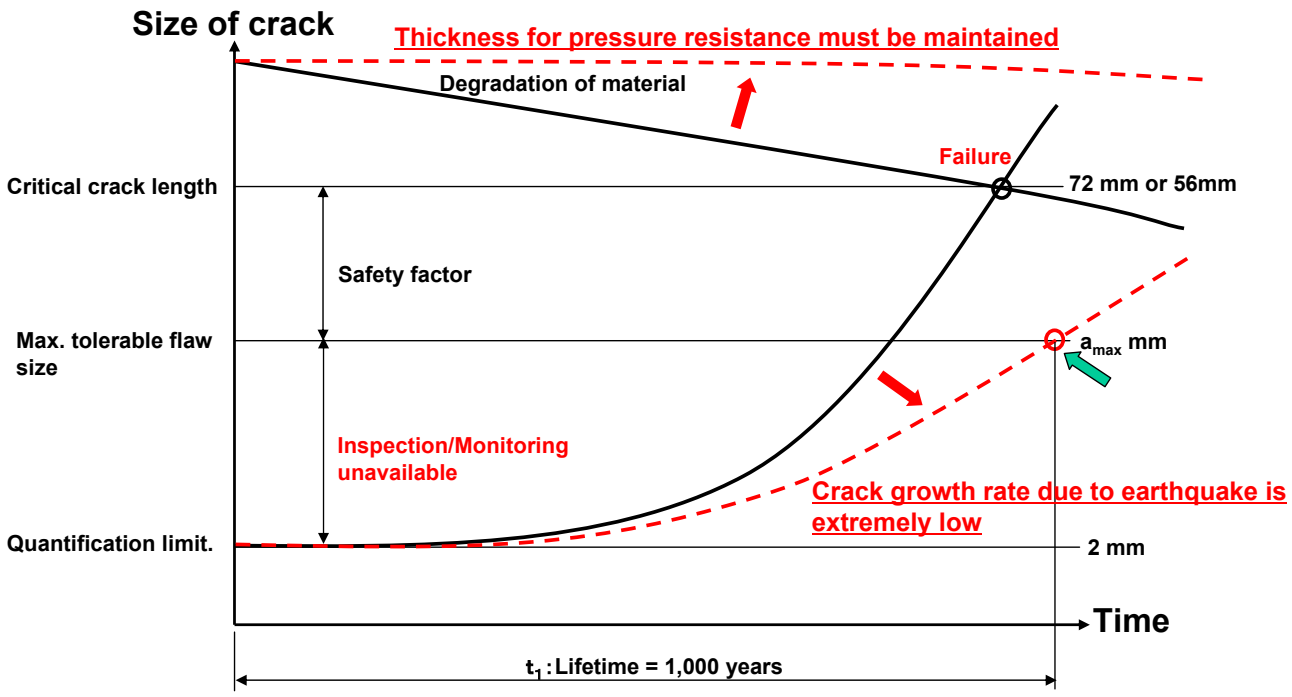
・・・・・please remember !

Assessment model for the long-term integrity of overpack closure weld, under consideration of・・・・・*

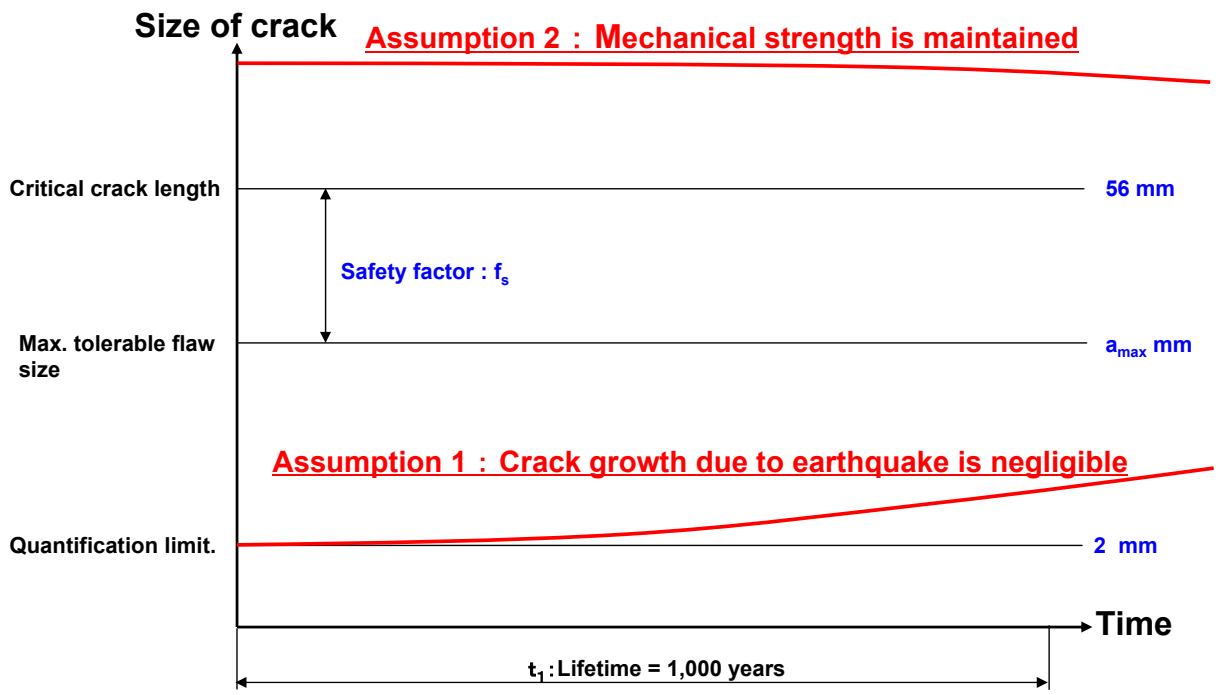
- Fitness-for-Service assessment for NPP components
- Fracture mechanics assessment

*Asano et al., Long-Term Integrity of Waste Package Final Closure for HLW Geological Disposal, (IV) Influence of Welding and Prediction of Long-Term Integrity of Weld Joint, J.Nucl.Sci.Technol., 43[8], (2006)

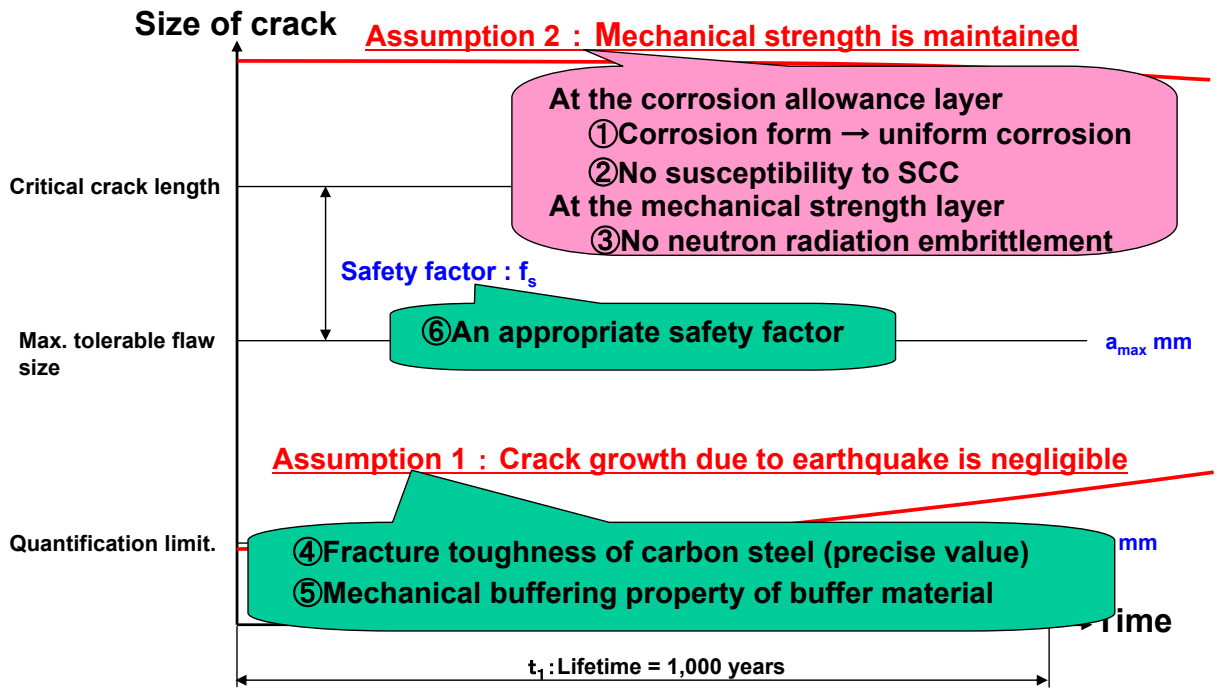
Overpack : Failure Model for the Closure Weld



Assessment Model of the Long-Term Integrity of Overpack Closure Weld



Prerequisite for the Long-term Integrity Guaranteeing Model



Assumption 2 : Mechanical strength is maintained

Corrosion allowance layer /uniform corrosion

Pressure resistance layer /No degradation

①Uniform corrosion must be kept

②No susceptibility to stress corrosion cracking (SCC)

③No neutron radiation embrittlement

Corrosion tests of weld joint

Estimation of neutron radiation embrittlement (analysis)

Today's presentation

1. Corrosion property of weld joint / Assumption 2 of the proposed evaluation method
   - (1) Uniform corrosion
   - (2) Stress corrosion cracking (SCC)
2. Neutron radiation embrittlement of weld joint / Assumption 2 of the proposed evaluatio method
3. Structural integrity of closure weld
   - (1) Max. principal stress & Critical crack length
   - (2) Relation between Lid structure, welding & NDE methods
4. Future works

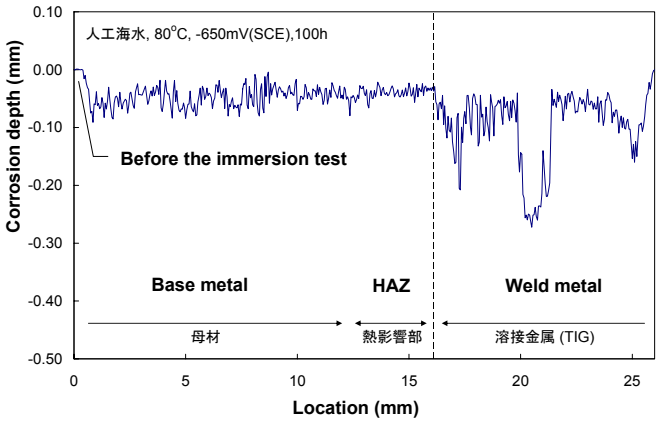
   (significant to the reliability of the long-term integrity of the closure weld)
5. Conclusions

1. Corrosion Property of Weld Joint

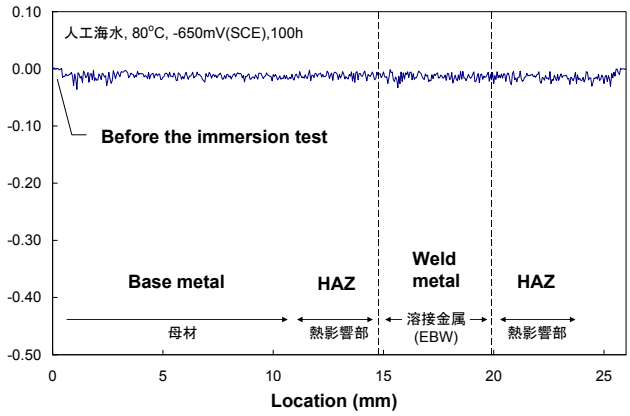
Uniform Corrosion?

Potentiostatic immersion test (80 ℃, synthetic sea water, -650mV(SCE), 100 h)

➡ **Preferential corrosion／TIG weld specimen**



TIG specimen



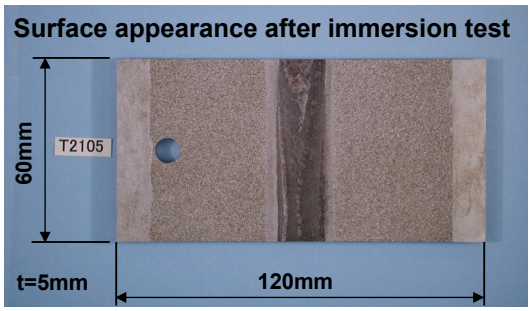
EBW specimen

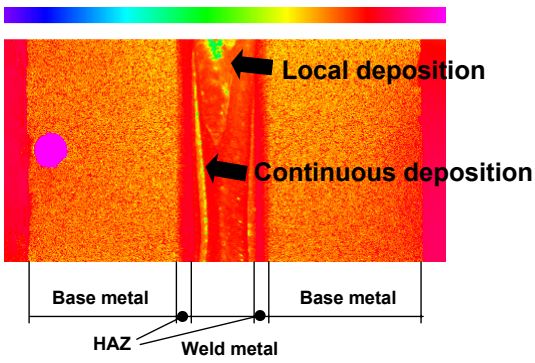
Corrosion depth measured by laser profile meter

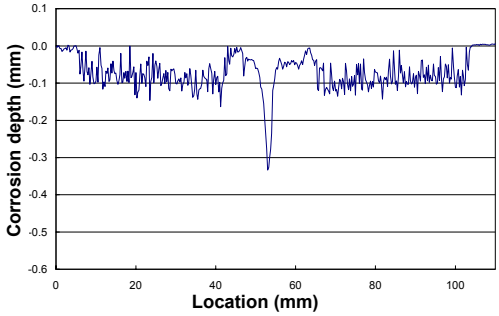
Uniform Corrosion?

Immersion test (80 ℃ synthetic sea water, 90 days, air bubbling)

➡ **Preferential corrosion ／ TIG weld specimen**







Corrosion depth measured by laser profile meter

Preferential corrosion

(1) Chemical composition

Materials \ Chemical composition		C	Si	Mn	P	S	Cu	Cr	Mo
TIG	Base Metal	0.15	0.19	0.36	0.006	0.002	0.01	0.05	0.01
	Weld Metal	0.11	0.67	1.29	0.009	0.011	0.25	0.02	<0.01
	Filler Metal	0.11	0.72	1.41	0.013	0.013	0.24	0.04	0.01
EBW	Base Metal	0.12	0.25	0.65	0.012	0.004	0.05	0.11	0.02
	Weld Metal	0.11	0.25	0.7	0.011	0.003	0.05	0.11	0.02
Base Metal (JIS Code)	SF340A (JIS G 3201)	≦0.60	0.15 ~0.50	0.3 ~1.20	≦0.030	≦0.035	—	—	—
Filler Metal (JIS Code)	YGT50 (JIS Z 3316)	≦0.15	≦1.00	≦1.90	≦0.030	≦0.030	≦0.50	—	—

(2) Deposition = Sulfide formation (MnS), Mn, Si, and S in weld metal *

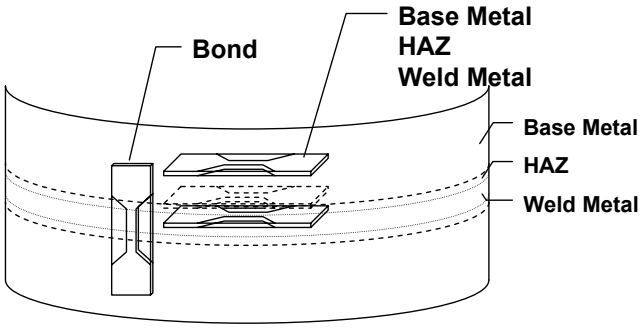
⇒local degradation of corrosion property

(3) Galvanic cell between weld metal and base metal

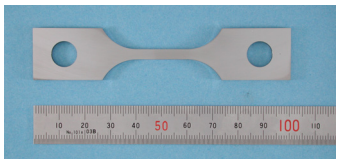
⇒acceleration of corrosion rate

＊Saarinen(1971), Malik et al.,(1990), Matsushima et al.,(1992), Miyuki(2000)

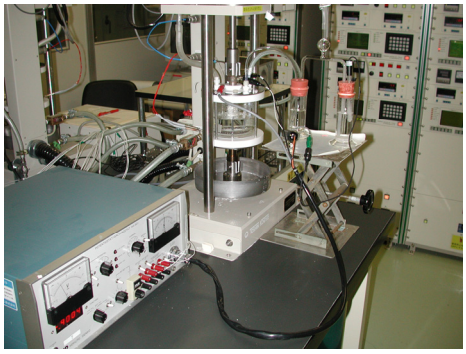
SCC Susceptibility ／ Weld Joint ／ Slow Strain Rate Test (SSRT)



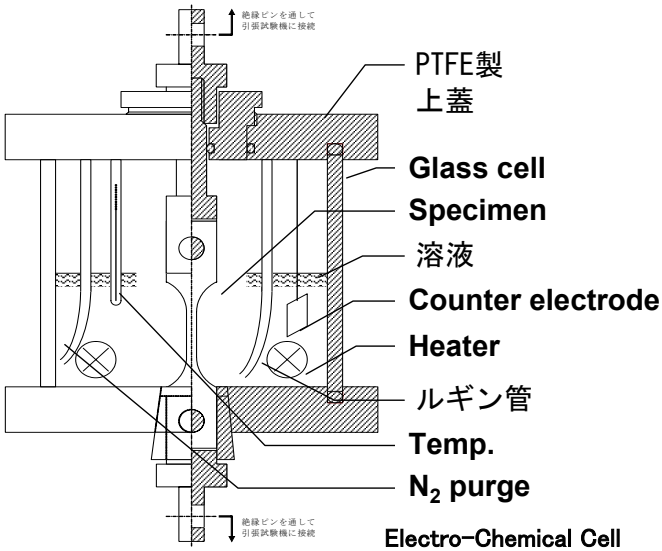
SSRT Specimen



SSRT Specimen



Test Equipment

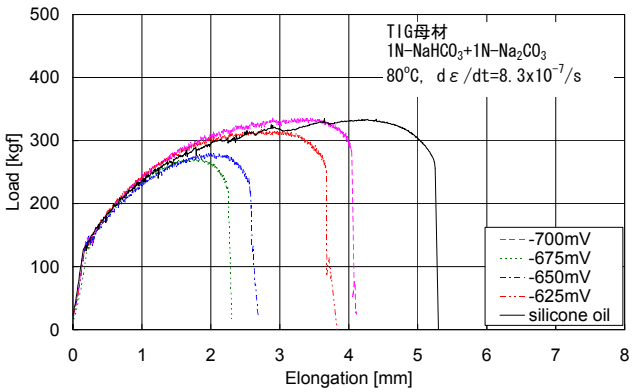


Electro-Chemical Cell

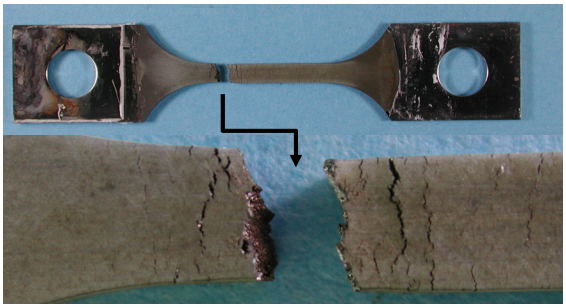
SSRT Condition

1. 1M $NaHCO_3$ + 0.5M Na_2CO_3 & Silicon Oil
2. 80℃
3. Extension Rate : 1 μm/min（strain rate 8.3 x 10^{-7} s^{-1}）
4. Potential : -700～-625mV(vs. SCE)

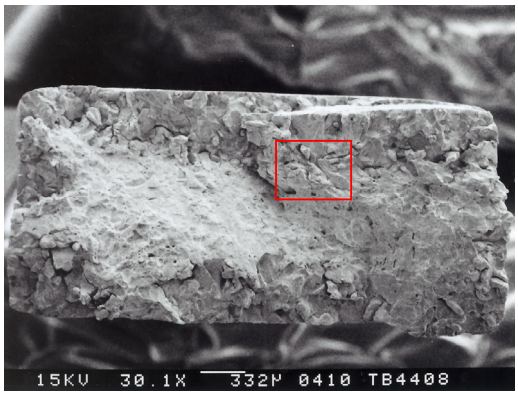
SCC Susceptibility –TIG specimen, Base Metal－

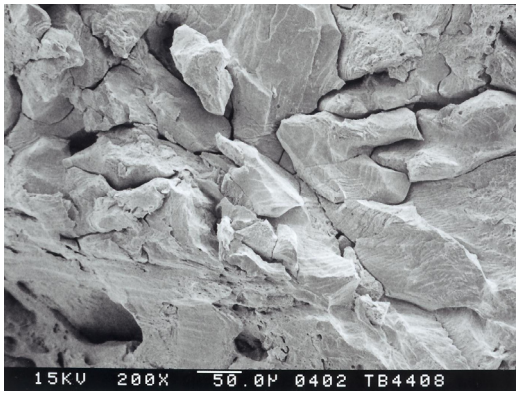


Elongation-Load Curve

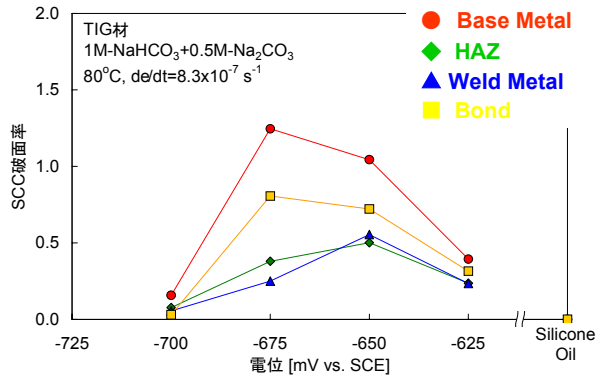


After SSRT(-675mV)

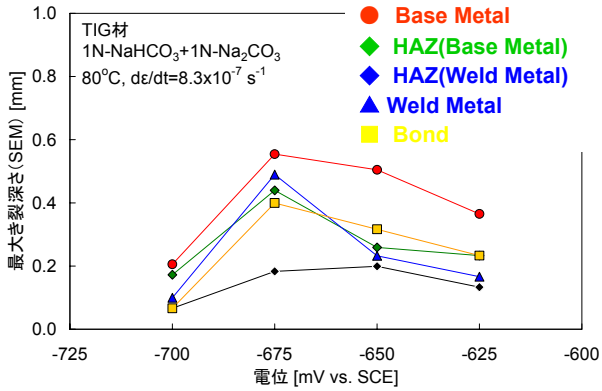




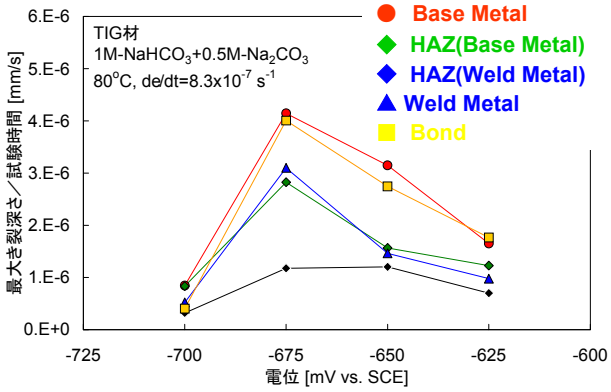
Fracture surface（-675mV）



Fracture Area



Max. Crack Depth (mm)



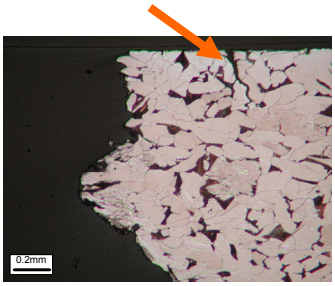
Crack Growth Rate (mm/s)

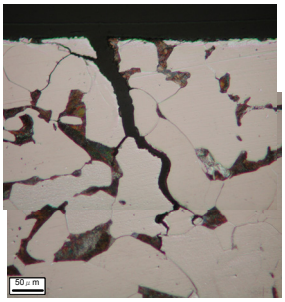
SCC Susceptibility

Base Metal≧Bond（Base Metal）＞HAZ（Base Metal）≧Weld Metal＞HAZ(Weld Metal)

Fracture Surface/TIG Specimen,-675mV

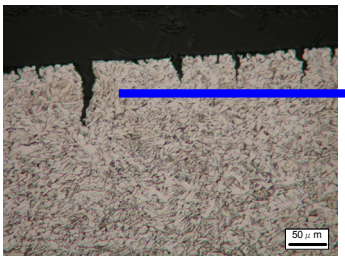
Base Metal

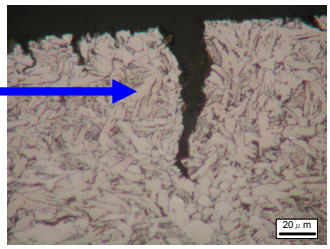




◆Trans-Granular (ferrite/ferrite、ferrite/pearlite)

Weld Metal





◆Ferrite based fine structure

◆Trans & Inter Granular (ferrite/ferrite)

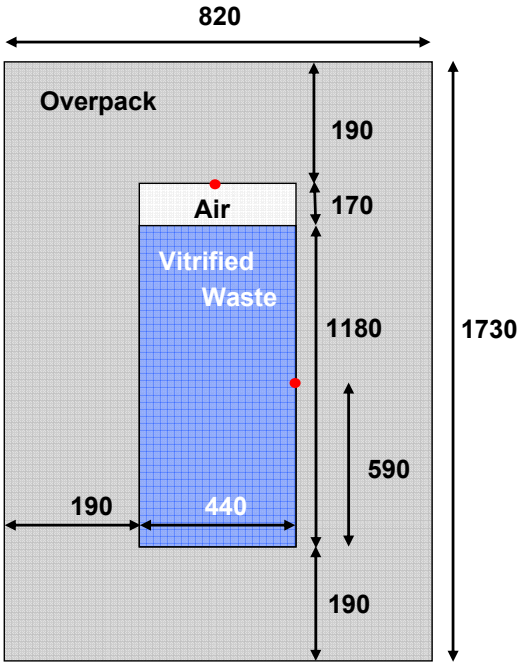
2. Estimation of neutron radiation embrittlement - calculation model and method -

1.Vitrified Waste/ Inventory
- ORIGEN Code
- B.U. : 45,000MWD/MTU
- Initial Enrichment : 4.5%
- Cooling Time : 4 years (Aft. Reactor operation)
- Vitrified waste storage : 30 and 50 years
- Aft. Disposal : 1,000 years

2.Neutron dose deposition (n/cm^2)
- 3-D Monte-Carlo Code, MCNP-4C2
- Neutron energy : ＞1Mev
- (α、n) and spontaneous fission neutron

National prediction formula of neutron radiation embrittlement for reactor pressure vessel (Transition of RT_{NDT})

(Japan Electric Association Code, JEAC 4201-2004、Method of Surveillance Test for Structural Materials of Nuclear Reactors, March 2005)



Neutron radiation embrittlement dose not occur within the 1,000 year containment duration required for overpack

Neutron dose deposition for 1,000 years

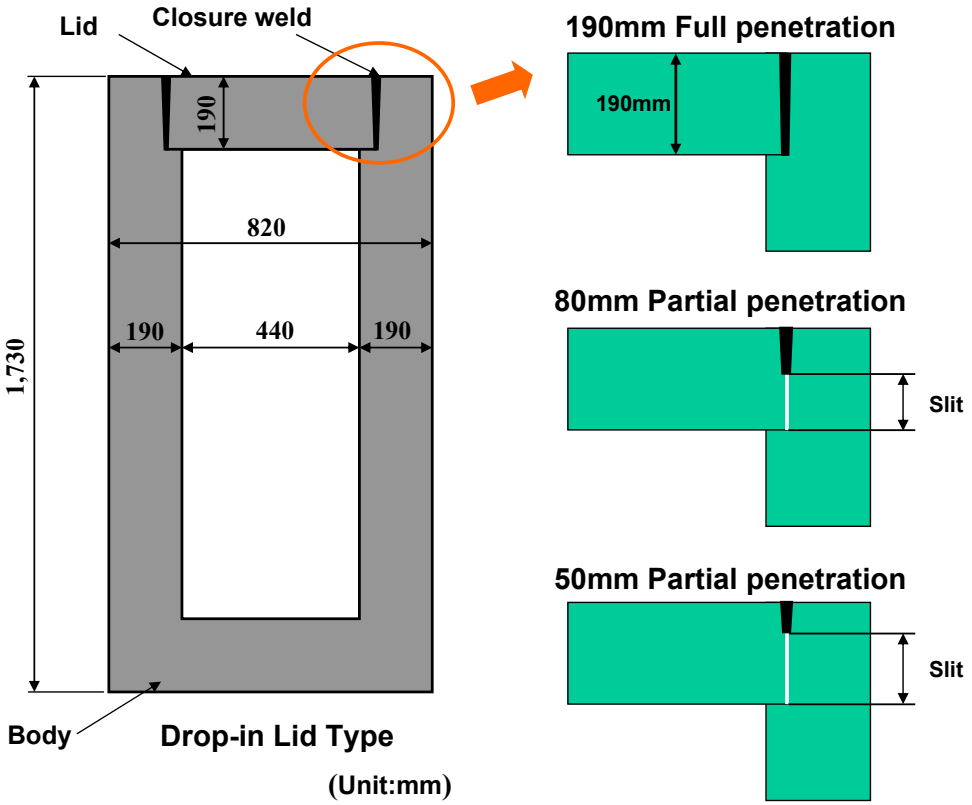
Coolin time (year)	Calculation	Neutron dose deposition (n/cm2)	
		Lid center/inside	Vitrified waste center /side wall inside
30	(α、n)	3.114E+13	8.153E+13
	Spontaneous fisson neutron	1.109E+13	2.896E+13
	(α、n)＋s.f.n	4.223E+13	1.105E+14
50	(α、n)	2.877E+13	7.539E+13
	Spontaneous fisson neutron	6.342E+12	1.657E+13
	(α、n)＋s.f.n	3.512E+13	9.196E+13

Comparison of neutron dose deposition between overpack and national prediction formula for NPP components

Evaluation poit	Neutron dose deposition (n/cm2、E≧1MeV)
Lid center/inside	4.223E+13
Vitrified waste center /side wall inside	1.105E+14
Availability	1.0×10E+17～1.0×10E+20

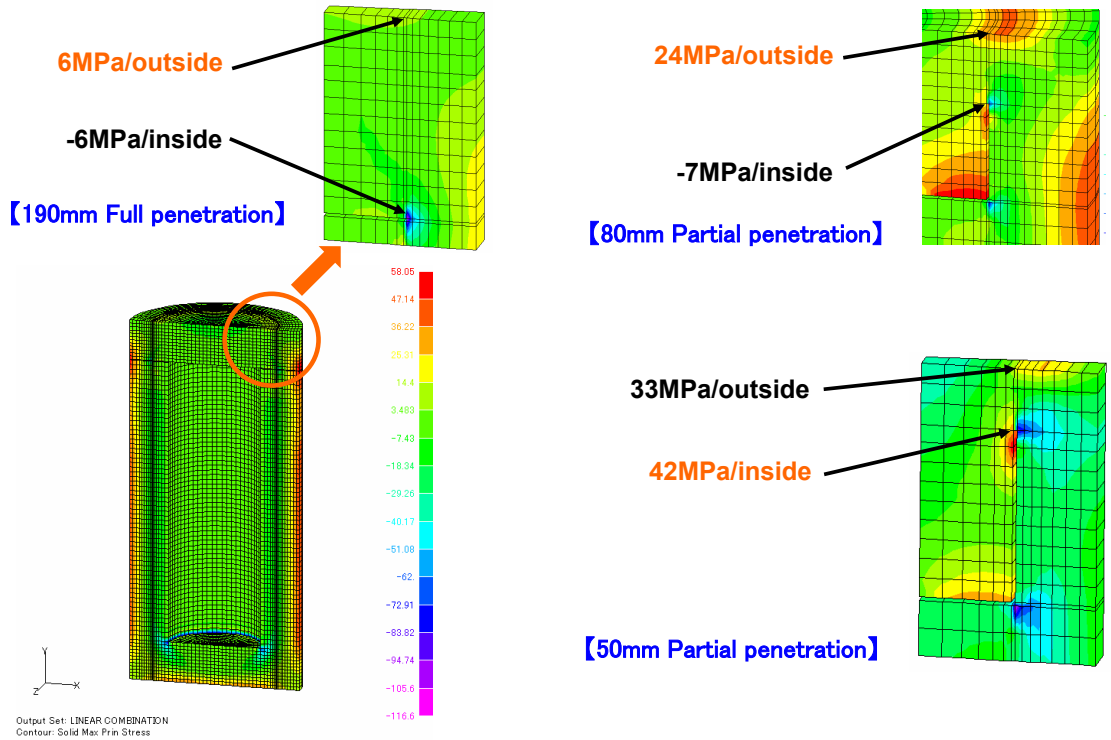
3. Structural Integrity

Max. Principal Stress & Critical Crack Length at Deferent Penetration Depth



Max. Principal Stress under (TS+EP+Eq+(RS)) Condition

(TS : Thermal stress, EP : External pressure, Eq : Earthquake, RS : Residual Stress)



Critical crack length under (TS+EP+Eq+(RS)) Condition

Crack Model \ Penetration depth(mm)		Full penetration	Partial penetartion	
		190	80	50
Critical crack length(mm)	Elliptical crack in an infinite body	88	72	59
	Semi-elliptical surface crack	69	56	46

- Penetration depth ⇒ decrease, Max. principal stress ⇒ increase
- Location of the Max. principal stress changes from weld outer surface to inner surface
- Effect of the slit at the weld joint
- Decision of the penetration depth ⇒ corrosion allowance & structural integrity

◆However, the estimated critical crack lengths are still larger(1 order of magnitude) than the Min. quantified flaw height by NDE.
◆Therefore, these three different lid structures could be built.

Comparison of different lid structures for overapck

Structure NO.		A-1	A-2	B-1	B-2	C-1	C-2
Lid Structure		Drop-in lid					
Penetration depth (mm)		190mm full penetration		80mm Partial penetration		50mm Partial penetration	
Welding method		TIG	EBW	TIG	EBW	TIG	EBW
Welding time (for 1 overpack)		23h.	30min.	10h.	9min.	6.3h.	4min.
Weld surface		As weld	Machining	As weld	Machining	As weld	Machining
NDE*	1	Surface CW(DTPT) Detection	←	←	←	←	←
	2	Surface CW(DPT) Sizing, ≧2mm	←	←	←	←	←
	3	Inside(5～10mm) CW(DPT) Sizing, ≧2mm	←	←	←	←	←
	4	Inside(～30mm) TOFD Sizing ≧2～5mm	←	←	←	←	←
	5	Inside(～110mm) TOFD Sizing, ≧2mm	←	Inside(～80mm) TOFD Sizing, ≧5mm	←	Inside(～50mm) TOFD Sizing, ≧5mm	←
	6	Inside(～190mm) TOFD Sizing, ≧5mm	←	-	Surface:ACFM Sizing, ≧0.5mm	–	Surface:ACFM Sizing, ≧0.5mm
	7	–	Surface:ACFM Sizing, ≧0.5mm	–	–	–	–
Critical crack length (mm)		69		56		46	

*NDE, CW(DTPT): Creeping Wave (Double Transducer Probe Technique), CW(DPT) : Creeping Wave (Double Probe Technique)
TOFD: Time of Flight Diffraction, ACFM: Alternation Current Field Magnetic

4. Future works need for more reliable evaluation of the closure weld integrity

Prerequisite No.	Items	Current Situation	Future work for getting more reliability	
			View point	Future work
1	General Corrosion	・Most Fundamental Property ・Influence of Filler Metal	Effect of filler metal on the corrosion propery of the weld metal	design of filler metal (chemical composition)
2	Susceptibility to SCC	・No effect of welding on the susceptibility	More realistic condition	Chemical environment ⇒groundwater composition (low carbonate concentration)
		・Appropriateness of evaluation method(SSRT)		Loading condition⇒Constant loading
		・appropriateness of evaluation time		Test period(Loading time)
3	Neutron Radiation Embrittlement	・Used estimation method for nuclear power plant component (high radiation／shor time)	Prediction method	Low & long-term radiation
4	Fracture Toughness	・Used low carbon steel for nuclear power plant component	More realistic estimation	Selected carbon steel
		・Available data for a geological formation on the estmation of critical crack length	Preciseness of geological condition	External Pressure
				Earthquake ・Max. Loading ・Fatigue fracture
		・LFM／NLFM	Fracture mode	Plastic-elastic farctute mechanics
5	Mecanical Buffering Property of Buffer Material	・Available data for a buffer material	Maintainng the mechanical buffering property	saturation, heat effect, stress condition, chemical reaction etc.,
6	Safety Factor	・referring from FFS of nuclear power plant component	Estimation basis	Appropriate value as a diposal container

5. Conclusions

1. Preferential corrosion induced on the weld metal provide by arc welding. Increasing of Si, Mn & S contents induces a change of corrosion property, which could be caused by a sulfide formation, MnS, etc. Attention should be paid on the chemical composition of the filler metal.
2. SCC susceptibility (crack growth phenomenon) was lower at the weld metal rather than at the base metal. The influence of welding - Rapid thermal cycle- makes microscopic fine structure and carbon dispersion into ferrite. That bring a resistivity toward SCC initiation.
3. Neutron radiation embrittlement was estimated not to occur at the closure weld
4. These results support the evaluation model proposed for the long-term integrity of overpack final closure.
5. Several penetration depth could be stand on the overpack lid closure. However, attention should pay to,
   (1)Structural integrity and its tendency, especially for the Max. P. S. and critical crack length.
   (2)Combination of welding and NDE methods.

End

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Laboratory of Engineering materials

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The time effects on the integrity of canisters for geological disposal

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Workshop on Design and Assessment
of Packages for Radioactive Waste

21st-22th November 2006 at Hotel Marijke, Bergen N-H
Organized by the Joint Research Centre,
Institute for Energy, Petten, The Netherlands

TIMESCALES OF THE RADIOTOXICITY OF THE SPENT NUCLEAR FUEL

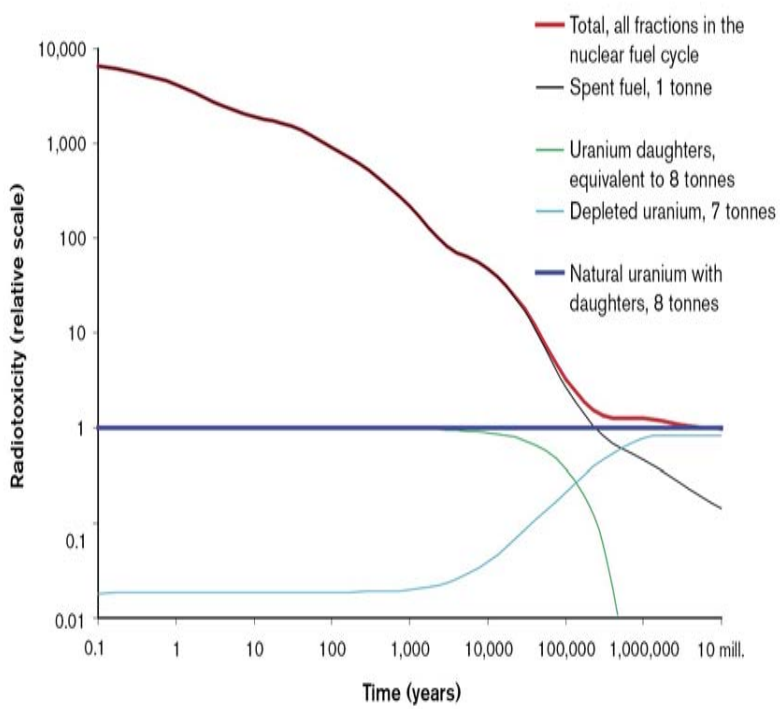
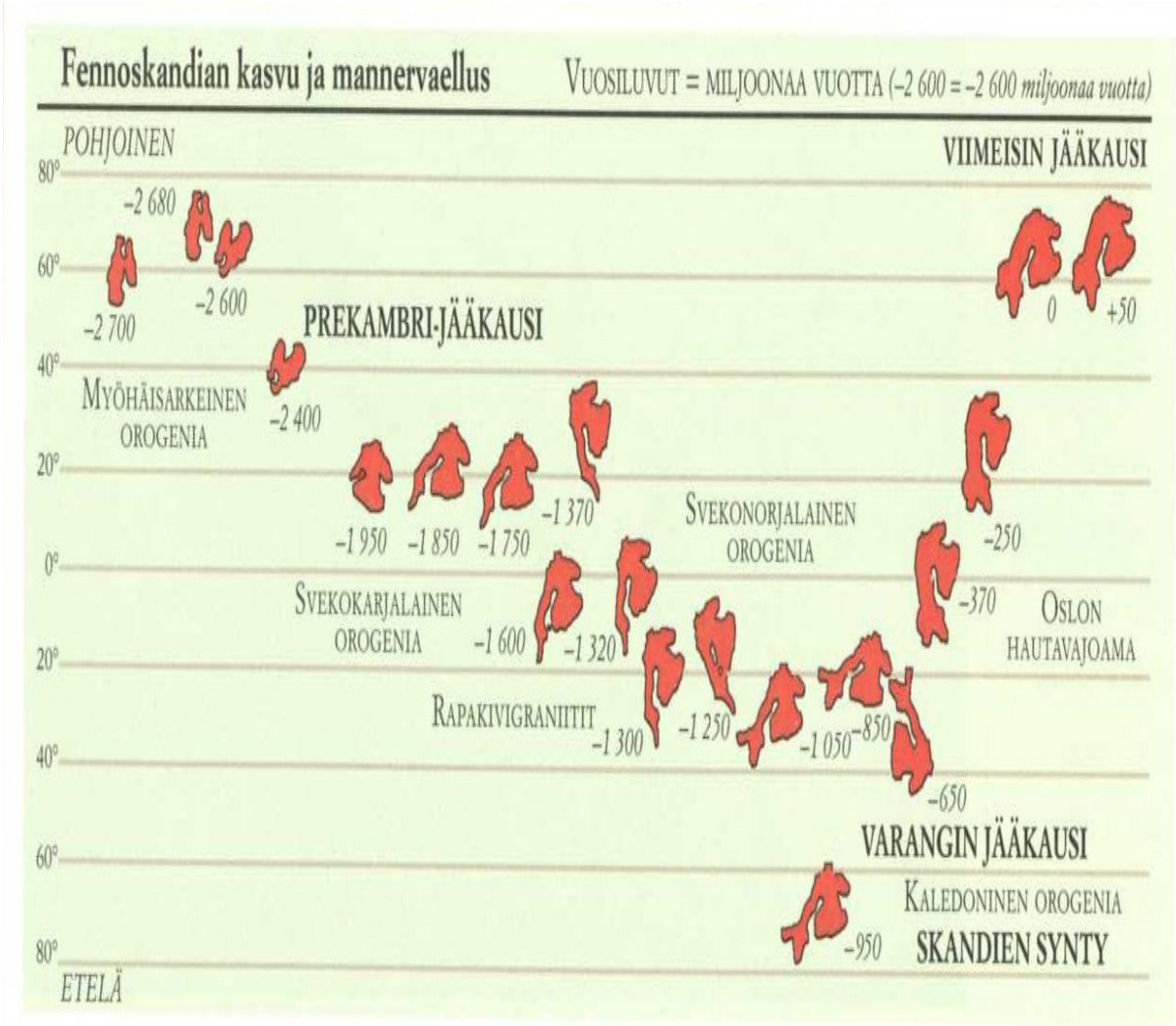


Figure 2-3. Radiotoxicity on ingestion of uranium ore (blue line), and of the sum of all fractions that arise when the same quantity of uranium mineral is used in the nuclear fuel cycle (red line). The time refers to the time after reactor operation. The different fractions comprise the spent fuel (38 MWd thermal energy/kg U of type SVEA 64 BWR), the depleted uranium and the uranium daughters that are separated in the uranium mill. From /Hedin, 1997/.

(SKB, TR-04-11, 2004)

- **- An assessment period longer than 100000 y. is required. After that period the radiotoxicity of spent nuclear fuel is comparable to that of the natural uranium ore once used to produce the fuel**
- **- However, the radiotoxicity of fission products and actinides remains after 100000 y. and therefore, for example, in Swedish SR-Can safety assessment the timescale is 1 m. y.**
- **- This means nine repetitions of the Weichselian type (last ice age) of glacial cycles!**

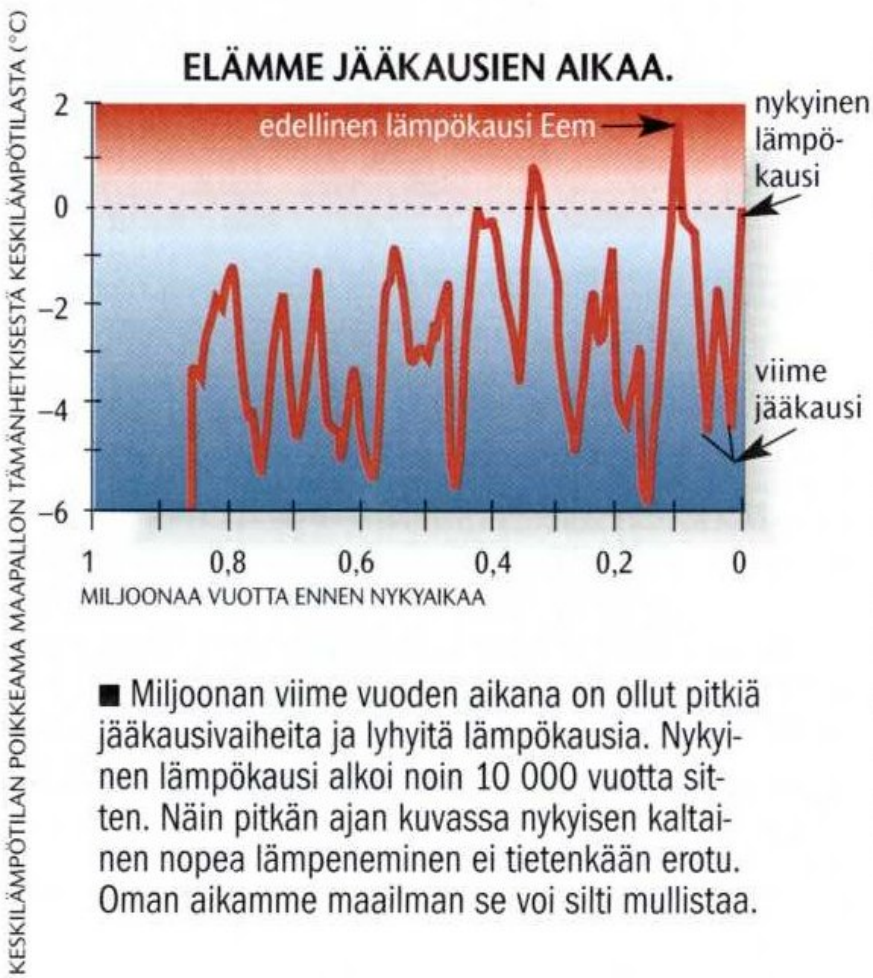
FORMATION OF FENNOSKANDIA AND ITS TRAVELLING ON EARTH



(M. Koivisto, Jääkaudet (Ice ages), 2004)

- **Ice age cycles occurred already during early geological times on earth**
- **For example, Huron ice age (2,3 mrd y. ago) lasted over 200 m. y.**
- **During the time of dinosaurs (250 – 65 m. y. ago) - Jurassic and Creataceous periods – average temperature of the earth was up to 14 ºC higher (peak at 100 m. y. ago) than now. As the result of the extremely high biological life main part of the oil and natural gas reserves were formed.**

THE EARTH LIVES THE TIME OF RECURRING ICE AGES



■ Miljoonan viime vuoden aikana on ollut pitkiä jääkausivaiheita ja lyhyitä lämpökausia. Nykyinen lämpökausi alkoi noin 10 000 vuotta sitten. Näin pitkän ajan kuvassa nykyisen kaltainen nopea lämpeneminen ei tietenkään erotu. Oman aikamme maailman se voi silti mullistaa.

(M. Eronen, Tiede 2004/4)

- **- During last 2 m. y. cyclic ice ages occurred globally and cold and warm periods have been repeated cyclically**
- **- Especially, during last 1 m. y. period long cold periods and shorter warm periods (interglacials) have cyclically evolved in 100000 y. periods**
- **- The present rapid warming due to greenhouse gas effects is not visible here. If the about 200 y. lasting human-induced fossile fuel period is able to change the cyclic climate change or even block the astronomic climate change is an important open question**

EEM INTERGLACIAL IN FENNOSKANDIA

- **- Eem interglacial was the warmest climate time – the average temperature in Fennoskandia was 4 ºC higher 125000 y. ago as at present lasting for 11000 y.**
- **- Eem sea was connected to the Northern Arctic Sea. Later because of rising land this was disconnected and only the Atlantic connection survived**
- **- Eem is the name of a river in Holland. The names of the previous ice ages are given by names of places in Holland**

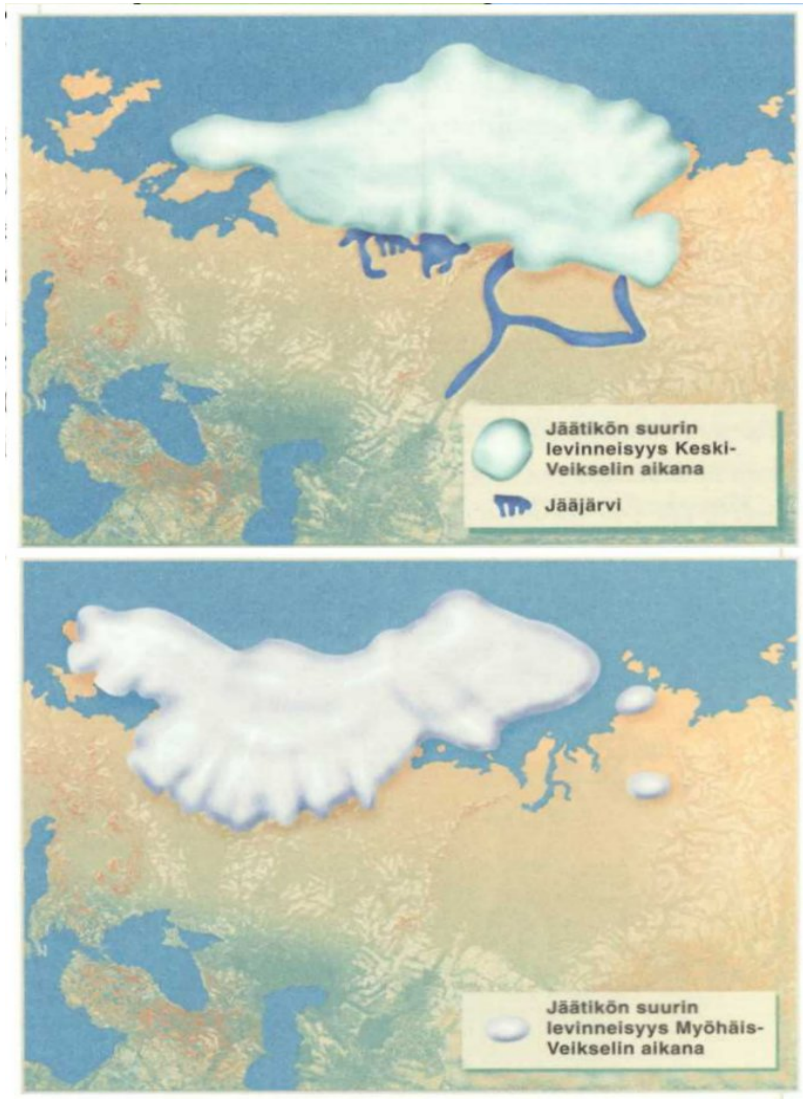
(M. Koivisto, Jääkaudet (Ice ages), 2004)

FENNOSKANDIA BEFORE LAST WEICHSELIAN ICE AGE

- **- Long and cold period before last Weichselian ice age (115000-105000 y. ago) together with rising of land after Saale ice age to a level much higher than now resulted in lake districts in the area of Eastern Sea**
- **- The end moraine formations in Lapland and rivers depict the advance and retreat of the ice front**
- **- In general, earth is having always an ice age; 30% of earth was covered by continental glacier in maximum and 10% at present**

(M. Koivisto, Jääkaudet (Ice ages), 2004)

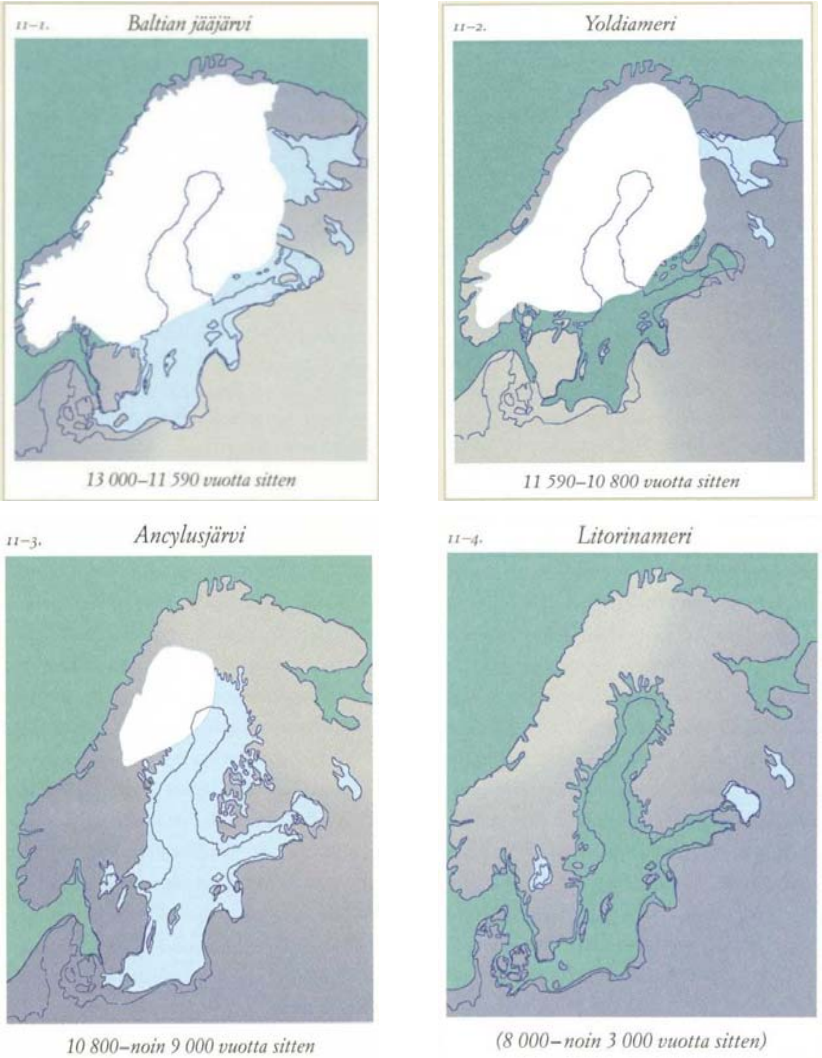
WEICHSELIAN ICE AGE EXTENSION IN ITS MIDDLE AND MAXIMUM PHASE



- **-Weichselian ice age (115000-11590 y. ago) had periods of warmer and colder climate causing the ice front to advance and retreat**
- **- 25000 y. ago started the late Weichselian phase and growth of the continental glacier resulting in 2-3 km thick ice sheet extending to Hamburg and Moscau 20000-18000 y. ago**
- **-Retreat started 18000 y. ago, Finnish coast was reached 13000 y. ago and 11590 y. ago cyclic period of ice growth ended and finally 10200 y. ago Finland was free of ice**

(M. Koivisto, Jääkaudet (Ice ages), 2004)

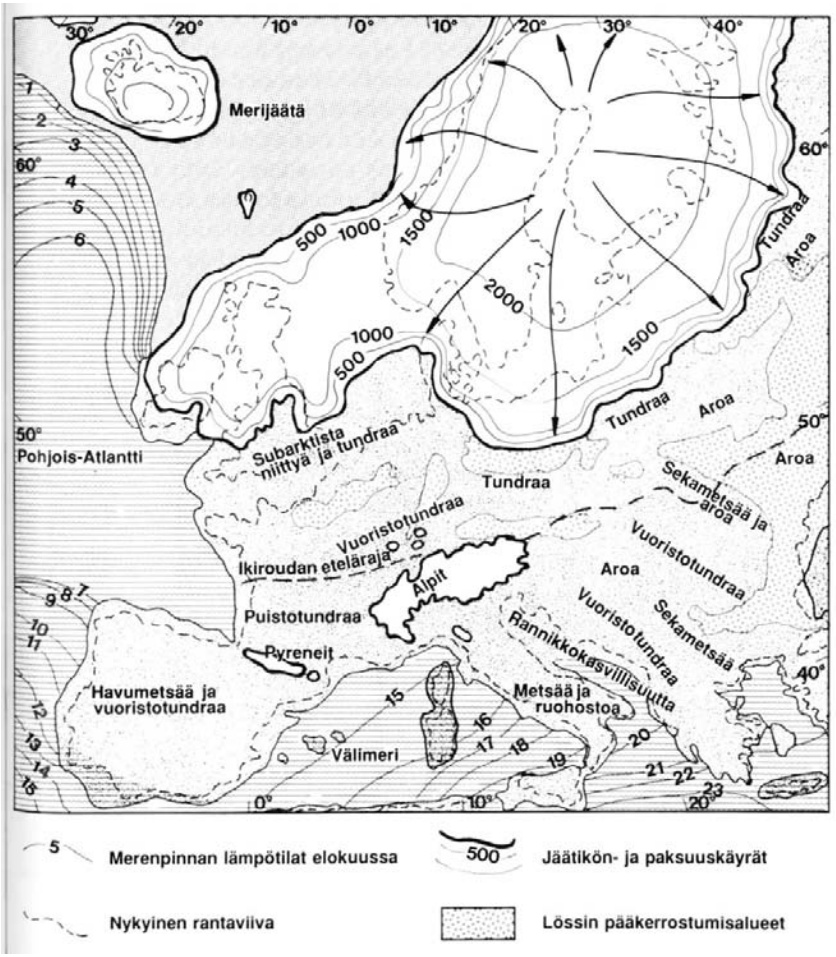
FORMATION OF PRESENT EASTERN SEA AFTER WEICHSELIAN ICE AGE

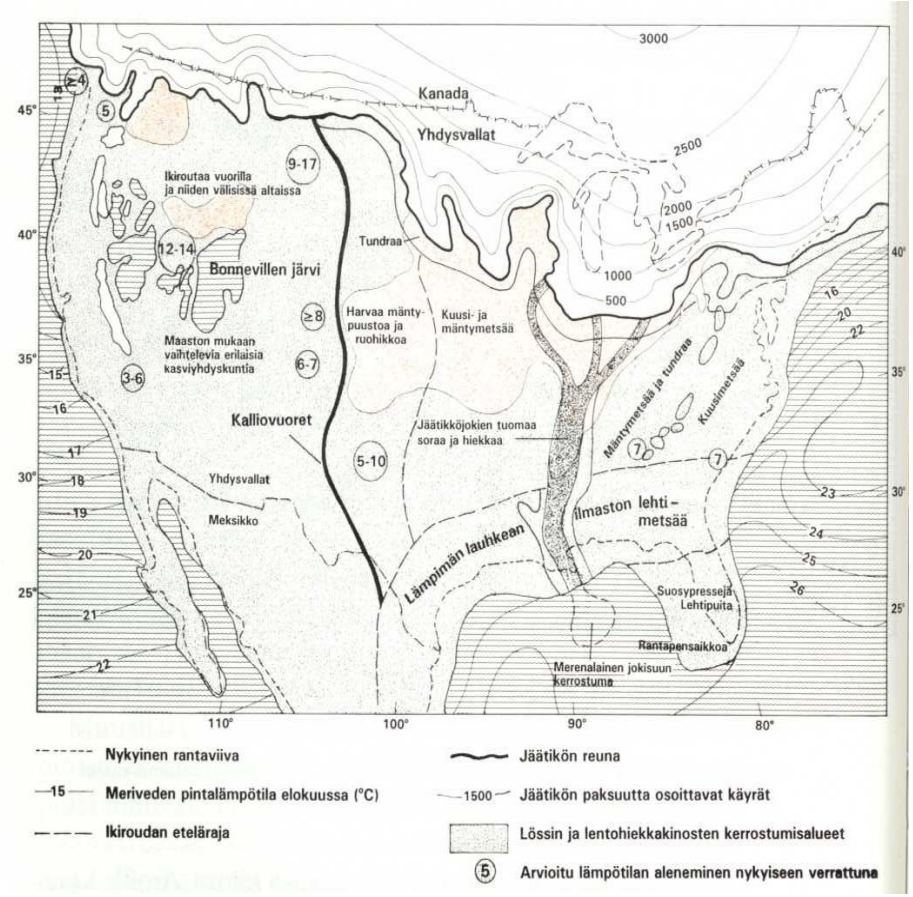


- **- Smelting of ice, rise of sea level and land have affected the phases of Eastern Sea. During last 2000-3000 y. only rising of land has occurred**
- **- After Weichselian ice age Eastern Sea has been periodically a lake and a sea with Atlantic connection; change from Baltic to Yoldia caused rapid drop of lake level by 28 m**
- **- Rise of land was in the beginning faster than rise of sea level (affected by climate) and they were competing. Rise of water level in Ancylus lake and in the sea (1 m in 100 y.!) resulted 9500 y. ago in the same sea level**

(M. Koivisto, Jääkaudet (Ice ages), 2004)

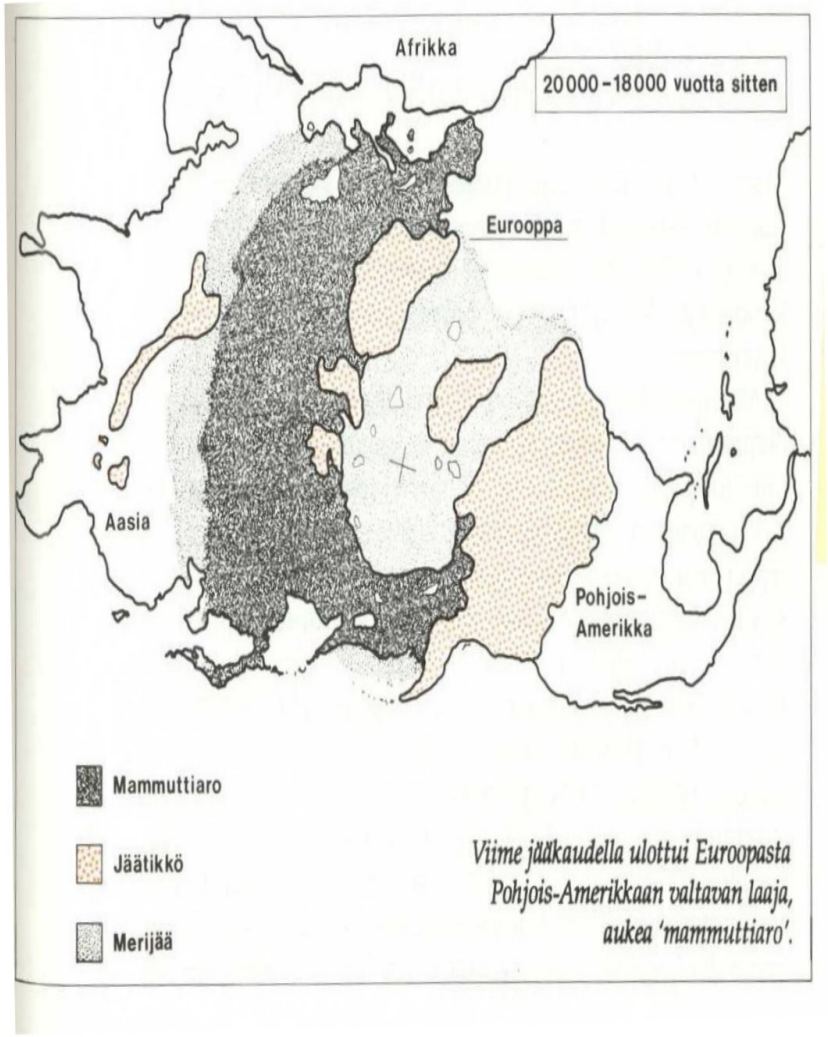
EUROPE AND NORTH AMERICA DURING THE COLDEST PHASE OF THE LAST WEICHSELIAN ICE AGE





(M. Eronen, Jääkausien jäljillä, 1991)

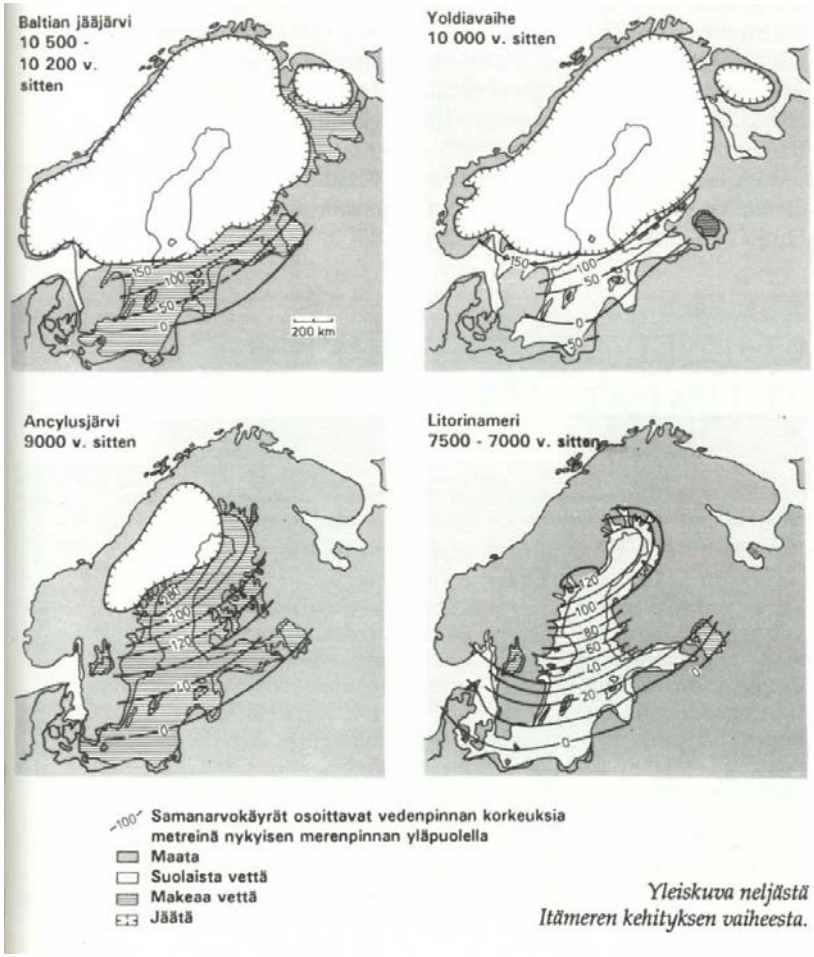
VAST MAMMOTH PRAIRIE EXTENDING FROM EUROPE TO NORTH AMERICA



- **- Vast mammoth prairie extended from Europe to North America during 20000-18000 y. ago**
- **- Temperature in Europe was 4-5ºC and in Eurasia 10-15ºC lower than at present**
- **- Permafrost in Siberia extends still over 1 km depth in the earth ground**

(M. Eronen, Jääkausien jäljillä, 1991)

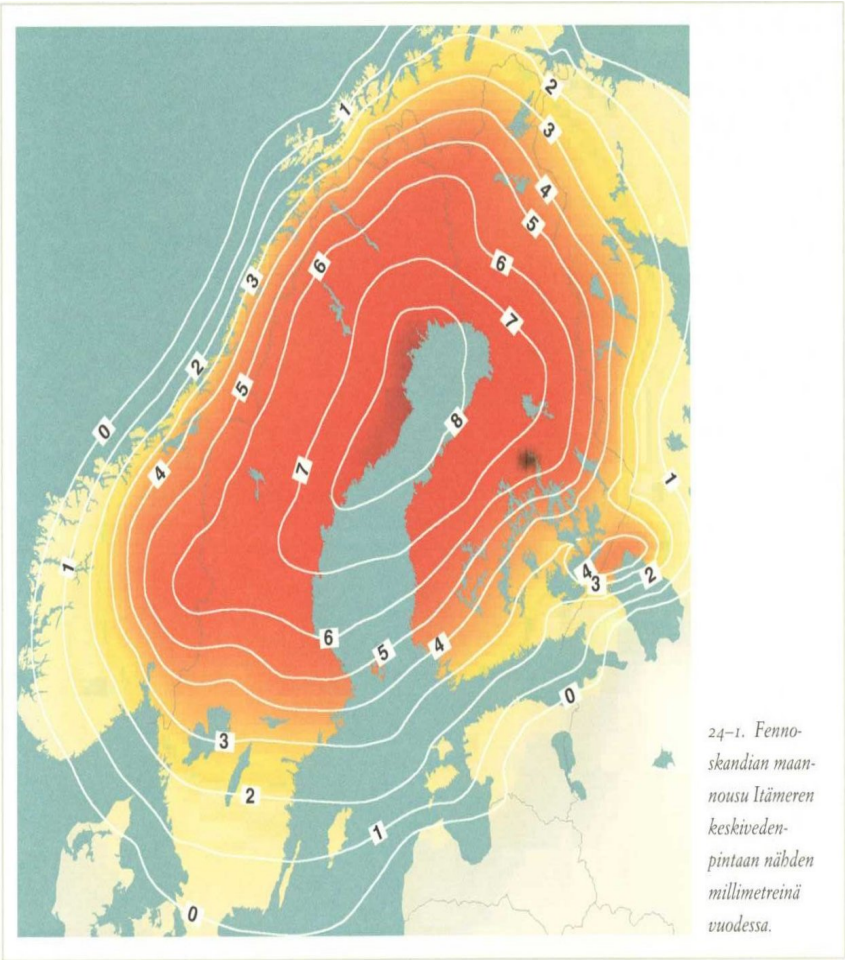
RISE OF LAND AS A RESULT OF SMELTING OF ICE



- **- Smelting of ice resulted in rapid rise of depressed land**
- **- In the beginning the rise is estimated to be even 10 times faster than at present**
- **- The development phases of Eastern Sea are mainly controlled by the fast rise of land after last ice age**
- **- The isoaltitude curves show the sea levels above the present sea level giving a relative figure of the rise of land**

(M. Eronen, Jääkausien jäljillä, 1991)

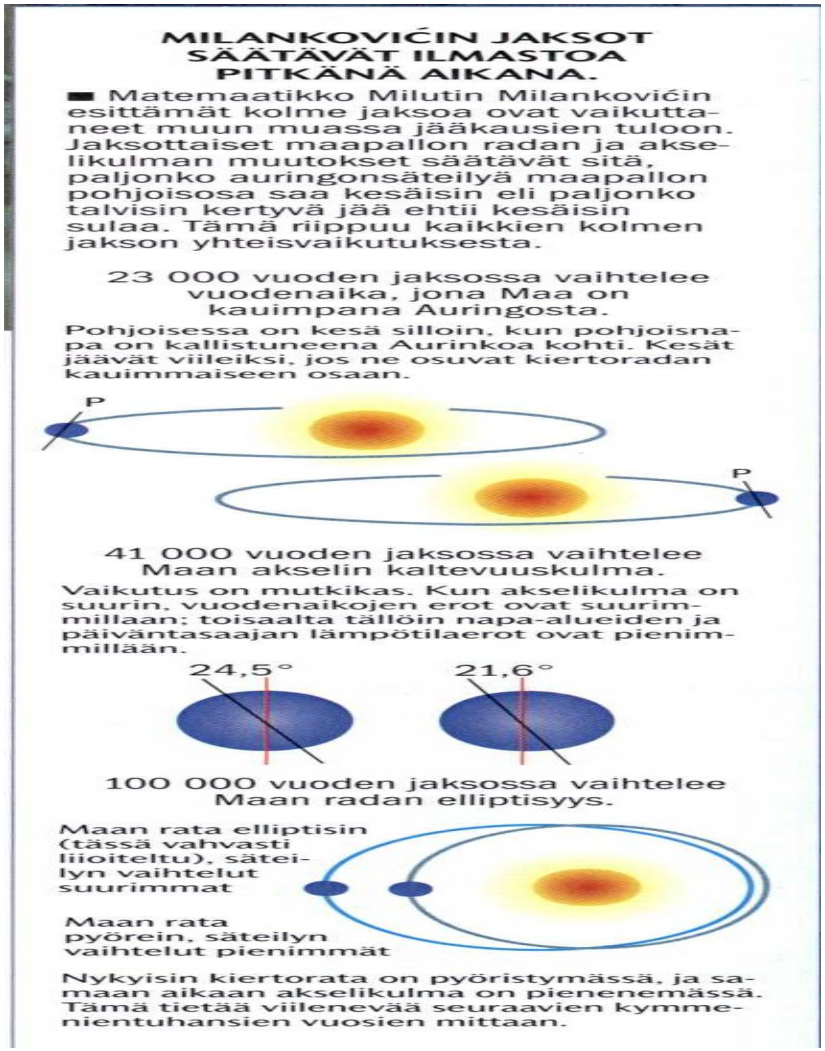
RISE OF LAND TODAY



(M. Koivisto, Jääkaudet (Ice ages), 2004)

- **- The rise of land in low land areas may move the coastline even 1 km in 100 years**
- **- The rise of land is shown in Fennoskandia in mm per year as compared to the average water level of Eastern Sea**
- **- The depth of the depression in the earth crust caused by ice sheet (3,3-3,7 km) was 900-1000 m. Main part of the rise, 500 m, occurred already under the smelting ice. The present depression is 80-120 m and its correction will take 7000-12000 more years**

ASTRONOMIC CLIMATE CHANGE THEORY OF MILUTIN MILANKOVITCH

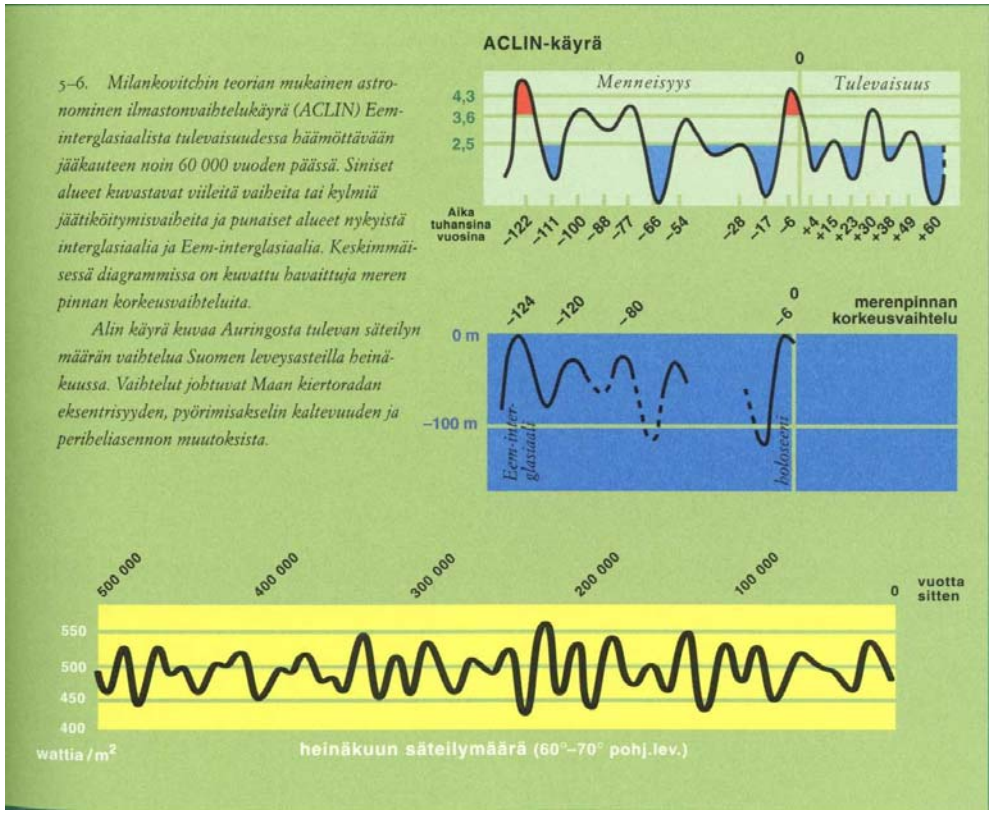


- **- According to theory continential glaciers form, when the earth is most far from the sun as the summer of northern hemisphere starts and the ecentricity of the earth orbit as well as the angle of the earth axis (22,0-24,5º) are simultaneously in minimum**

- **- Under these conditions the summers in the northern hemisphere are long and cool (snow from winter does not smelt completely) and the winters are very cold**

(M. Eronen, Tiede 2004/4)

PREDICTION OF THE FUTURE ICE AGES



- Based on the climate change theory of Milankovitch (ACLIN theory) starting from Eem interglacial the next ice age starts 60000 y. from now and the ecosystem processes are expected to be similar to the present

- Blue color depicts the cold periods of ice growth and red the present interglacial phases. The sea level variation is also shown as well as the radiation of sun in Finland in July

- We are living the late interglacial period (average temperature 2ºC lower than 6000 y. ago). The human role in the present rapid warming due to greenhouse gas effects may affect the climate development drastically

(M. Koivisto, Jääkaudet (Ice ages), 2004)

CYCLIC EVOLUTION OF THE CLIMATE IN FENNOSKANDIA

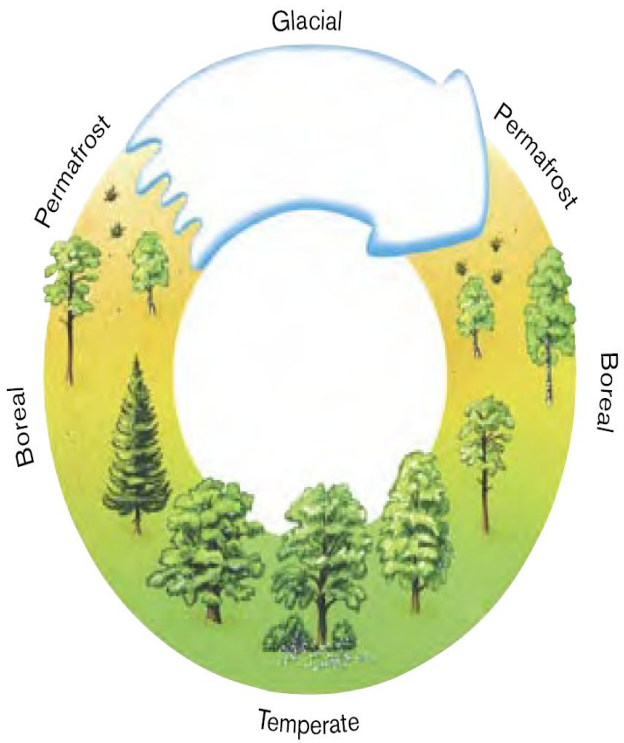
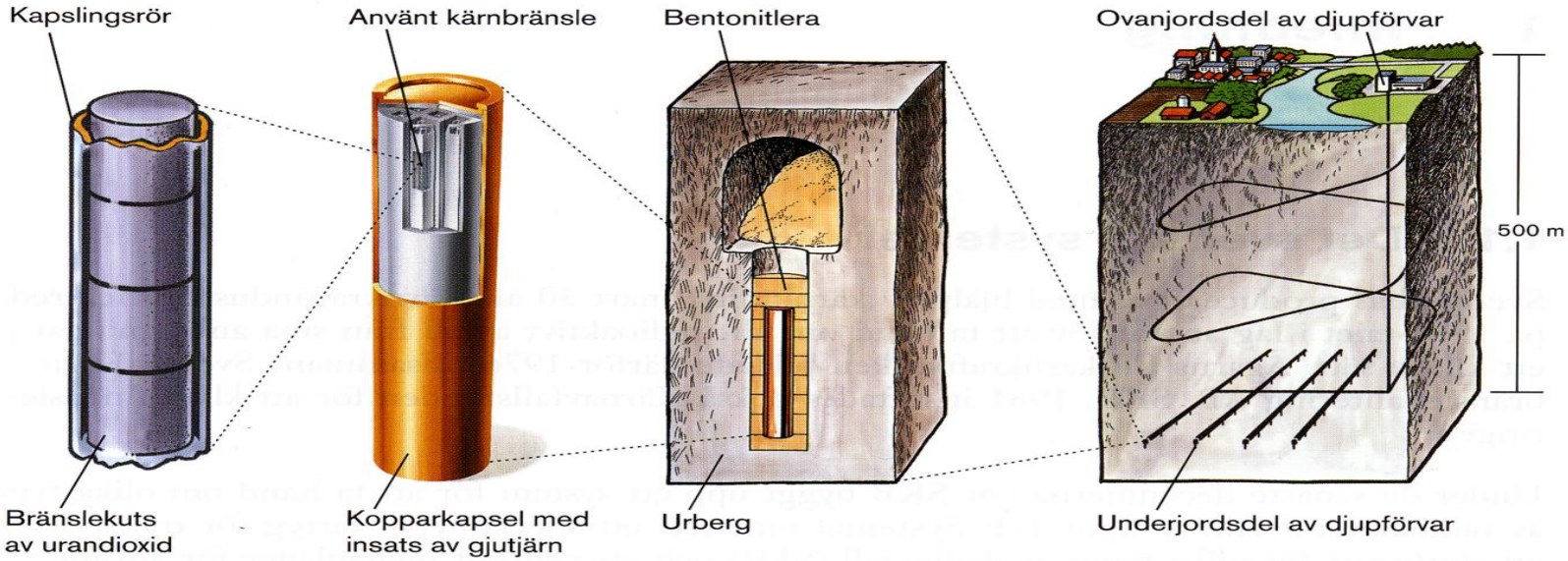


Figure 21-2. Climate-related changes can be viewed as a cycle with successive transitions between periods of different length when different climate domains prevail.

- **- The climate evolves in roughly 100000 y. cycles each cycle including several phases of permafrost and various glacial conditions**
- **- The mechanical, hydraulic and groundwater chemical conditions in the bed rock vary according to the climate evolution**
- **- The effects of glacial loading are, thus, very important and it can also be debated if the human-induced climate changes have to be accounted**

(SKB, TR-04-21, 2004)



Figur 1-2. *KBS-3-metoden är SKB:s referensmetod för förvaring av använt kärnbränsle.*

(SKB, TR-04-21, 2004)

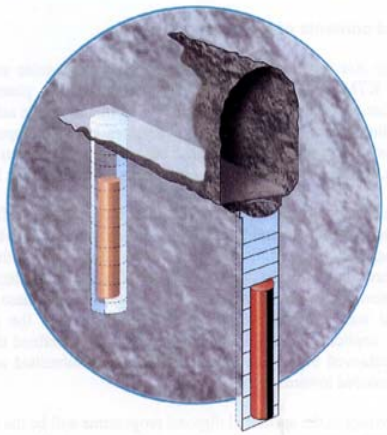


Figure 1-3. *The KBS-3 concept for the disposal of spent fuel.*

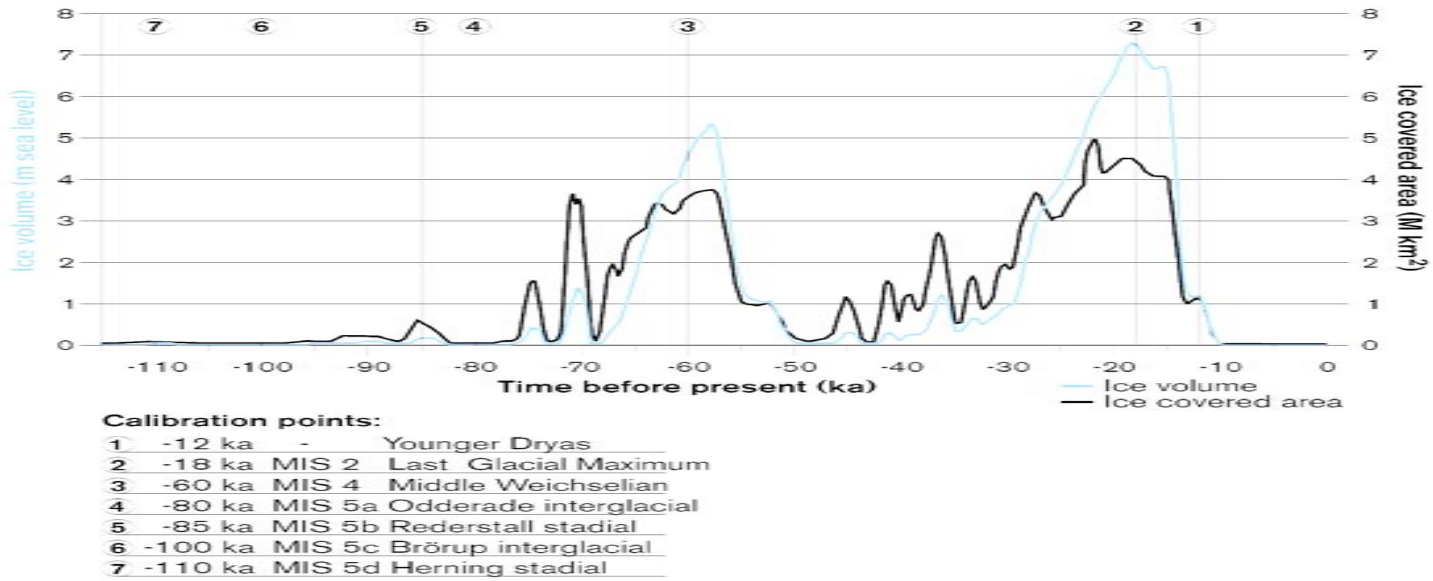


Figure 11-7. *Reconstructed total ice volume and ice covered area throughout the Weichselian, and the calibration points for the reconstruction.*

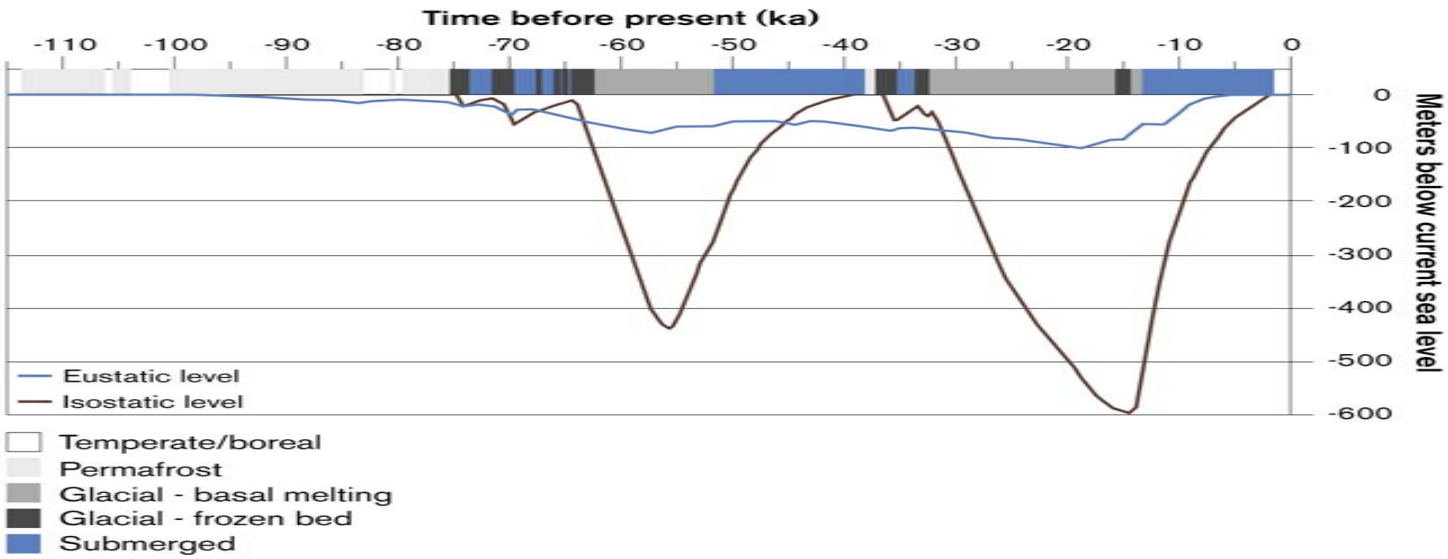


Figure 11-9. *Time series of climate domains and submerged periods and simplified eustatic and isostatic evolutions from ice sheet modelling.*

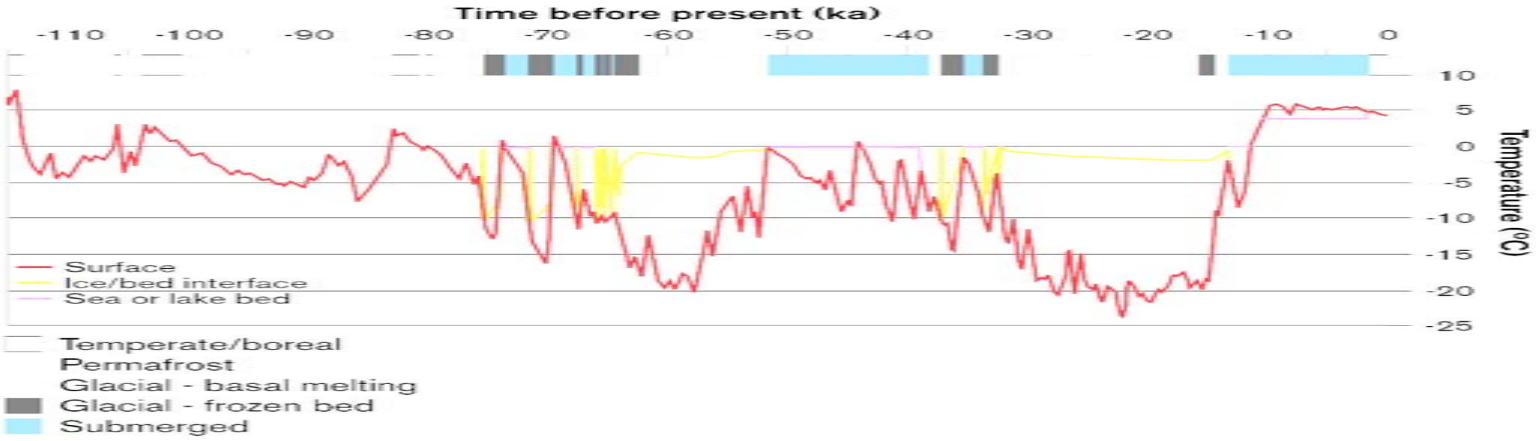


Figure 11-10. *Time series of climate domains, submerged periods and temperatures. Surface temperature during ice free, non-submerged periods is that at the bed surface and during submerged periods that at the sea or lake surface (red). During ice-covered periods both the temperatures at the ice surface (red) and at the ice/bed interface (yellow) are shown. Assumed temperature at the sea/lake bed is show in purple.*

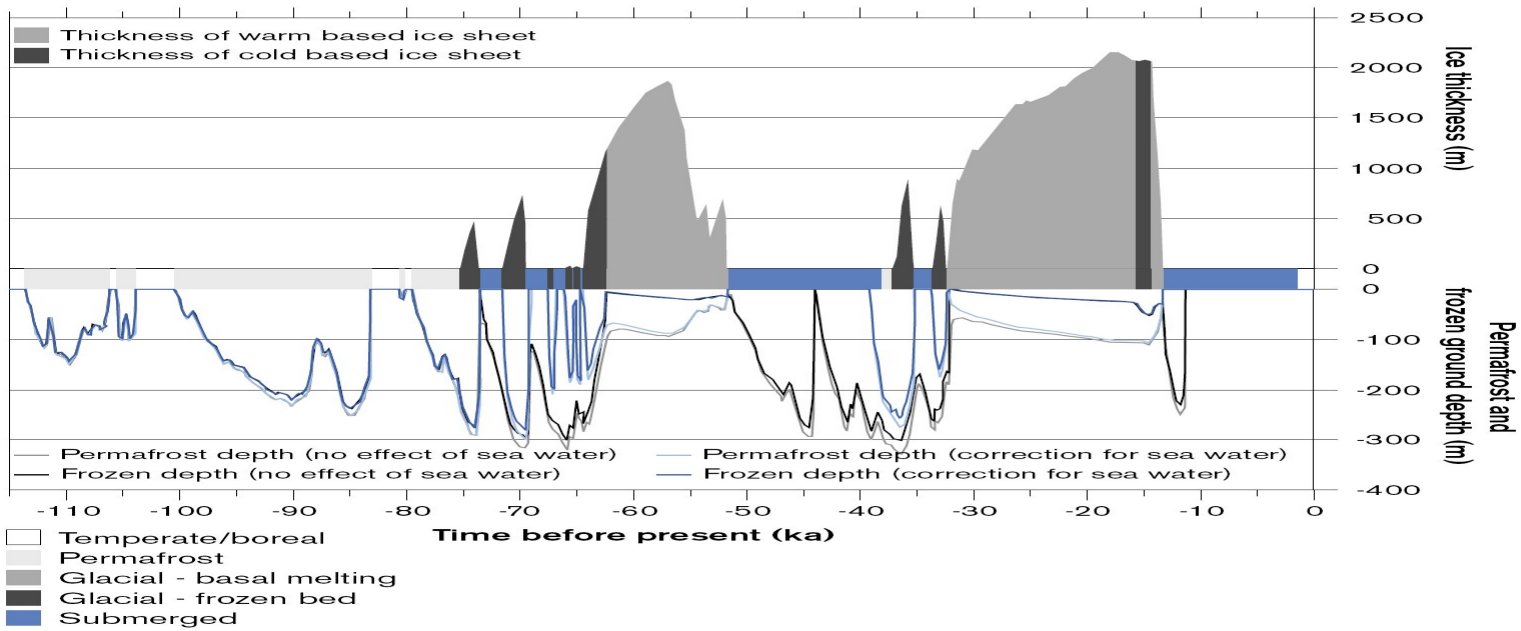


Figure 11-11. *Time series of climate domains, ice thickness and depth of permafrost and perennially frozen ground (as permafrost is defined by temperature the occurrence of permafrost does not always mean that the ground is frozen).*

PERMAFROST DEVELOPMENT

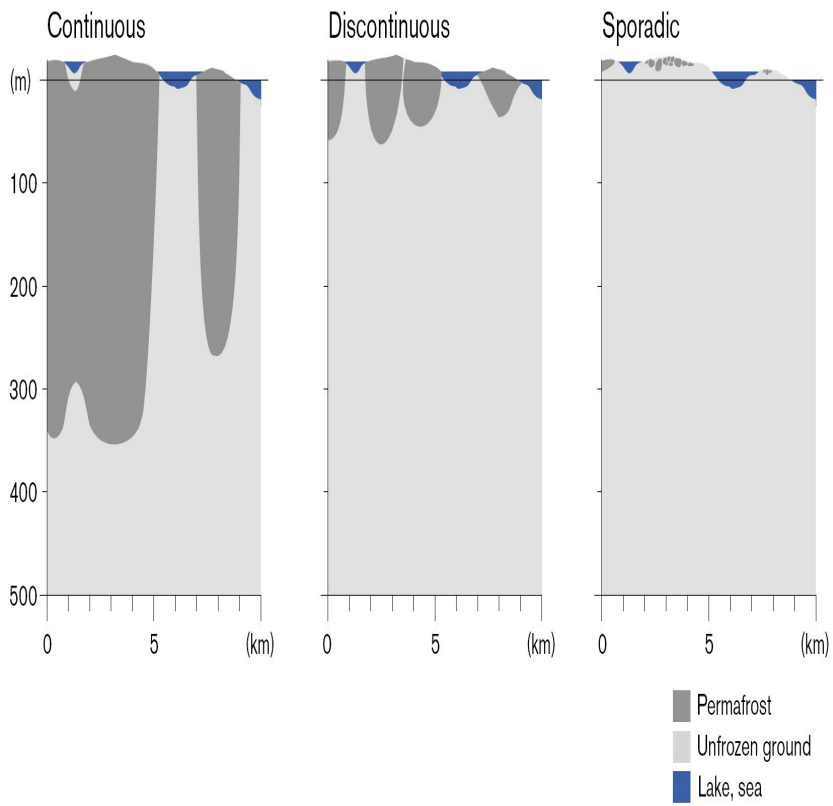


Figure 21-4. Illustration of sporadic, discontinuous and continuous permafrost.

(SKB, TR-04-21, 2004)

- **- Periglacial permafrost develops during cold periods when the area is ice free or lies above the sea surface level. Subglacial permafrost develops under the cold ice sheet**
- **- The extension of permafrost induces formation of saline groundwaters due to the exclusion of salt during groundwater freezing. Gravitational effect causes gradual downward movement of salinity causing a salinity gradient below permafrost**
- **- Saline ice front and non-saline melting zone and oxygen-rich melt waters may affect the corrosion of canisters**

SHORELINE DISPLACEMENT EFFECTS ON DEEP REPOSITORY CONDITIONS

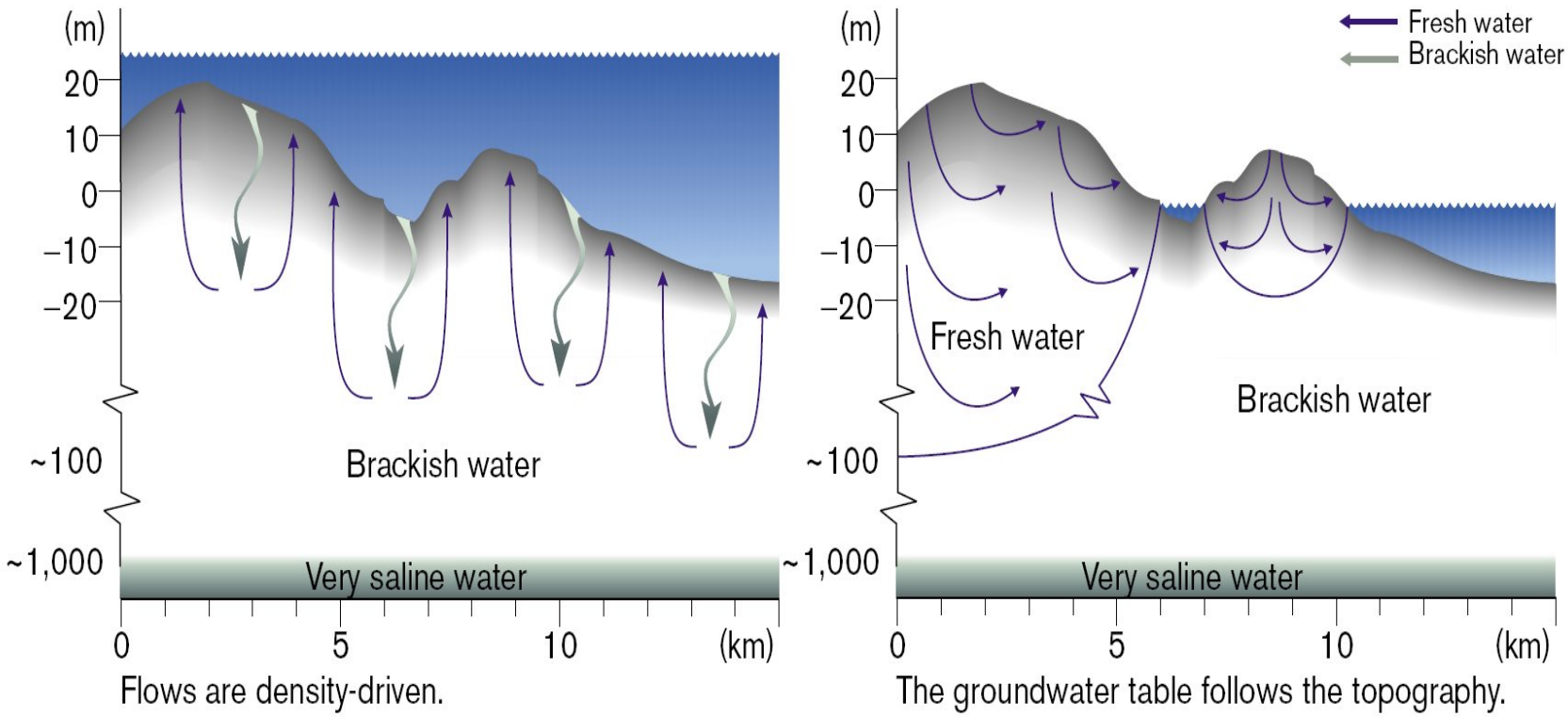


Figure 21-5. *Schematic illustration showing variations of hydrology and salinity in connection with regression on a hypothetical site on the present-day coast of the Baltic Sea.*

(SKB, TR-04-21, 2004)

IMPORTANT QUESTIONS UNDER GLACIAL PERIODS

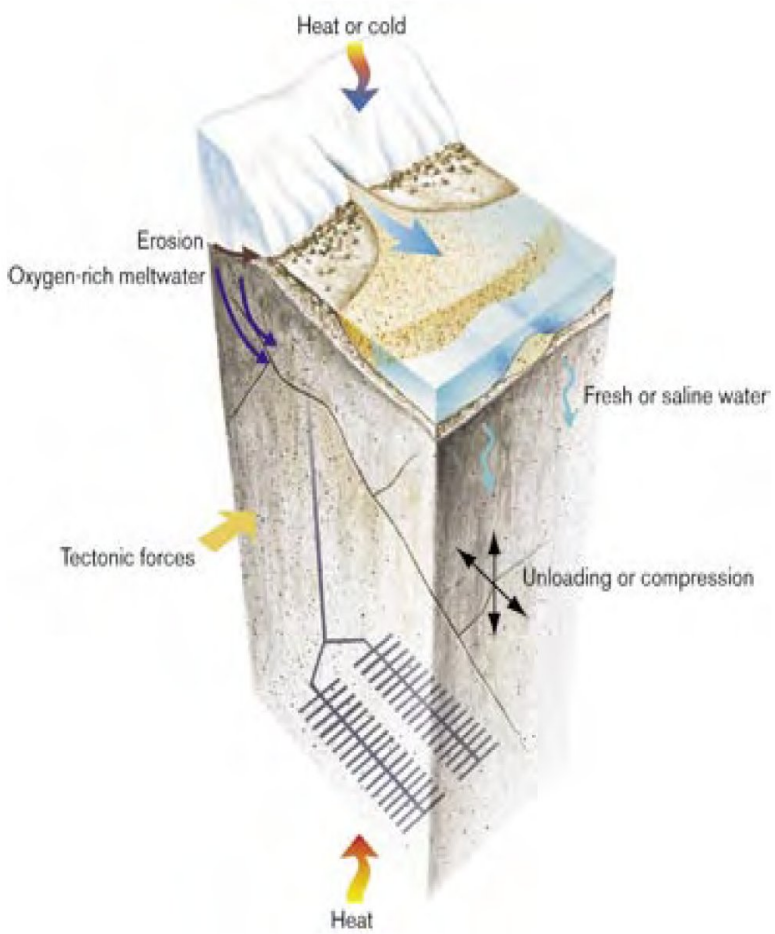


Figure 21-3. As an ice sheet advances and retreats, changes occur in temperature, hydrological conditions, rock stresses and groundwater composition.

(SKB, TR-04-21, 2004)

- **- As an ice sheet advances and retreats, changes occur in temperature, hydrolocical conditions, rock stresses and groundwater composition**
- **- Knowledge is needed from: depth of permafrost, water pressure and flows, infiltration of deep saline groundwater and oxygen-containing glacial meltwater, changes in rock stresses and movements**

DESIGN PREMISES FOR THE CANISTER

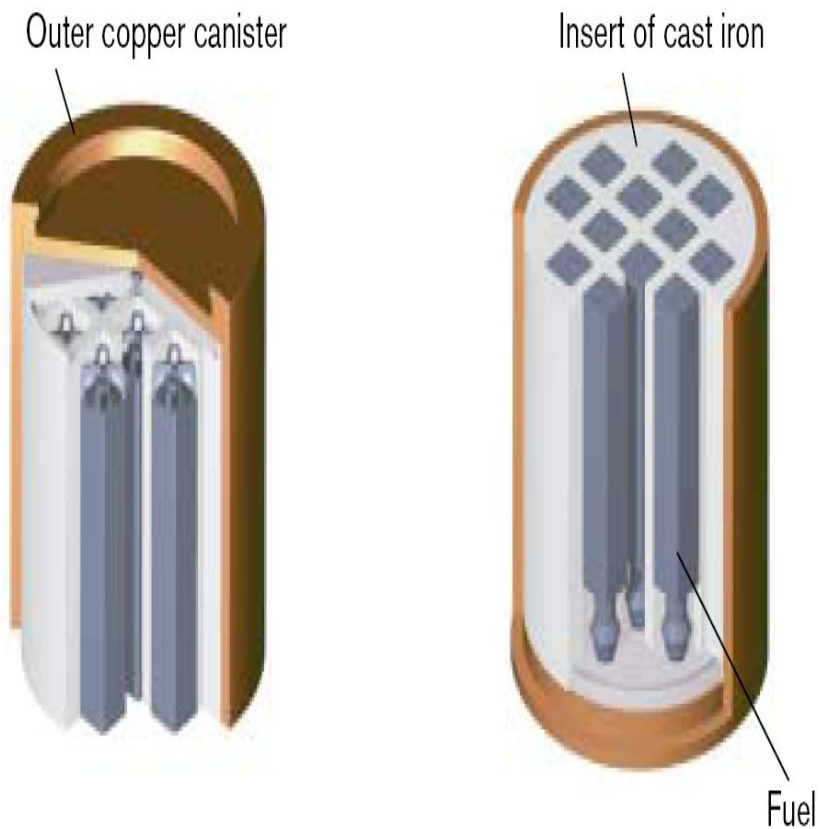
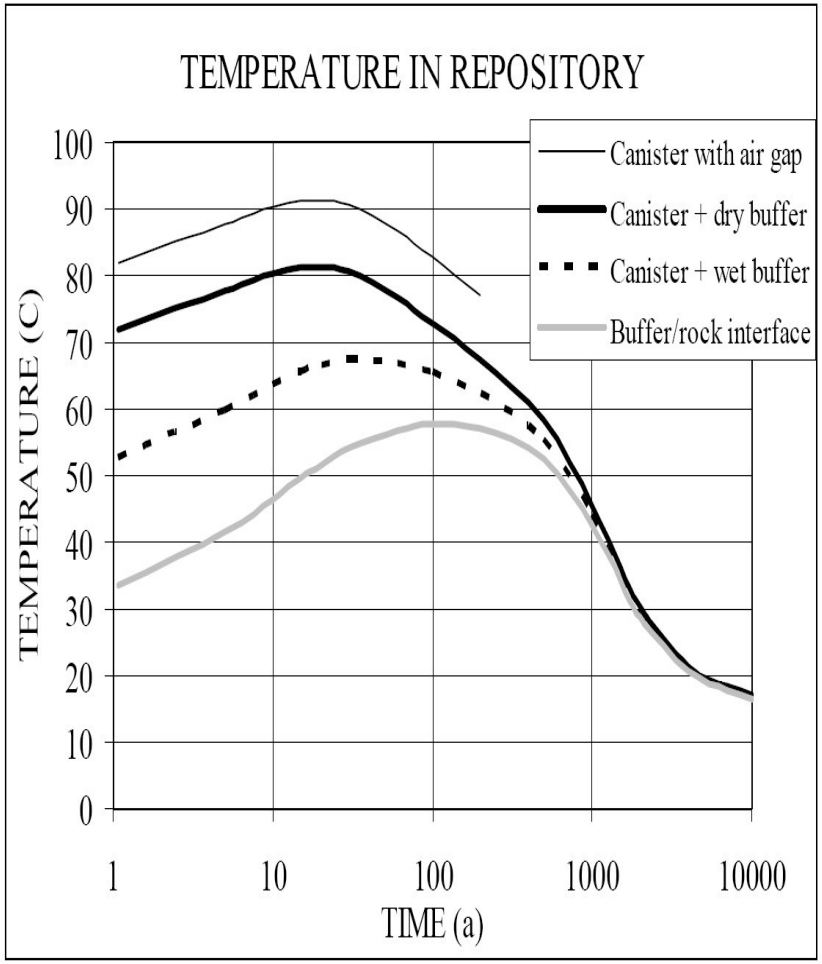


Figure 1-1. Exploded view of spent BWR fuel disposal canister.

(SKB, TR-04-11, 2004)

- **- The canister must withstand all known corrosion processes and it is expected to remain intact in the repository for at least 100000 years**
- **-The canister must withstand the mechanical stresses that arise in the repository up to the depth of 700 m**
- **- The strength of the canister must withstand the loads that can be expected during an ice age (or even 9 ice ages!)**
- **- The surface dose rate on the outside of the canister must not exceed 1 Gy/h**

CANISTER SURFACE TEMPERATURE DEVELOPMENT



- **- Maximum temperature of the canister is reached within 10-15 years after disposal**
- **- Thermal conductivity of the metal body of the canister is two orders of magnitude higher than that of the surrounding bentonite and rock. The metallic canister is in a uniform temperature**
- **- It is very likely that the canister surfaces will not experience temperatures exceeding 100ºC and thus no water boiling and salt deposit formation occur**

(Posiva, 2006-02)

CREEP DEFORMATION AND FRACTURE OF COPPER CANISTER

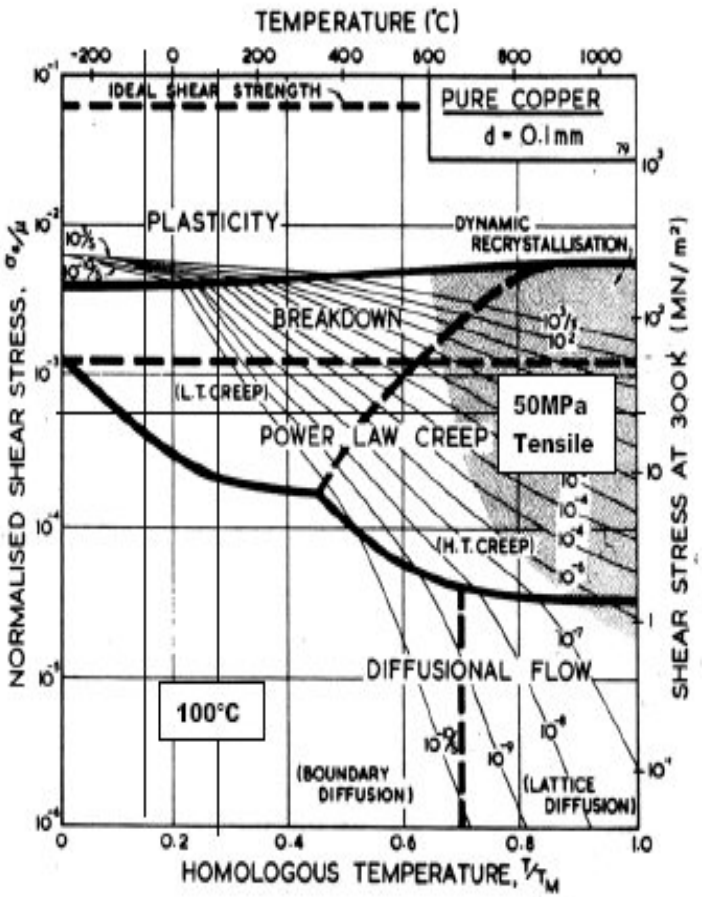
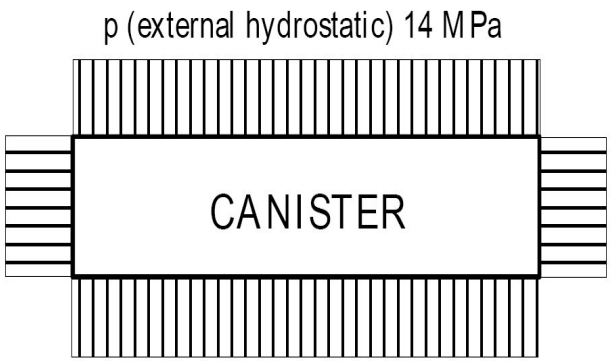


Figure 1. Deformation diagram for pure copper (after Frost and Ashby, 1982)

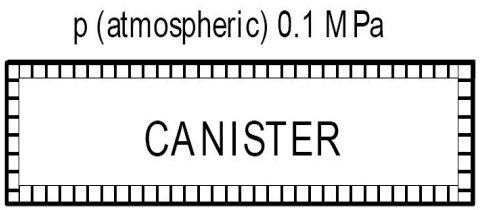
- **- Canister is subjected to mechanical loads due to the hydrostatical and swelling pressure of the bentonite**
- **- The external pressure of about 15 MPa will deform copper canister until a full contact to cast iron insert closing the 1-2 mm radial gap and resulting in 3-7% creep strain, which is less than the minimum 10% creep rupture strain. The dominant creep deformation mechanism is grain boundary controlled diffusion**
- **- Creep data should also be generated in groundwater in addition to air and at decreased strain rates**
- **- The temperature decrease will later cause tensile strain in copper exceeding the yield stress, but not the creep rupture strain unless accidental shear movements occur in the bed rock**

(SKi Report 2005: 18)

MECHANICAL DESIGN LOADS OF THE CANISTER



Case 1. Basic design load, external pressure of 14 MPa



Case 2. Internal pressure of 0.1 MPa

- **- In the normal condition the mechanical design load is 14 MPa consisting of 7 MPa hydrostatic pressure and maximum of 7 MPa swelling pressure of bentonite**
- **- The design pressure is assumed to be evenly distributed and acting hydrostatically on the canister**
- **- Internal pressure of 0,1 MPa is assumed**

(Posiva, 2006-02)

SPECIAL SWELLING PRESSURE LOADS OF BENTONITE

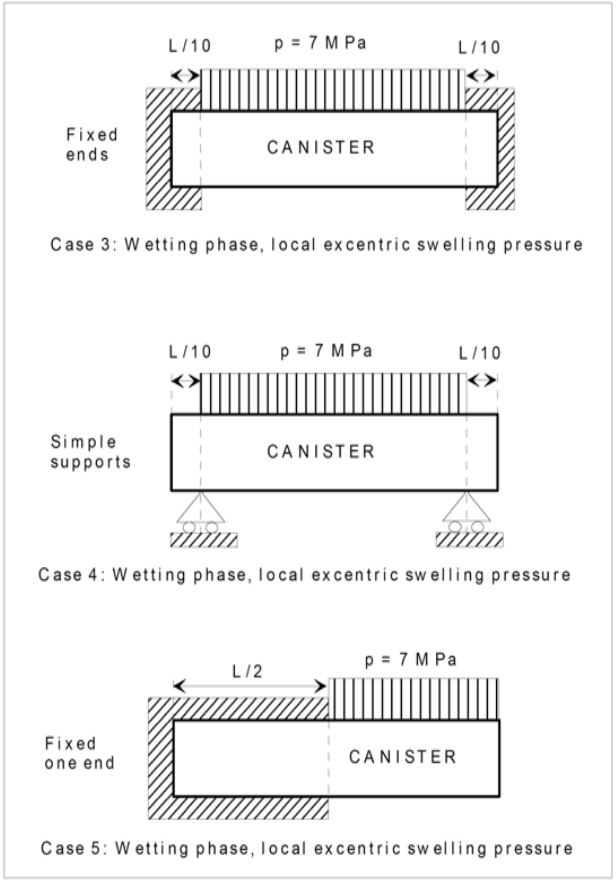


FIGURE 2. Special local swelling loads postulated during wetting phase.

- **- Swelling pressure of bentonite may have irregularities, especially during early years of the repository, when bentonite starts to wet**
- **- Conservatively defined different types of loading conditions have been analysed**
- **- The analyses have to be considered based on the detailed creep mechanisms and behavior of the canister**

(Posiva, 2006-02)

SPECIAL SWELLING PRESSURE LOADS OF BENTONITE

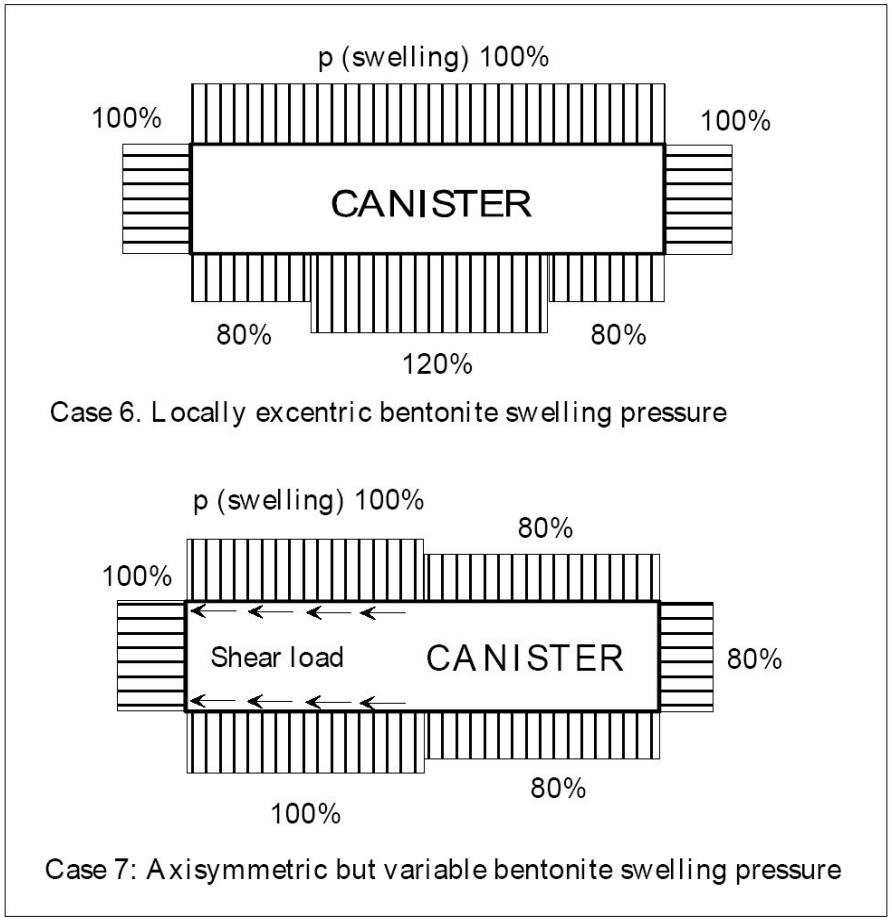


FIGURE 3. *Special swelling pressure loads postulated in fully saturated phase. Ground-water hydrostatic pressure can be present simultaneously.*

(Posiva, 2006-02)

EFFECTS OF ROCK SHEAR MOVEMENTS ON PLASTIC STRAIN IN CANISTER

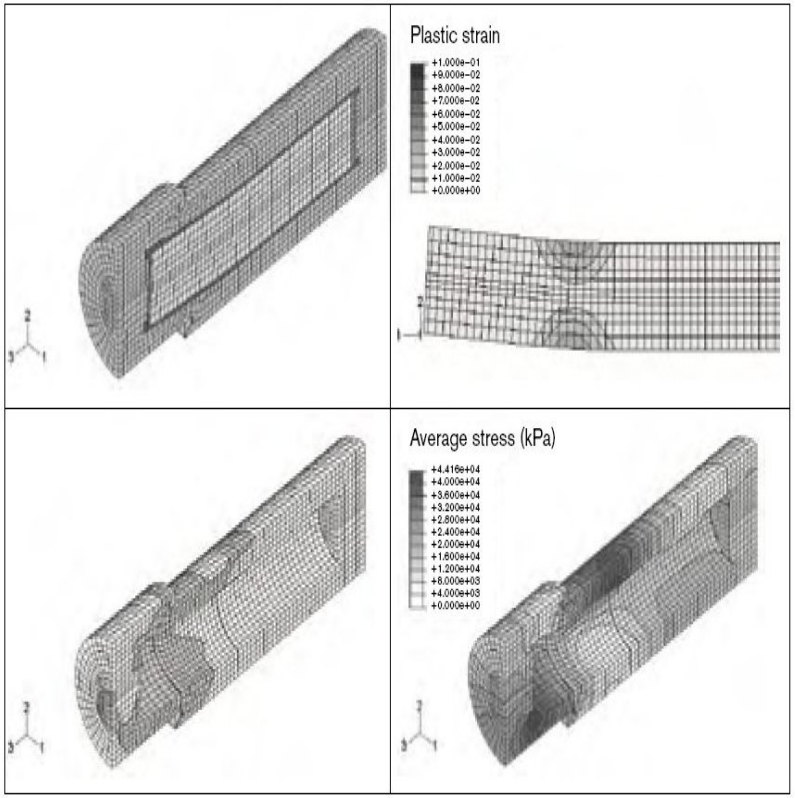
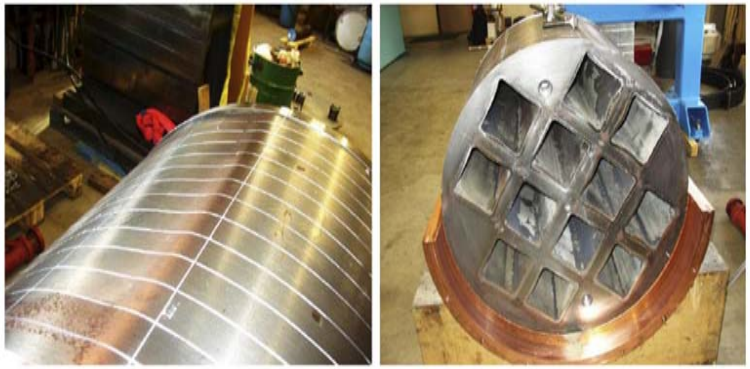


Figure 17-2. Examples of results from a calculation with 20 cm eccentric rock shear (density at water saturation 2,000 kg/m^3 and shear rate 1 m/s). Deformed structure (upper left), plastic strain in the cast iron insert (upper right), plastic strain in the buffer (lower left) and average stress (kPa) in the buffer (lower right).

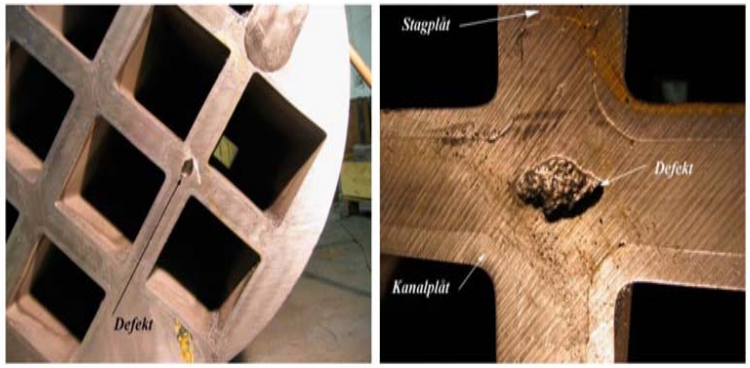
(SKB, TR-04-21)

- **- Shear displacements, i.e. faulting, in the rock intersecting deposition holes may cause damage on the canister and the buffer**
- **- Copper has ductility over 40% and creep ductility of at least 10% and for the cast iron insert, ductility of 11% is required. Based on the analyses the rock displacement of 200 mm is still acceptable.**
- **- The imposed strains and stresses may, however, result in canister failure by creep rupture both in copper and the cast iron during long times**

DEFORMATION OF THE CANISTER INSERT IN THE PRESSURE TEST



Figur 5-1. Bilder som visar deformationen av insatsen och bränslekanalerna vid tryckprovet.

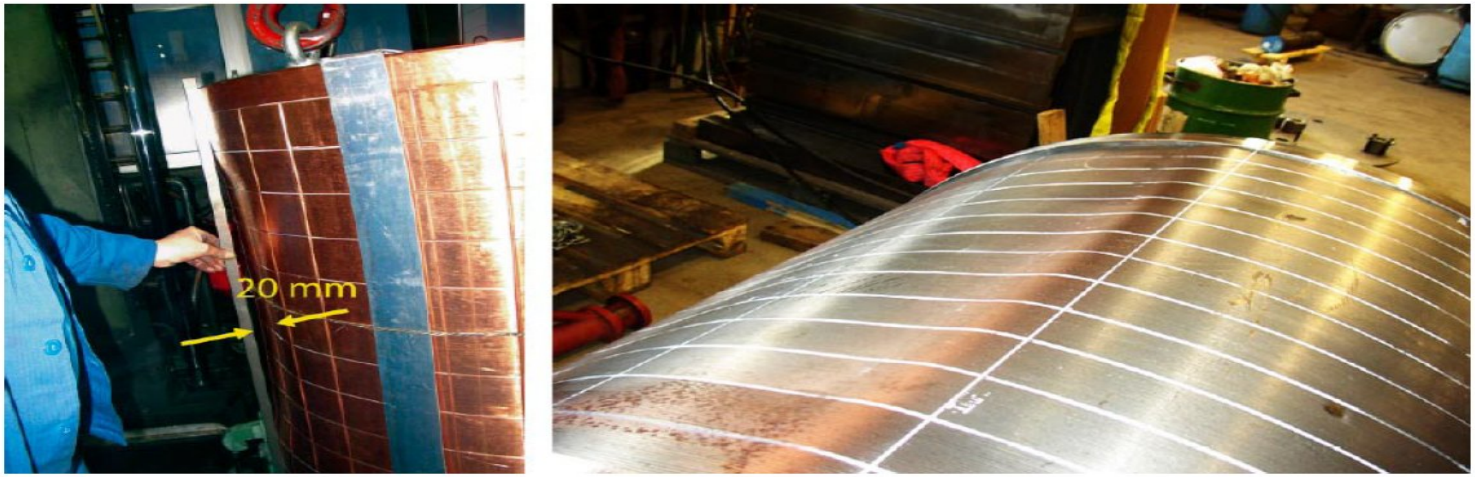


Figur 5-2. Exempel på stor diskontinuitet av typen sugning mellan kanalrören i en insats.

- **- During the ice age hydrostatic pressure loading may increase up to 45 MPa (design pressure)**
- **- The mock-up tests (950 mm long) up to 130 MPa showed only deformations, but the importance of defect-free and well-centralized insert was realized. Copper was tight**
- **- Canister collapsed at the pressure of 139 MPa**
- **- Creep behavior of coppar and cast iron insert material has also to be known because of long timescales**

(SKB, R-06-01)

DEFORMATION OF THE CANISTER INSERT IN THE PRESSURE TEST



Figur 2-4a och b. *Deformationen av provkapseln efter belastning till 130 MPa.*

Figur 2-5a och b. *Provkapsel nr 2 efter kollaps vid tryckbelastningen 139 MPa.*

(SKB, R-05-70)

HYDROCHEMICAL CONDITIONS AT OLKILUOTO SITE: INITIAL STATE

Depth (m)	Water type	Cl (mg/L)	pH	Alkalinity (meq/L)	Redox
0	Ca-Na-Mg-HCO_3-SO_4-(Cl)	<10	5.5	<0.5	Postoxic
	Na-(Ca)-Cl-HCO_3	10	7	2.5	
50 ≈	Na-Cl-(HCO_3)	400	8	6.5	Sulphidic
150 ≈	Na-Cl	1 000	7.8	4	
200 ≈	Na-(Ca)-Cl-(SO_4)	4 500	7.5	1.0	
	Na-Cl	2 700	8.2	0.4	
400 ≈	Na-Ca-Cl	8 000	8.2	0.1	Methanic
600 ≈		14 000			
	Ca-Na-Cl	45 000	7.8	<0.1	
1 000					

Figure 2-1. Vertical variation of the main hydrochemical constituents at Olkiluoto. Vertical lines represent smooth changes in groundwater.

- **- The ground water compositions show great variability, which is seen in the chemical data, notably in salinity (fresh – brackish – saline), water type, and contents of various constituents**

- **- There are five water types affecting the groundwater composition: infiltrate water from surface (carbonate-rich), sea water from Gulf of Botnia, Litorina Sea water (sulfate-rich, 2500-7500 y. old), Ancylus Lake water (fresh glacial melt water, 7500-10000 y. old) and saline ancient water (Ca-Na-Cl brine)**

(SKB, TR-01-23)

Table 2-3. Potential hydrogeochemical conditions at repository depth (400–700 m) in the Olkiluoto site. The concentrations are given both in mg/L and mol/L.

Constituent		Initial, indisturbed conditions	At closure, infiltration into unsaturated bentonite	After closure and saturation** (i.e. up to 100 years)	After closure up to 10,000 years
pH		7.8–8.3	6–8	7–8	7.5–8.3
Redox	mV	−230 to −280	Oxic to −250	−150 to −250	−200 to −280
DIC	mol/L	$(0.7–0.1)\cdot10^{-3}$	$(0.1–10)\cdot10^{-3}$	$(0.5–10)\cdot10^{-3}$	$(0.1–7)\cdot10^{-3}$
Cl^-	mg/L mol/L	7,500–15,000 $(2.1–4.2)\cdot10^{-1}$	1,000–22,000 $(0.3–6.2)\cdot10^{-1}$	1,000–5,000 $(0.3–1.4)\cdot10^{-1}$	200–15,000 $(0.06–4.2)\cdot10^{-1}$
Na^+	mg/L mol/L	2,500–5,000 $(1.1–2.2)\cdot10^{-1}$	500–6,500 $(0.2–2.8)\cdot10^{-1}$	500–2,000 $(0.2–0.9)\cdot10^{-1}$	300–5,000 $(0.1–2.2)\cdot10^{-1}$
Ca^{2+}	mg/L mol/L	1,500–4,000 $(0.4–1.0)\cdot10^{-1}$	100–6,000 $(0.03–1.5)\cdot10^{-1}$	100–1,000 $(0.03–0.2)\cdot10^{-1}$	20–4,000 $(0.005–1.0)\cdot10^{-1}$
Mg^{2+}	mg/L mol/L	30–70 $(1.2–2.9)\cdot10^{-3}$	10–250 $(0.04–1.0)\cdot10^{-2}$	10–250 $(0.04–1.0)\cdot10^{-2}$	10–250 $(0.04–1.0)\cdot10^{-2}$
K^+	mg/L mol/L	10–20 $(2.6–5.1)\cdot10^{-4}$	5–30 $(1.3–7.7)\cdot10^{-4}$	5–30 $(1.3–7.7)\cdot10^{-4}$	5–20 $(1.3–5.1)\cdot10^{-4}$
SO_4^{2-}	mg/L mol/L	0–20 $0–0.2\cdot10^{-3}$	0–500 $0–5.2\cdot10^{-3}$	10–500 $(0.1–5.2)\cdot10^{-3}$	0–500 $0–5.2\cdot10^{-3}$
HS^-	mg/L mol/L	0–3 $0.9\cdot10^{-4}–0$	0–3 $0–0.9\cdot10^{-4}$	0–10 $0–3.0\cdot10^{-4}$	0–3 $0–0.9\cdot10^{-4}$
NH_4^+	mg/L mol/L	<0.05 $<0.03\cdot10^{-4}$	<0.1, but if marine <3* $<5.5\cdot10^{-6}$, $<1.7\cdot10^{-4}$	0.05–3 $(0.03–1.7)\cdot10^{-4}$	<1 $<0.6\cdot10^{-4}$
$CH_4(g)$	ml/L mol/L	50–400 $(0.2–1.8)\cdot10^{-2}$	<0.1, but if saline <600 $<4.5\cdot10^{-6}$, $<2.7\cdot10^{-2}$	<100 $0.4\cdot10^{-2}$	0.1–400 $(0.004–17.9)\cdot10^{-3}$
$H_2(g)$	ml/L mol/L	<0.5 $<2.2\cdot10^{-5}$	<0.5, but if saline <20 $<2.2\cdot10^{-5}$, $<8.9\cdot10^{-4}$	<0.1 $<4.4\cdot10^{-6}$	<0.5 $<2.2\cdot10^{-5}$
DOC#	mgC/L mol/L	<2 $<1.7\cdot10^{-4}$	<2 $<1.7\cdot10^{-4}$	<10 $<8.3\cdot10^{-4}$	<2 $<1.7\cdot10^{-4}$
Microbes		SRB, methanogens	Aerobic bacteria, SRB, IRB methanogens	SRB, IRB,	IRB, SRB, methanogens

*) Based on Hästholmen results.
**) Probably marine water will dominate.
#) Most of current DOC data is unreliable. Mostly, samples with high pump rate have only a few mg/L.

Table 2-7. Estimated constituent ranges at different times in bentonite pore-water in the case of Olkiluoto.

Constituent		Infiltrating groundwater at closure	Pore-water in saturated bentonite (up to100 years)	Por-water after closure up to 10,000 years
pH		6–8	7–9	7–9
Redox	mV	Oxic to −250	−150 to −250	−200 to −280
DIC	mol/L	$(0.02–1.6)\cdot10^{-4}$	no estimate	no estimate
Cl^-	mol/L	$(0.3–6.2)\cdot10^{-1}$	$(0.3–6.2)\cdot10^{-1}$	$(0.06–4.2)\cdot10^{-1}$
Na^+	mol/L	$(0.2–2.8)\cdot10^{-1}$	$(3\text{-}5)\cdot10^{-1}$	$(3–4)\cdot10^{-1}$
Ca^{2+}	mol/L	$(0.3–1.5)\cdot10^{-1}$	$(0.4–4.0)\cdot10^{-2}$	$(0.4–4.0)\cdot10^{-2}$
SO_4^{2-}	mol/L	$(0–5.2)\cdot10^{-3}$	$1.4\cdot10^{-1}$	$1.4\cdot10^{-1}$
HS^-	mol/L	$(0–0.9)\cdot10^{-4}$	$(0–3)\cdot10^{-4}$	$(0–0.9)\cdot10^{-4}$
NH_4^+	mol/L	$<5.5\cdot10^{-6}$ $<1.7\cdot10^{-4}$*	$(0.03–1.7)\cdot10^{-4}$	$<5.5\cdot10^{-5}$
$CH_4(g)$	mL/L	$<4.5\cdot10^{-6}$ $<2.7\cdot10^{-2}$**	$<4.5\cdot10^{-3}$	$(0.004–17.9)\cdot10^{-3}$

*) constituent value in the case of marine water
**) constituent value in the case of saline water

PREDICTED TIME DEPENDENCE OF OXYGEN FLUX TO THE CANISTER

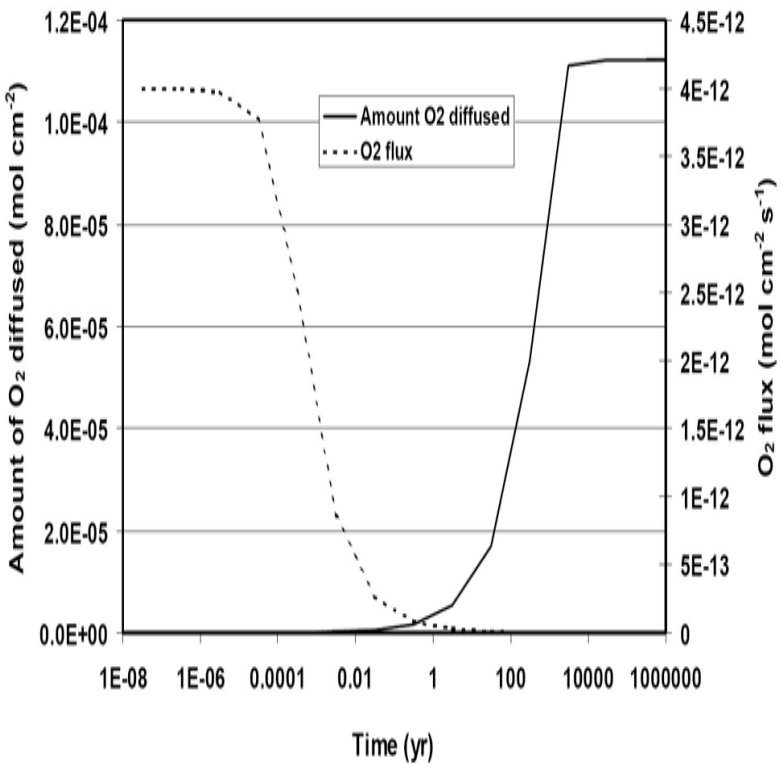


Figure 4-2. Predicted time dependence of the flux of O_2 to the canister surface and of the time-integrated amount of O_2 that has diffused (see text for details).

- **Copper dissolution is supported by cathodic reduction of oxygen on the external surface of the canister and eventual pit or crack**
- **- Oxygen flux to canister wall is time dependent. All oxygen is in the borehole either dissolved or gaseous**
- **- Based on the analysis 99% of the oxygen is consumed within first 3000 years, which can be used as the definition for the ultimate length of the aerobic phase (oxygen is mainly consumed in 200 y.)**

(SKB, TR-04-05)

ASSESSMENT OF CORROSION BEHAVIOR OF COPPER CANISTERS

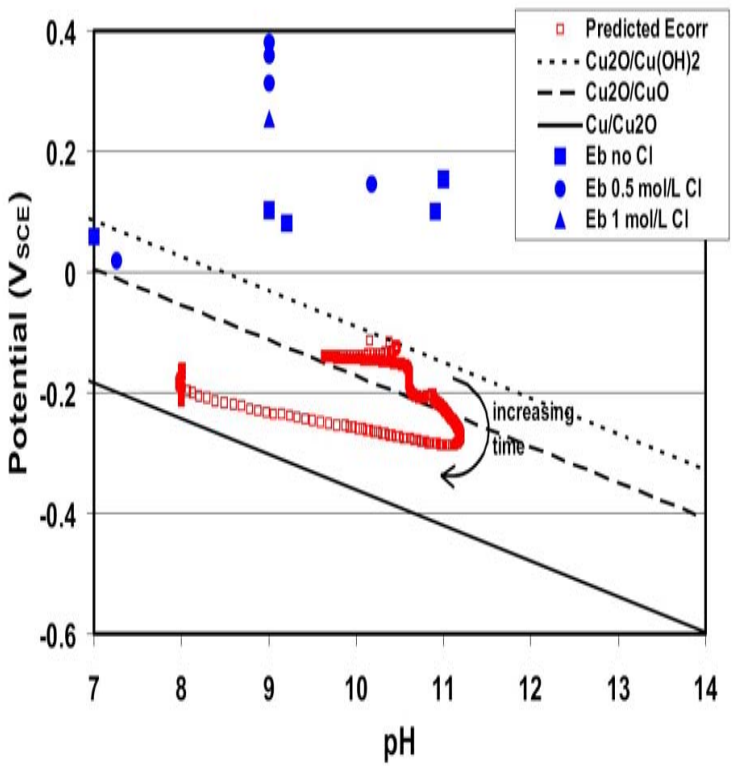


Figure 4. Predicted evolution of the corrosion potential with time. The comparison with the pitting breakdown potential is made by assuming that it coincides with the Cu_2O/CuO equilibrium potential line. The corrosion potential is initially above the Cu_2O/CuO equilibrium potential line but stabilises well below for subsequent stages (source: SKB workshop presentation).

- **Various corrosion mechanisms of copper such as general, localized, stress corrosion cracking and microbial corrosion depend on local electrochemical conditions in the repository with time**
- **Average rate of corrosion to penetrate 50 mm thick copper in 1 m. y. is 50 nm/y (5 nA/cm^2), which is too slow to be measured. Thus, the corrosion mechanisms have to be understood for proper modelling**
- **Oxygen causes oxidation and affects the corrosion potential, chlorides form strong complexes and facilitate mass transfer, sulphides cause general corrosion and nitrogen compounds may cause stress corrosion cracking**
- **Microbial activity may increase sulphides and nitrogen compounds, but also reduce the oxygen content in the repository**

(SKi Report 2006: 11)

EXCLUSION PRINCIPLE OF STRESS CORROSION CRACKING OF COPPER FOR NITRITE AND AMMONIA

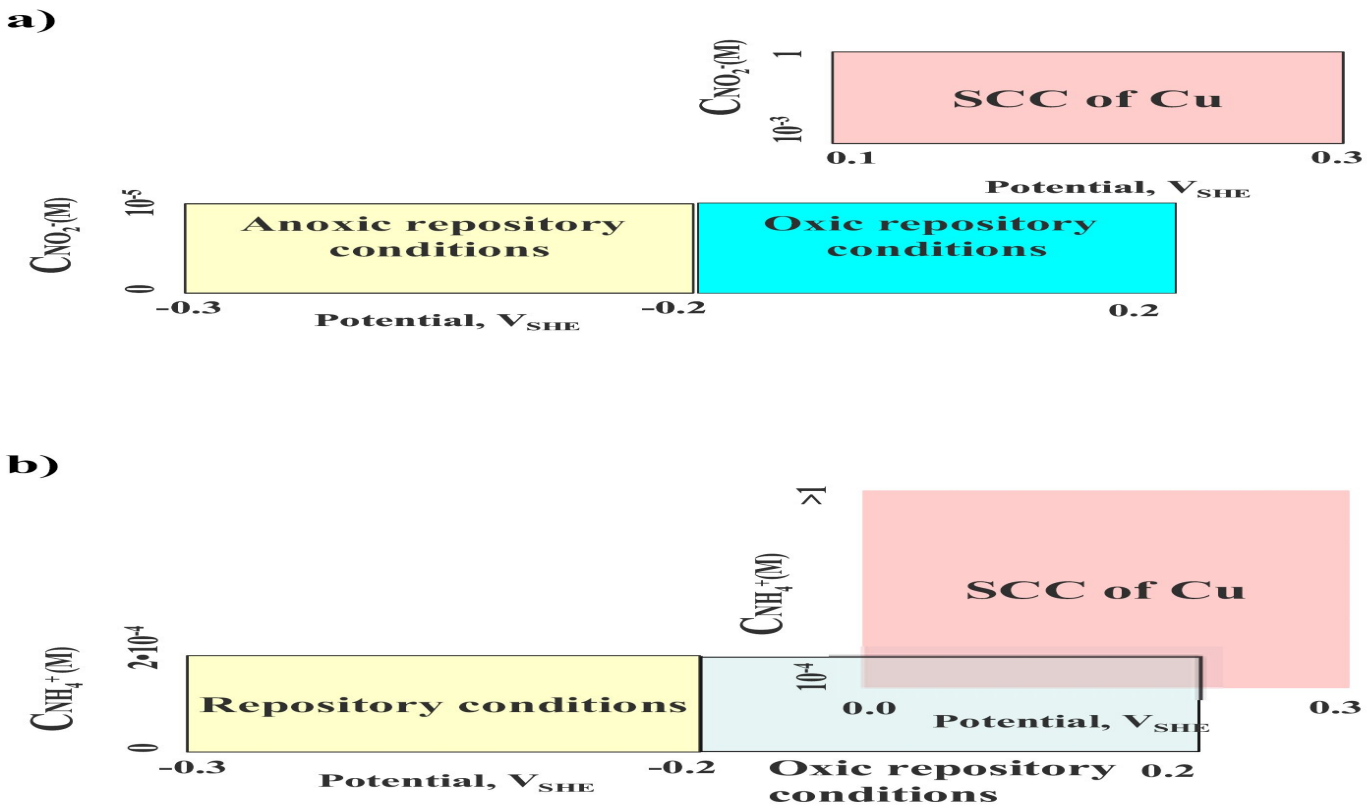


Figure 5. Assessment of likelihood of stress corrosion cracking (SCC) for a) nitrite and b) ammonia by application of the exclusion principle (Saario, see Abstract C). These figures show the concentration of SCC agent (Y-axis) plotted against potential (X-axis). In figure b) there is an overlap between the areas of possible SCC and the repository conditions, which suggest that SCC with ammonia needs to be further studied.

(SKi Report 2006: 11)

Table 1. Threats to be considered for the evaluation of canister corrosion.

Threat	Time frame in EBS evolution of main relevance	Description
Oxygen	Initial phase and possibly glacial	Rapid reaction of oxygen with copper with formation of copper oxides
Chloride	Throughout	Complexation with copper ions
Sulphide	Throughout	Reaction with copper by reduction of water and formation of copper sulphides
Nitrogen compounds	Initial phase	Ammonia and nitrite are along with acetate most important agents facilitating stress corrosion cracking Nitric acid is an aggressive reactant formed by radiolysis in the interior of the canister
Microbes	Throughout	Microbes catalyse otherwise relatively slow inorganic processes, such as consumption of oxygen and reduction of sulphate
Surface salt	Resaturation phase	Evaporation of water on canister surface leading to accumulation of salt deposits in micro-environments on the canister surface
Sulphide whiskers	Reducing phase	Rapid formation of whiskers on copper surfaces during pitting corrosion
General corrosion	Oxygenated phase (by O_2) Reducing phase (by HS^-)	Uniform thinning of the canister wall
Pitting corrosion	Initial and resaturation phases	Localised attack with persistent pits under oxidizing conditions. Also general rough corrosion caused by unstable pit formation. Uneven swelling of bentonite might lead to under-deposit corrosion.
Stress corrosion cracking	Initial phase and possibly glacial	Cracking at tensile parts of canister wall
Radiolysis and radiation influenced corrosion	Initial phase	Creation of locally oxidizing conditions due to gamma radiation field
Galvanic coupling	Early inside canister and after canister failure	Different potentials for iron and copper in presence of water lead to corrosion. Earth currents leading to external corrosion
Environmentally controlled creep	Throughout	Corrosion might influence creep and result in creep failure

(SKi Report 2006: 11)

MODEL FOR GENERAL CORROSION OF COPPER IN OXIC SALINE GROUNDWATER

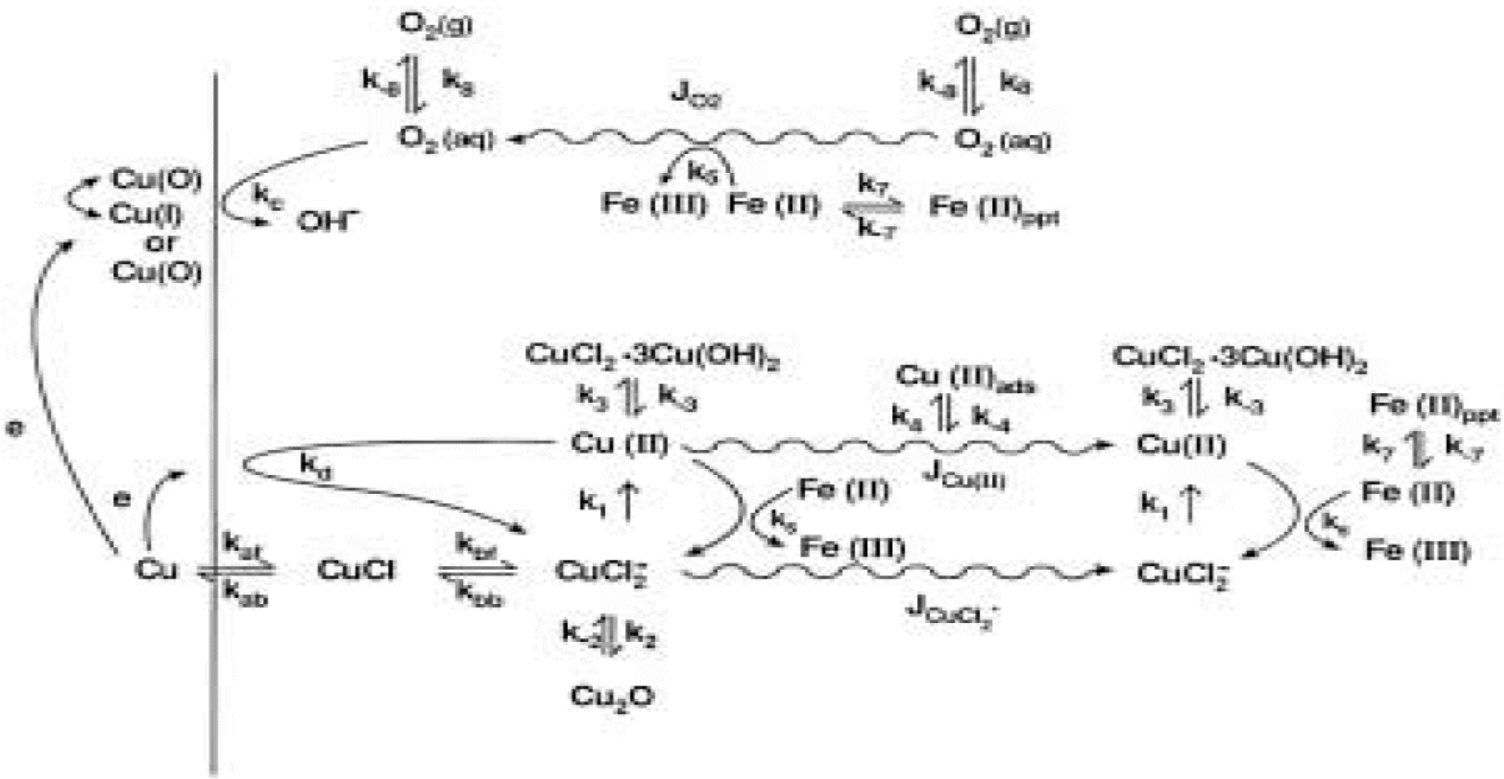


Figure 6-14. Mechanism for the general corrosion of copper in compacted buffer material with O_2-containing saline groundwater /King 1996b/.

(SKB, TR-01-23)

MICROBIAL ACTIVITIES IN REDUCING THE AMOUNT OF OXYGEN IN THE REPOSITORY IN DIFFERENT CLIM ATIC CONDITIONS

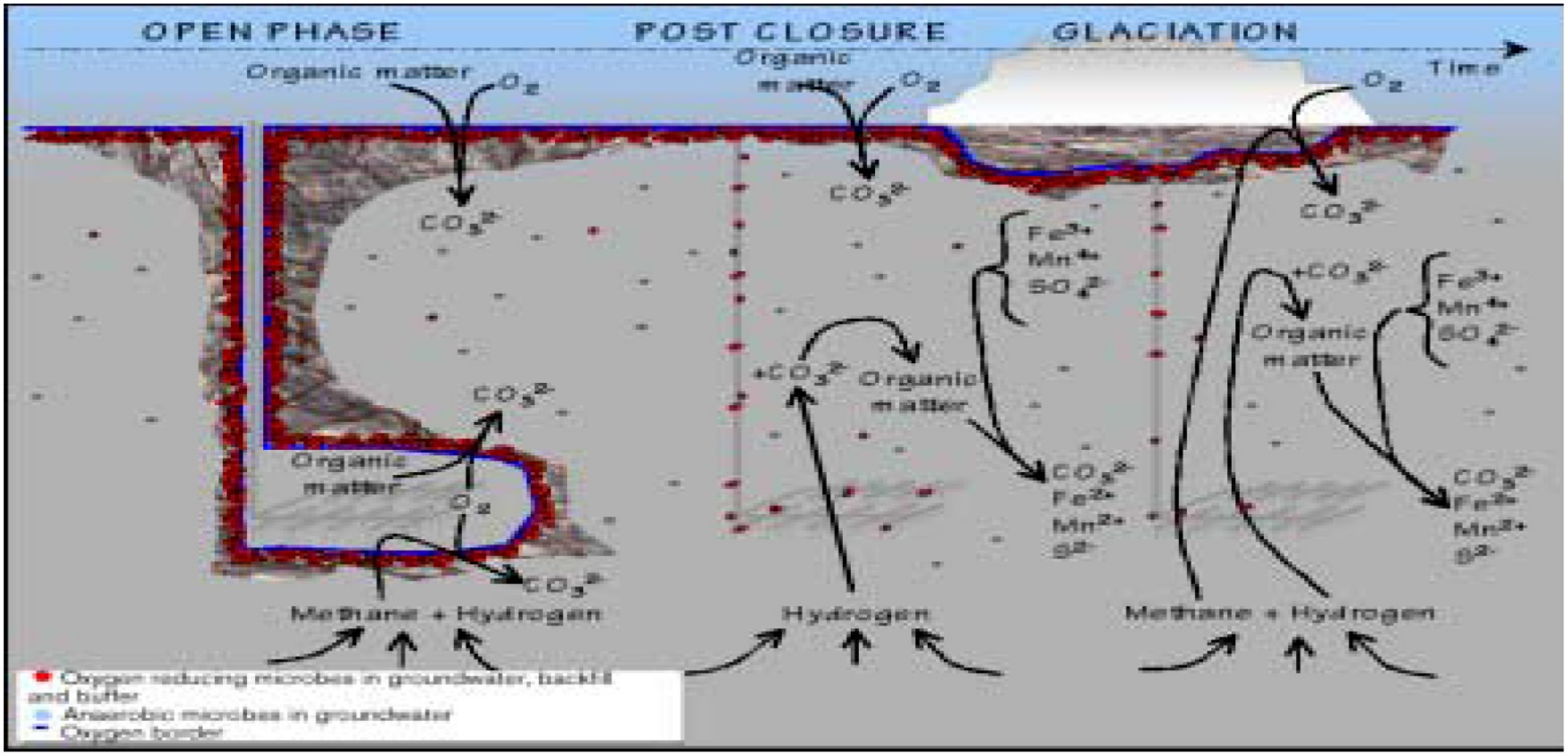


Figure 2-7. *A schematic model of how microbes in the geosphere would stop oxygen from reaching a HLW repository and keep the groundwater redox potential at low levels. See text for an explanation.*

(SKB, TR-01-23)

Table 9-1. Comparison of predictions of long-term corrosion behaviour and canister lifetimes.

Country	General Corrosion	Localized Corrosion	Microbially Influenced Corrosion	Stress Corrosion Cracking	Predicted Lifetime	Reference
Sweden/ Finland[1)]	0.05 mm in 10^6 yrs (realistic) 4 mm in 10^6 yrs (conservative)	0.05 mm in 10^6 yrs (realistic) 18 mm in 10^6 yrs (conservative)	–	–	$>10^6$ yrs	Wersin et al. /1994/
Sweden/ Finland[1)]	0.35 mm in 10^6 yrs	0.35 mm in 10^6 yrs (realistic) 1.4 mm in 10^6 yrs (conservative)	SRB assumed to reduce SO_4^{2-} to HS^-	Maximum possible nitrite concentration below threshold for SCC	$>10^6$ yrs	Werme et al. /1992/, Swedish Corrosion Institute /1983/, SKB /1983/
Sweden/ Finland[1)]	0.33 mm in 10^6 yrs	0.33 mm in 10^6 yrs (realistic) 1.3 mm in 10^6 yrs (conservative)	SRB assumed to reduce SO_4^{2-} to HS^- in tunnel and groundwater only	SCC does not occur based on threshold potential and concentrations of SCC agent, and because creep is faster than SCC	$>10^6$ yrs	This report
Canada[2)]	0.011 mm in 10^6 yrs	6 mm in 10^6 yrs	Limited impact. Maximum additional wall loss of 1 mm in 10^6 yrs	SCC not included because of limited period of stress, absence of SCC agents, general lack of oxidant and inhibitive effects of Cl^-	$>10^6$ yrs	Johnson et al. /1996/
Japan	9–13 mm in 10^3 yrs, depending on repository design	18–26 mm in 10^3 yrs based on pitting factor of 3, 2 mm in 10^3 yrs based on extreme-value analysis	SRB assumed to reduce all SO_4^{2-} to HS^-	Maximum concentrations of ammonia, nitrite and acetate less than threshold concentration	None given	JNC /2000/

[1)] Reference canister wall thickness of 50 mm.
[2)] Reference canister wall thickness of 25 mm.

(SKB, TR-01-23)

CONCLUSIONS

- **Next ice age starts 60000 y. from now if the ecosystem processes are expected to be similar to the present. (The present human-induced fossile fuel period may be able to change the astronomically-dependent cyclic climate development.)**

- **After 60000 y. the radiotoxicity of spent nuclear fuel is almost comparable to that of the natural uranium ore once used to produce the fuel and the canister surface temperature has decreased to ambient.**

- **The repository sites will be because of slow land rising (less than 100 m in next 10000 y.) and shoreline movement in land far away from shores and not covered by brackish or sea water.**

- **During the coming 60000 y. period the present knowledge and models are able to predict copper canister creep and corrosion and the other processes in the repository site.**

- **When the repository site is covered with thick retreating and advancing ice sheet, very slowly changing cyclic hydrostratic loading affects the repository site and the waste packages. The corrosion conditions may change because of infiltration of deep saline groundwater and oxygen-containing glacial meltwater. Changes in rock stresses and movements may cause shear displacements, i.e. faulting, in the rock intersecting deposition holes and cause damage on the canisters and the buffer.**

- **The coming ice age(s) result in loading and environmental conditions in the repository, which are still vague and much more difficult to predict than those during the present interglacial period.**

Shielding Integrity Testing of Radioactive Material Transport Packaging

Neil Carr – Technical Secretary for TCSC / Packaging Assessment Manager at Nirex

21st November 2006

nirex

TCSC - ToR

“ to examine the requirements for the safe transport of radioactive material with a view to standardisation and, as appropriate, produce and maintain guidance in the form of Standards documentation ”

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TCSC 1056

- Shielding Integrity Testing of Radioactive Material Transport Packagings – December 2005
- combines content of safety standards
  - AESS 6067 Application/testing
  - AECP 1056 Methodology/approach
- updated to include guidance on
  - ultrasonics
  - depleted uranium

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Scope

- Specification for designers
- Instructions to manufactures
- Procedures for testing
- Suitable for testing of other gamma shielding
  - re-fuelling machines
  - shielded facilities – stores and hot cells

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Transport Requirements

- Regulatory – Before first shipment [501(b)],
- verification of design following manufacture
- source for test need not be material intended for transport
- attention paid to increased levels in cracks and gaps – lids, vents & drains
- consideration of mechanical and thermal stresses – under NCT

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Guidance on Methods

- Gamma radiography - source and detector
- X-radiography – source and film
- Ultrasonics - probe
- Conventional weight and dimensional checks

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Personnel

- Gamma and X-ray
  - ionising regulations, classified person, controlled area and approved system of work
- Ultrasonics
  - qualified to EN473 or equivalent
- Physical measurement
  - suitably qualified and experienced personnel

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Equipment

- Gamma and X-ray
  - sealed source should suitably housed
  - calibrated radiation monitor
  - use of mechanical linkage and elimination of scatter
- Ultrasonics
  - detection to BS EN 12668-1
  - probes to BS EN 12668-2
  - use of calibration and test blocks
- Physical measurement
  - appropriate scale to size of packaging

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Testing with Gamma Sources

- Radiological scanning
  - defined system of work
  - selection of source
  - scanning grid pattern
  - consideration of scatter
- Testing
  - derive scanning rates
  - minimise operator dose

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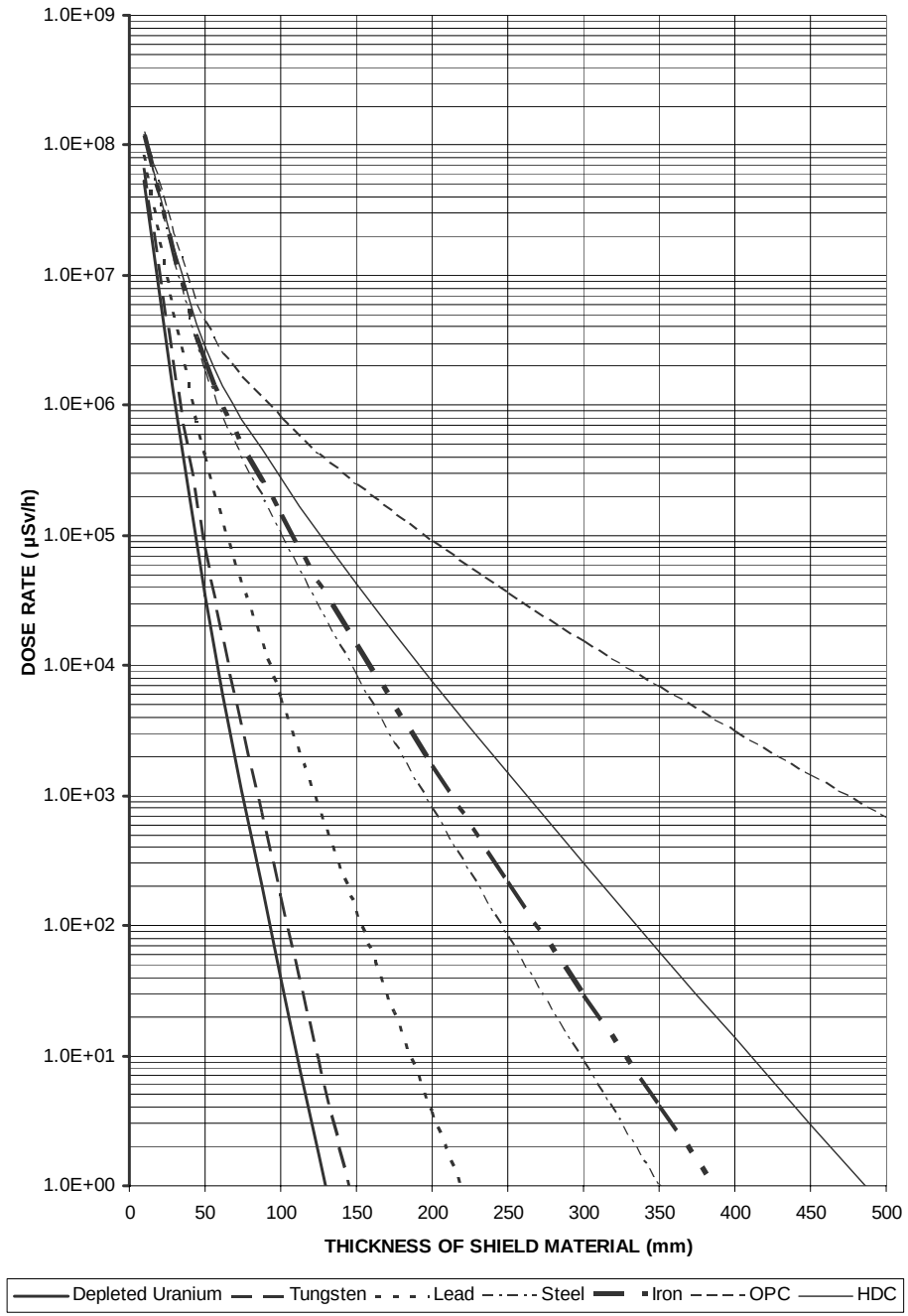
37 GBq
Colbalt-60

Source



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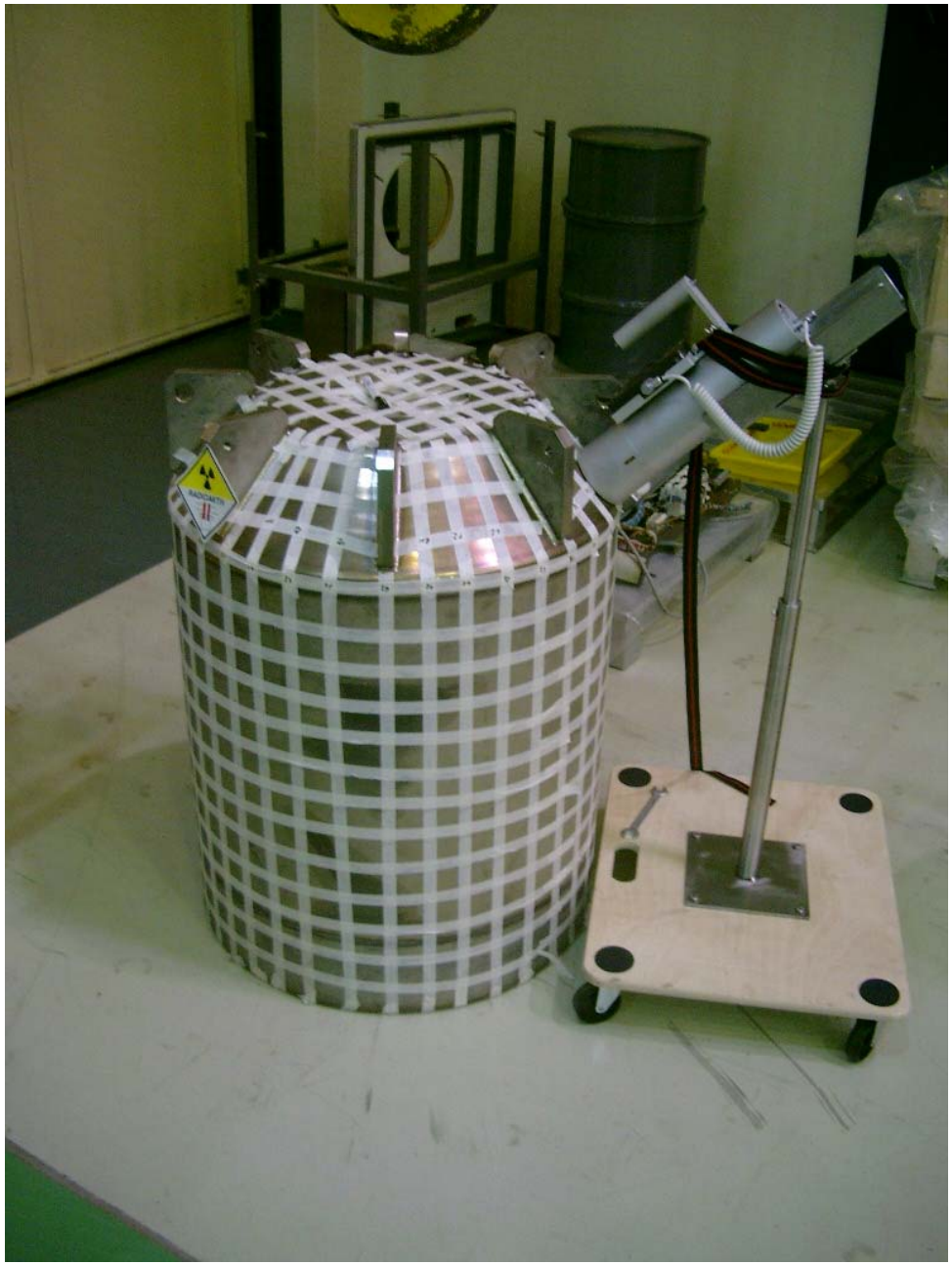

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Types of Flaw

- Shrinkage cavities
- Air locks
- Gaps between successive pours
- Blowholes
- Porosity
- Inclusions

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Reporting

- criteria to agreed with design authority
- recommend criteria for <10% increase
- flaws that result 20% increase, require re-design and or repair
- report to include
  - contract details, date, location, method, equipment and personnel, findings and appropriate authorisation

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UK Reference HLW/SF Concept
Nirex Report N/125

- Development of reference HLW/Spent Fuel Concept
- Geological disposal based KBS-3 developed by SKB
- Co-operation with Swedish and Finnish programmes

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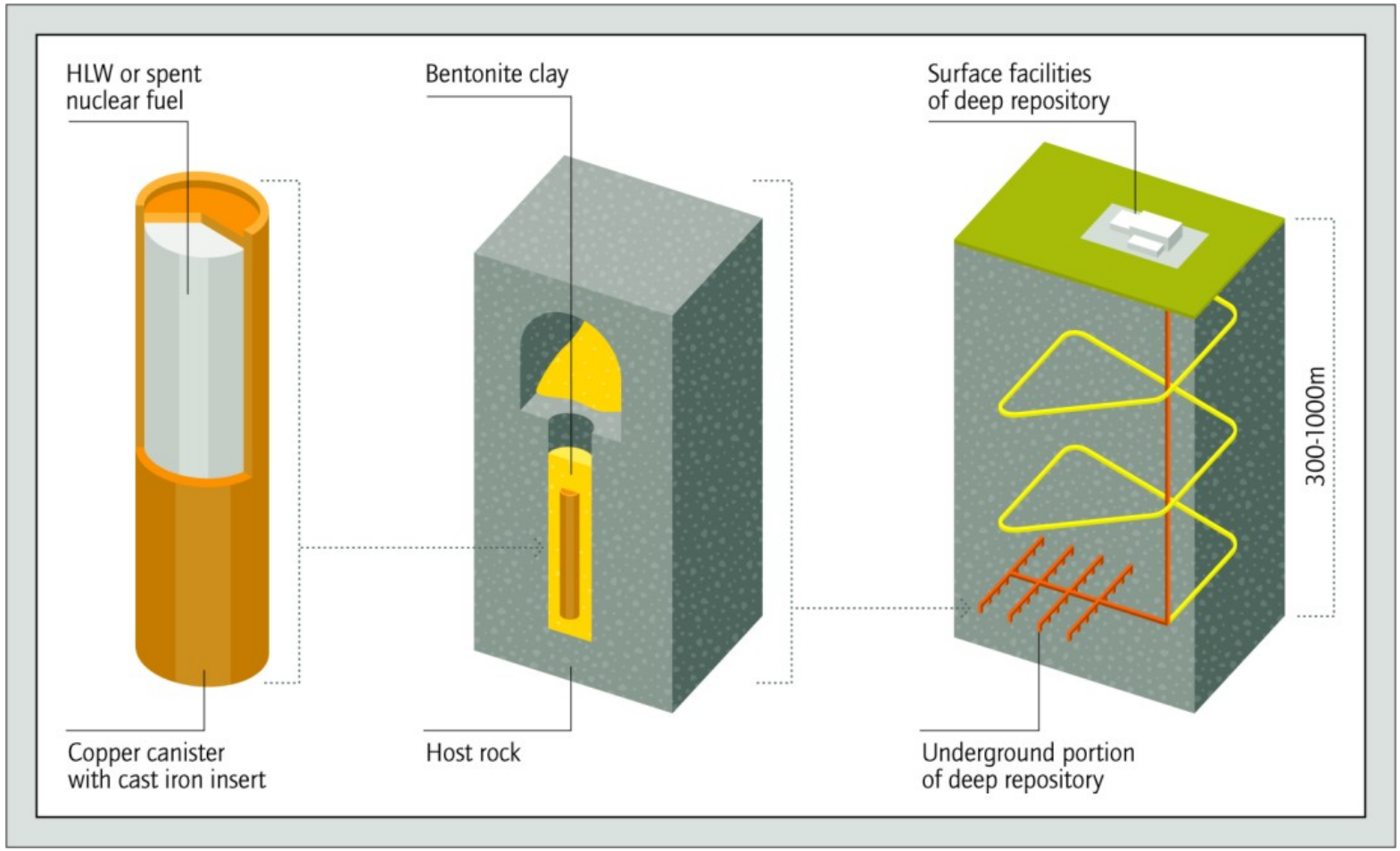
HLW/SF Concept



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HLW/SF Concept

- Aim to demonstrate viability
- Adopted KBS-3V on the basis of:
  - Credibility
  - Information base – extensive R&D programme
  - High-level of regulatory scrutiny and international review
  - Stakeholder buy in

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Current Status

- Identified KBS 3 as a viable concept for HLW/spent fuel
  - information provided to CoRWM
- Stakeholder and regulator feedback reinforced the need to:
  - analyse reference concepts against alternatives
  - justify why one concept is preferred over another
  - be flexible to respond to new information

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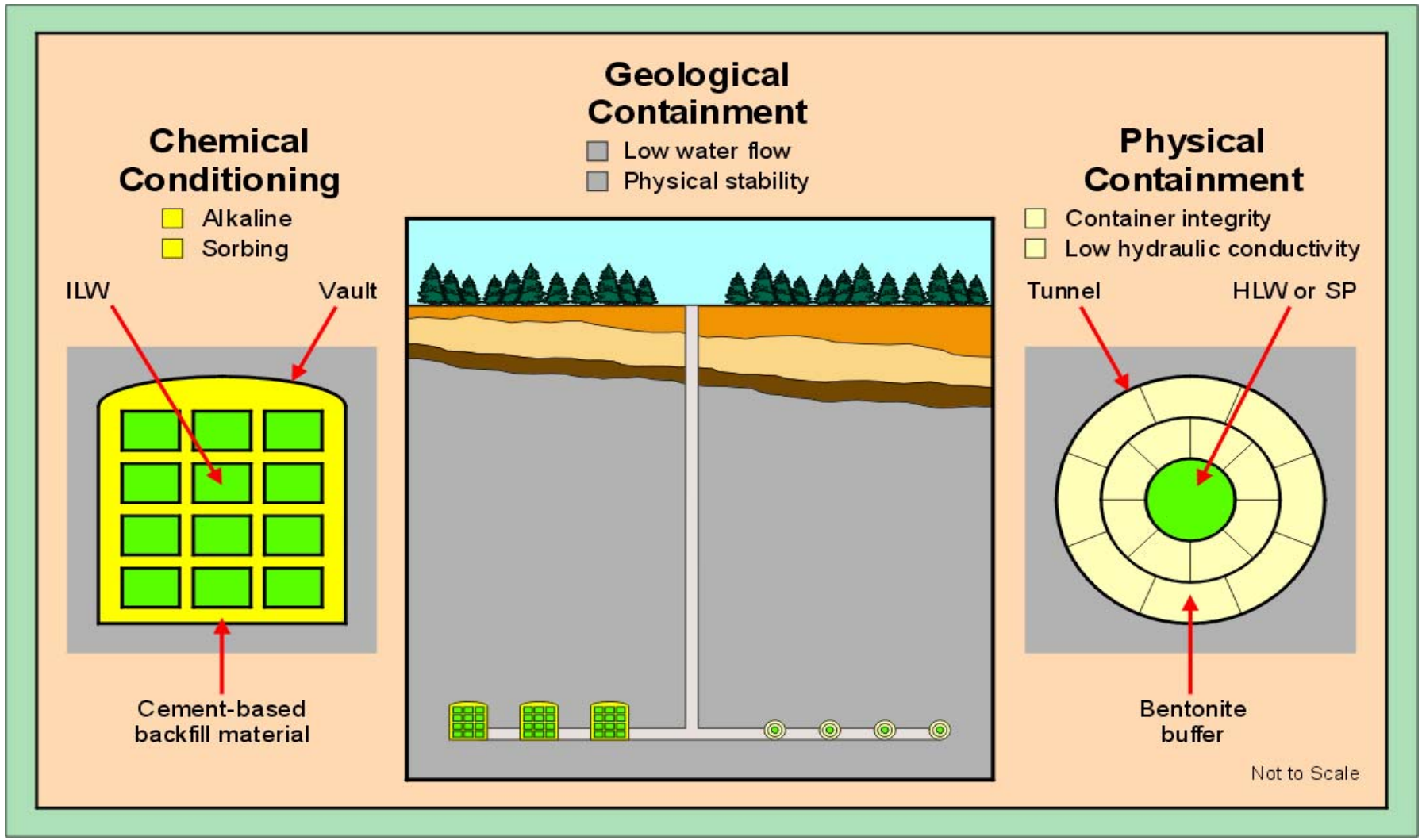
Co-location Concept



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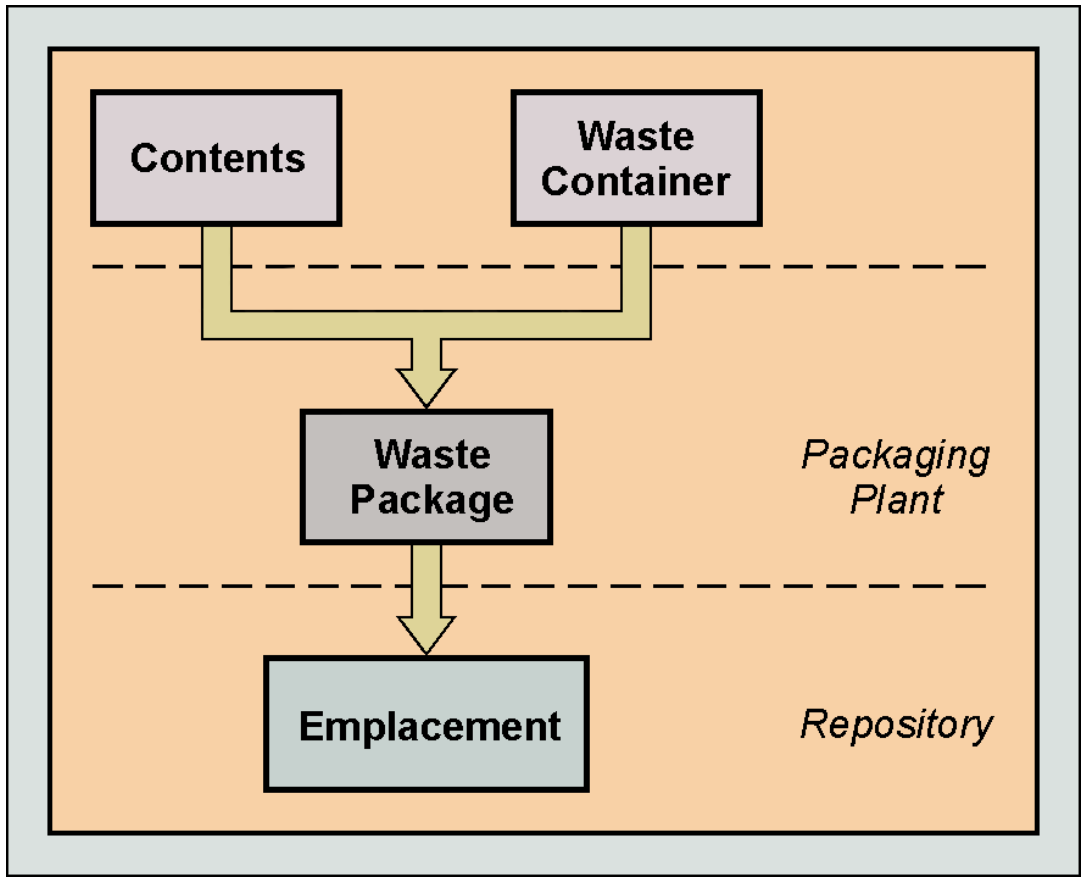
Packaging process for HLW/SF



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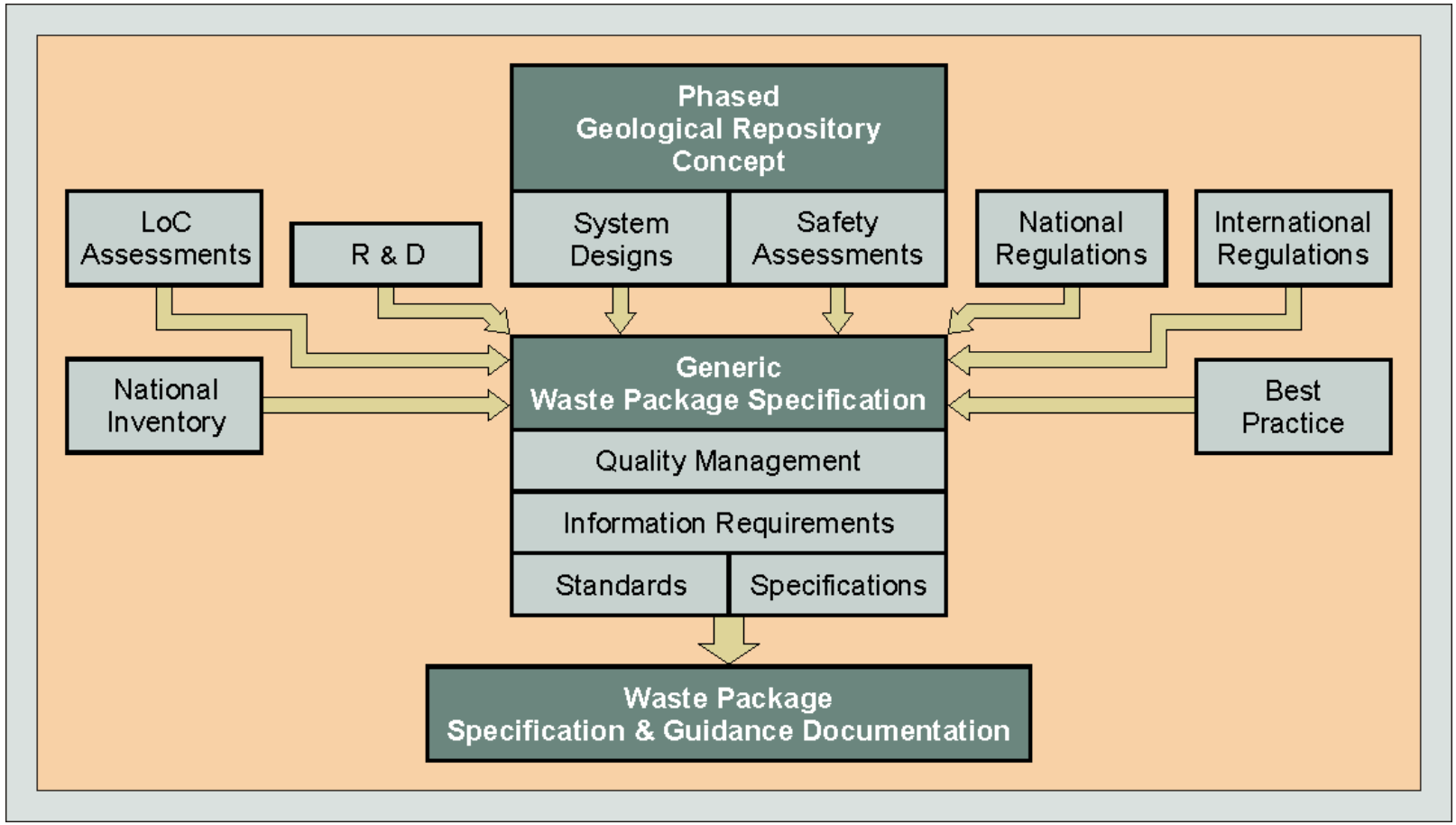
Phased
Geological Repository
Concept
LoC
Assessments
R & D
System
Designs
Safety
Assessments
National
Regulations
International
Regulations
National
Inventory
Generic
Waste Package Specification
Best
Practice
Quality Management
Information Requirements
Standards
Specifications
Waste Package
Specification & Guidance Documentation

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Nirex Report N/124

Defines standards and specifications which:

- Take account of the different phases of waste management, as defined by UK Reference HLW/SF Concept
- Acknowledge the different types of waste to be dealt with
- Provide a starting point for dialogue with waste ‘owners’
- Allow scrutiny of Nirex WPS by stakeholders

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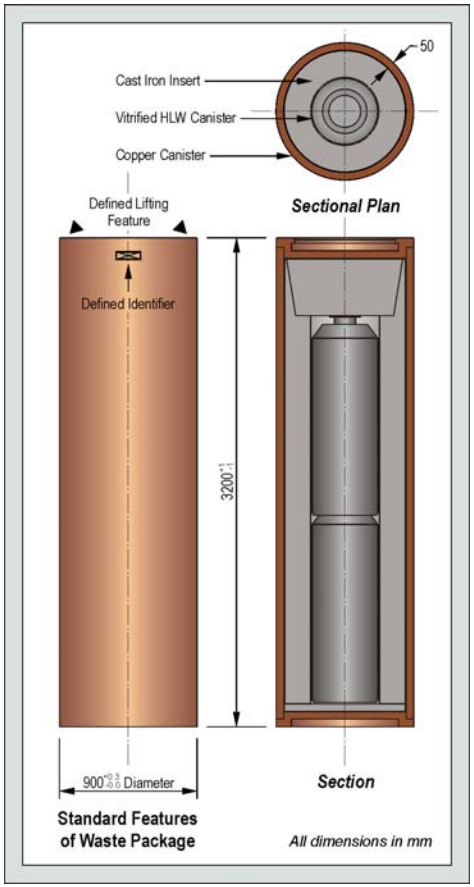
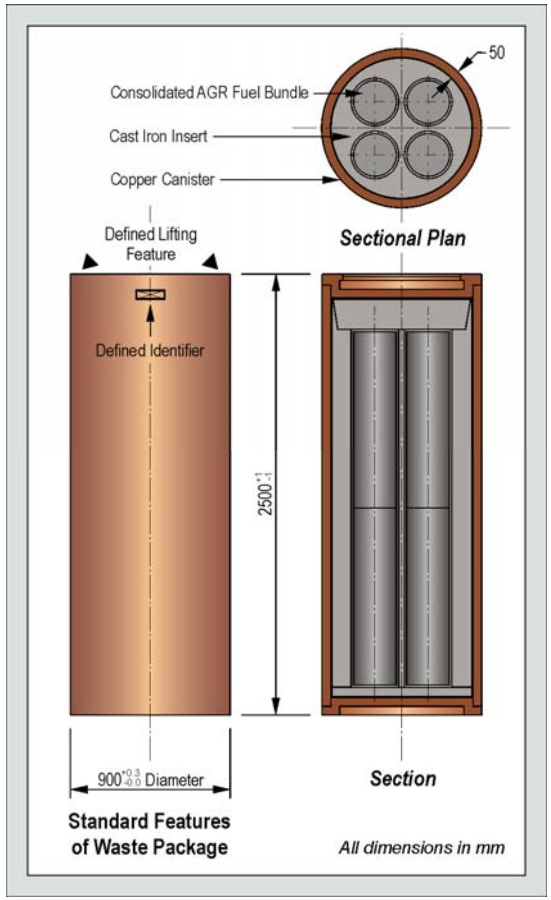
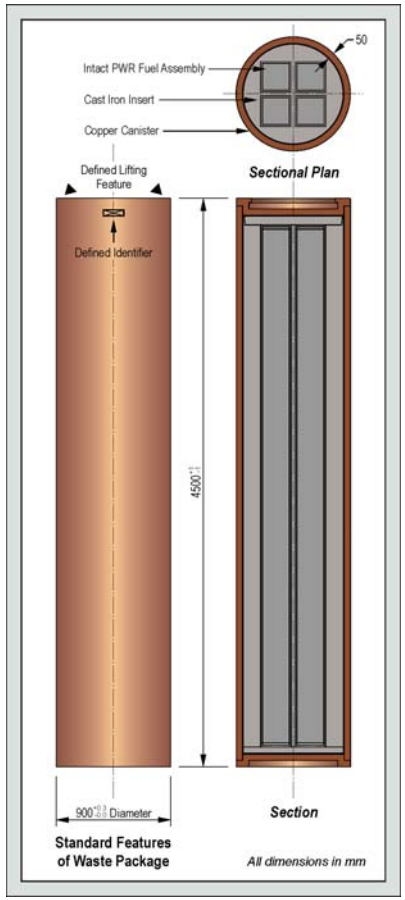
Standard Waste Packages

(Not to Scale)







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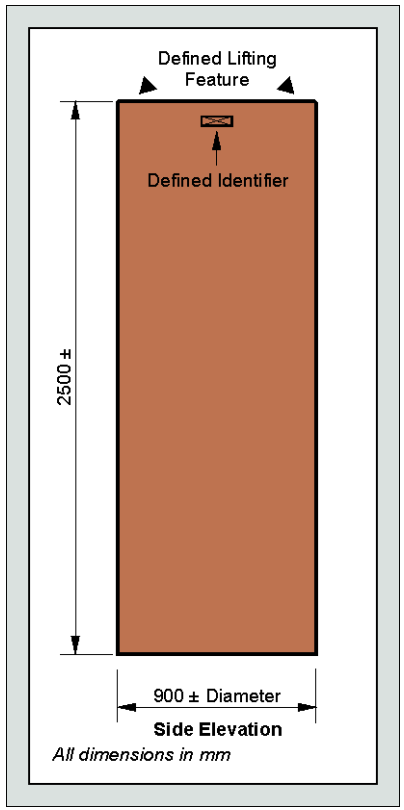
Standards



- Dimensional Envelope
- Maximum Gross Mass
- Lifting Feature
- Identifier format and location
- Containment System

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Specifications

- activity content
- dose rate
- heat output
- surface contamination
- integrity
- gas generation
- criticality safety
- impact performance
- fire performance
- nuclear security
- safeguards

- Quality Management System
- Data Recording Requirements

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Purpose of Waste Package Specifications

- Define standards for waste packages to ensure passive safety and eventual disposability
- Promote standardisation
- Provide a basis for waste package assessment and allows plans to be made for packaging of wastes
- Set out quality management and data recording requirements
- Preliminary WAC

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Further work (1)

- Further develop safety and environmental assessments and underpinning R&D
- Evaluate repository concept alternatives
- Discussing with UK regulators how the process of regulatory scrutiny for ILW can be extended to cover HLW/SF concept

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Further work (2)

- Address key challenges
  - establish views on the degree of retrievability and examine technical approaches to retrievability
  - optimising configuration of underground openings
  - investigate combined repositories (co-location)
  - identify siting requirements
  - investigate options for other radioactive materials

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NDT of Structures for Nuclear Waste Storage, Transportation and Disposal

2nd Workshop

21-22 November 2006

Inspection Challenges

- Large areas to inspect
  - Outside for atmospheric corrosion
  - Inside if the product may cause corrosion
- Detection of small pits or SCC cracks
- Awkward geometries
  - some flat areas but also edges, corners, raised sections etc
- Stainless steel welds - significant grain structure and magnetic changes
- Temperature
- May only be possible by remote access - large number of containers to inspect

Recent and Current Studies

- Review and evaluation of NDT techniques for detecting corrosion in general
- Evaluation of techniques for detection of stress corrosion cracking
- Inspection statistics for choice of inspection interval / sampling strategies
- Case Study: difficult double wall inspection
- Weld inspection by phased array ultrasonics
- Robotic, automated & high temperature NDT

Inspection for Corrosion - Overview

- TWI has data on the performance of over 10 different methods for inspecting corrosion from:
  - the European Project RACH (Reliability Assessment of Containers of Hazardous materials)
  - Follow-up Group Sponsored Project CRIS (Corrosion Reliability Inspection Scheduling)

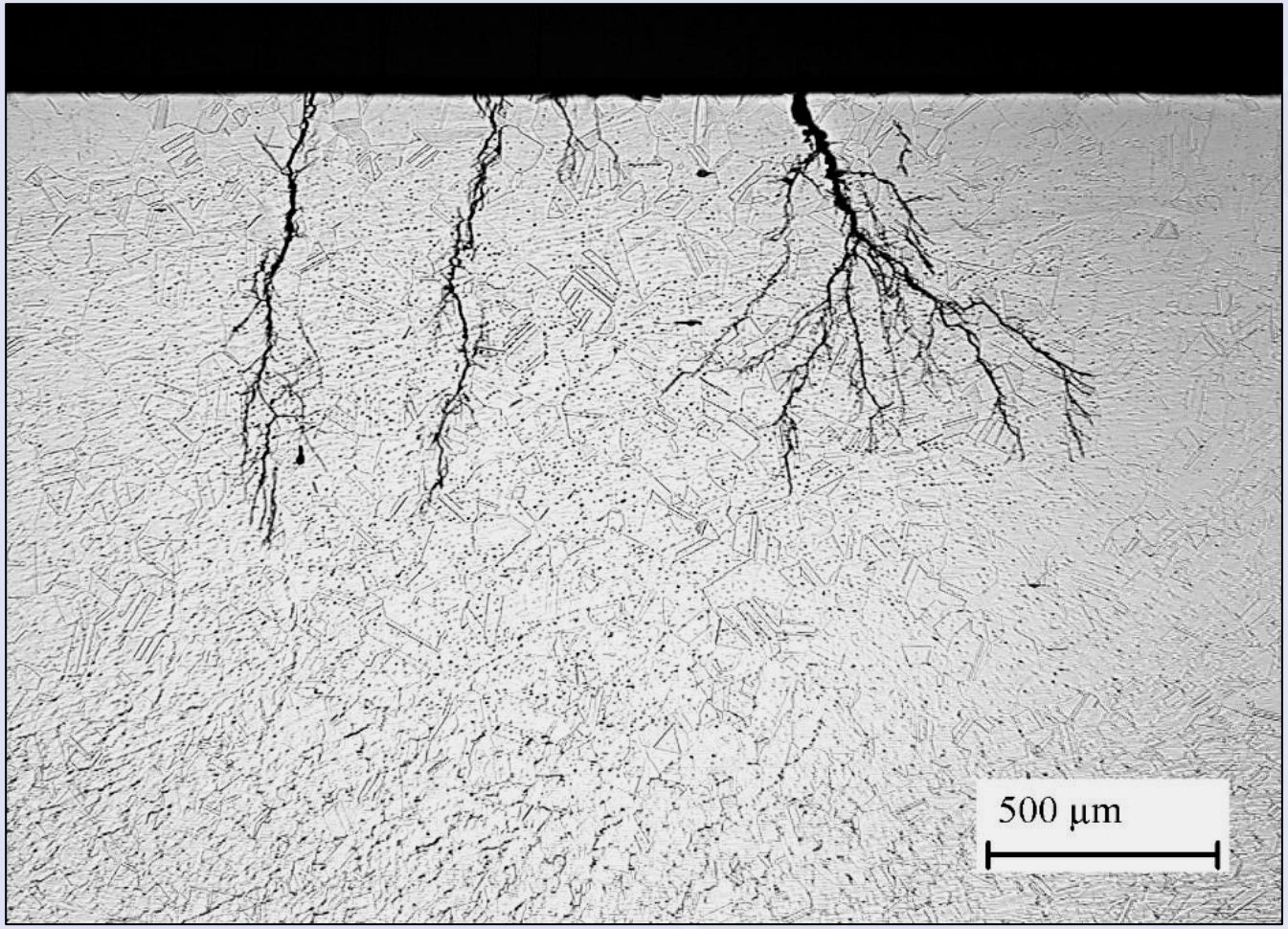
Inspection for Corrosion

- Ultrasonics – 5 variants of basic technique
- X/gamma Ray (methods for under insulation inspection)
- Magnetic/Electromagnetic – 4 variants
- Others – Thermal, strain, laser shearography

Example - Detection of SCC

- Detection of chloride induced SCC in Duplex stainless steel at temperatures around 150°C
- Detection of SCC in 316 type stainless steel at around 60°C

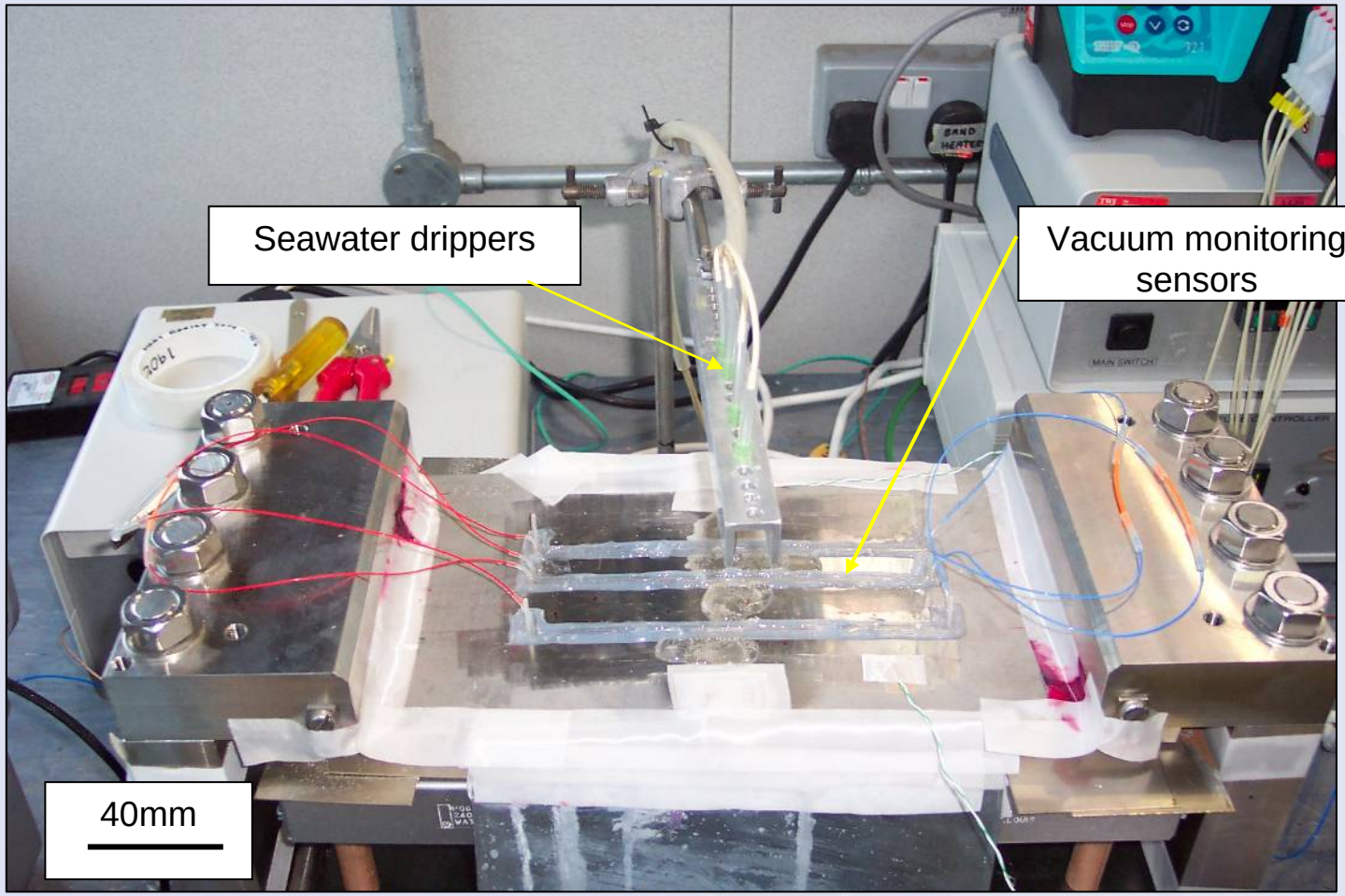
Section of 316 SCC



NDT Techniques Evaluated

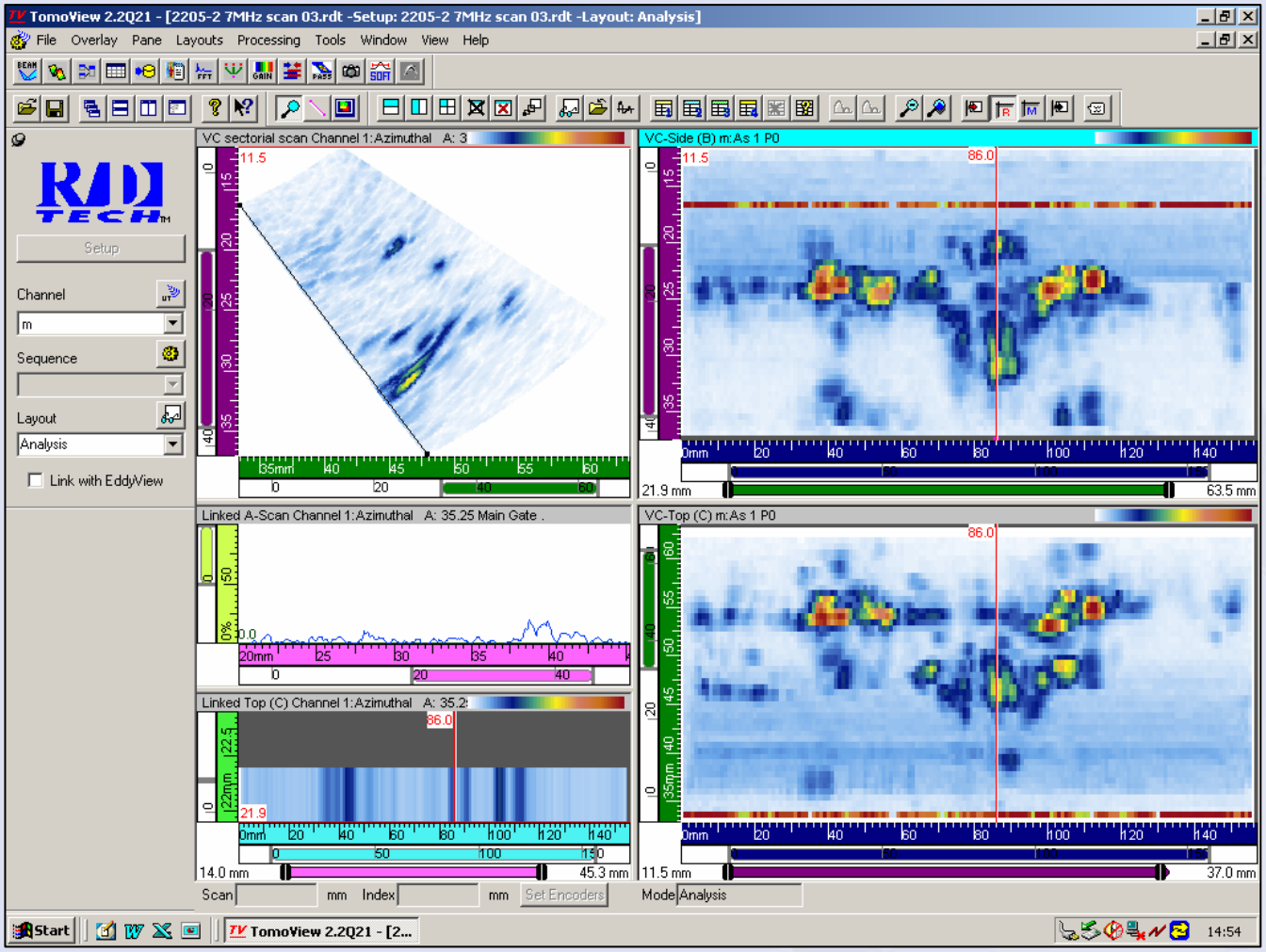
- Acoustic Emission
- Comparative Vacuum Monitoring (CVM)
- Penetrant/MPI
- Eddy Current (probes/arrays)
- Pulsed Eddy Current
- AC Field Measurement
- AC Potential Difference
- Phased Array Ultrasonic Testing (PAUT)
- Electromagnetic Acoustic Transducers (EMAT)
- Thermography
- Shearography

Sample Production - Plates



SCC on 316 Plate (penetrant test)

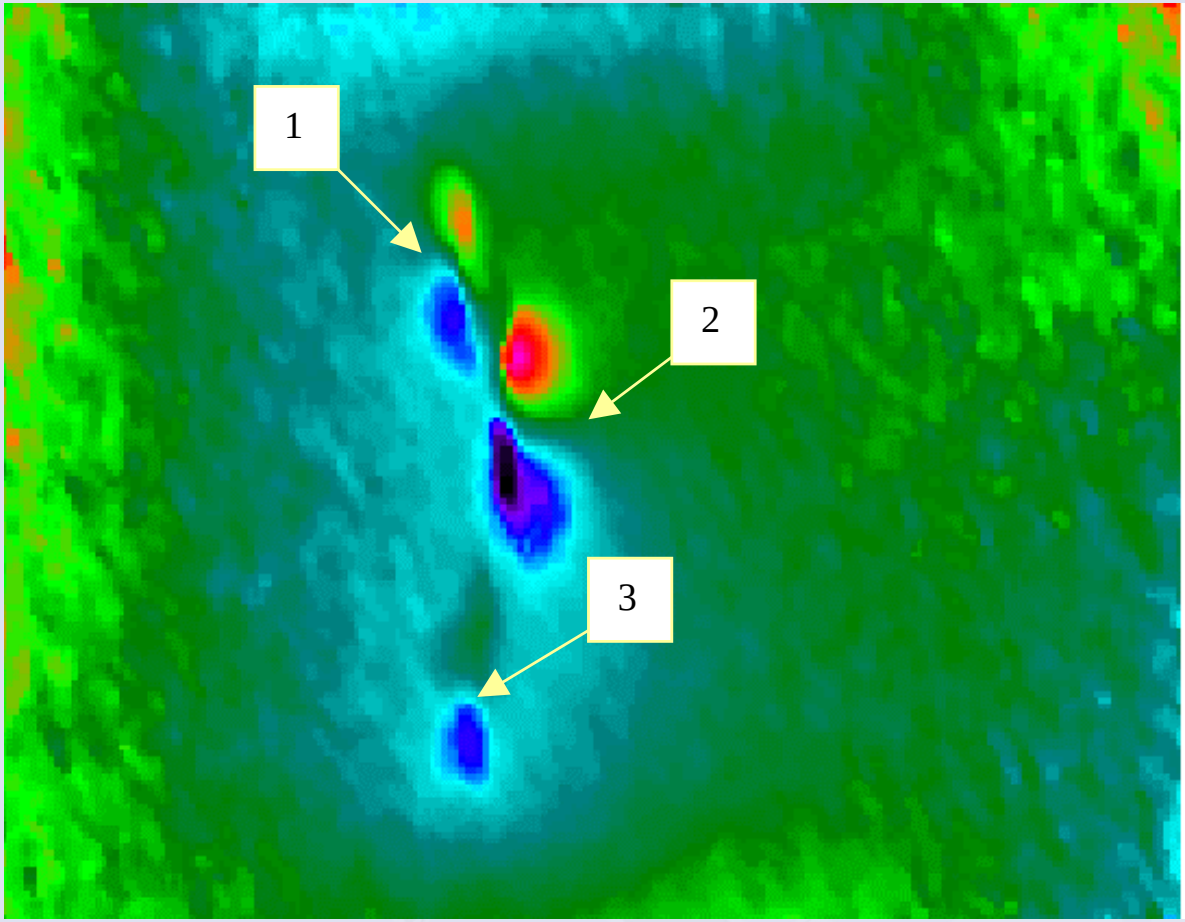
PAUT SCC Detection on Plate



Results - PAUT

- Detection of cracking
- Difficult to size because of:
  - multiple reflections
  - wide extent of cracking (several passes needed)
- Not yet tested at high temperature
  - temperature is likely to further affect sizing/characterisation

Shearography Image



Results - Shearography

- Shown to be capable of detecting cracks of around 3mm deep
  - at room temperature
- Would be capable of hot deployment
- Requires stress application and vibration free conditions

Conclusions for 316

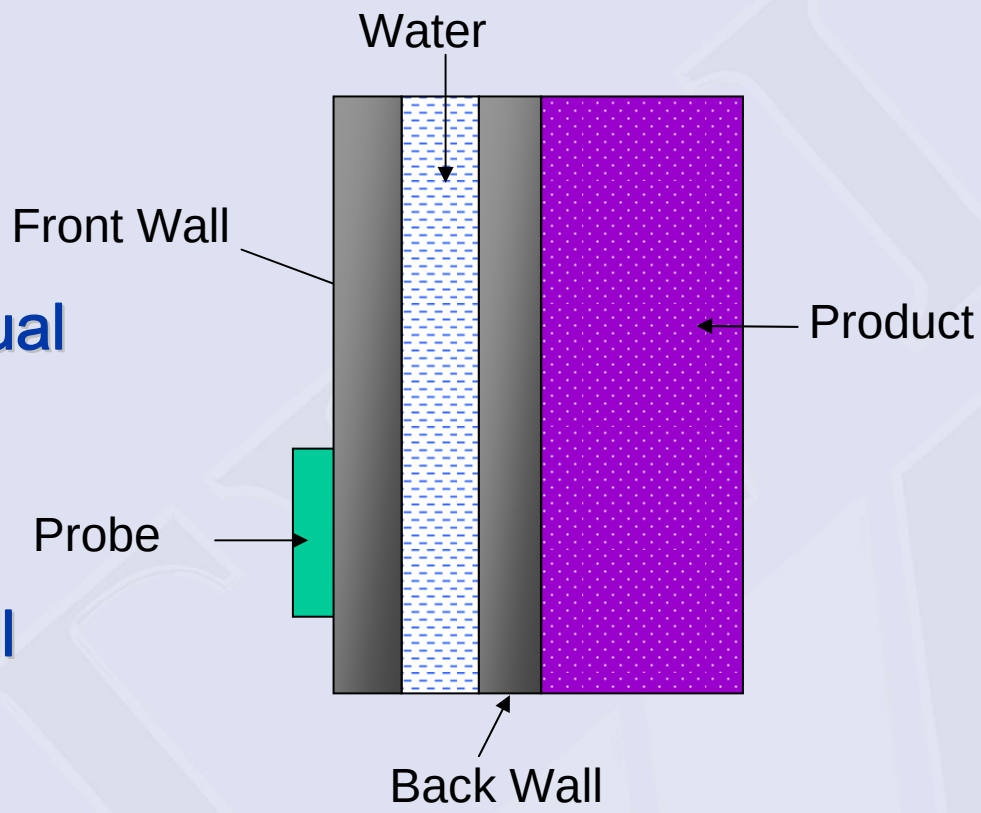
- Penetrant testing satisfactory
- High frequency eddy currents good alternative
  - Found all flaws detected by penetrant testing
  - Can operate with some lift off
- Phased array UT will detect SCC on far side but requires contact and couplant
- Shearography good for remote inspection but requires applied stress and vibration-free environment

Inspection for Corrosion - Statistics

- Extrapolation of sample inspection data to uninspected areas by extreme value statistics
  - assumes same corrosion conditions as in corresponding inspectable area
  - fits thickness distribution to inspectable area
- Technique for estimating future probability of failure based on successive inspections
- Suitable for general corrosion and general pitting situations

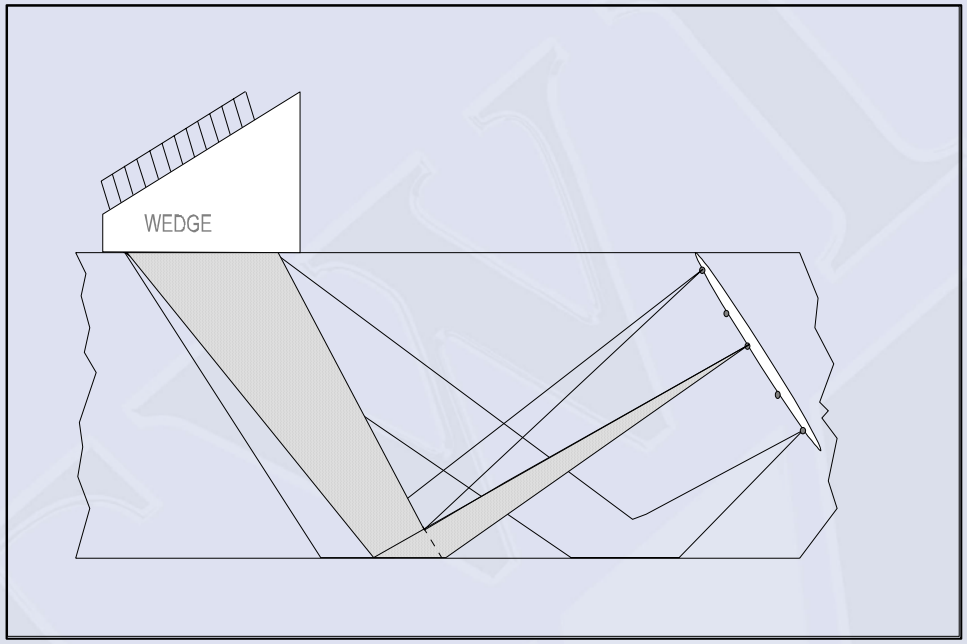
Double Wall Inspection for Corrosion

- Project to measure back wall thickness from front wall in nuclear reprocessing facility
- Established that manual and phased array UT capable of detecting and measuring corrosion on back wall



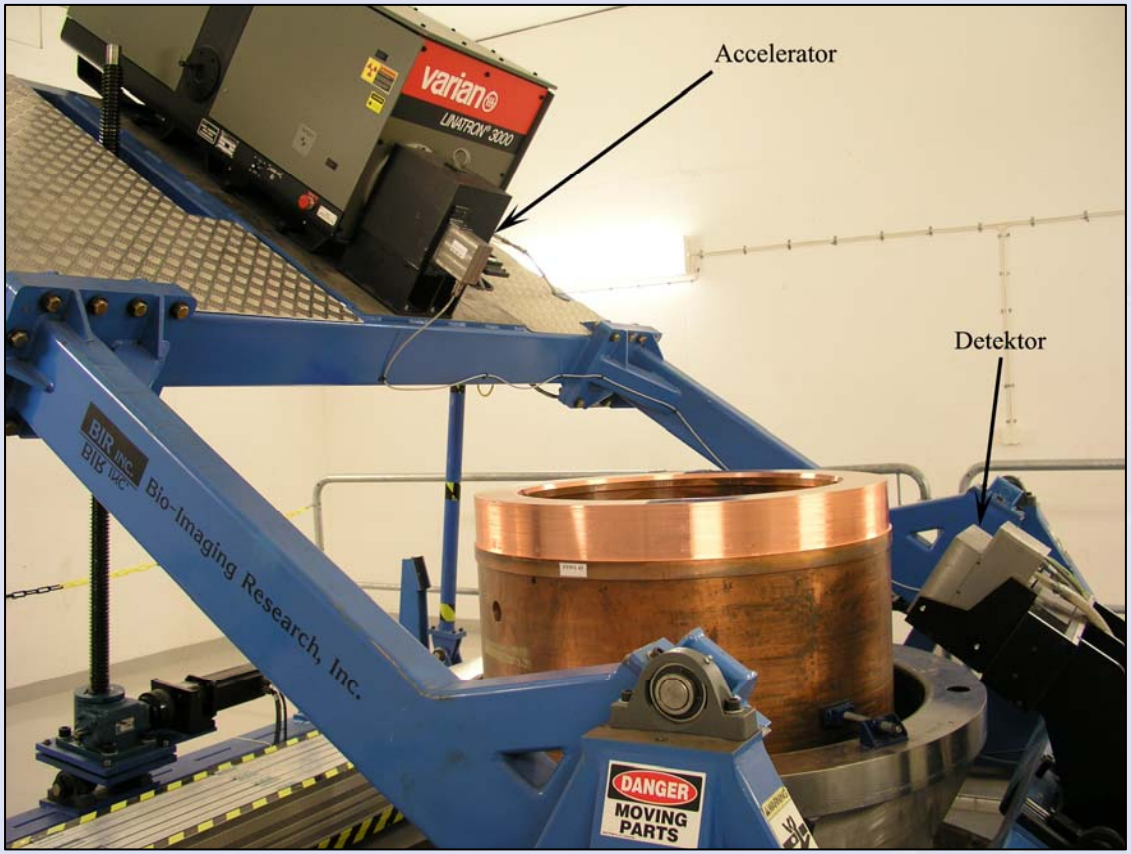
Stainless Steel Weld Inspection

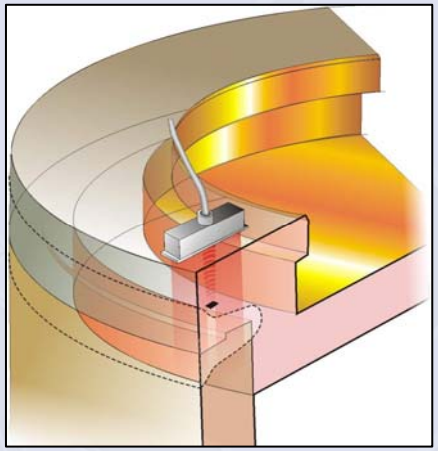
- Review concluded that standard Ultrasonic Testing procedures not suitable for stainless steel welds
- SS welds need to be inspected by angled compression waves and with an imaging system to determine signal sources
  - phased array scanning is most recent method available
- Defect sizing capability yet to be fully quantified. Expect sizing +/- 1.5mm but will be worse than in conventional ferritic welds.

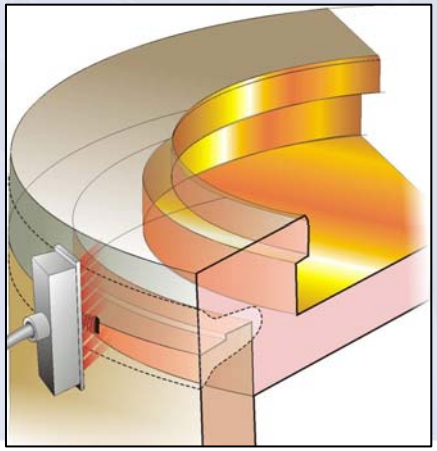


NDT of Copper Canisters at SKB

Digital radiography (X-ray) and phased array ultrasonic

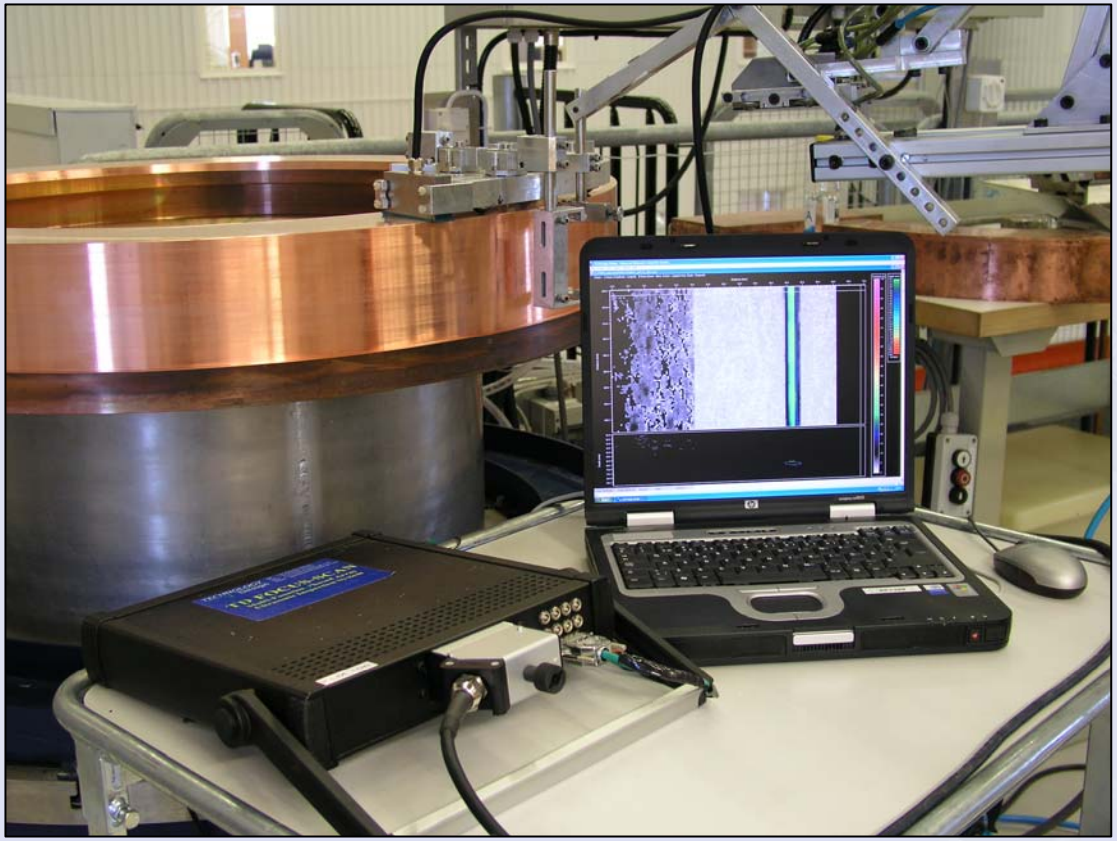


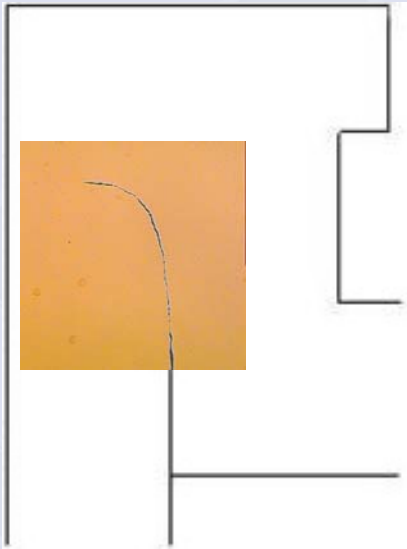




Process Comparison

- NDT: No defects by X-ray, Joint line hooking by UT

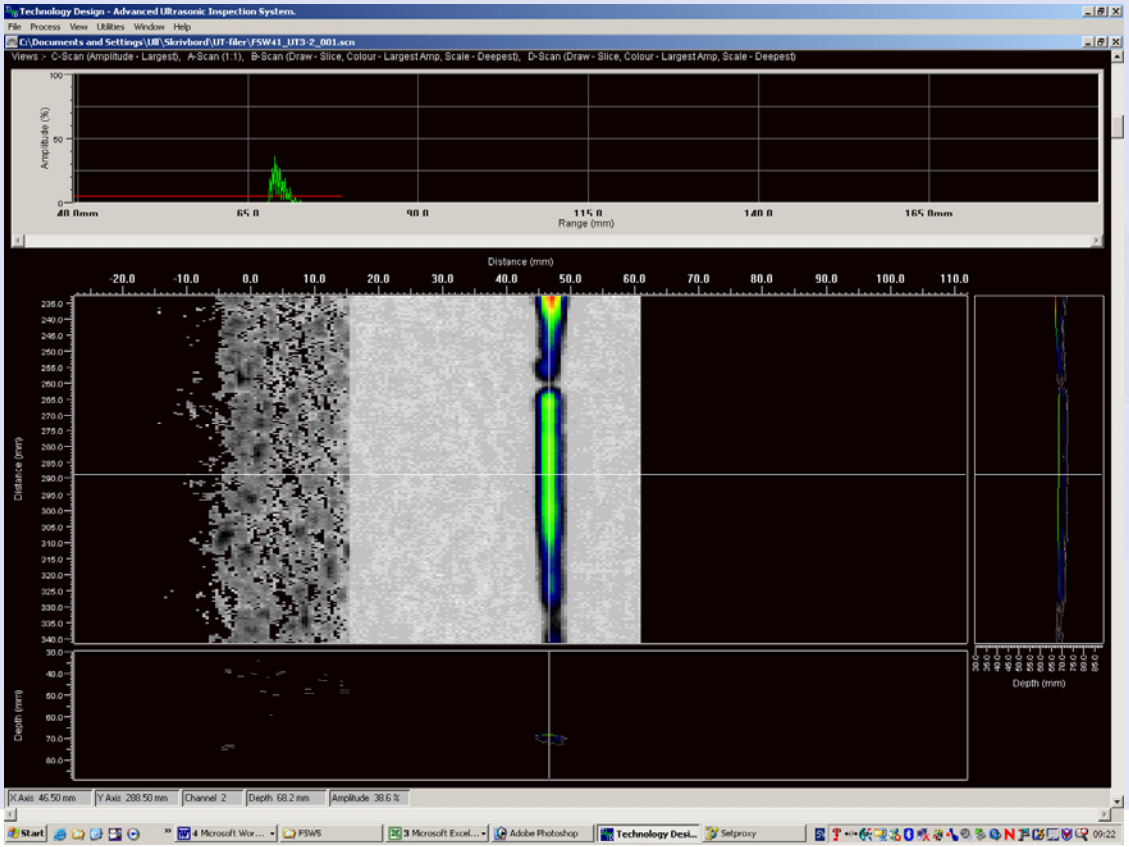




Not to scale, maximum 2 mm joint line hooking

PAUT Results

- Close up of ultrasonic results of joint line hooking during overlap sequence



Remote Inspection Methods

- Robotic deployment
- Recently projects commenced for deployment of robots for inspection on aircraft components, steel plates and under-oil in storage tanks.
- Also methods developed for ROV (remotely operated vehicle) underwater inspection

ROBAIR - Robotic Inspection

- Climbing robot using suction cups to attach to surface
- Developed for aircraft structures but applicable to other large area structures
- Alternative approach uses magnetic wheels for vehicle for inspection of ferritic steel structures
- Both developed in European collaborative projects

High Temperature UT

- Site Work to 250°C (cooled probes)
- Lab Work to 350°C
- Special couplants
- Calibration
- Research to 550°C
  - Lithium niobate crystals
  - Brazed to steel block

Concluding Comments

- Nuclear structures present a range of challenges for inspection:
  - Area, number of parts
  - Geometry
  - Access
  - Temperature
  - Flaw size
- Many candidate NDT techniques but no universal solution
- EU projects are proven route to develop, optimise and demonstrate processes and procedures for specific tasks

Future Project Work

- TWI has an ongoing interest in collaborating with other organisations on NDT research projects
  - EC Framework 7
  - UK DTI programmes
  - etc

BAM GUIDELINES FOR THE SAFETY ASSESSMENT OF TRANSPORT AND STORAGE CASKS FOR RADIOACTIVE MATERIALS

B. Droste, U. Zencker, H. Völzke, F. Koch, K. Müller

Federal Institute for Materials Research and Testing (BAM), Berlin, Germany

Workshop on Design and Assessment of Packages for Radioactive Waste

Bergen, The Netherlands, November 21 – 22, 2006

Organized by Joint Research Centre, Institute for Energy, Petten, NL

Contents:

BAM Guidance to support the design process and the assessment procedure of transport and storage casks for radioactive materials

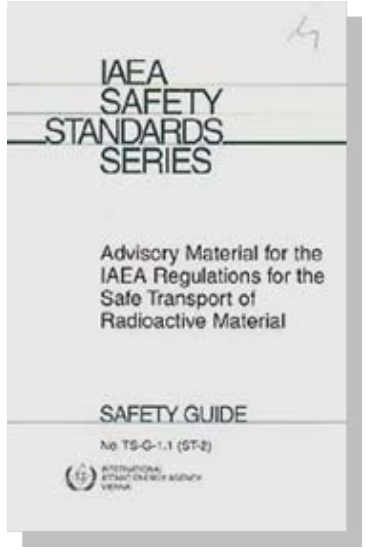
- **BAM GGR 007: Guideline for the Application of Ductile Cask Iron Casks.**
- **BAM GGR 008: Guideline for Numerical Safety Verifications within Design Assessment.**
- **BAM GGR XYZ: Guideline for the Calculation and Assessment of Lid and Trunnion Systems (draft).**

BAM Competence, Experience and Guidance in Drop Test Performance

BAM GGR 007 – Fracture Safety Evaluation Methods

IAEA recommendations for the prevention of failure by brittle fracture are contained in the Appendix VI to the Advisory Material for the IAEA Regulations:

- Ductile materials:
  - simplest method: material cannot fail by brittle fracture because of the material properties throughout the required service temperature range
  - no explicit limitation of stress or absence of flaws required
- T_{NDT} Approach:
  - based on the investigation of the arrest of a propagating crack in dependence of the temperature by means of the drop weight test
  - Charpy impact energy used as indicator of material toughness
- Fracture Mechanics Assessment
  - comparison of fracture mechanics load parameters with material properties characterizing the material resistance to crack initiation
  - detailed analysis of interaction between flaws, maximum stresses, fracture behaviour and deformation properties of the material
  - generally based on the laws of linear elastic fracture mechanics
  - used e.g. in standards for construction of nuclear components, but stress level and loading rate higher under accident conditions+



App. VI: Guidelines for safe design of shipping packages against brittle fracture

BAM GGR 007 - Concept for Design Approval from 1985

BAM Concept for Design Approval of DCI casks from 1985 was based on:

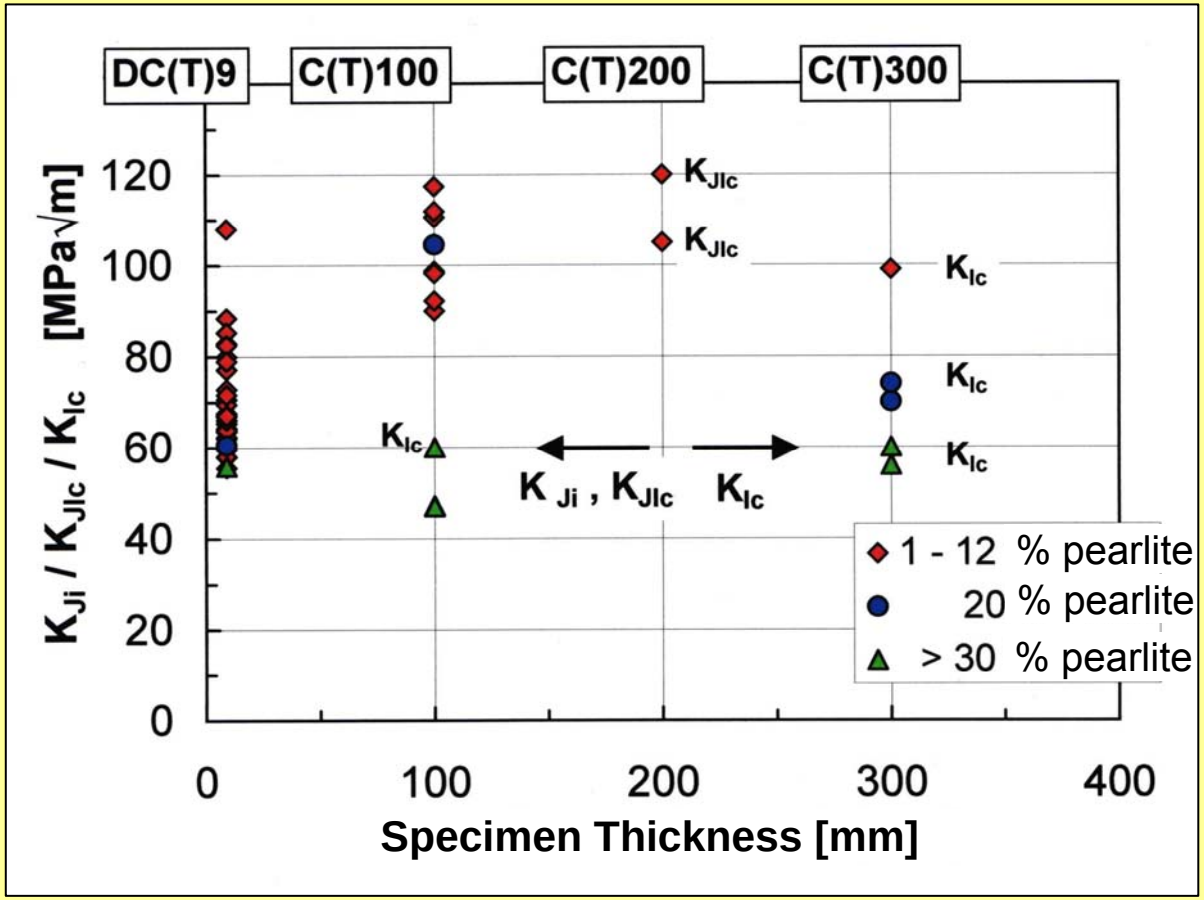
- **Limitation of** *maximum permissible stresses* **by design**
- **Definition of** *permissible material properties*
  - Microstructure
  - Deformation properties
    (yield stress, tensile strength, elongation at fracture)
  - Fracture toughness: $K_{Ic} \geq 50$ MPa√m
- **Limitation of the** *size of flaws* **by non-destructive examination**
- *Quality assurance measures*

Approval of

- **Plans for cask production**
- **Plans for test sequences**
- **Material specifications**

BAM GGR 007 - Evolution of DCI Properties since 1985

- **Fracture mechanics characteristics of DCI in dependence of wall thickness, pearlite content and specimen size:**

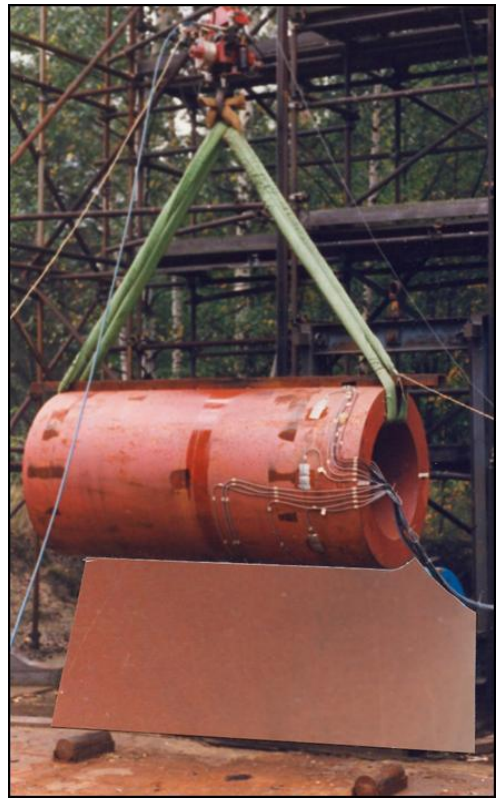


- **Lower bound fracture toughness value could be confirmed.**

BAM drop test with a thick-walled pipe of ductile cast iron

- **corresponding to the 1:2.5 scaled model of a large cylindrical CASTOR V cask**
- **drop height 9 m**
- **drop onto steel cylinders located on an unyielding IAEA target**
- **equipped with an artificial crack-like defect (40 mm in 150 mm wall)**

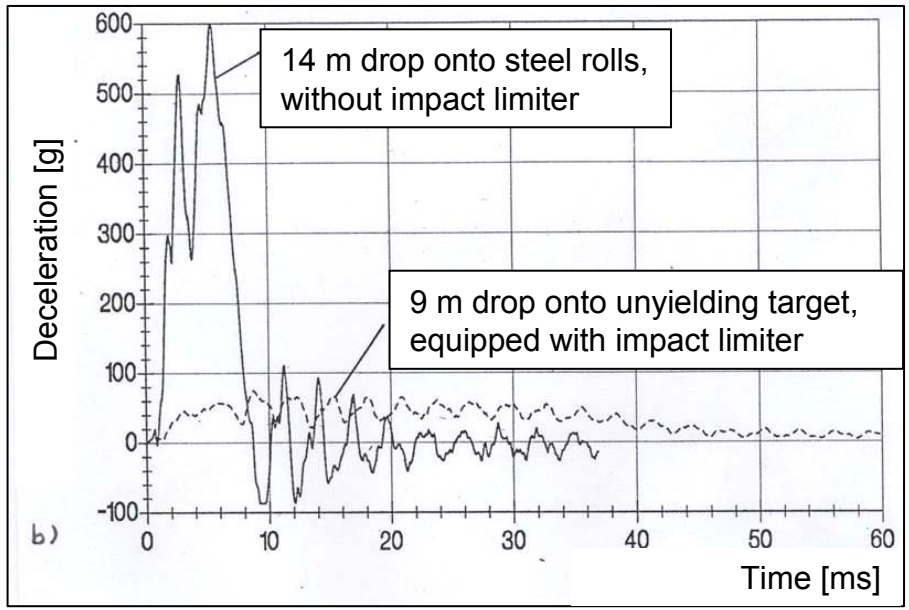
No failure by fracture.



BAM drop test with a CASTOR VHLW cask

- **drop height 14 m**
- **drop onto steel cylinders located on an unyielding IAEA target**
- **no impact limiters**
- **equipped with a 120 mm deep artificial defect (compared with wall thickness of 260 mm)**
- **defect located in the position of highest bending stress**

No failure by fracture.



BAM GGR 007 - A new Guideline for Cask Design Assessment

- **Cask designs were optimized on the part of the manufacturers.**
- **New cask features were introduced.**
- **Extended applications, e.g. at storage.**

- **Rise of the stresses in the structure.**
- **Prerequisites for the old concept could increasingly be satisfied only with very lavish impact limiters.**

BAM

Bundesanstalt für Materialforschung und -prüfung

Fachgruppe III.3 „Sicherheit von Transport- und Lagerbehältern"

BAM – GGR 007

Leitlinie zur Verwendung von
Gusseisen mit Kugelgraphit
für Transport- und Lagerbehälter
für radioaktive Stoffe

Rev. 0

Juni 2002

Diese Leitlinie besteht aus den Seiten 1 bis 35.

Guidelines for the Application of Ductile Cast Iron for Transport and Storage Casks for Radioactive Materials (BAM - GGR 007)

- **Higher stresses than before can be permitted. Comprehensive safety proof based on the methods of fracture mechanics is required.**
- **Consideration of highly dynamic load cases.**

BAM GGR 007 – A new Guideline for Cask Design Assessment

- **Concept bases on principle of preclusion of crack initiation (acc. to IAEA).**
- **Postulated material defects are modelled as sharp cracks.**
- **Methods of linear elastic or elastic plastic fracture mechanics have to be used in dependence of the material behaviour.**
- Criteria: **Failure by fracture can be avoided if**

$$K_{appl} < K_{mat} \quad \text{or} \quad J_{appl} < J_{mat}$$

- Crack tip load parameter must be an upper bound **with regard to**
  - size of crack
  - most unfavourable shape and orientation of the crack
  - maximum permissible stress
  - loading rate
- Material resistance to crack initiation must be a lower bound **with regard to**
  - material specification
  - loading rate
  - temperature
  - wall thickness of the cask

  Alternatively the material property can be found out for the individual case.

BAM GGR 007 - Calculation of Crack Tip Load Parameter

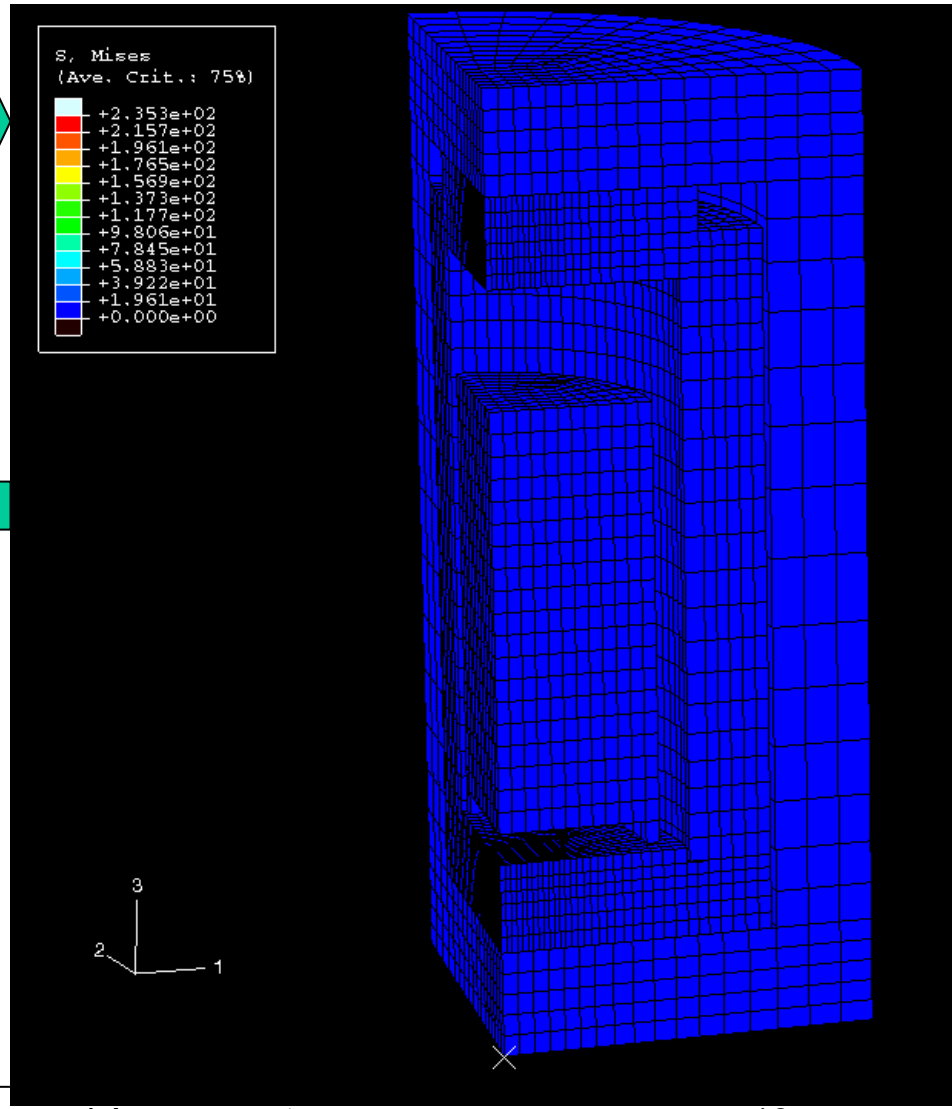
- **Computational fracture mechanics analysis by simulation of the mechanical tests:**

Find positions of high fracture mechanics load (in most cases by numerical stress analysis)

Find the crack tip load parameter as temporal and local maximum from all values along the crack front:

$$K_{I,appl} = \max\{K_{I,appl}(\vec{x},t)\}$$

$$J_{I,appl} = \max\{J_{I,appl}(\vec{x},t)\}$$

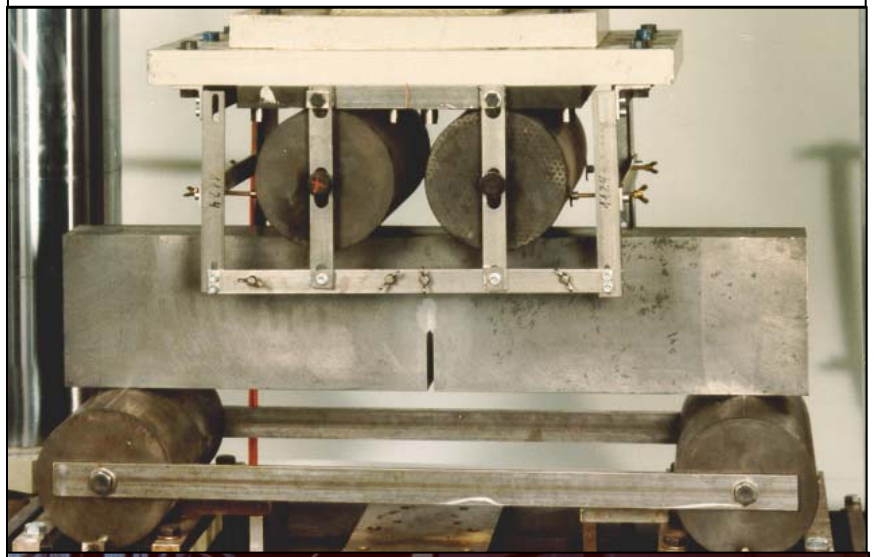


- **sub-model technique allowed, if quasi-static crack problem**
- **"BAM-Guidelines for the Numerical Safety Analyses for the Approval of Transport and Storage Casks for Radioactive Materials", GGR 008, 2003**

- **Measuring of valid fracture toughness values (K_{Ic}, K_{Id}) requires test of sufficiently large specimens.**
- **In the context of the approval of casks measuring generally carried out at small specimens.**
- **Small size and elastic-plastic material behaviour require application of J integral concept (J_i or J_{id}) with determination of crack resistance curve: J–Δa.**

- **Determination of static and dynamic initiation values acc. to standards ASTM E 399, ESIS P2, ASTM E 1820.**
- **Quasi-static conditions are assumed for strain rates $\dot{\varepsilon}$ < 0.1 s^{-1}.**

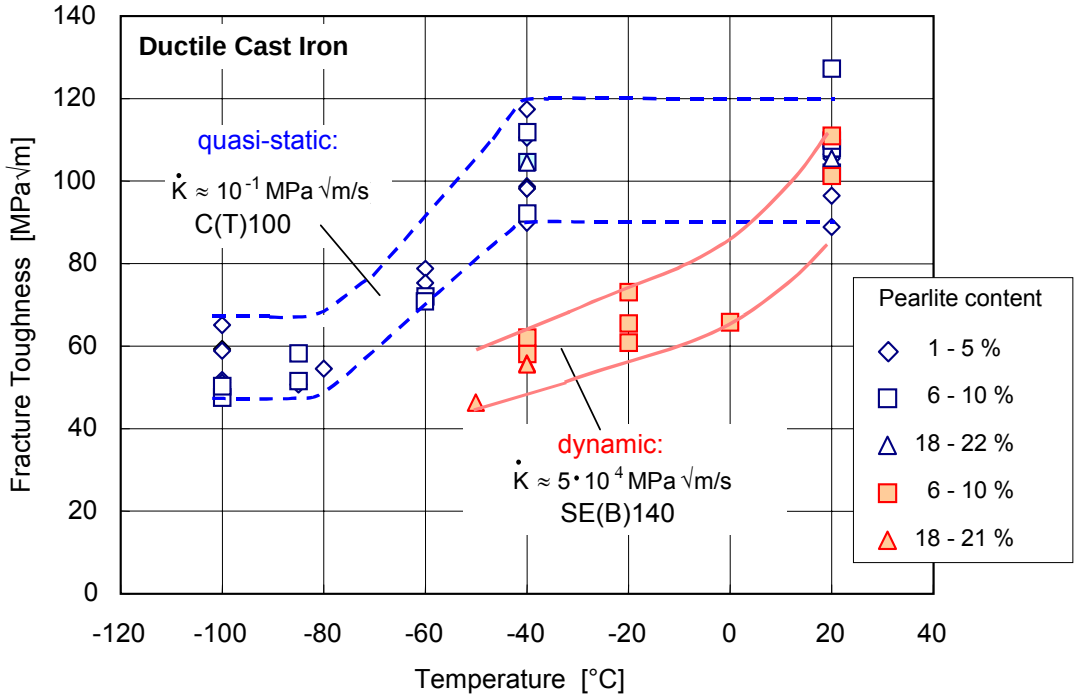
Test of SE(B) 140 specimen



BAM GGR 007 - Material Properties, EN GJS 400-15

- **No significant reduction of fracture toughness expected for static loading conditions down to temperatures of approximately -70 °C.**
- **Under dynamic loading conditions the transition region of the fracture toughness is passed in the temperature range from -20 °C to -40 °C.**
- **Crack initiation toughness of DCI can decrease on comparatively low values because of the changing of the failure mode from ductile to brittle fracture.**

- **BAM investigated bending specimens of type SE(B) 140 from container walls to find the dynamic fracture toughness for large specimens.**
- **Pearlite contents up to 20 %.**
- **Minimum value of the fracture toughness (lower bound: 50 MPa•m$^{1/2}$) could be confirmed for large specimens also under dynamic loading conditions.**



BAM Guidance to Finite Element Methods Applications → a New Guideline : BAM GGR 008

Title:

Guidelines for Numerical Safety Verifications within the Scope of the Design Assessment of Transport and Storage Casks for Radioactive Materials **(GGR 008: www.bam.de)**

Purposes of the Guideline:

- **Definition of Accepted Quality Assurance Measures for**
  - **Preparation,**
  - **Checking and**
  - **Evaluation of Finite Element Safety Assessments for Transport and Storage Casks**
- **Common Basis for Elaboration and Checking of FE Safety Assessments to Minimise the Iteration Process between Applicants and Responsible Authorities**
- **Guarantee Adequate Precision of Calculation Results with Respect to the Reality**

BAM Guidance to Finite Element Methods Applications → a New Guideline : BAM GGR 008

Structure of the Guideline:

- Definition of Technical Terms
- Requirements towards Numeric Safety Verifications

Definition of Technical Terms :

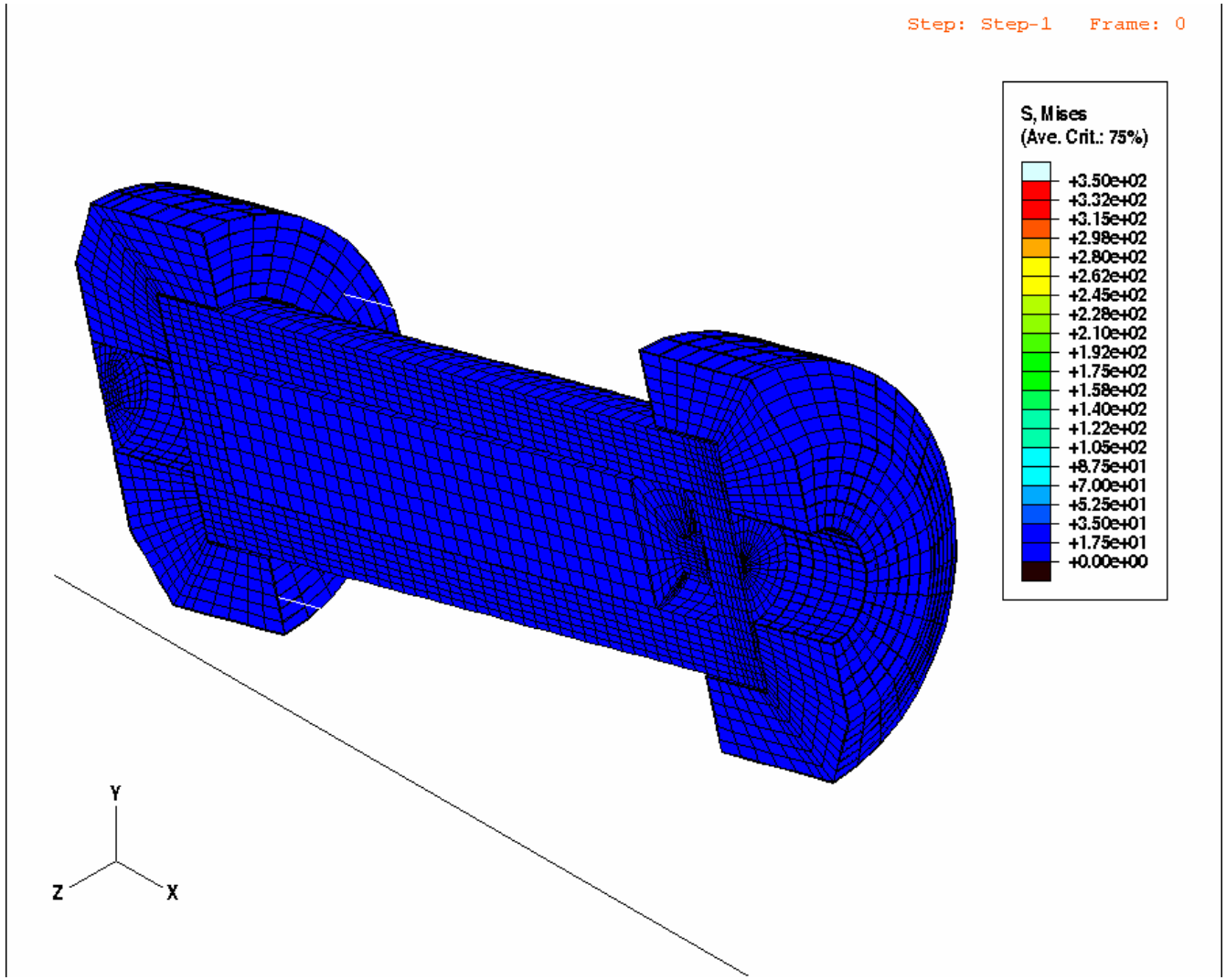
- Verification and Validation
- Benchmark investigation
- Parameter study
- Conservative design calculation
- Quality assurance
  - with Regard to the Used Finite Element Code
  - with Regard to the Users

BAM Guidance to Finite Element Methods Applications → a New Guideline : BAM GGR 008

Requirements towards Numeric Safety Verifications :

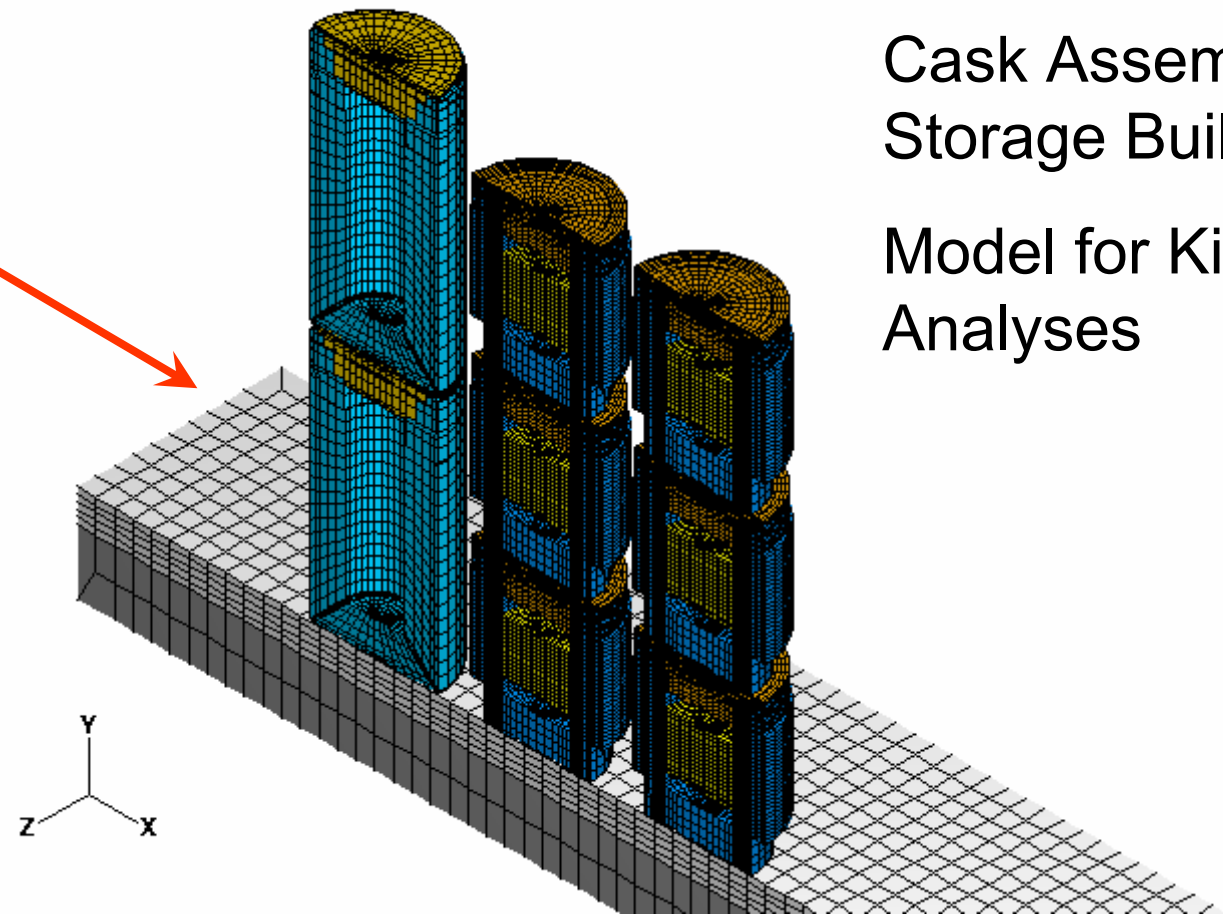
- **Formal Requirements**
- **Completeness**
- **Programme/FE Code Documentation and Description**
- **Modelling:** **e.g. load, geometry, material properties**
- **Data Documentation** **(input data sets, decisive results and further results)**
- **Data Presentation** **(graphical presentation preferred)**
- **Evaluation of Calculation Results:**
  - **analytical or numerical checks with a simplified structure,**
  - **checking by comparison with experimental results,**
  - **checking by grid refinement or**
  - **comparisons with reference examples**
  - **precision should be within ± 5%**

BAM FE-Analysis of a 9m Spent Fuel Cask Drop Test



Application of Finite Element Analyses: Aircraft Crash onto Spent Fuel Storage Casks

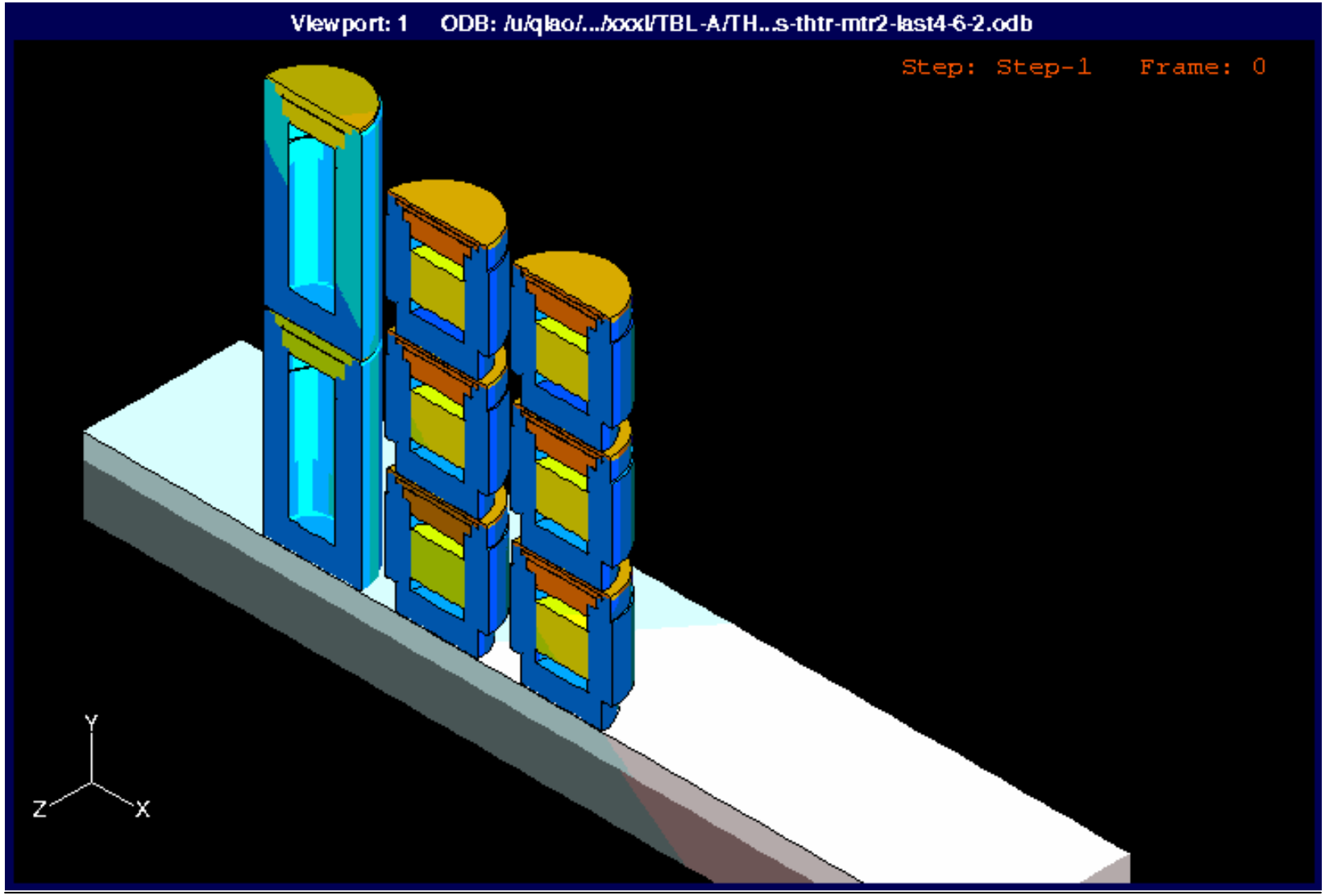
Aircraft Wrackage Load



Cask Assembly in a Storage Building

Model for Kinematic Analyses

Moving of the Casks after Aircraft Crash (0 … 1 sec.)

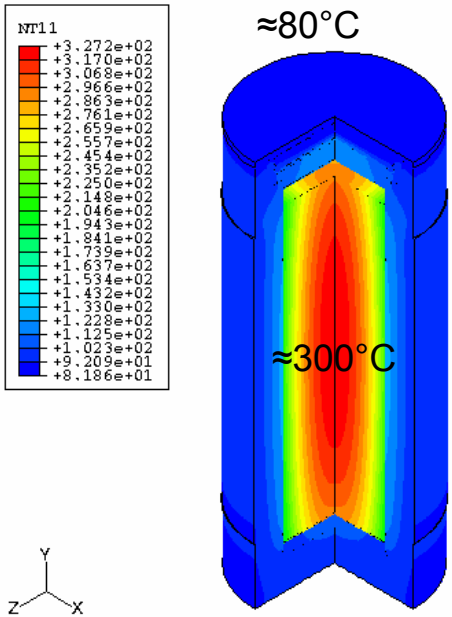


Finite Elemente Analyses of Casks in Fire Scenarios

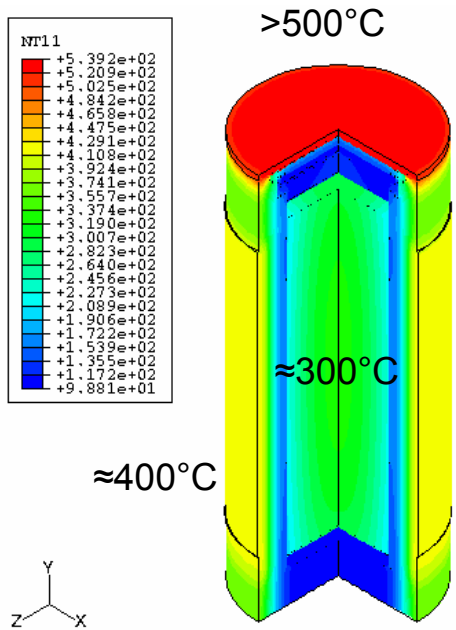
BAM FE-calculation example:

Temperatures of a CASTOR cask with 40 kW spent fuel decay heat

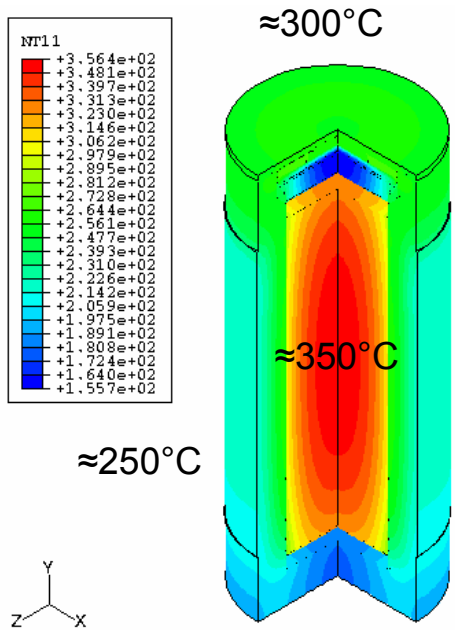
Balanced thermal conditions before fire



After 20 minutes fire duration



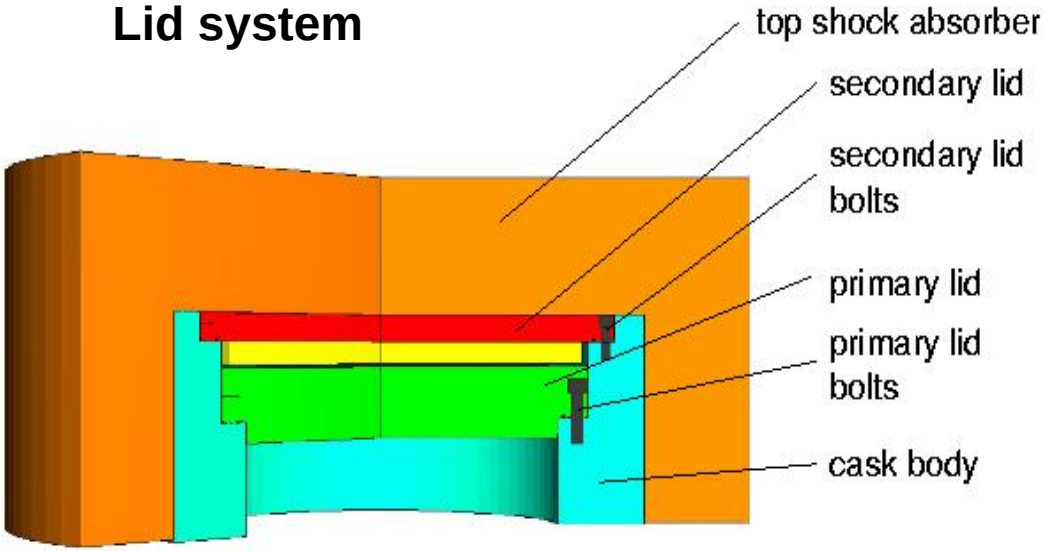
2 hours after end of fire



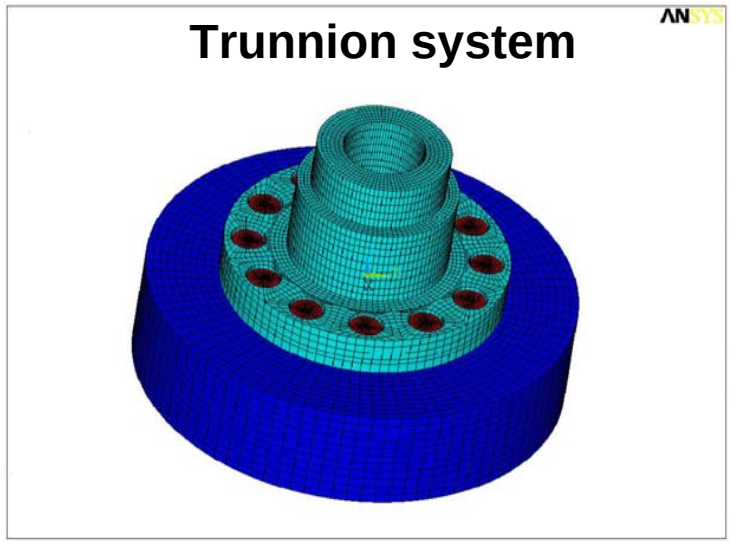
BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

Adressed Objects:

Lid system



Trunnion system



In the Past:
Analytical approaches based on nominal stress limitation concepts

Current Developments:
Numerical approaches based on local stress limitation concepts

BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

Adressed Verifications:

Lid system

- strength of lid bolts
- compression of seals
- strength of lids
- sliding of lids

Trunnion system

- strength of trunnions
- fatigue strength of trunnions
- strength of trunnion bolts
- fatigue strength of trunnion bolts

Both systems

- surface pressure at bolt head
- length of engagement

Local and Nominal Stresses

FEA, experiments:
loads for routine, normal and accident conditions

VDI 2230:
bolt prestress

nominal loads

FE-model for lid system

local stresses

$$F = \iint_A \sigma \, dA \qquad M = \iint_A s \cdot \sigma \, dA$$

nominal forces

$$\sigma_z = \frac{F}{A_s} \qquad \sigma_b = \frac{M}{W}$$

nominal stresses

VDI 2230:

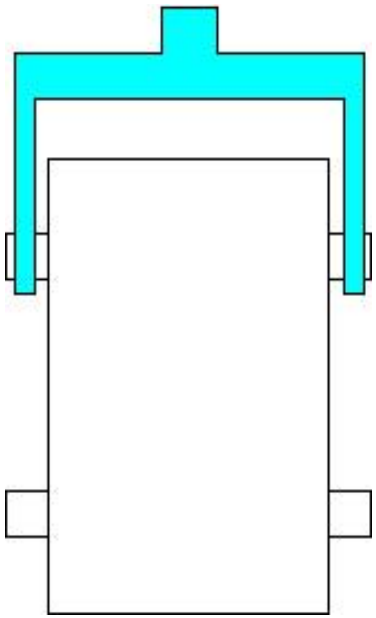
$$\sigma_v = \sqrt{(\sigma_z + \sigma_b)^2 + 3 \cdot (f_M \cdot \tau_{G,Mon})^2}$$

von Mises stress

BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

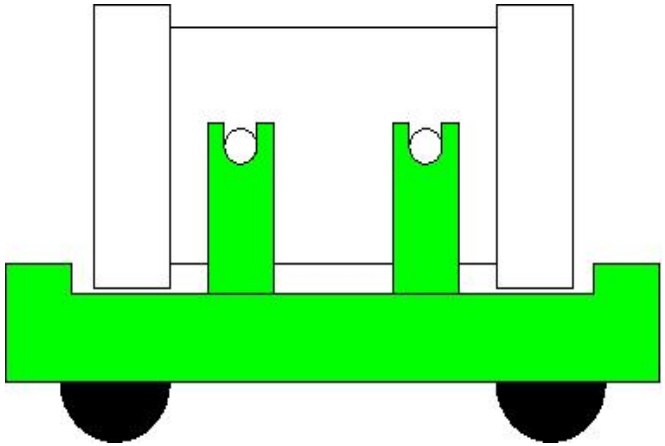
Trunnion Problem

Handling



- DIN 15018-1
- KTA 3905

Transport



- ISO/TC85/SC5/WG9

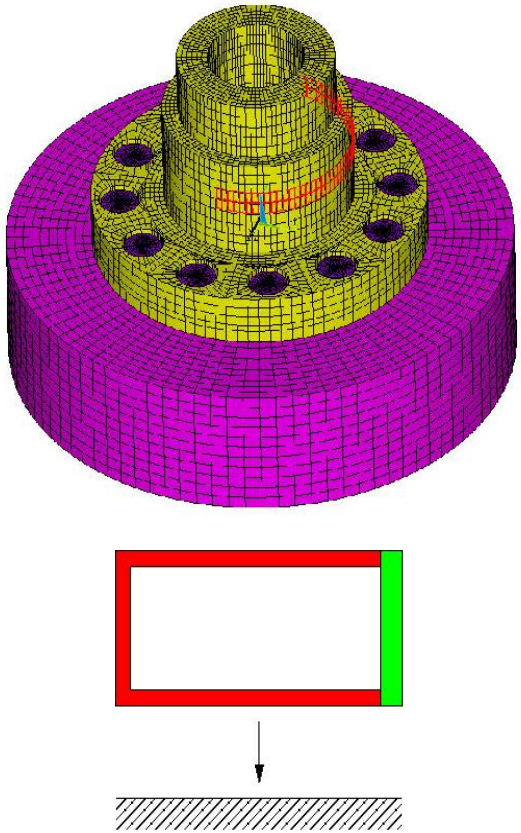
Provided Solution

- different load assumptions combined with load factors
- many FE-analyses using one FE-model with different loads
- results assigned to strength and fatigue strength verification
- assessment according to KTA 3905 safety concept

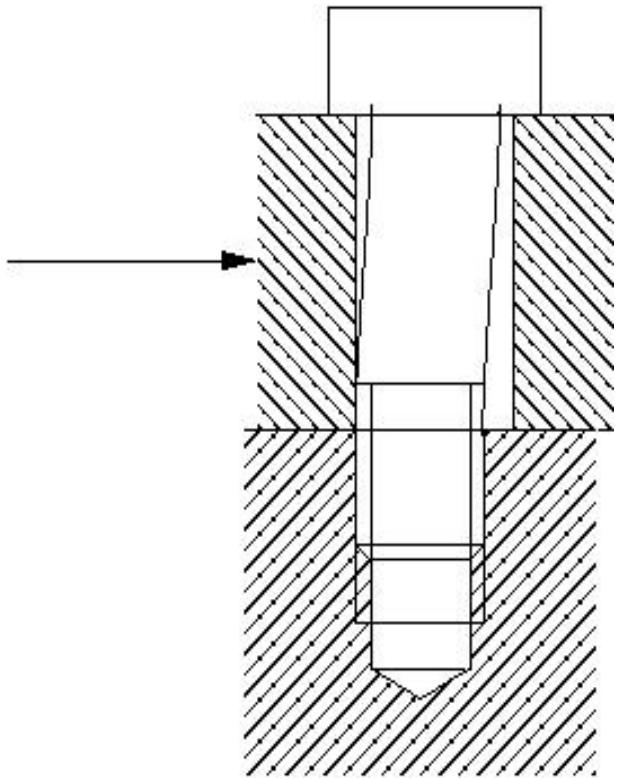
BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

Interactions

Sliding effects



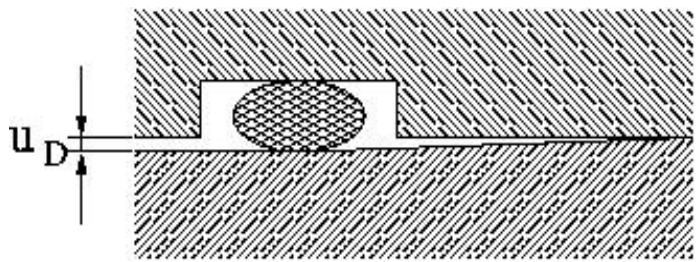
Bending of Bolts



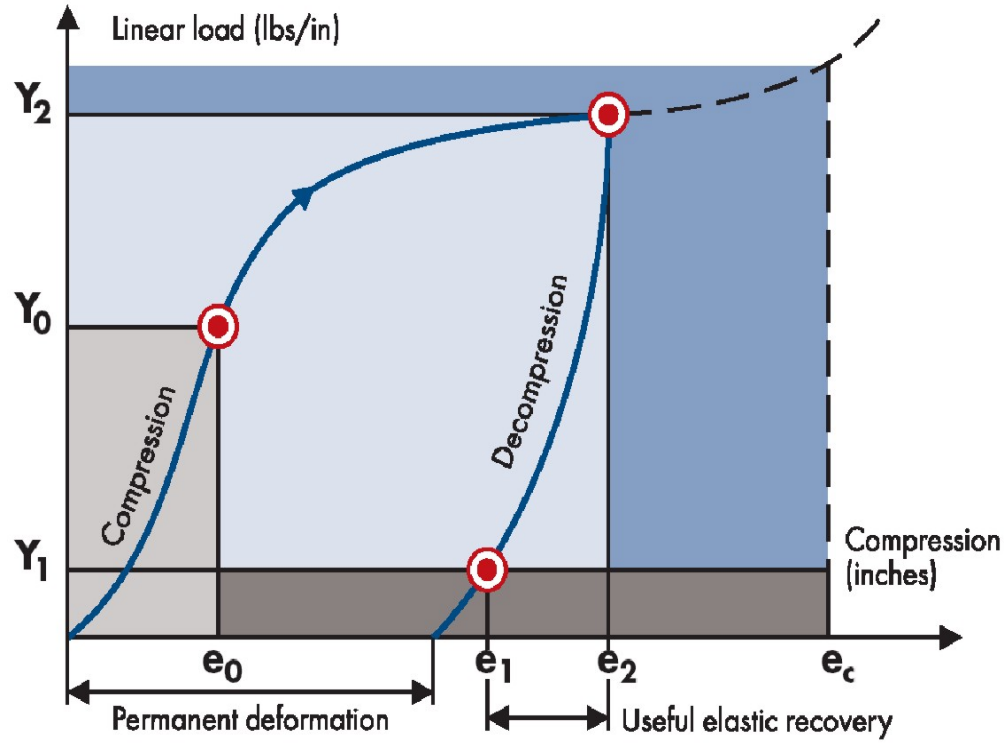
BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

Tightness

- No modelling of seals itself
- Modelling of seal force
- Assessment of gap between flanges
- Comparison with useful elastic recovery



Gap between flanges



Seal characteristic (Garlock Inc.)

Bird`s-eye View of BAM`s Drop Test Facility in Horstwalde

The new large BAM Drop Test Facility

Drop Tower with Test Hall

- Height: 36 m
- 200 ton Hoist,
  80 ton Portal Crane
- Test Hall with
  24 m x 20 m Area

Unyielding Target

Steel reinforced Concrete Foudation:
14m x 14m x 5m

2,450,000 kg	Concrete
103,000 kg	Steel Reinforcement

Preparation of the CONSTOR V/TC Cask before Drop Test

Connection of the Measurement Cables to the CONSTOR V/TC Full-Scale Cask

BAM
Bundesanstalt für Materialforschung und -prüfung

Full-scale Drop Test CONSTOR weight: 181 t

III.3 / 0994

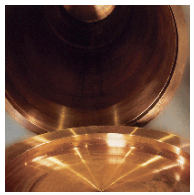
21.09.2004

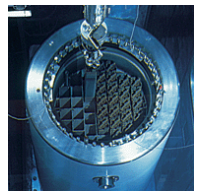
Drop Hight: 9m

Defect types, **NDT** and preliminary **acceptance criteria** approach based on the damage tolerance analysis and international standards **in the manufacturing of nodular cast iron insert**

Pitkänen, J*, Leskinen, N., Rydén, H.** and Emilsson, G.*****

* Posiva, ** SKB, *** Bodycote

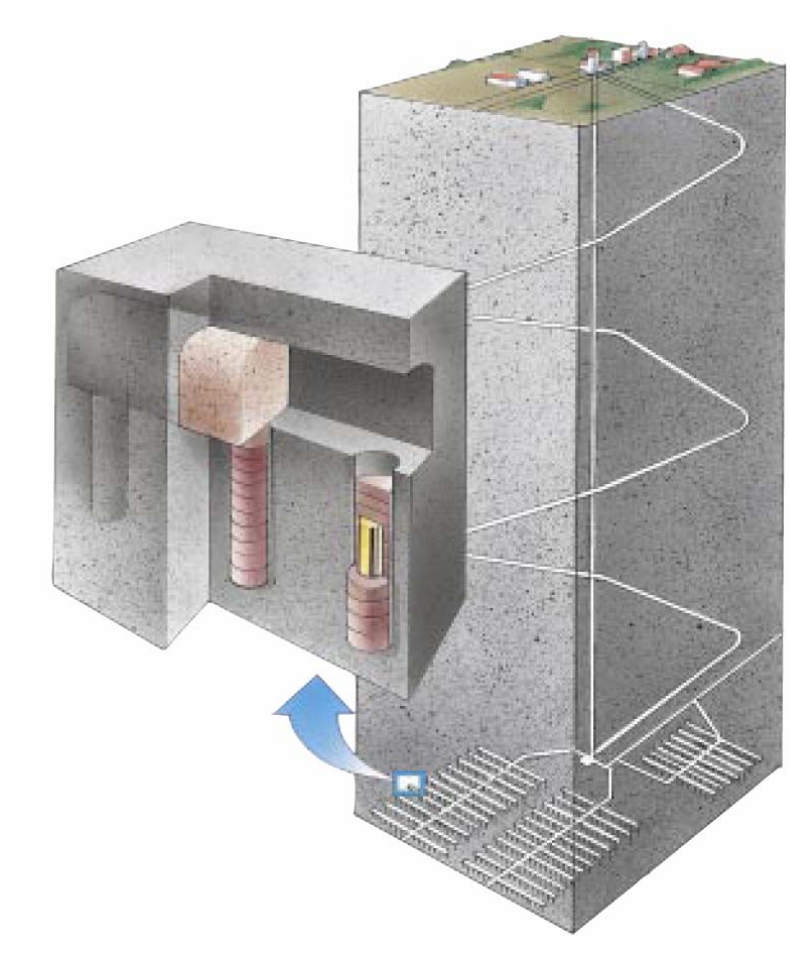



Content

- Disposal plan for nuclear fuel
- General Requirements
- Manufacturing of disposal canister
- Nuclear Fuel Disposal Canister Principle Design in Finland and Schweden
- Defect types in manufacturing of Nodular cast iron insert
- Inspection – Acceptance Criterial – Standards
- Preliminary acceptance criteria for detected defect sizes
- Inspection techniques
- Interference areas
- Surface zone inspection
- Intermediate zone inspection
- Inspection of Zone between Steel channels
- Nearest Corner to Surface Zone Inspection
- Effect of Nodularity to inspection
- Qualification items for NDT
- Input data for insert
- Conclusions

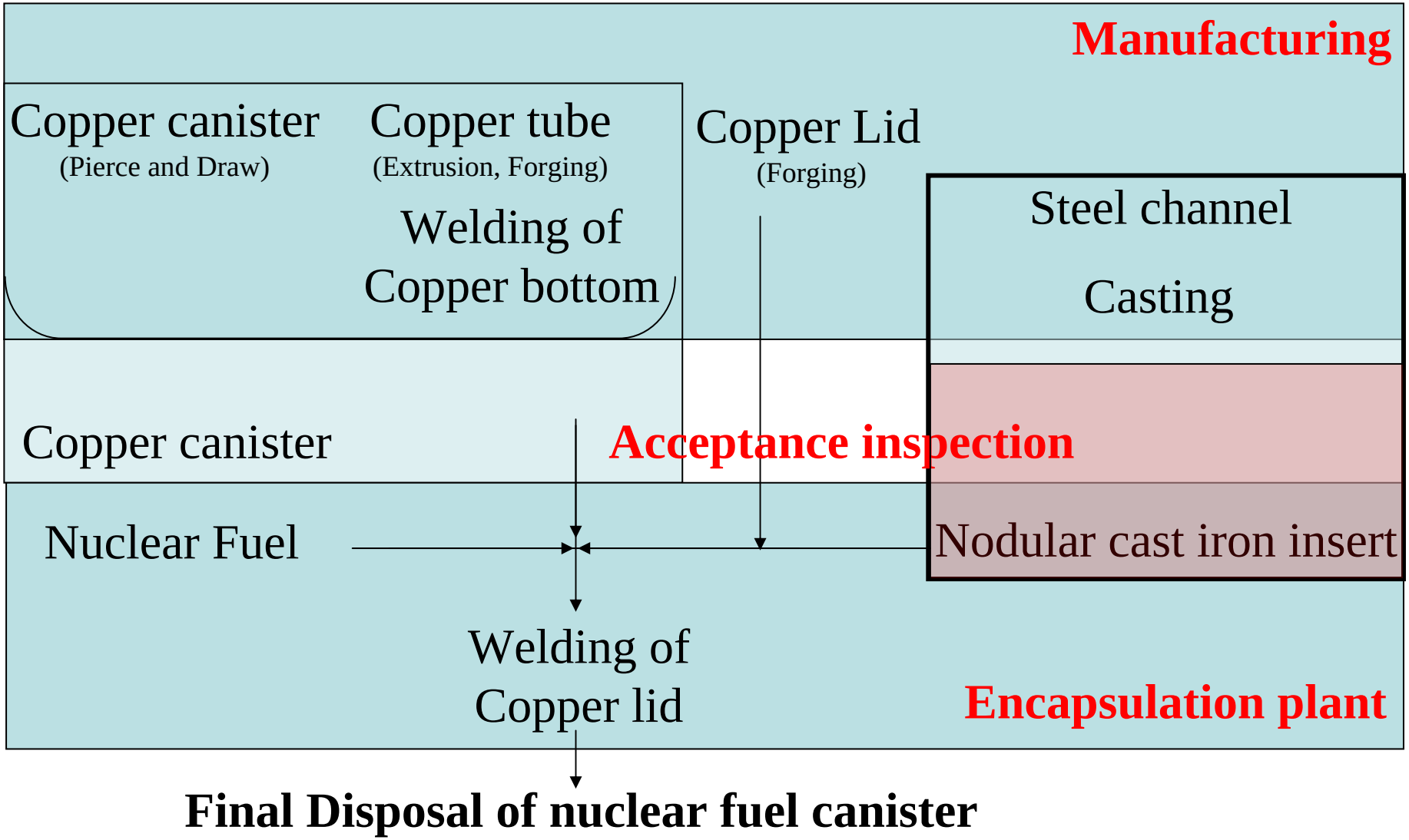
Disposal Plan for Nuclear Fuel



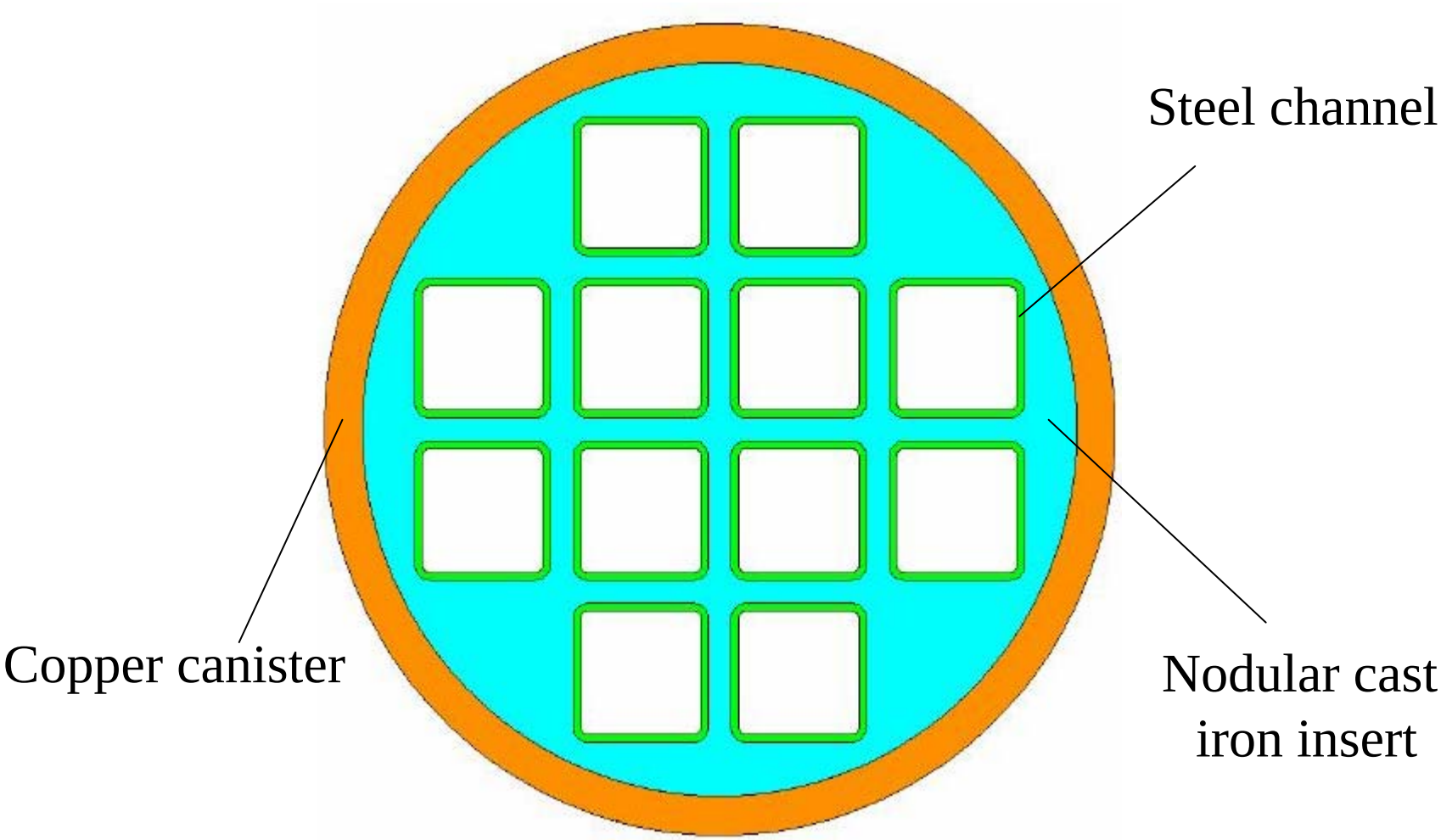
General Requirements for Nuclear Fuel Disposal

- 100 000 Years (1 milj. Years)
- 3 km thick ice scale have been in consideration
- Total design pressure is estimated to be 44 MPa

Manufacturing of Nuclear Fuel disposal canister

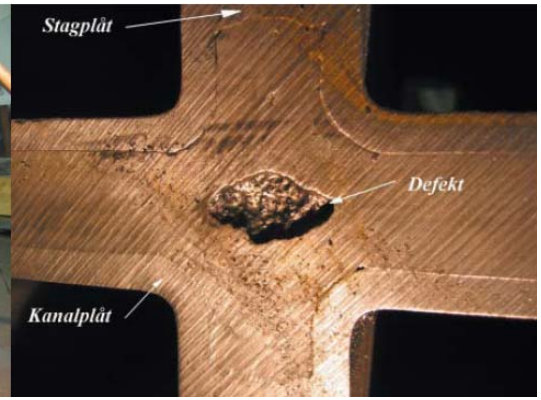


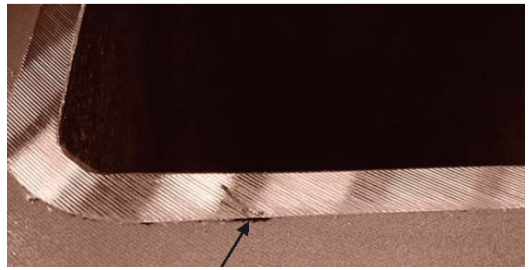
Nuclear Fuel Disposal Canister Principle Design in Finland and Schweden



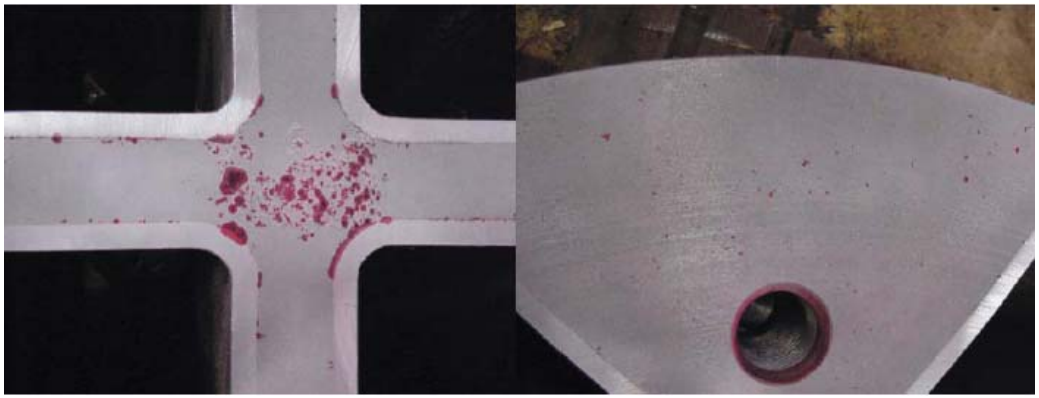
Defect types

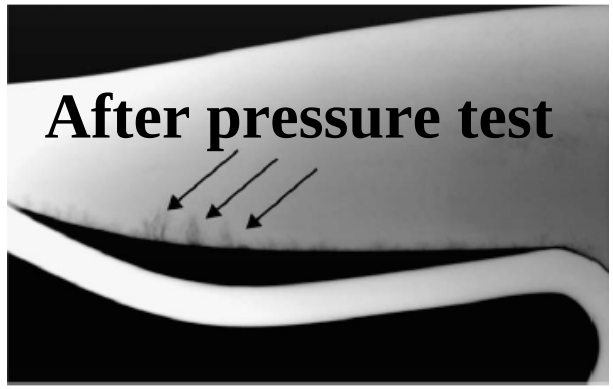
Pipes, pores, inclusions





Incomplete fusion between steel channel and cast iron base material





Most of defects originated from manufacturing are concentrated to upper part of the cast iron insert during casting

Inspection – Acceptance Criterial – Standards

NDT methods

Detection of a defect

↓

Sizing of a defect

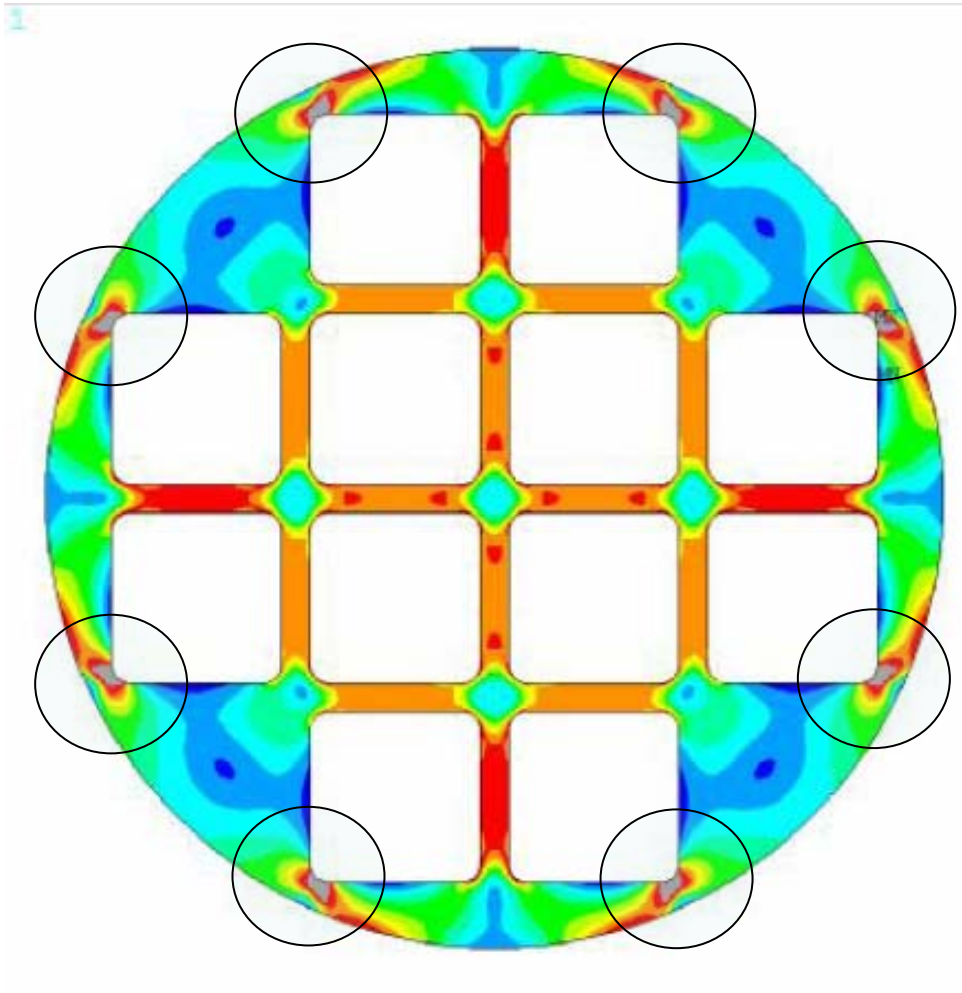
↓

Fracture Mechanics (UT) EN 12680-3

Acceptance criteria

Accepted size | **Non accepted size**

Stresses concentration in the nearest corner to surface during the isostatic pressure 45 MPa (Effective stresses (von Mises) Dillström)



Preliminary acceptance criteria for detected defect sizes

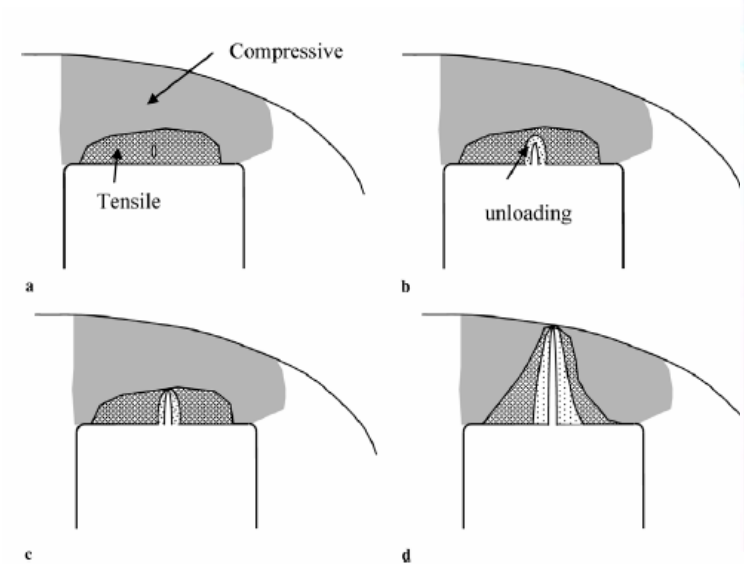
- Planar (Crack-like) type defects (1:6 / height :length)

PWR Zone	PWR Height [mm]	PWR Length [mm]	BWR Zone	BWR Height [mm]	BWR Length [mm]
A	37	222	A	53	318
B1	65	390	B	112	672
B2	50	300			
C	24	144	C	104	624
D	32	192	D	32	192

- Volumetric type defects (round holes through whole canister)

PWR Zone	PWR Diameter [mm]	BWR Zone	BWR Diameter [mm]
A	40	A	80
B1	60	B	100
B2	20		
C	20	C	100
D	20	D	20

Crack growth scenario according to JRC



Acceptance levels EN 12680-3

Severity level 1

Depth range [mm]	Decrease of BWE [dB]	Minimum FBH [mm]	Areas where no BWE Defect echo S/N [dB]
10 -20	12	3	6
20 -100	20	5	6
100 -250	20	8	12
250 - 500	20	10	12

Acceptance levels 12680-3

Severity level 2	Defect ≥level 1
Depth zone [mm]	200 -
% of WT	20
Largest defect in the surface zone [cm^2]	10
Largest defect in the core zone [m^2]	0.015
Total area of inspected surface %	15 (1.8 m^2)

Acceptance criteria EN 12680-3

Severity level 3	Defect ≥level 1
Depth zone [mm]	200 -
% of WT	25
Largest defect in the surface zone [cm^2]	20
Largest defect in the core zone [m^2]	0.03
Total area of inspected surface %	20 (2.4 m^2)

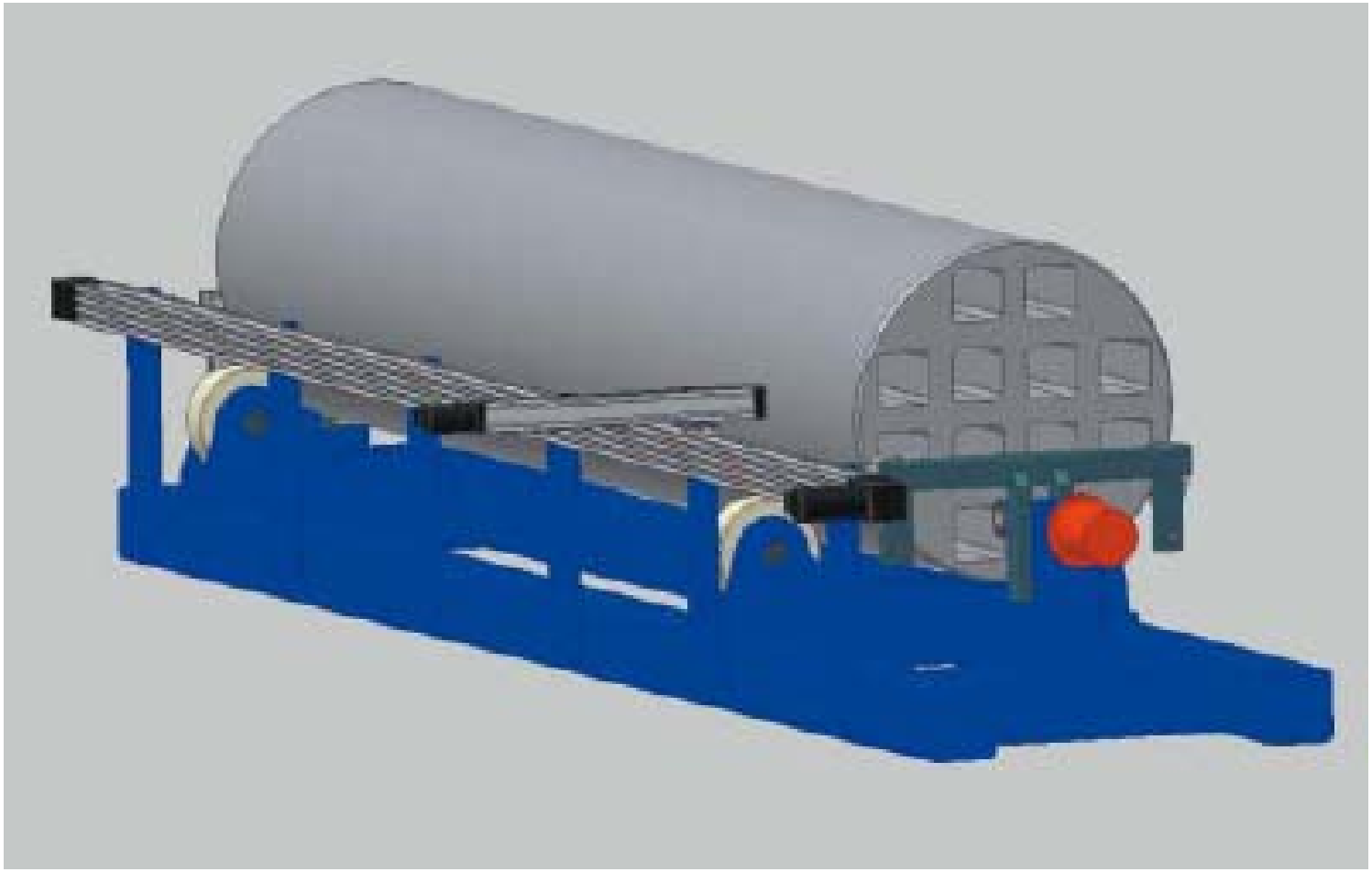
Acceptance criteria EN 12680-3

Severity level 4	Defect ≥level 1
Depth zone [mm]	200 -
% of WT	30
Largest defect in the surface zone [cm^2]	-
Largest defect in the core zone [m^2]	-
Total area of inspected surface %	30 (3.6 m^2)

Acceptance criteria EN 12680-3

Severity level 5	Defect ≥level 1
Depth zone [mm]	200 -
% of WT	35
Largest defect in the surface zone [cm^2]	-
Largest defect in the core zone [m^2]	-
Total area of inspected surface %	40 (4.8 m^2)

Inspection system for nodular cast iron insert at SKB Encapsulation laboratory (Rotator)



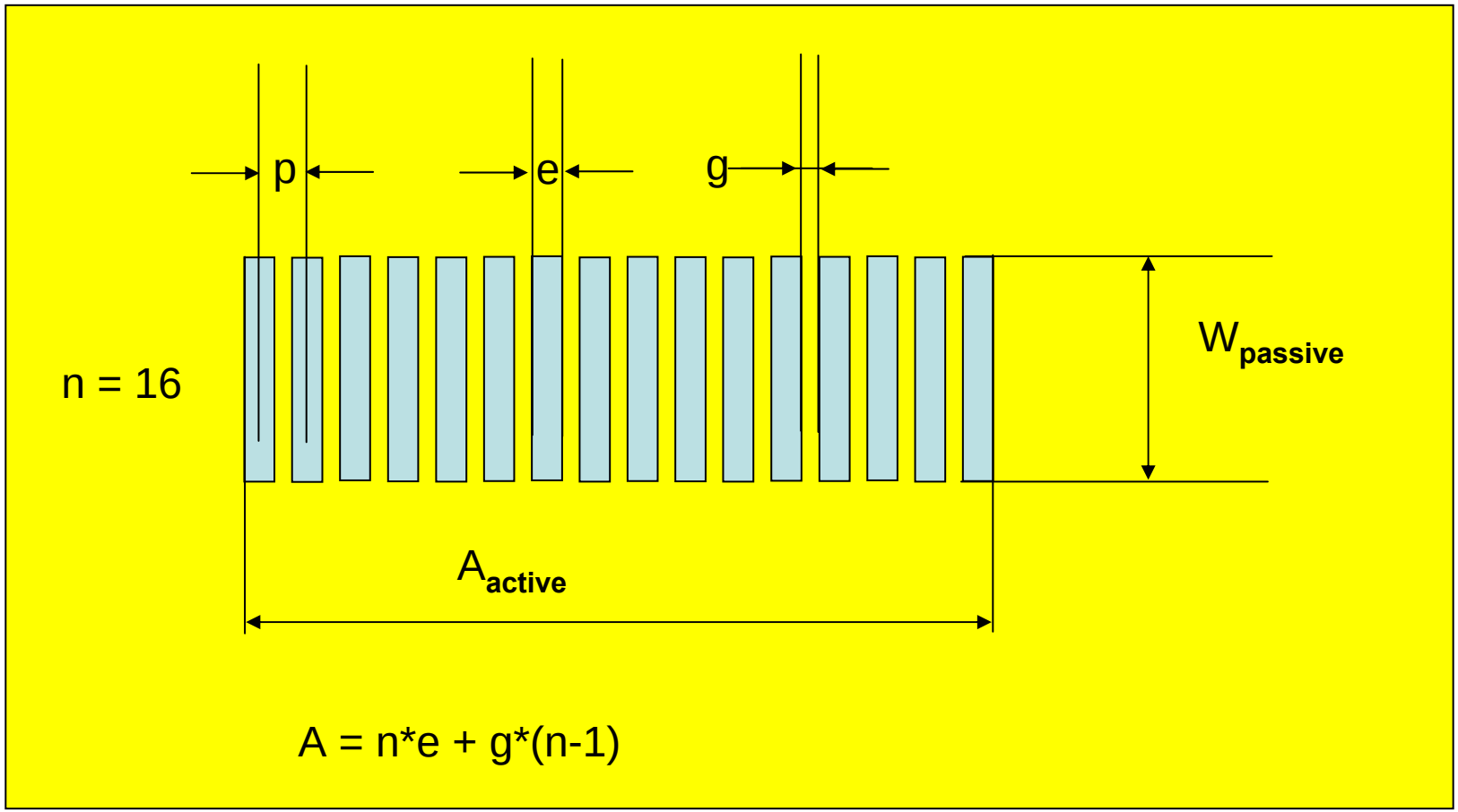
Inspection tehniques for nodular cast iron insert

Zone	Setup	Tested objects	Comments
A (surface zone)	70º TRL 2 MHz (5-40mm) Axial, Circumference Directions, Probes matched to curvature	Preliminary measurements has been started with reference specimen, One whole insert component	Reference specimen available Analysis software not capable to handle different directions
B (Intermediate zone)	PA 0º 2MHz, probe is not curved 112 elements in axial directions	**I 49 BWR** **I 49-2 BWR** **I 49-3 BWR** **IP5-3 PWR** **IP5-Top PWR**	
C(Channels)	Curved UT-PA 0º 1MHz, needs calculated specific delay laws because of geometrical changes	First measurements will be carried out in December	Calibration blocks has been designed, Probe holder is ready in 2 weeks
D (Near Corner areas)	PA 30º - 70º 3 MHz Several wedges	First measurements in December	Probeholder is ready in 2 weeks, calibration blocks has been designed and manufactured
E (Support plate areas)	Isotope + detector plates	Will be tested at the end of 2006 or at the beginning of 2007	Reference specimen designed

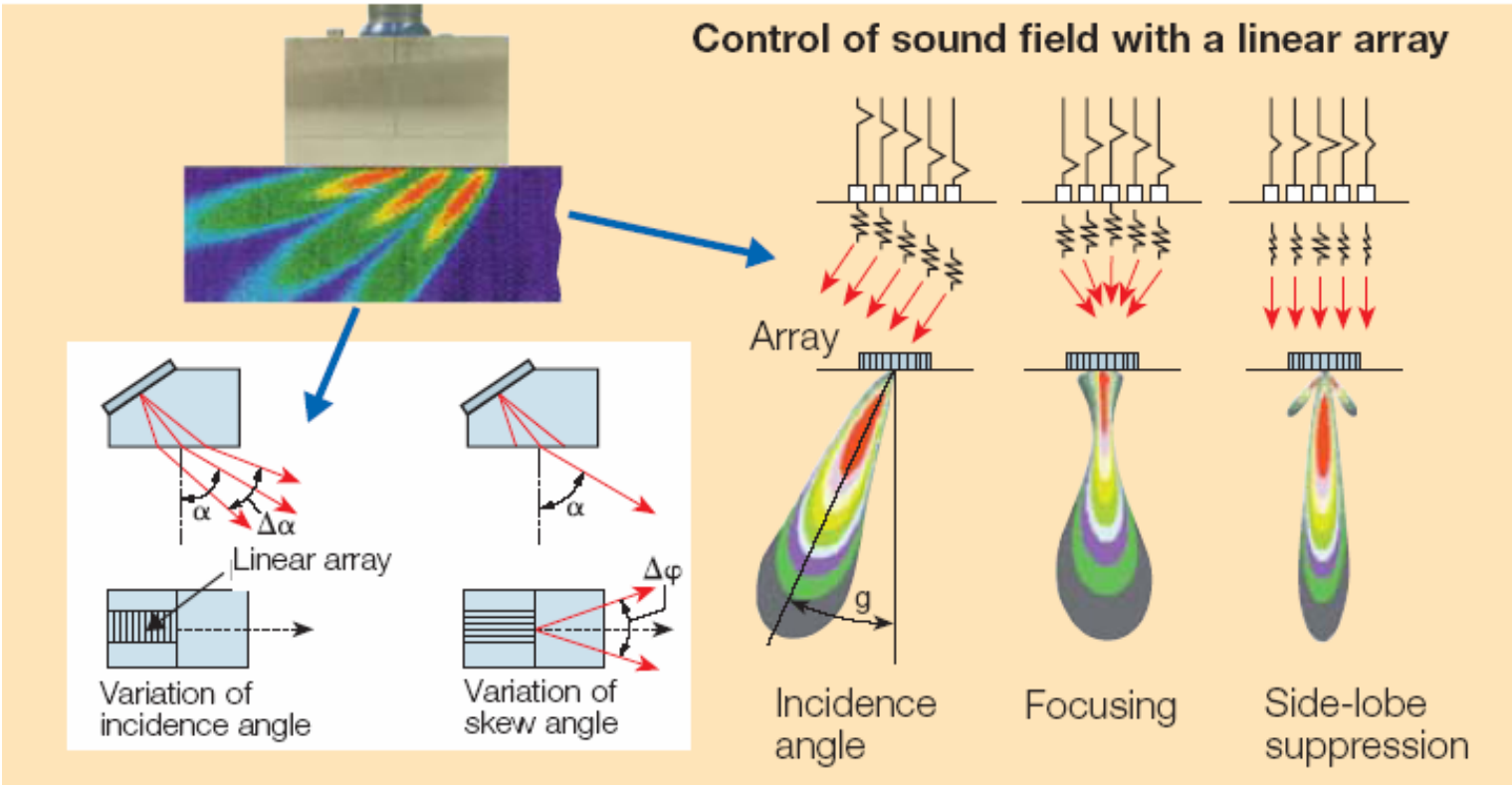
Interference areas (Areas which cannot be inspected properly)

- Several support plate areas (length 50 mm) which has been point welded for holding the steel channel in a shape during casting
- Partly top and bottom areas

Linear ultrasonic **p**hased **a**rray probe construction UT- PA

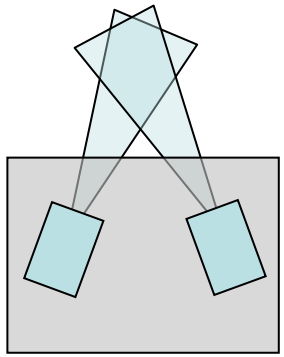


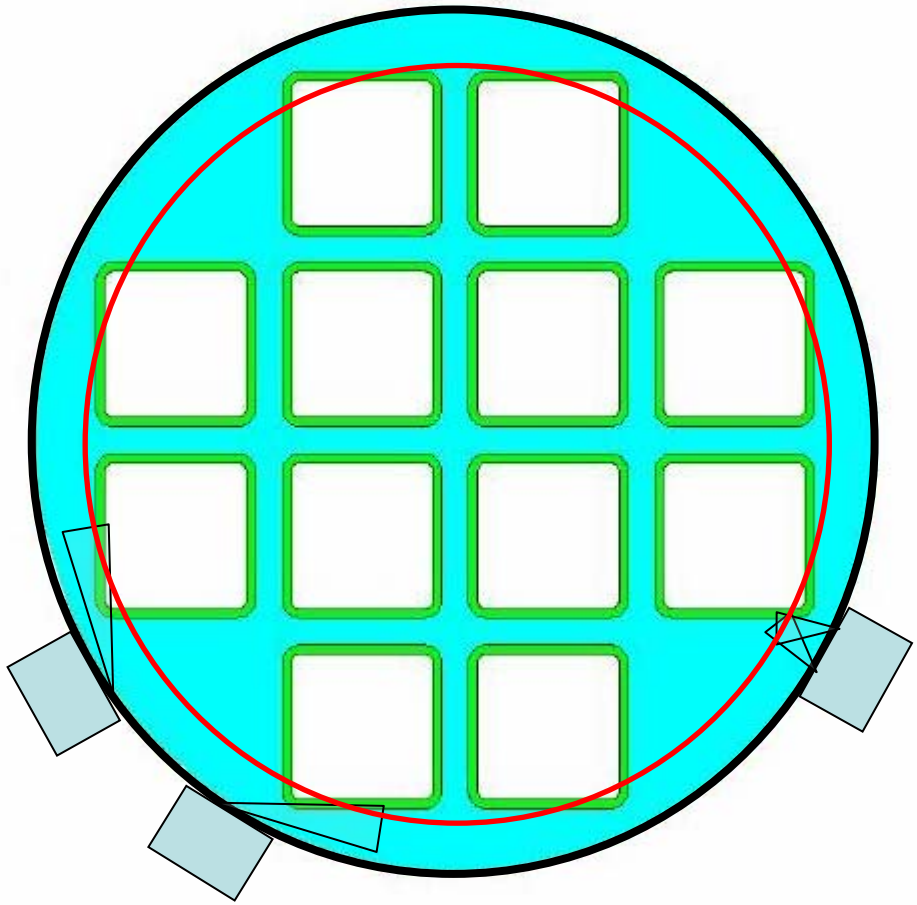
Linear phased array characteristics (ref. Intelligent NDT)

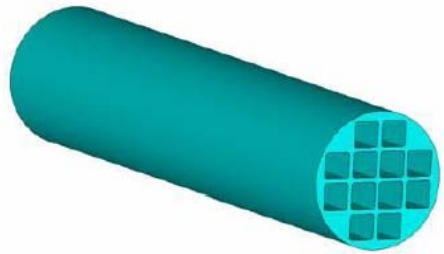


Surface Zone Inspection (Zone A)

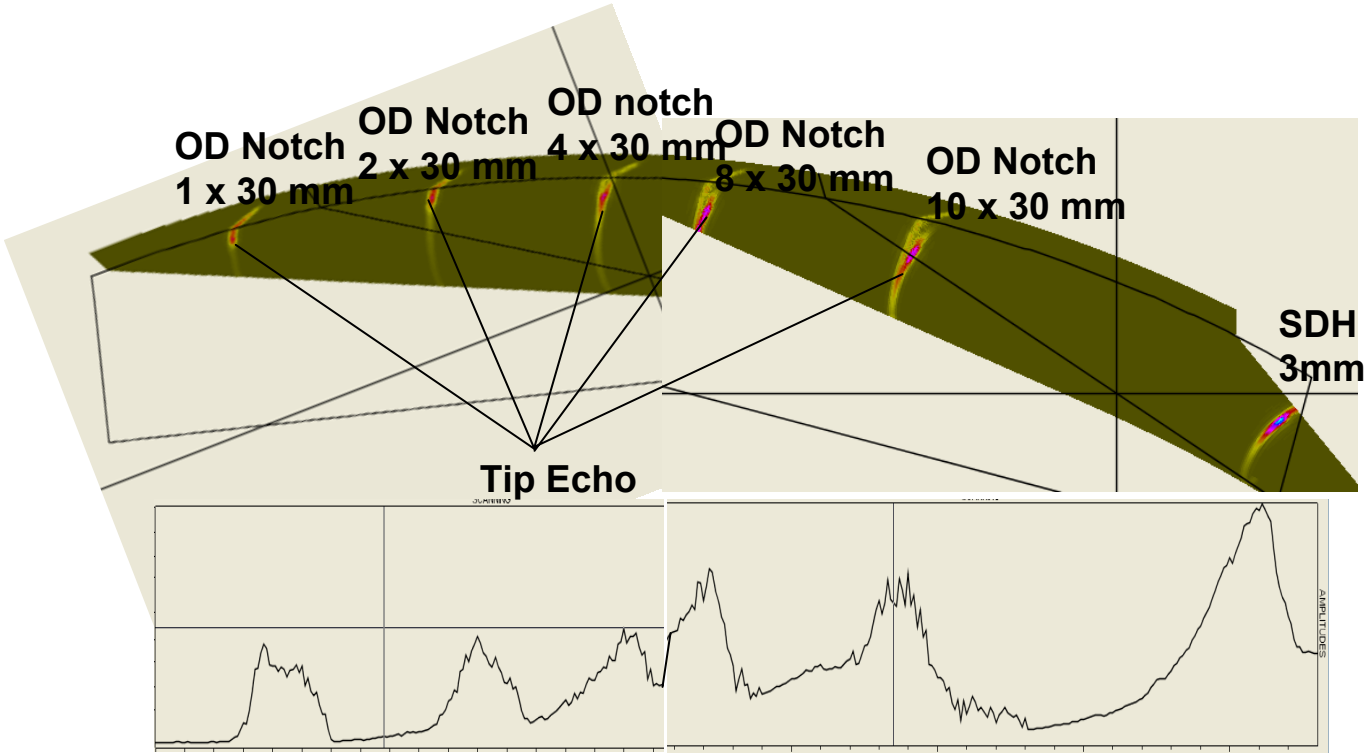
TRL70 2MHz PE, 4 Directions (2 x axial and 2 x circumferential)







Surface zone inspection modelling with Civa

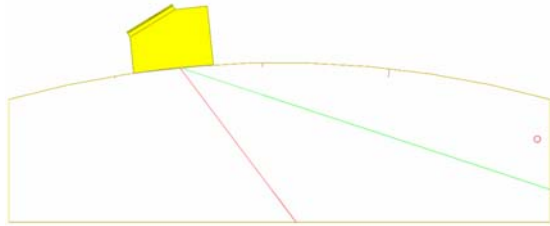


Postioning: normal to surface

Reflector	dB
SDH	0
1	-7.3
2	-6.6
4	-6.1
8	-2.7
10	-2.7

Postioning: perpendicular to surface (real positions)

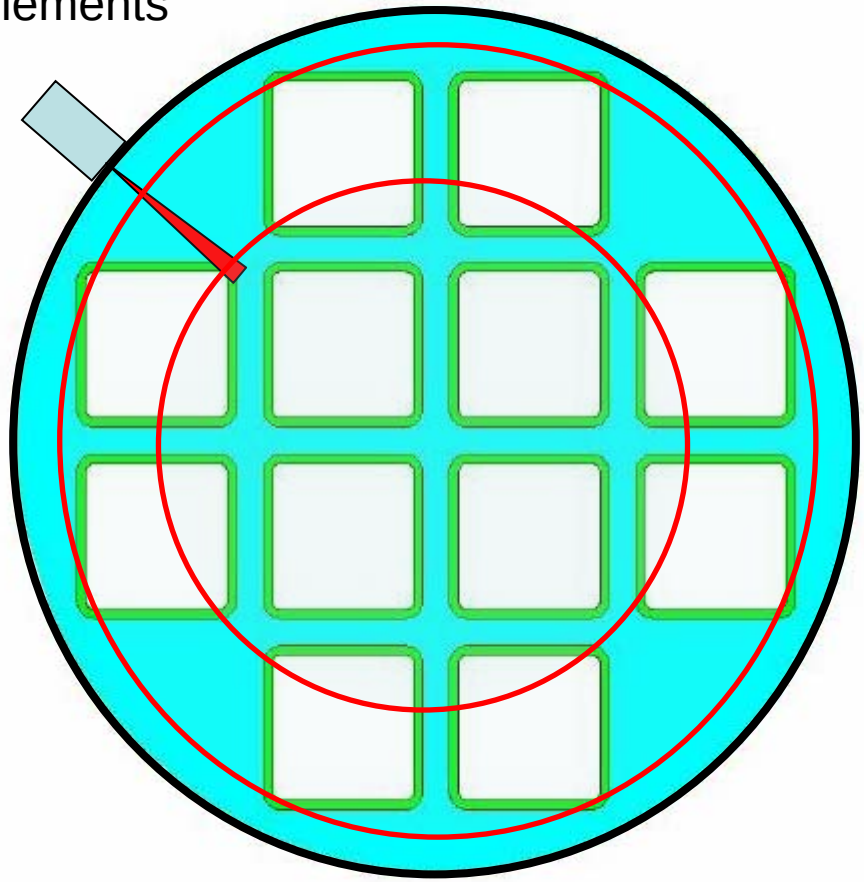
Reflector	dB
SDH	0
1	-5.9
2	-6.2
4	-9.5
8	-2.3
10	-2

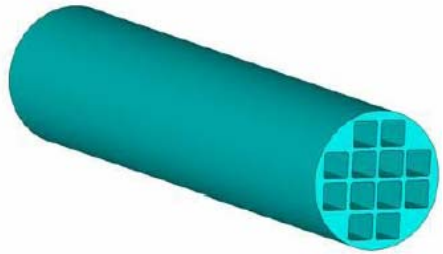


1 mm:n notch position 6.1

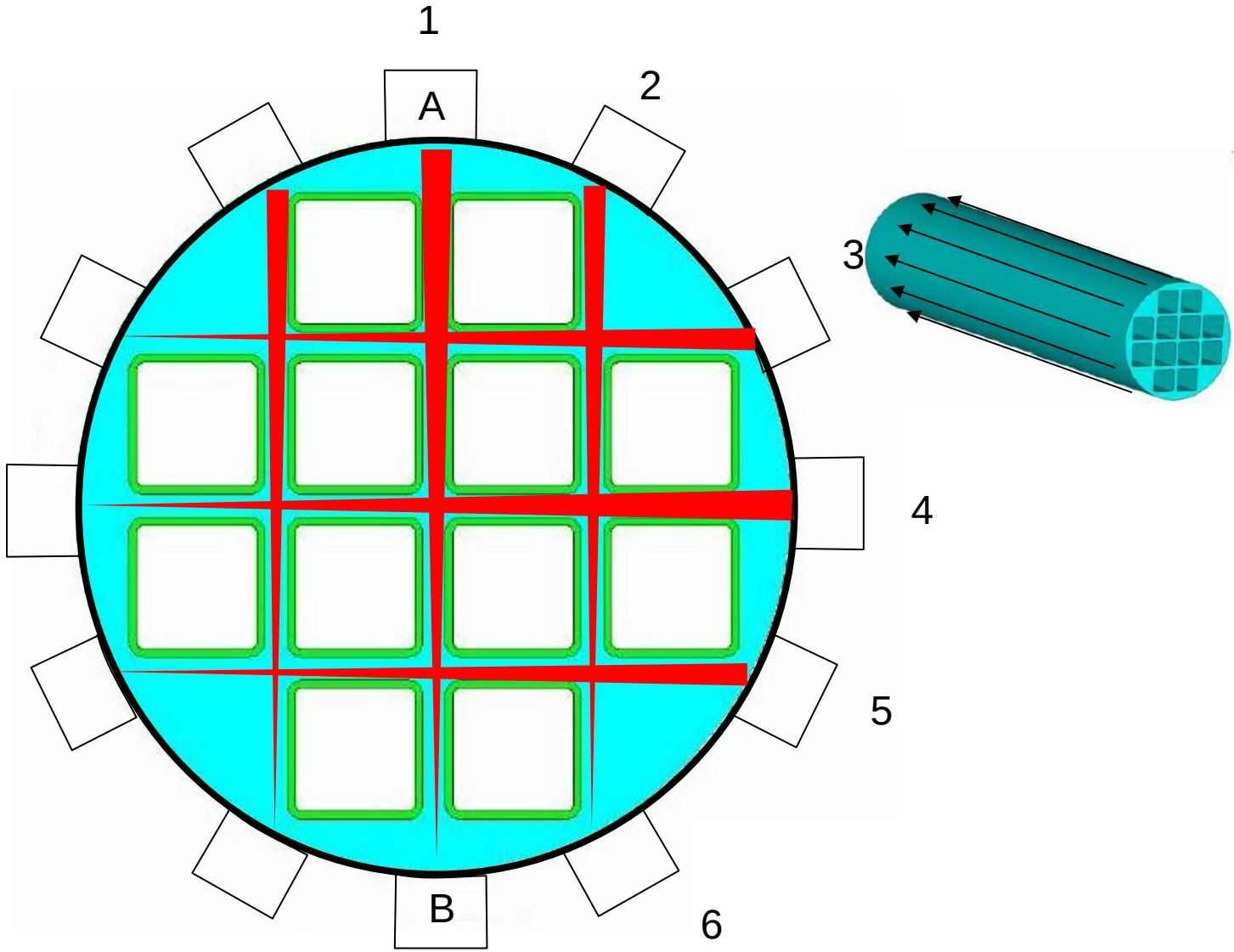
Intermediate Zone (Zone B1, B2, B)

PA 2MHz 0º linear scanning PE, 360º scanning along circumferential direction
1 scanning, 112 elements

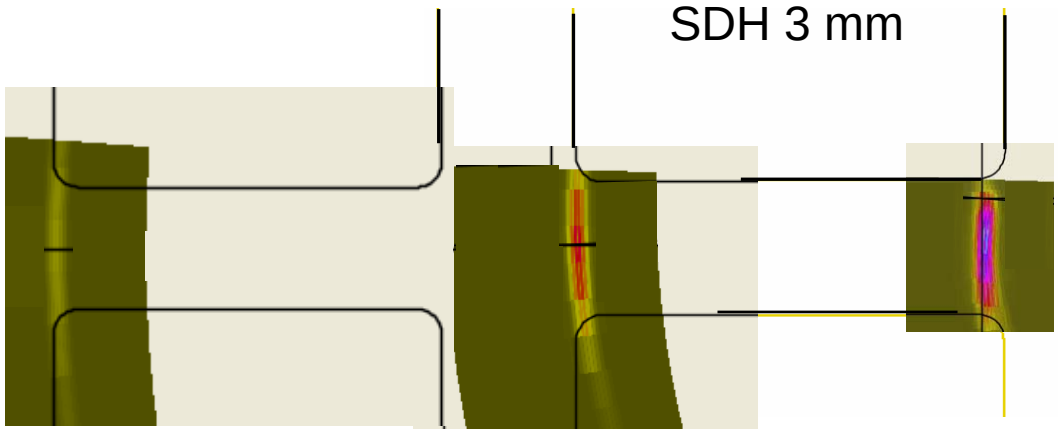


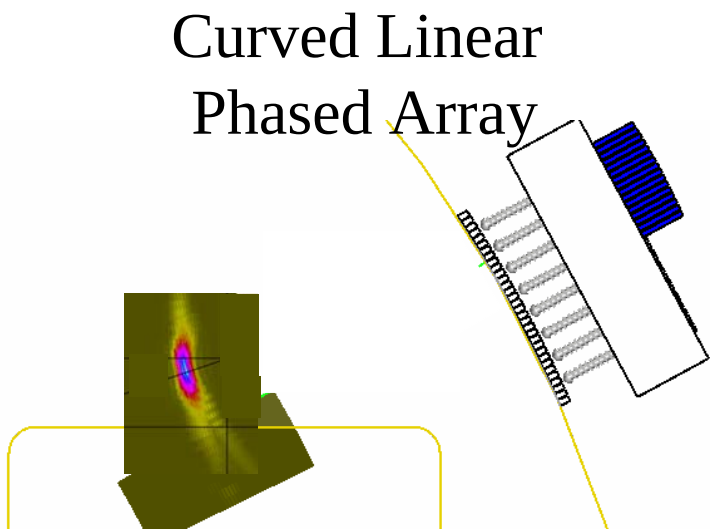


Zone between steel channels (Zone C)



Channel zone inspection modelling with Civa (Puls-echo)



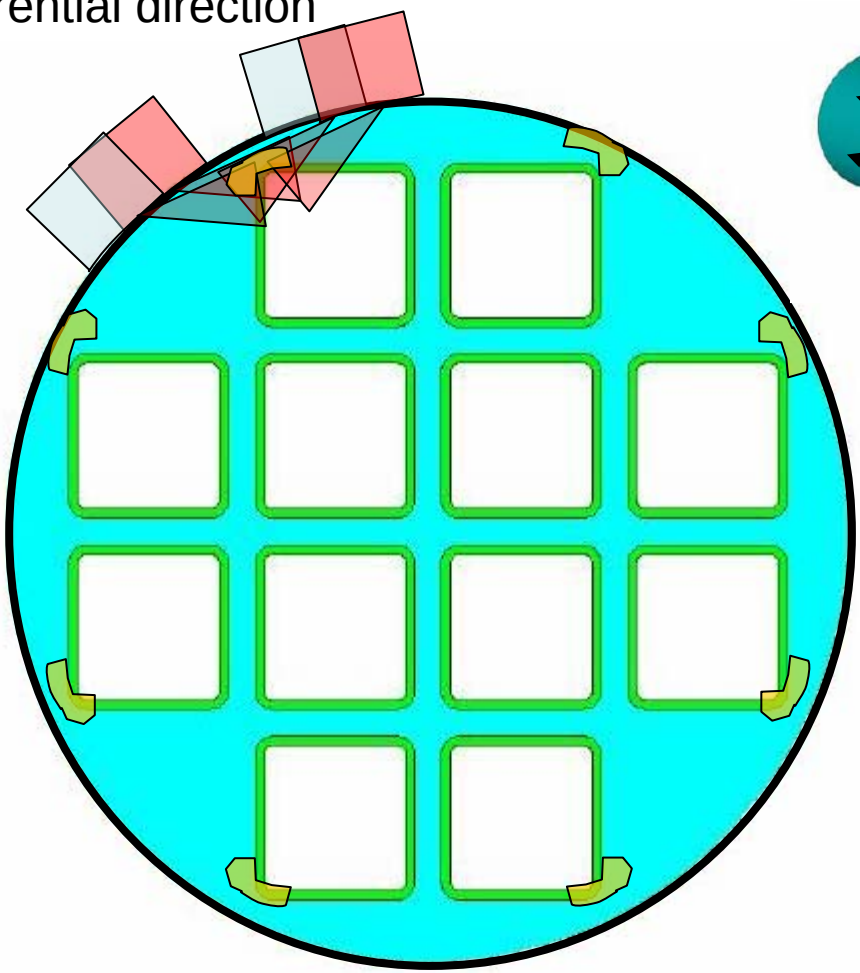


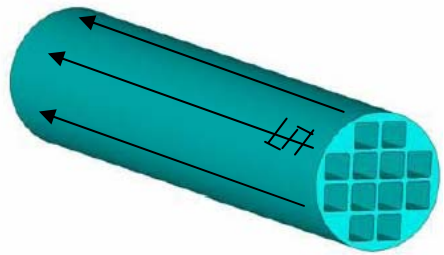
-20 dB -14 dB -8 dB 0 dB

Sound beam opens fast
Defect outer limits definition difficult – Sizing in the inner part of channel
The echoes from corners and walls of the steel channels disturb the analyses of the defects
Possibilities to solve this task: SAFT, OPT

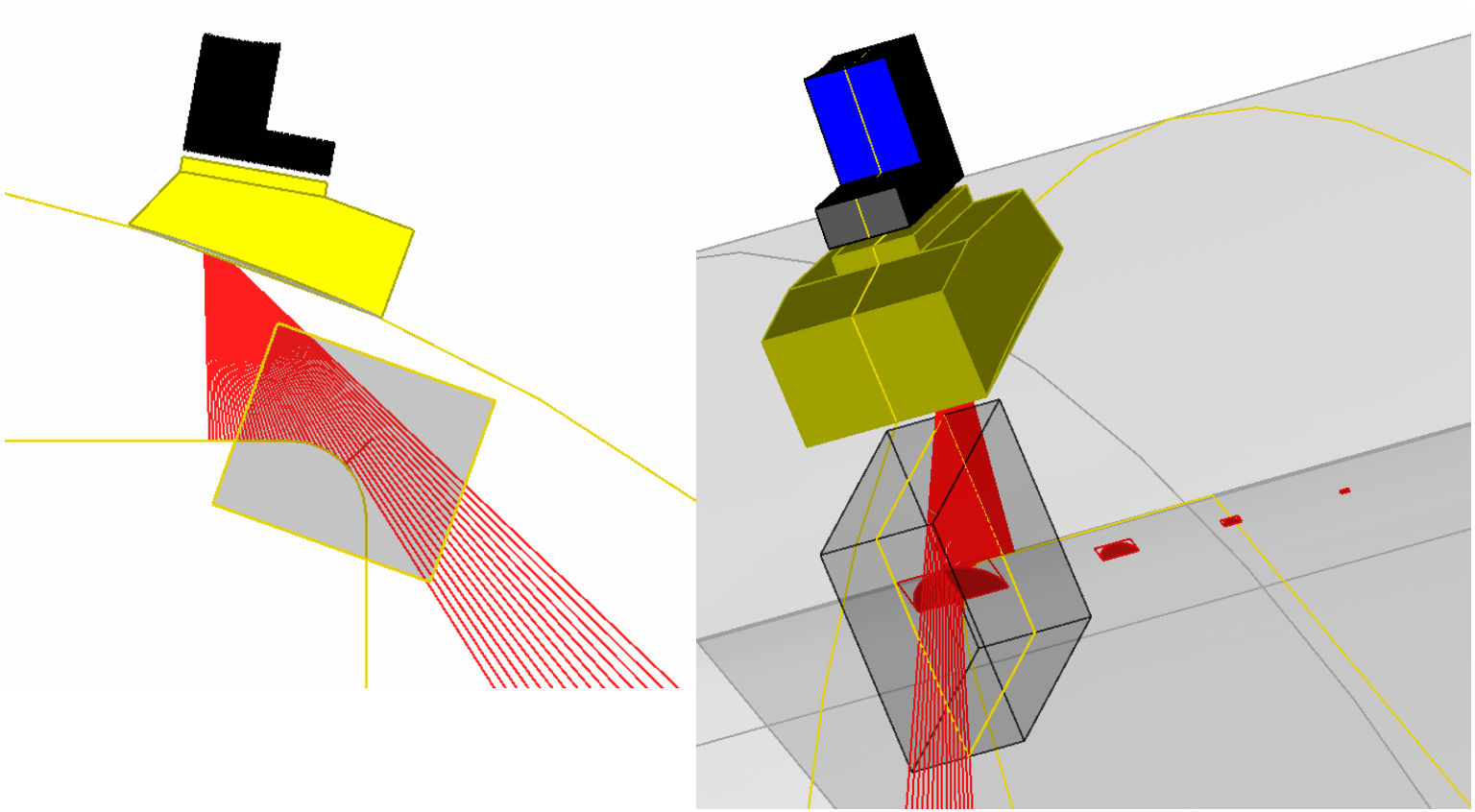
Nearest corner to surface zone (Zone D)

PA 3 MHz 25º - 70º angular scanning 2 x PE,
8 scanning along axial direction
Small scanning in circumferential direction
Fast measurement,
angular scanning will
slower the measurement,



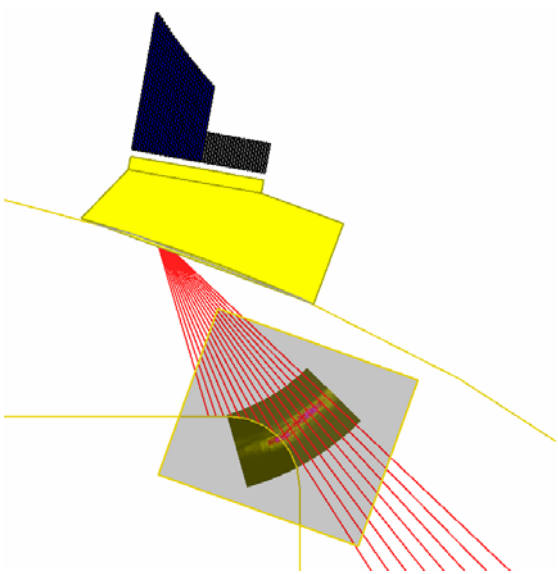


Nearest Corner to surface Inspection modelling with Civa

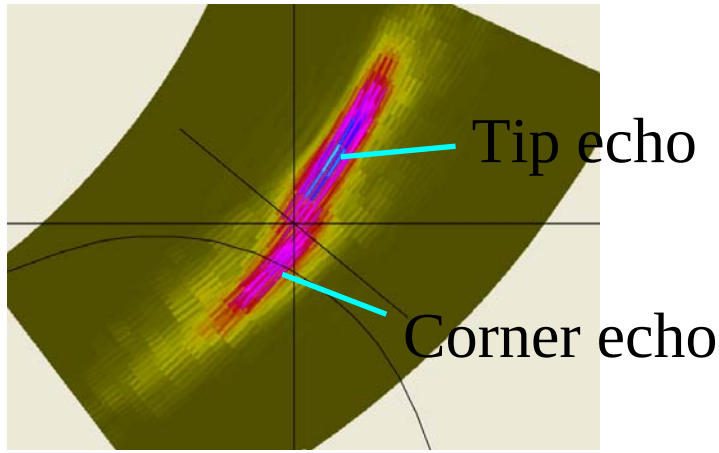


3MHz linear phased array 48 elements
Angular scanning will be tested from both sides of corner

Nearest Corner to surface zone inspection modelling with Civa



10 mm deep round Notch



Electronic scanning and small linear scanning combined
This method is under developing, first measurements will be carried out on December
Analysis will be carried out using suitable reference specimen
Improvements to more accurate analysis gives more advanced data analysis software like SAFT, Aousctic holography, Sampling Phased Array or similar

Effect of Nodularity to Ultrasonic Inspection

- **Velocity**
- **Defect location**
- **Defect sizing**
- **Thickness evaluation**
- **Channel location changes**
- **Measurement of nodularity**
- **- on the surface (surface waves)**
- **- back wall echo if received**

Qualification items for NDT

- ENIQ
- Input data for qualification
- Training of inspectors
- Qualification specimen (real or simulated defects types in the qualfication specimen)
- Open / Blind trials
- Inspection instruction / Technical Justification
- Scope of the inspection
- Shperodization of nodules during cooling in the casting
- Inhomogenity in the castings (which areas?)
- Local regulations for qualifications (Finnish, Swedish intructions for qualifications)
- General standards (EN-12680-3, EN-473, etc.)
- Authority instructions (SKI (SKIFS) SSI, STUK (YVL, or comparable instructions)

Input-Data for Insert

- Manufacturing defects types
- (Handling defects)
- POD-Curves (Probability of Detection) which are investigated in co-operation project with the BAM, SKB and Posiva
- Acceptance criteria for NDT (preliminary results at the end of 2006)

Conclusion

- Preliminary acceptance criteria will be ready to the end 2006,The results showed that the accepted defect sizes were quite large and the for detection this should not be a problem
- Preliminary inspection techniques has been chosen in order to study their suitability for detection and sizing of the defects
- Preliminary Inspection procedure will be produced when all the reference specimen are available
- During 2007 – 2009 the POD curves will be produced

POSIVA

The ONDRAF/NIRAS Waste Overpack and Supercontainer design

(post-conditioning the Belgian HLW and SNF)

*Presentation by **Chris De Bock** (O/N)*

Workshop and Technical Meeting on
Design and Assessment of Packages for Radioactive Waste
Hotel Marijke, Bergen (NL)
November 21st 2006

ONDRAF/NIRAS

Presentation Overview

- PART 1: introduction (the context)
  - ✓ national context
  - ✓ waste inventory
  - ✓ reference disposal site

- PART 2: disposal concept for HLW / SNF
  - ✓ safety strategy and long-term safety functions
  - ✓ Supercontainer

- PART 3 : the design and fabrication of the overpack
  - ✓ functions and constraints of the overpack
  - ✓ components
  - ✓ overpack design for:
    - vitrified HLW
    - UOX SNF (options CRD and IOD)
    - MOX SNF
  - ✓ Closure weld

Concluding thoughts

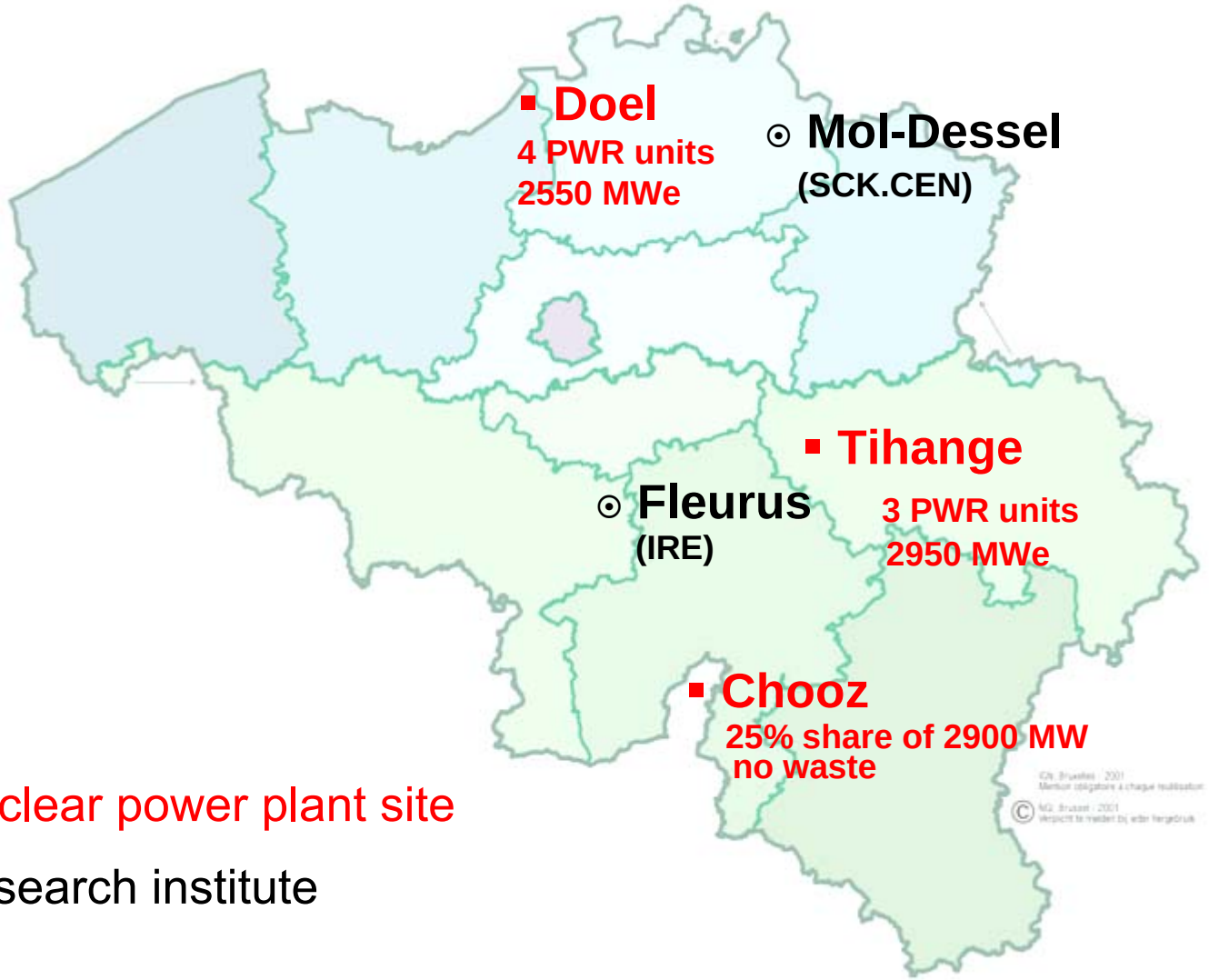
ONDRAF/NIRAS

PART 1

INTRODUCTION (CONTEXT)

ONDRAF/NIRAS

BELGIUM: MAIN NUCLEAR SITES



- nuclear power plant site
- ⊙ research institute

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WASTE INVENTORY: 3 CATEGORIES

	Low level	Medium level	High level
Short half-life (30 years or less)	A	A	C
Long half-life (30 years or more)	B	B	C

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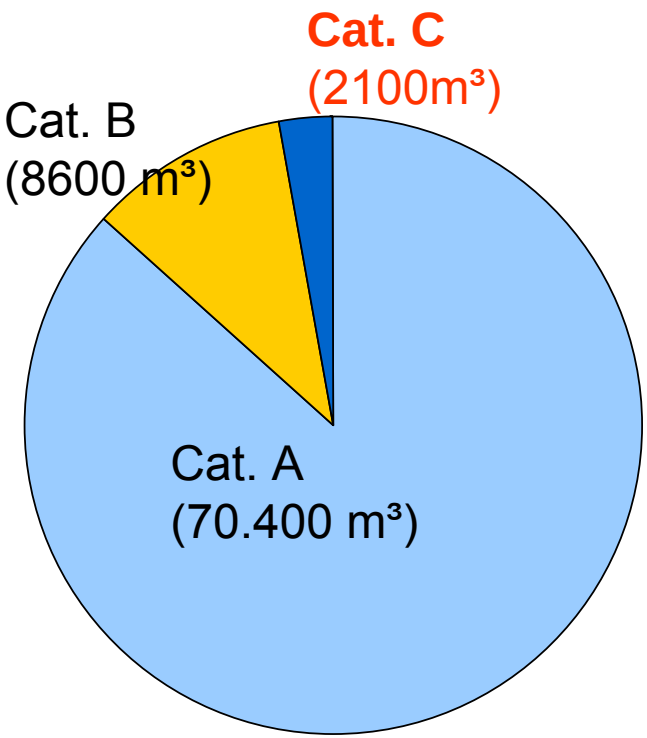
WASTE INVENTORY: 2 SCENARIOS

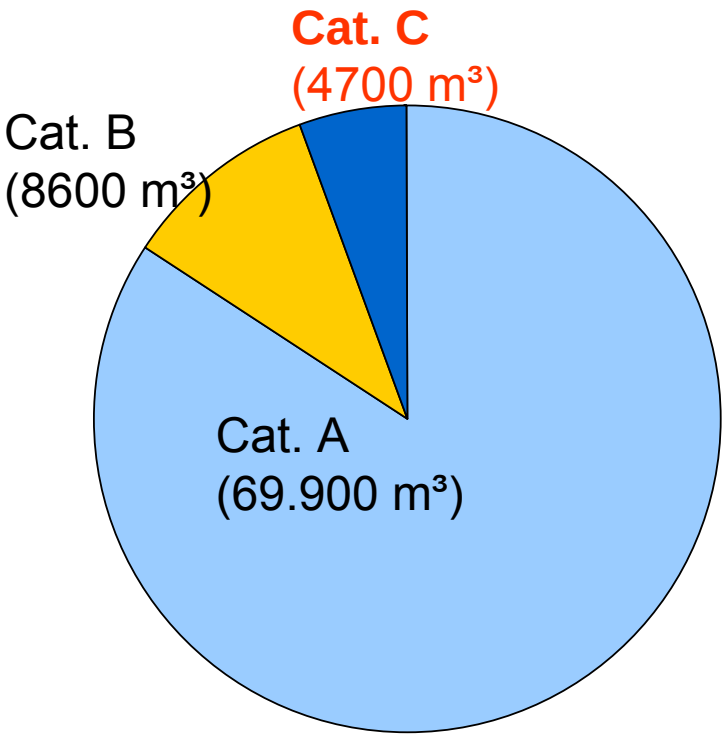
Resumption of SNF reprocessing



4,000 vitrified HLW canisters

6,500 hulls & ends canisters (= cat.B)

No further SNF reprocessing



10,000 spent fuel assemblies
400 vitrified HLW canisters

800 hulls & ends canisters (= cat.B)

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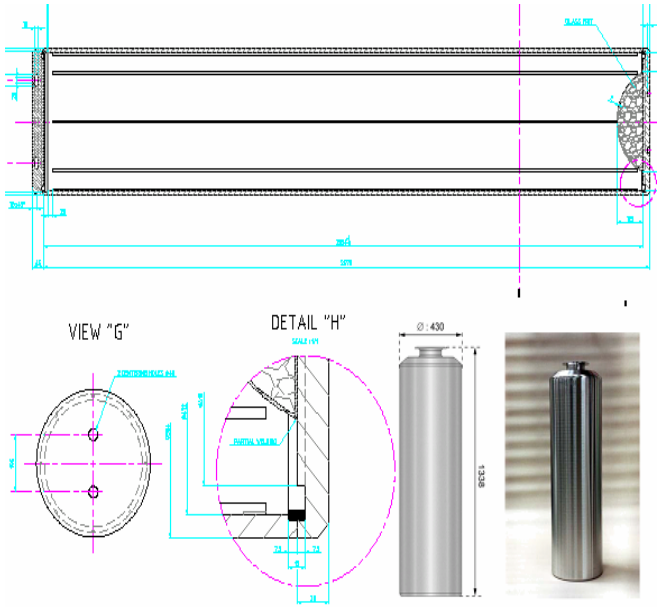
HLW INVENTORY DETAILS

- Vitrified in borosilicate glass at the La Hague plant



- Inventory:
  - No further reprocessing: 420 canisters (Doel 1 and 2)
  - Resumption of reprocessing: 3919 canisters

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SNF INVENTORY DETAILS

FUEL MATRIX	FUEL DIMENSIONS		NPP of origin	INVENTORY (# of assemblies)
	GRID	ACTIVE FUEL LENGTH		
UOX	14 X 14	8 ft	Doel 1 and 2	1615
	15 x 15	12 ft	Tihange 1	4690
	17 x 17	12 ft	Doel 3, Tihange 2	
	17 x 17	14 ft	Doel 4, Tihange 3	3187
MOX	17 x 17	12 ft	Doel 3, Tihange 2	145

Note: “exotic” fuel from test reactors and BR-3 pilot NPP will not be reprocessed (19 assemblies)

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BELGIUM: MAIN PLAYERS

- Nuclear industry/research:
  - Utility owning the NPPs: ELECTRABEL (owned by Suez)
  - Architect/Engineer: TRACTEBEL (owned by Suez)
  - Engineering:
    - BELGONUCLEAIRE (50% Suez, 50% Belgian State)
    - BELGATOM (80% Suez, 20% Belgonucleaire)
  - Ownership of the nuclear fuel: SYNATOM (50% Suez, 50% Belg. State)
  - Commercial production of fuel: FBFC (owned by AREVA)
  - Nuclear energy research institute: SCK.CEN (state-owned)
- Licensing authority:
  - Federal agency for nuclear control: FANC / AFCN (founded 1994)

    Technical support provided by FANC/AFCN: AVN (private non-profit)
- Nuclear waste management:
  - Long-term management: ONDRAF/NIRAS (public body, founded 1980)
  - waste conditioning: BELGOPROCESS (owned by ONDRAF/NIRAS)

BELGIUM: DEVOLUTION OF AUTHORITY

- Federal state: 3 Regions
  - Flanders: 6.0 million inhabitants
  - Wallonia: 3.4 million inhabitants
  - Brussels: 1.0 million inhabitants
- 10 Provinces
- 589 Municipalities

Devolution of fields of authority:

- Federal state: occupational safety, nuclear industry and safety
- Regions: environmental protection (except nuclear aspects)
- Provinces: limited authority
- Municipalities: high authority on all matters on municipal territory

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IMPORTANT GOVERNMENT DECISIONS

1. No new SNF reprocessing contracts until further notice (1993)
   ⇒ ONDRAF/NIRAS should take account of both reprocessing scenarios on an equal basis
2. Law on nuclear phase-out (2003)
   ⇒ Belgian NPPs stop between 2015-2025
3. Decision to go ahead with the construction of a cat. A disposal facility in the zone reserved for nuclear activities in the municipality of Dessel (June 2006)

! For cat. B and C, there is no official position regarding
- Disposal or long-term interim storage
- Host rock layer for disposal
- Site of the repository

⇒ To continue its R&D work, ONDRAF/NIRAS has opted to work with a "reference" site and expects that results will be, to some extent, transferable to the finally selected site.
Belgium has only clay formations as possible host rock.

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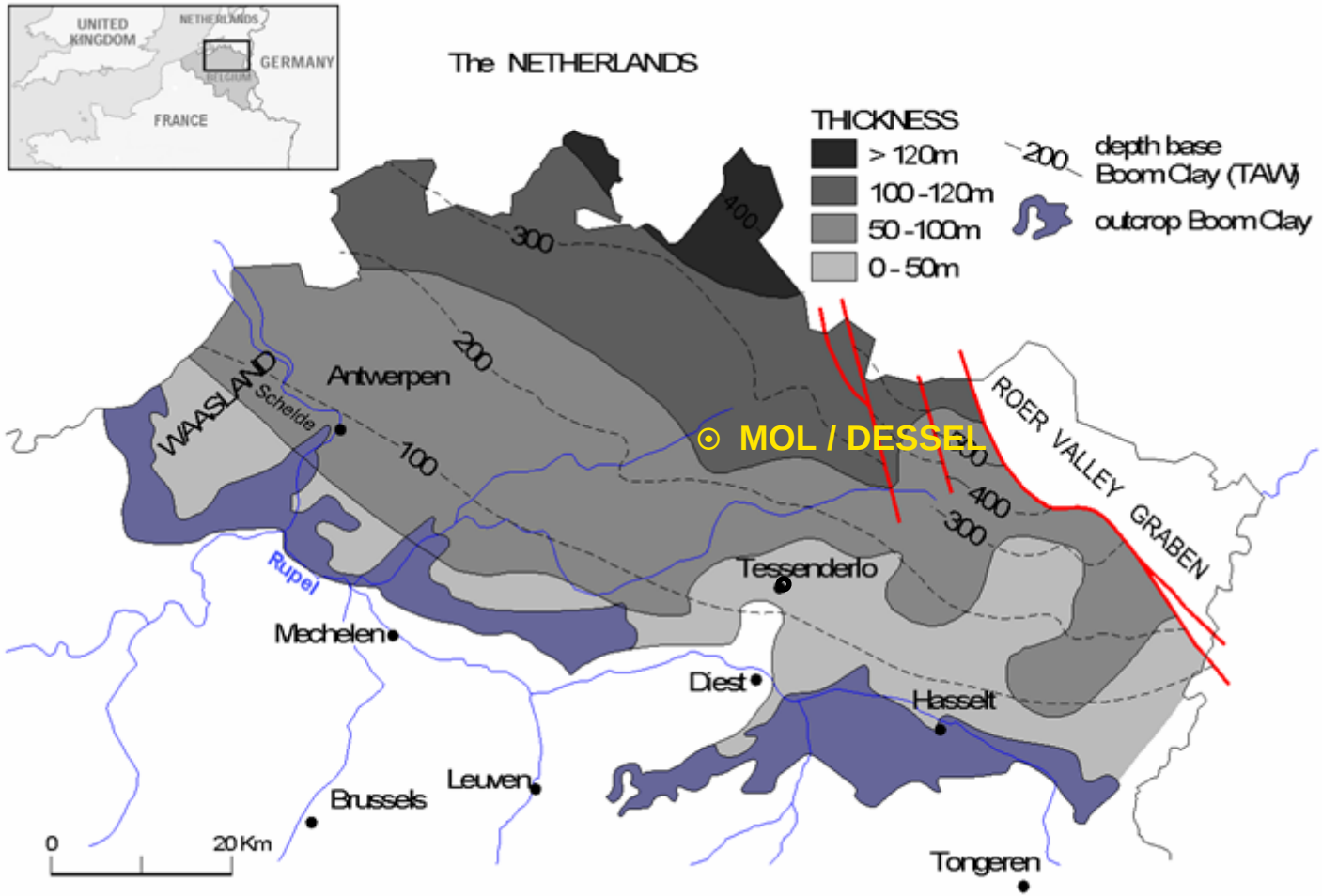
REFERENCE SITE AND HOST ROCK



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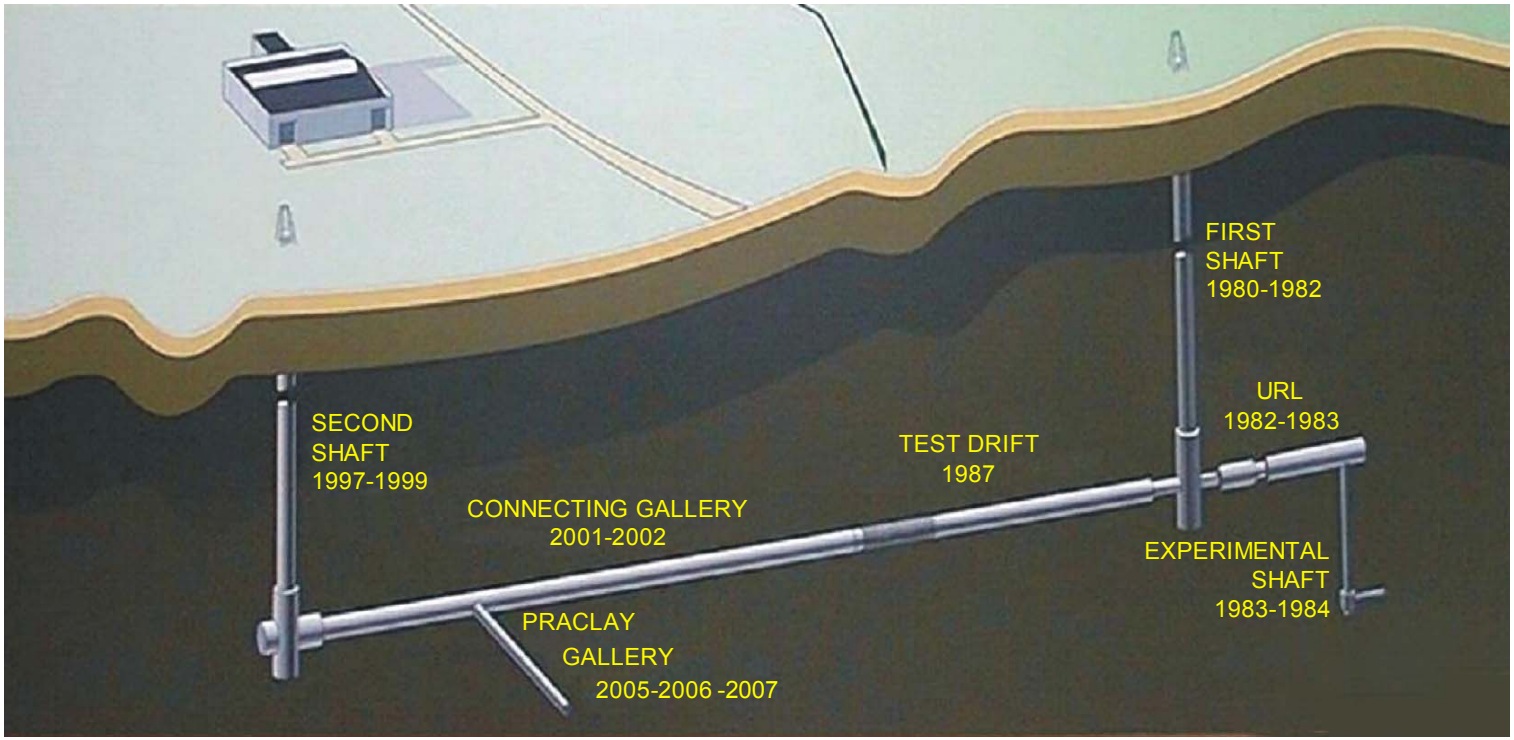
UNDERGROUND RESEARCH LAB

- Situated on the SCK.CEN site in Mol/Dessel
- In the median plane of the Boom clay (230 m deep)
- Managed by EURIDICE
- URL construction and research since 1980
- Presently major infrastructure expansion: PRACLAY gallery for investigation of thermal effects on host rock and gallery structure



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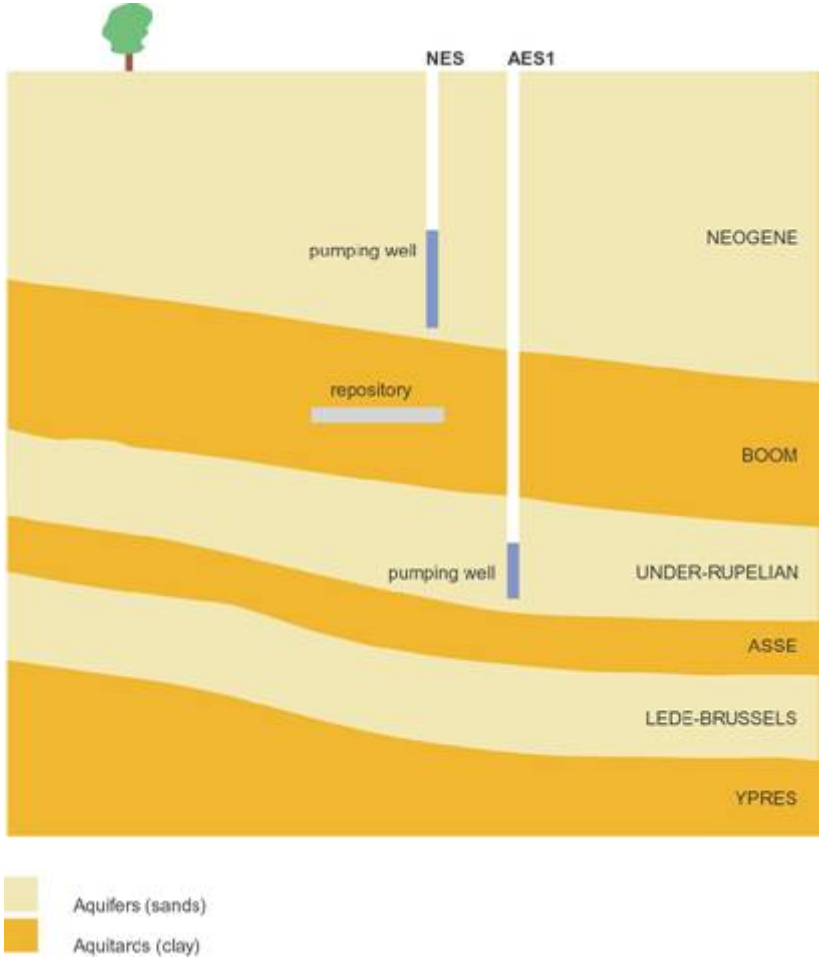
BOOM CLAY MAIN CHARACTERISTICS

- General host rock characteristics:
  - poorly-indurated (plastic) clay of about 20 million years old
  - clay mineralogy: mainly illite + smectites (20%), kaolinite (15%)
  - high pyrite content (2%)
  - high organics content (5%)

- At Mol/Dessel:
  - depth: 230 m (median plane)
  - thickness: 100 m
  - 1% NNE downward slope
  - aquifer used for drinking water
  - lithostatic pressure: 4.5 MPa
  - hydrostatic pressure: 2 MPa



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PART 2

DISPOSAL CONCEPT FOR HLW / SNF

ONDRAF/NIRAS

HISTORY OF DESIGN OF REPOSITORY

- SAFIR 2 concept (2001)
- NEA Peer Review
- Re-design period in GTA working group (2002-2003)
- Selection of Supercontainer as disposal concept for HLW and SNF, based on multi-criteria analysis
- Work-out of the design of integrated B&C repository

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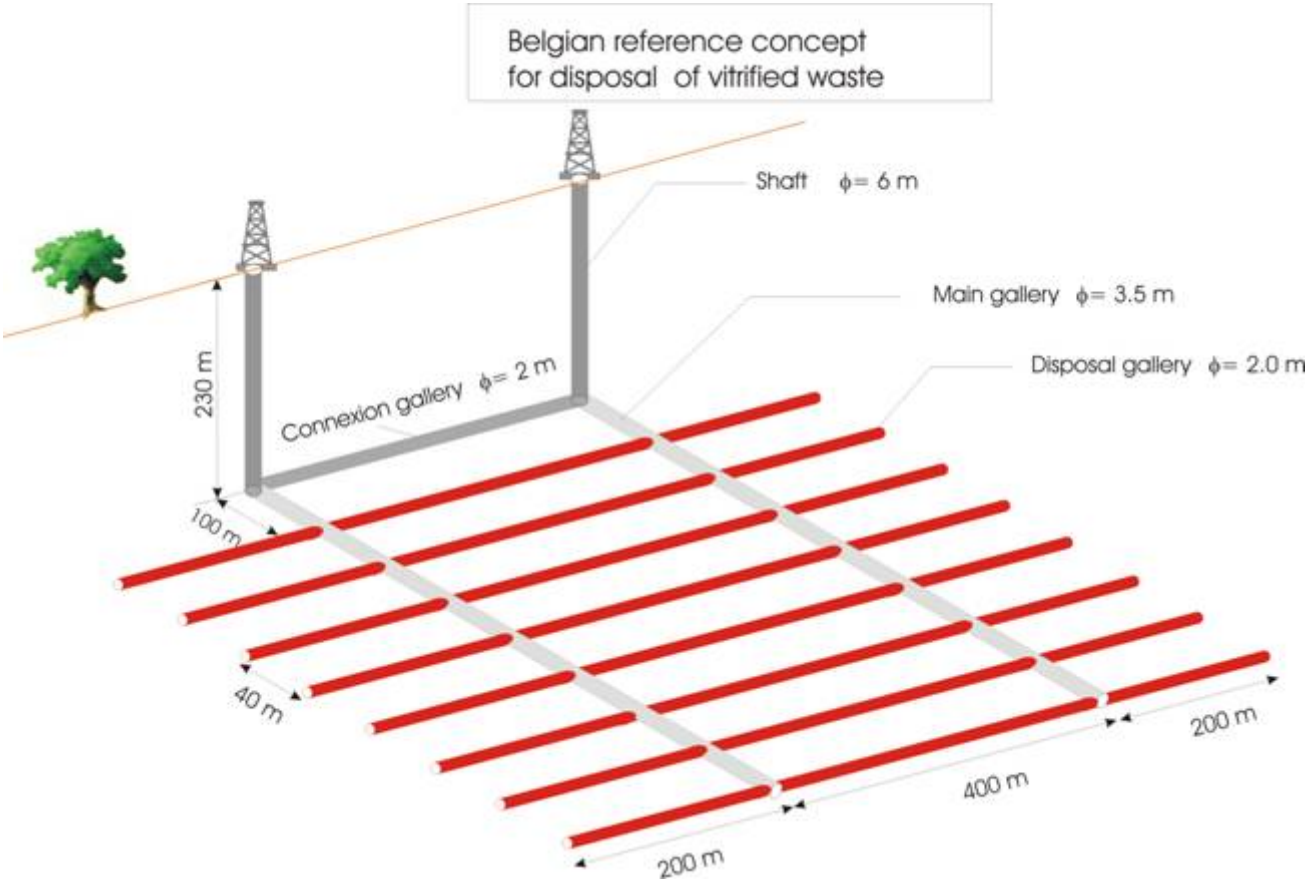
SAFIR 2 REPOSITORY CONCEPT



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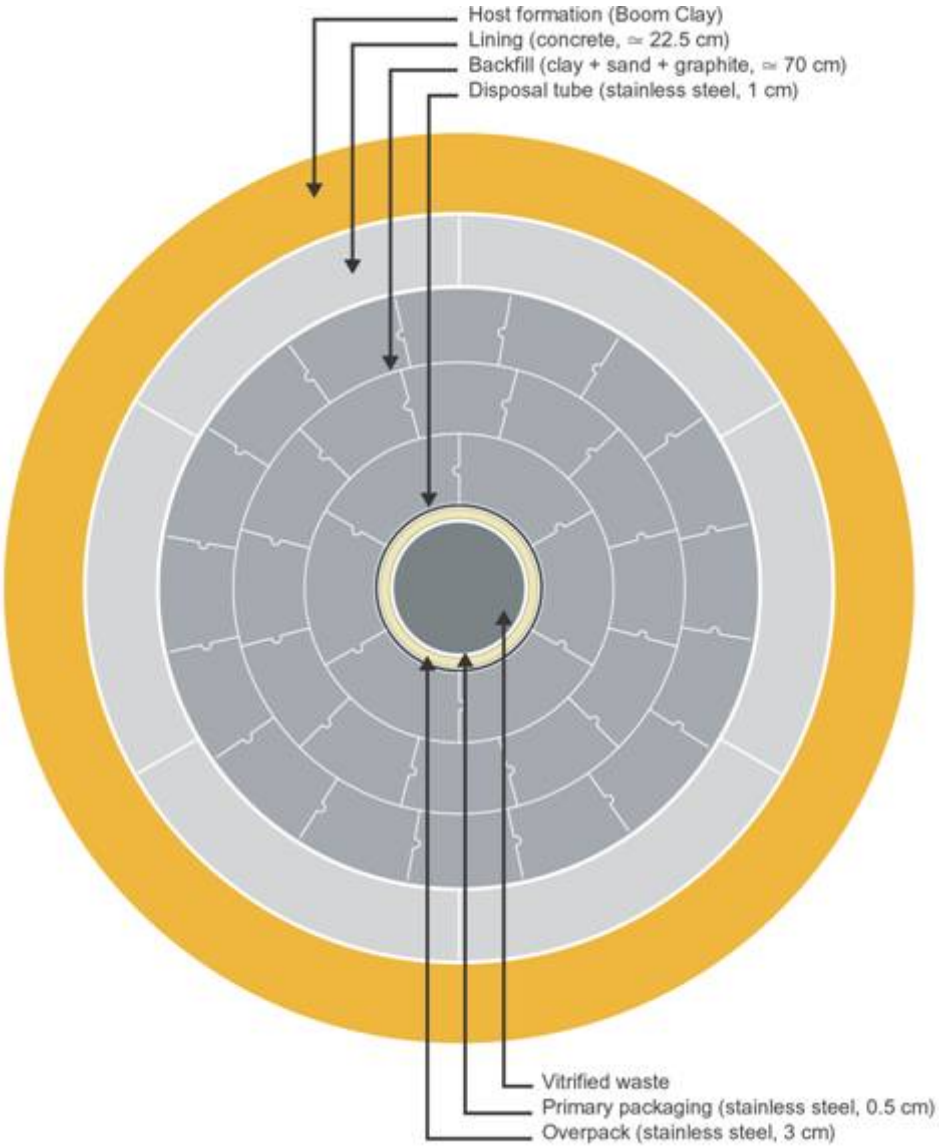
SAFIR 2 VHLW DISPOSAL CONCEPT



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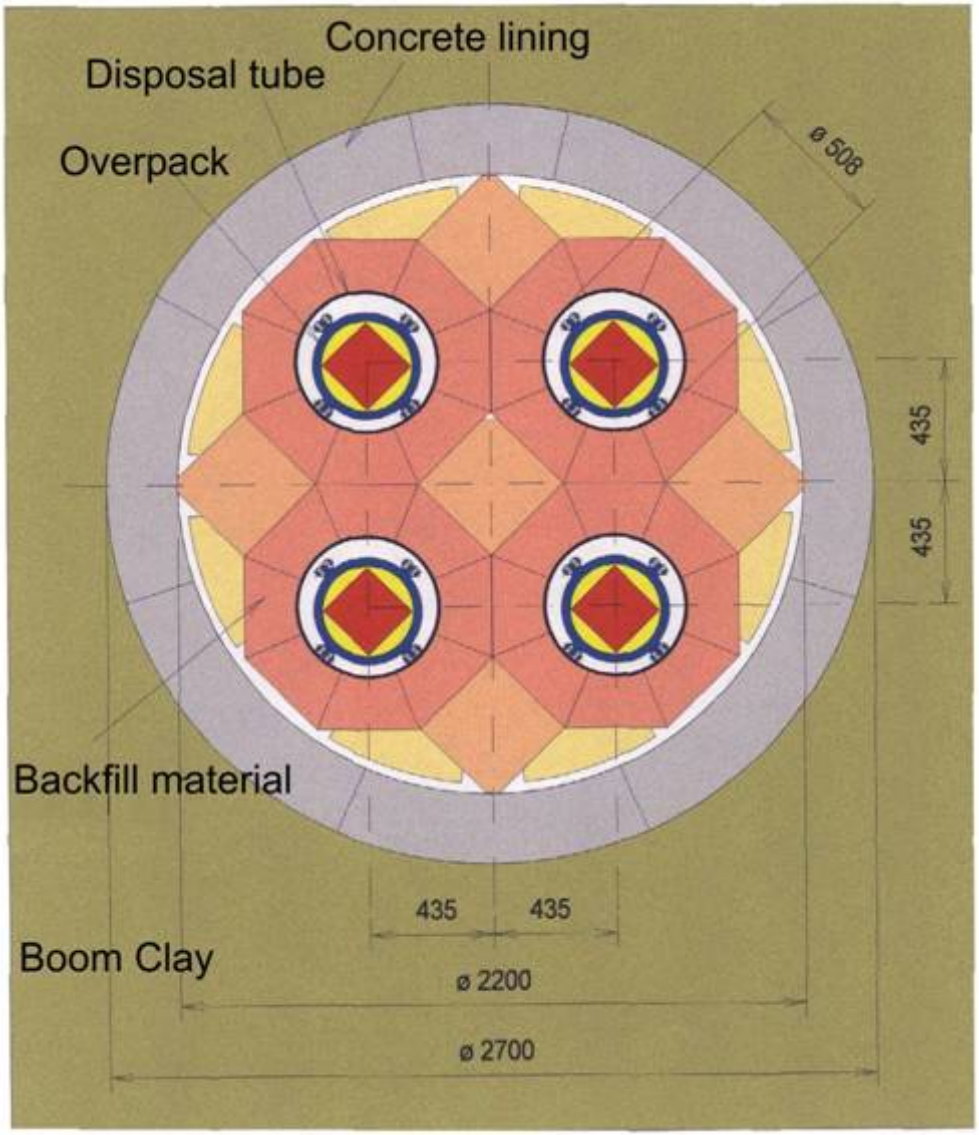
SAFIR 2 SNF DISPOSAL CONCEPT



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NEA PEER REVIEW

- *"The studies made so far are relevant and provide an excellent platform for continuing with the program."*
- *"The Belgian program for the disposal of high-level and long-lived radioactive waste is well developed and sufficiently advanced to address the siting issue."*
- *"However, there are some doubts about the reliability of the system chosen (disposal tube, hydration system) for emplacing the wastes and the associated engineered barriers. These doubts were confirmed by ONDRAF/NIRAS and their experts"*

  ⇒ Technical feasibility issues

  ⇒ Re-engineer the disposal concept

ONDRAF/NIRAS

RE-DESIGN OF THE DISPOSAL CONCEPT

- Repository design requires a systematic approach:

  Step 1: What do we want the system to do ?

  Is based on long-term safety functions (reflect high level safety, boundary conditions and strategic choices)

  ⇒ disposal concept of the waste types

  Step 2: How do we want the system to do this ?

  Is based on operational concerns

  ⇒ repository design

- Safety functions in Boom Clay:
  - Uncertainty of Boom Clay characteristics at elevated temperature
  - need for a watertight isolation during the thermal phase

    ⇒ C function (see next slide)

- Redesign the disposal concept for vitrified HLW (VHLW)
  - the work of the GTA team in 2002-2003
  - Development of 3 alternative disposal concepts for VHLW
  - Consideration of technical variants based on basic disposal concepts

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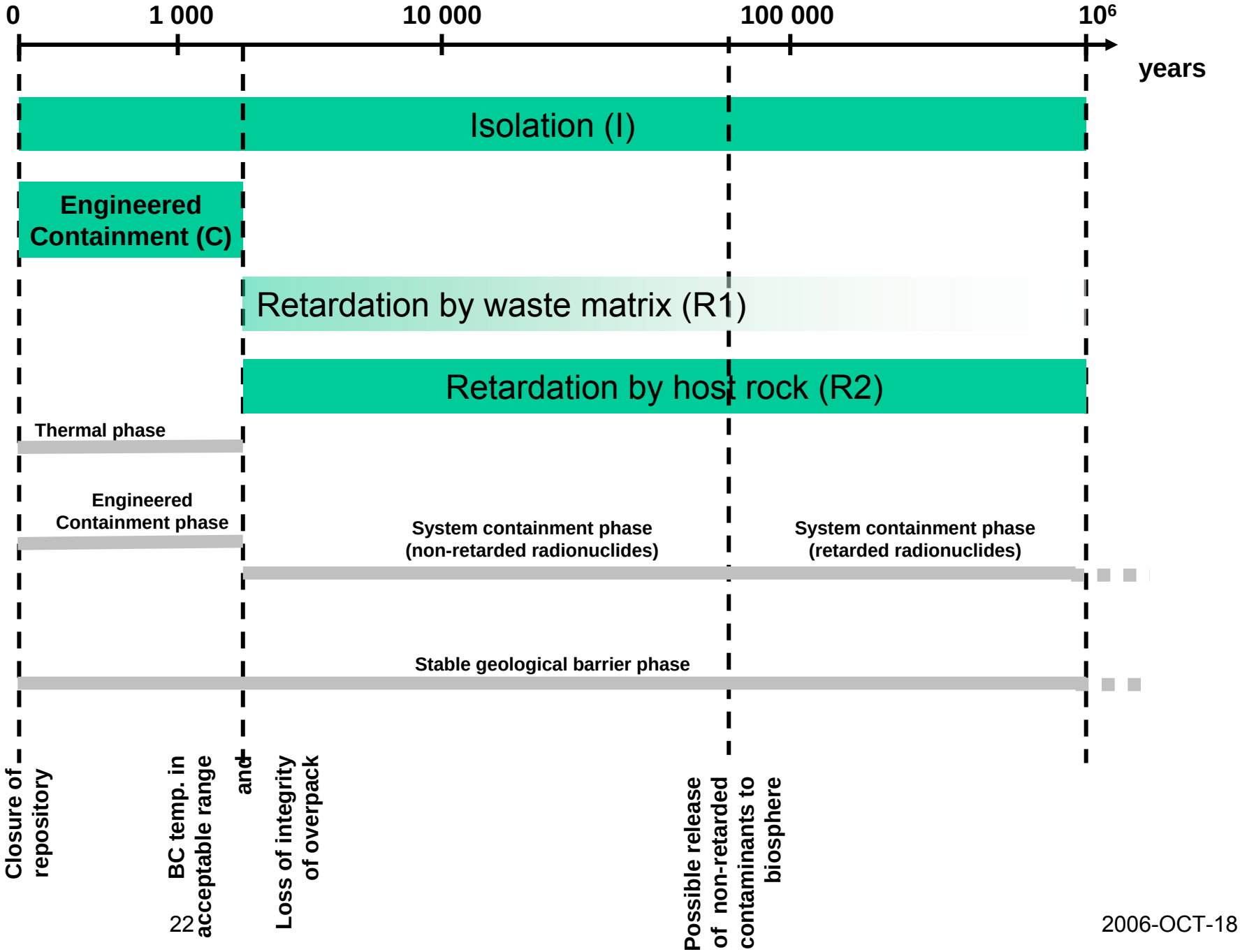
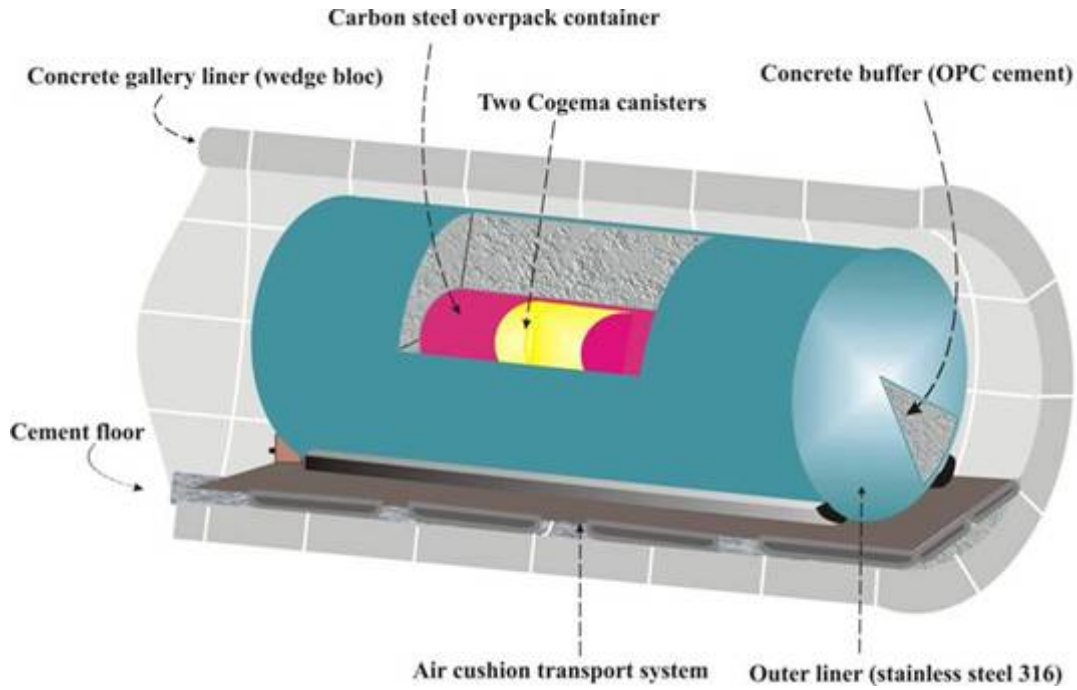

0
1 000
10 000
100 000
10⁶
years
Isolation (I)
Engineered Containment (C)
Retardation by waste matrix (R1)
Retardation by host rock (R2)
Thermal phase
Engineered Containment phase
System containment phase (non-retarded radionuclides)
System containment phase (retarded radionuclides)
Stable geological barrier phase
Closure of repository
BC temp. in acceptable range
and
Loss of integrity of overpack
Possible release of non-retarded contaminants to biosphere

SELECTION OF THE DISPOSAL CONCEPT

- Basis for selection process; multi-criteria analysis
- Criteria based on:
  - long-term safety functions
  - disturbance of their performance
  - uncertainty on their performance
  - other considerations (operational, cost, …)
- Selected concept: Supercontainer with OPC(1) buffer



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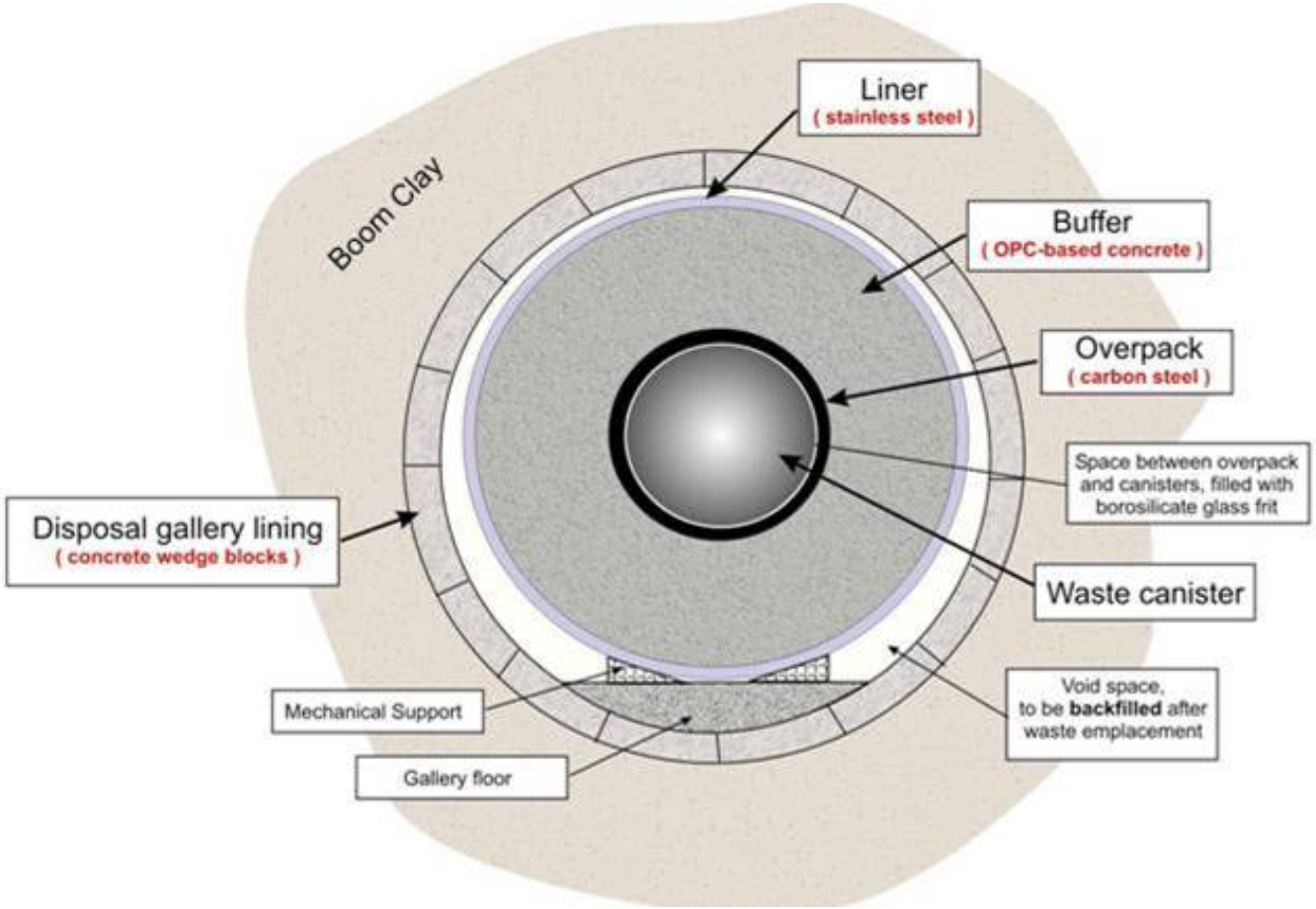
(1) OPC = Ordinary Portland Cement

SUPERCONTAINER (OPC variant)



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STRENGTHS OF OPC SUPERCONTAINER

- Long-term safety advantage;
  excellent corrosion protection because of OPC environment ⇒
  - better assurance of watertightness during thermal phase in corrosive Boom Clay environment (high pyrite content)
- Operational safety advantage:
  permanent radiation shielding ⇒
  - no need for shielding casks
  - no absolute need for remote control (although it may still be applied, to comply with the dose limitation principle)
  - allows workers to approach the disposal package in case of malfunction during transportation and emplacement
- Technical Feasibility advantage
  construction of engineered barrier in surface building ⇒
  - better quality assurance and control
  - reduction of “delicate” underground operations

  use of OPC-based concrete for the buffer ⇒
  - well known characteristics (extensive use in industry)
  - cost-effective
  - widely available, also in the future

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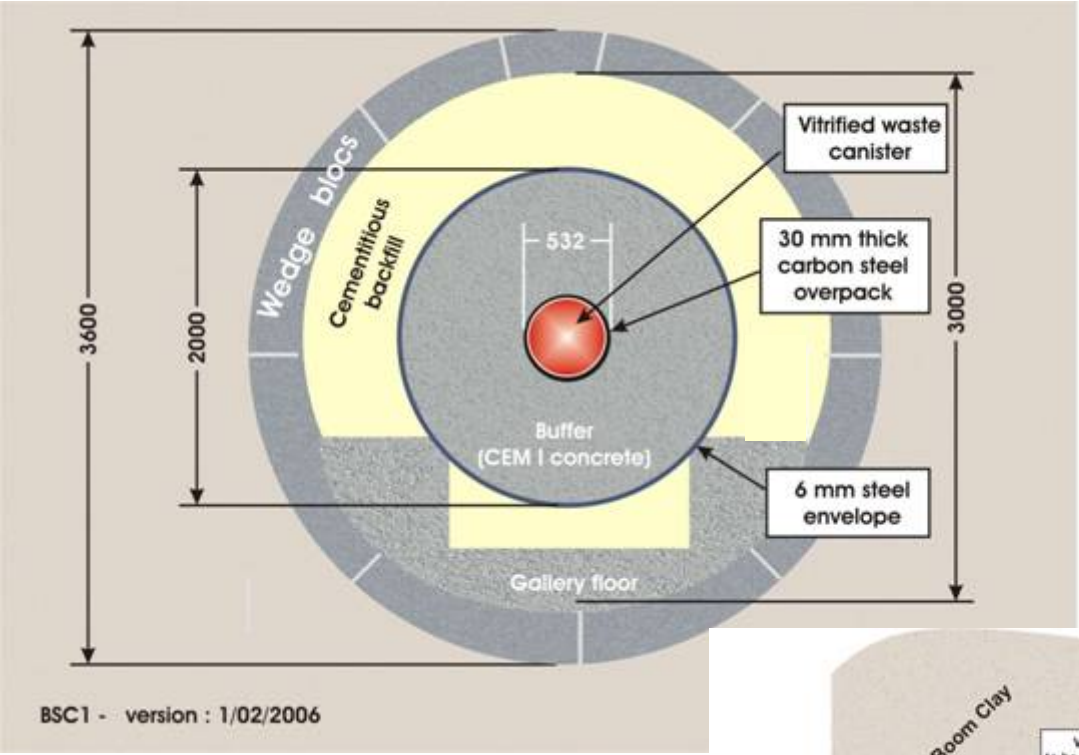
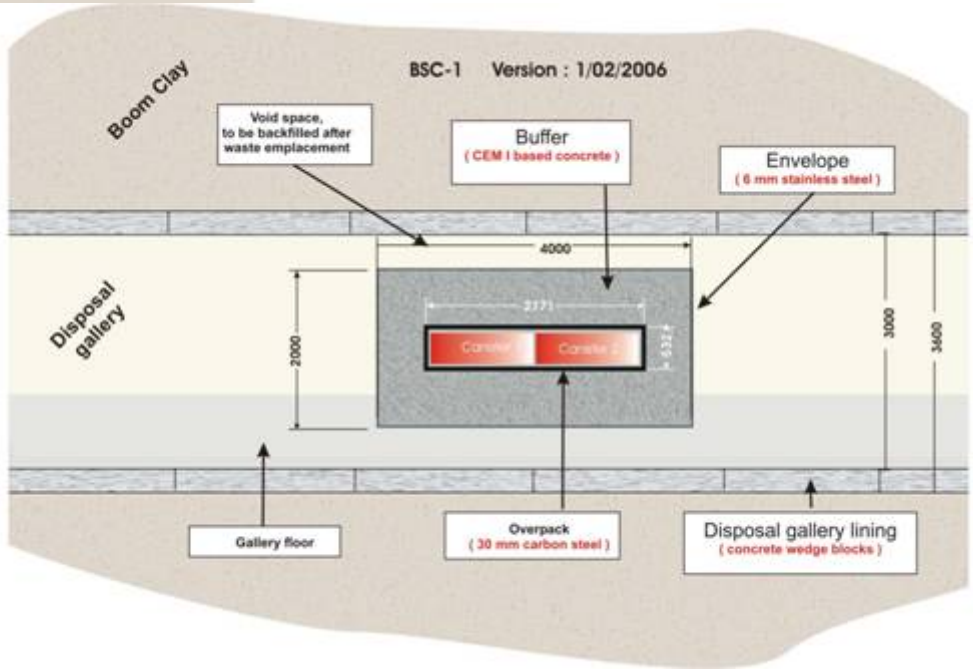
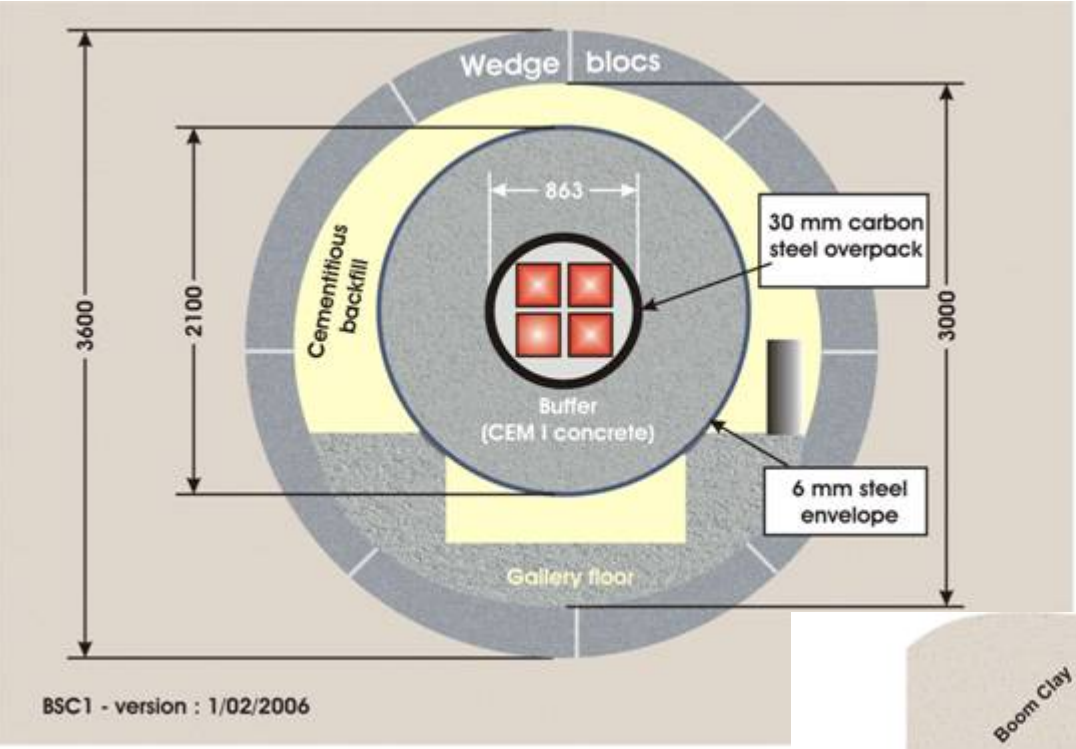
SUPERCONTAINER DESIGN for Vitrified HLW

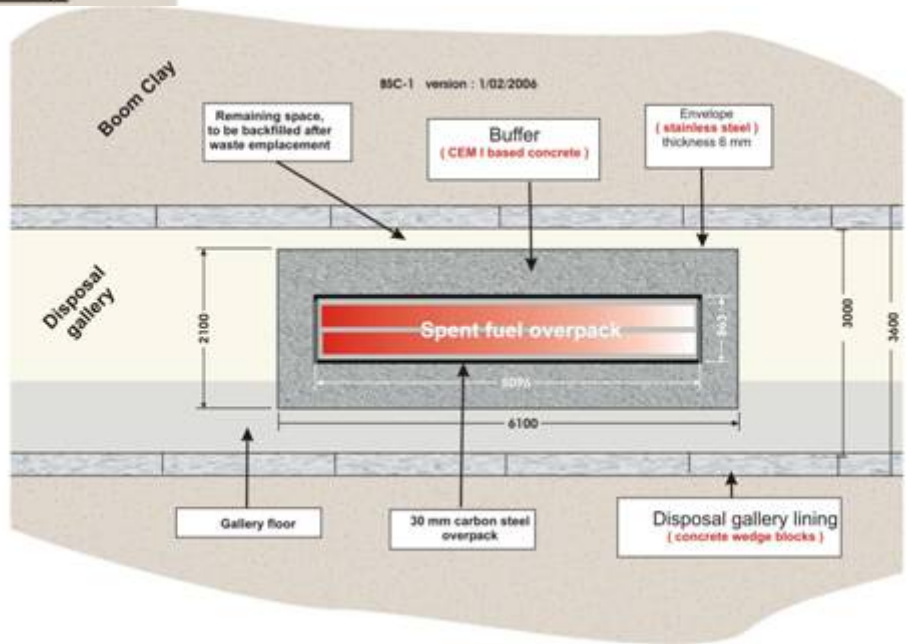


Characteristics of the Supercontainer for vitrified HLW	
Outer diameter	2.000 m
Outer length	4.000 m
Approximate weight	30 t
Number of primary waste packs per Supercontainer	2



SNF SUPERCONTAINER DESIGN for SNF (with CRD overpack for 14' fuel)





Characteristics of the Supercontainer for SNF (CRD overpack)	
Outer diameter	2.100 m
Outer length	6.100 m
Approximate weight	60 t
Number of primary waste packs per Supercontainer	4

PART 3

DESIGN AND FABRICATION OF THE OVERPACK

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FUNCTIONS AND CONSTRAINTS FOR THE DESIGN OF THE OVERPACK

FUNCTIONS

1. watertight containment of the primary waste during at least the thermal phase, i.e.
   - 500 years for Vitrified HLW
   - 2000 years for SNF
2. Provide mechanical support for fixation of primary waste packs

CONSTRAINTS

1. uniform corrosion rate of carbon steel of 1 μm/year (given by the Supercontainer and host rock boundary conditions)
2. mechanical load coming from host rock hydrostatic pressure (anisotropic load 2 MPa from top-bottom / 1.8 MPa on sides)
3. thermal transient after disposal, whereby the overpack surface does not surpass 100°C, so accommodate for thermal expansion and thermally-induced mechanical stress of components
4. Allow for transportation and handling, also in hot cell conditions

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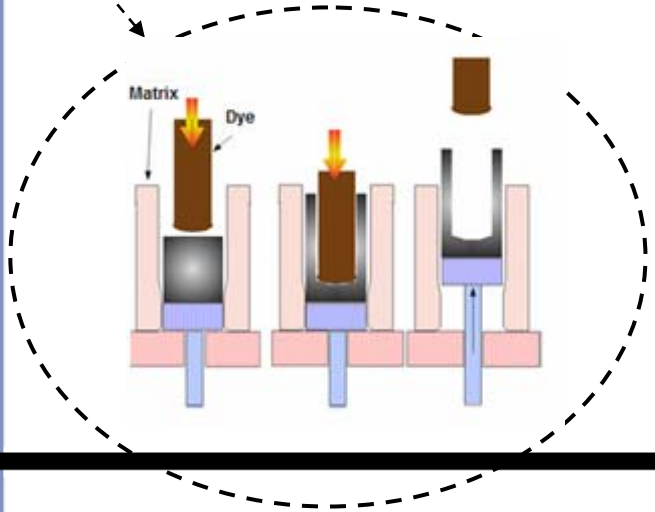
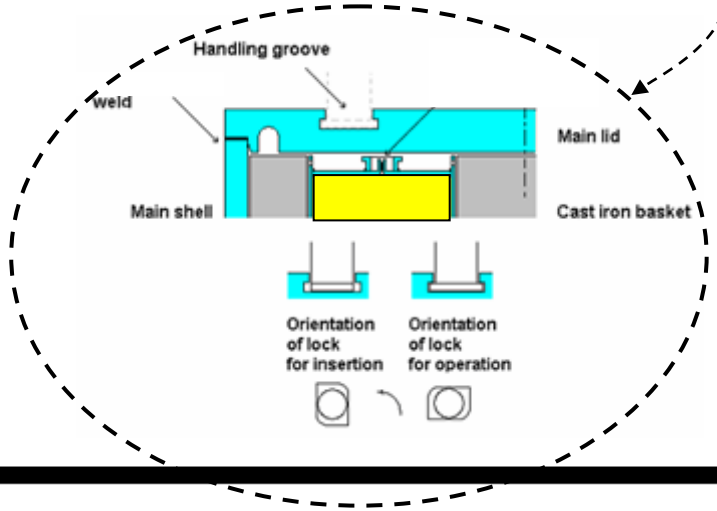
GENERAL OVERPACK COMPONENTS

1. Vessel
   - cylindrical shell, in carbon steel
     - *made of bended and welded plate, or*
     - *by direct forging*
   - bottom, in carbon steel
     - *contains **centering holes**, to fix overpack and basket*
2. Lid, in carbon steel
   - *contains **groove**, for handling the overpack*
3. Closure weld (lid to vessel)
4. Basket (mechanical fixation of waste packs inside overpack)
   - *variety of designs, dep. on waste type and technical solution*





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DESIGN CASES

1. Overpack for Vitrified HLW
2. Overpack for SNF
   - UOX
     - 8 ft fuel
     - 12 ft fuel
       - 15 x 15 grid
       - 17 x 17 grid
     - 14 ft fuel
   - MOX (12 ft)

most challenging case of UOX SNF

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DESIGN OF OVERPACK for Vitrified HLW

CHARACTERISTICS

1. Shell thickness:
   - assume 4 mm corroded after 500 years (= 8 µm/y > 1 µm/y)
   - Mechanical and thermal stress analyses ⇒ 16 mm needed (12+4)
   - Conservative thickness: 30 mm (for uniformity with other designs)
2. Fixation: basket into which 2 canisters are inserted
3. Filling of empty space with borosilicate glass frit
4. Dimensions:

Characteristics of the overpack for vitrified HLW	
Shell thickness (m)	0.030 m
Outer diameter (m)	0.532 m
Outer length (m)	2.771 m
number of primary waste packs	2

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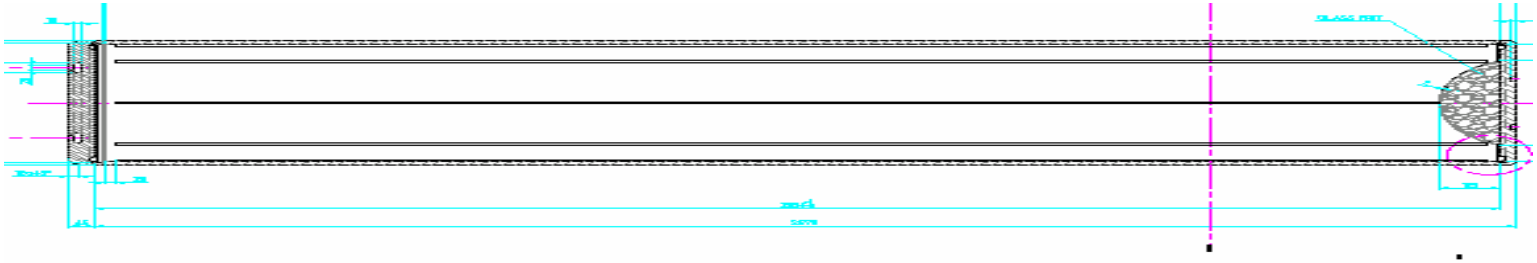
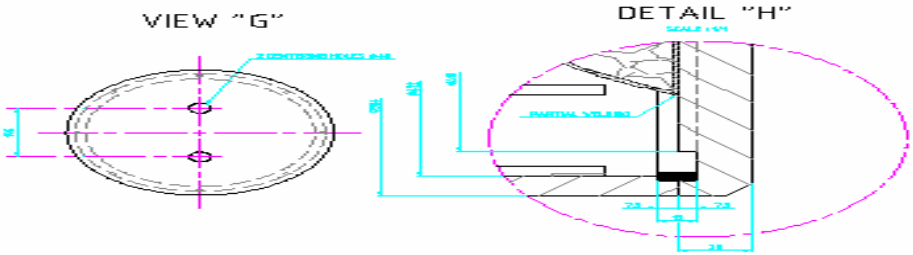
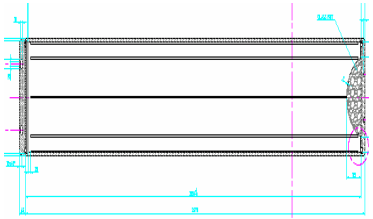
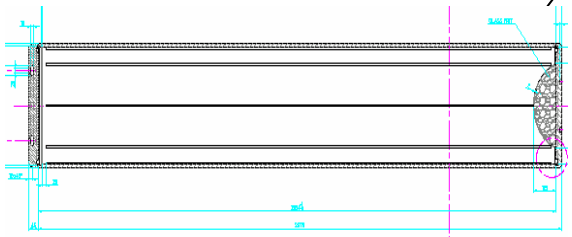
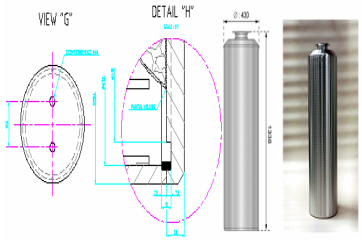
DESIGN OF OVERPACK for Vitrified HLW

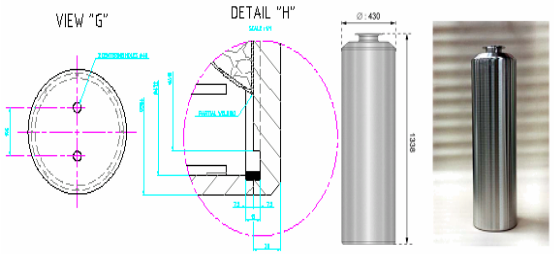












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DESIGN OF OVERPACK for UOX SNF

1. Fixation: basket into which 4 assemblies are inserted.
   Still open acceptability of "burnup credits" for criticality ⇒

   TWO BASIC DESIGN OPTIONS:

   - "**Conservative Reference Design**" (CRD):
     - Based on proven technologies and conservative assumptions
     - Each assembly is placed and fixed inside tight square box
     - Sand filling can be added and/or inert gas
     - Basket made of 2 pieces in spheroid ductile cast iron
   - "**Innovative Option Design**" (IOD):
     - Cylindrical shell and its bottom are forged
     - Sand filling is not considered
     - Shell is used as formwork to directly cast the iron basket

2. Shell thickness:
   - assume 6 mm corroded after 2000 years (= 3 µm/y > 1 µm/y)
   - Mechanical and thermal stress analyses ⇒
     - 20 mm needed for CRD (14+6)
     - 26 mm needed for IOD (20+6)
     - Conservative thickness: 30 mm (uniformly for CRD and IOD)

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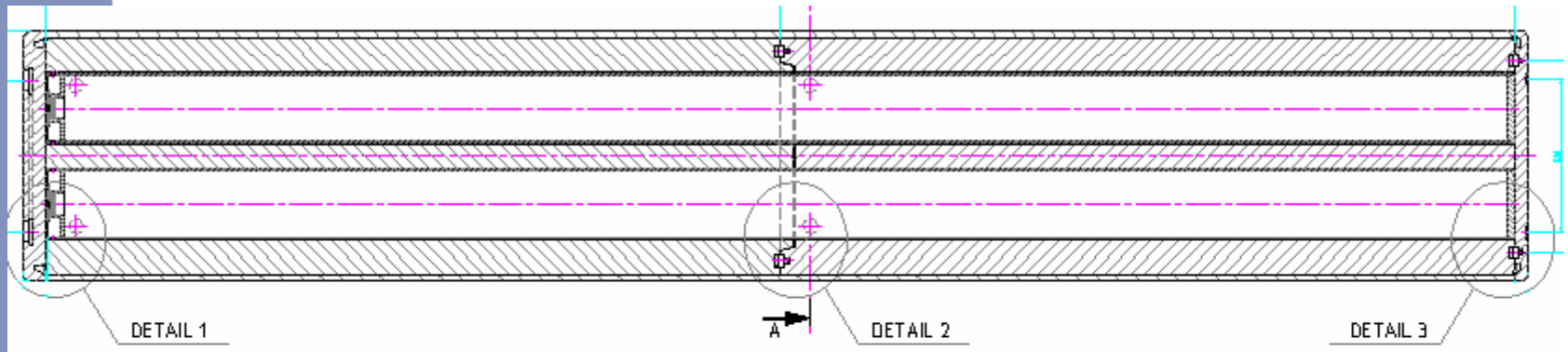
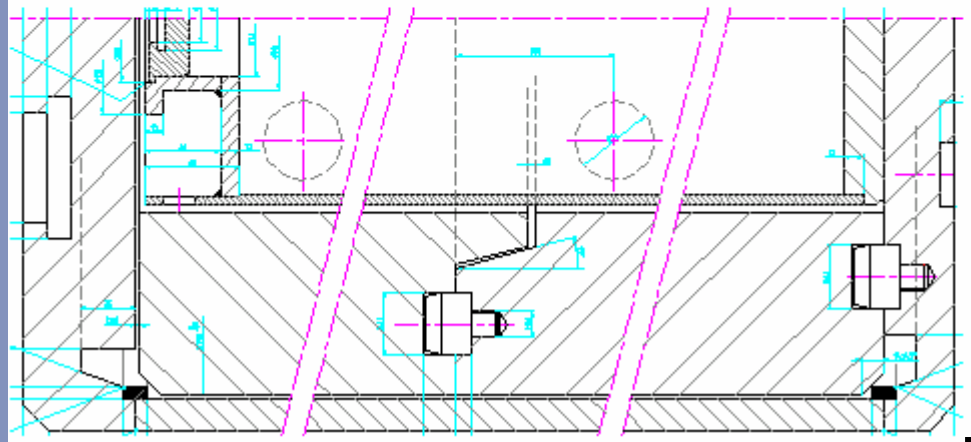
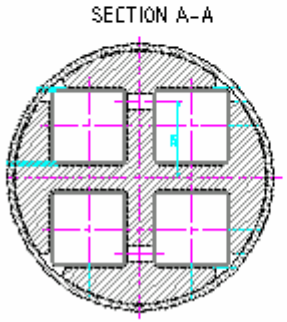
DESIGN OF CRD OVERPACK for UOX SNF







Characteristics of the CRD overpack for 14 ft UOX SNF	
Shell thickness (m)	0.030 m
Outer diameter (m)	0.863 m
Outer length (m)	5.096 m
number of primary waste packs	4

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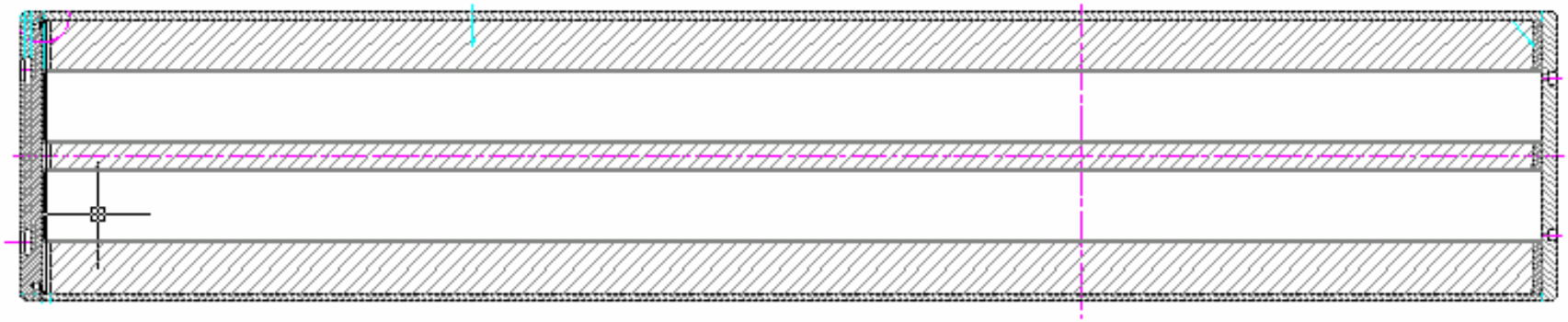
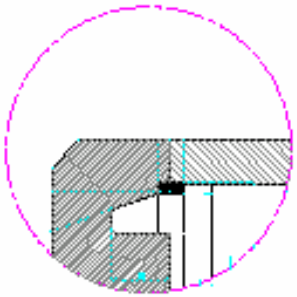
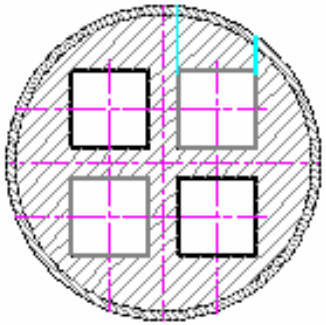
DESIGN OF IOD OVERPACK for UOX SNF







Characteristics of the IOD overpack for 14 ft UOX SNF	
Shell thickness (m)	0.030 m
Outer diameter (m)	0.948 m
Outer length (m)	5.005 m
number of primary waste packs	4

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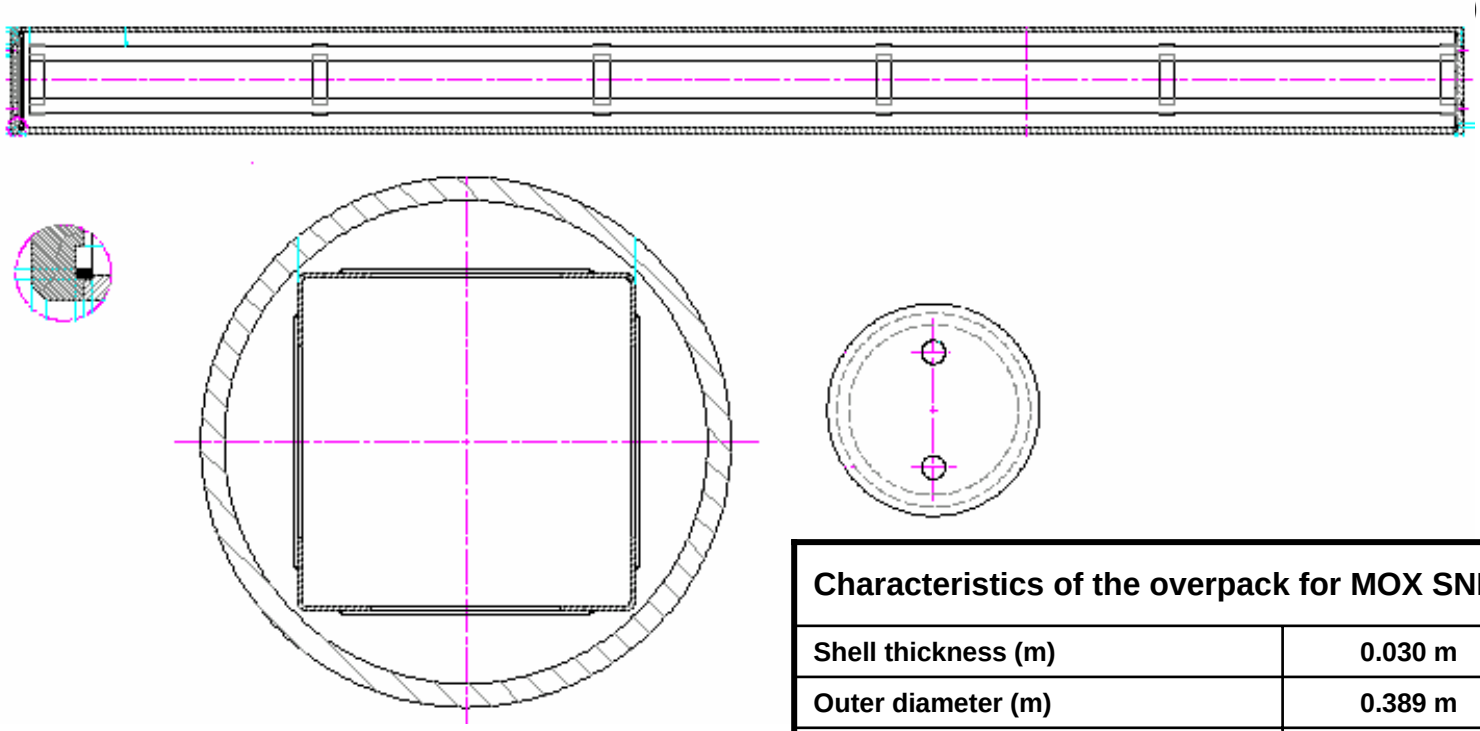
DESIGN OF OVERPACK for MOX SNF

1. Fixation: basket into which 1 assembly is inserted.
   Because of this design (only 1 assembly), large criticality margin exists ⇒ selection of the IOD design option
2. Shell thickness:
   - assume 6 mm corroded after 2000 years (= 3 µm/y > 1 µm/y)
   - Mechanical and thermal stress analyses ⇒ 16 mm needed (10+6)



Characteristics of the overpack for MOX SNF	
Shell thickness (m)	0.030 m
Outer diameter (m)	0.389 m
Outer length (m)	4.261 m
number of primary waste packs	1

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OVERPACK CLOSURE WELD

- Closure weld is difficult because:
  - Radiation prevents direct human intervention
  - Radiation prevents X-ray check of weld
  - High quality is required (need for long-term watertightness)
- Investigated options:
  - Electron beam welding
  - Laser beam welding
  - Arc welding
    - TIG (Tungsten Inert Gas)
    - MIG (Metal Inert Gas)
    - SAW (Submerged Arc Welding)
  - Friction welding
    - Stir friction welding
    - Component friction welding

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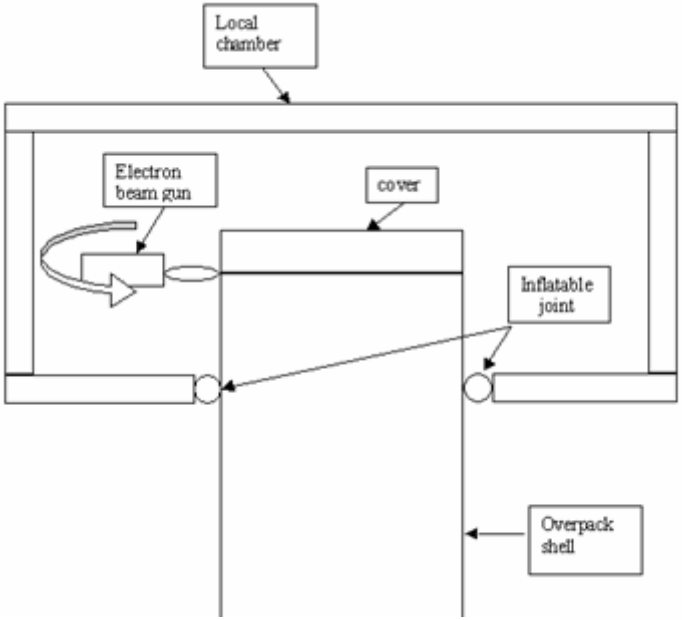
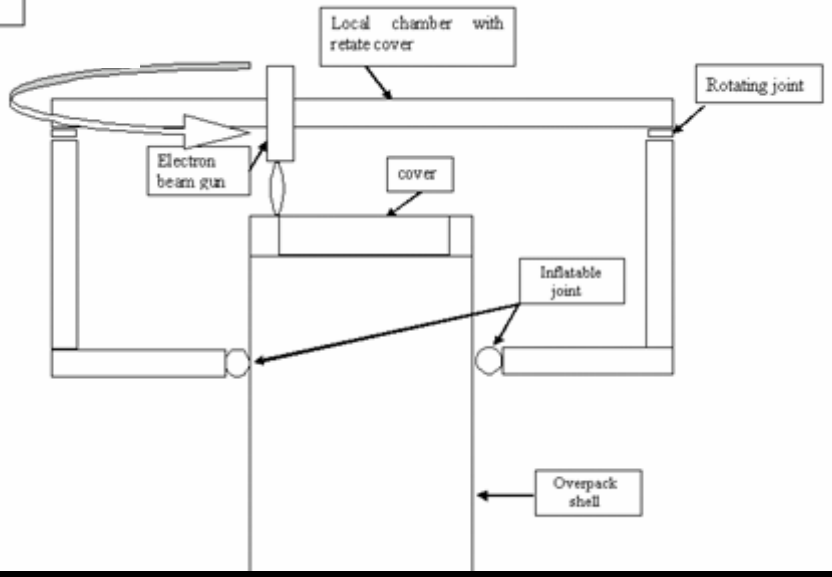
ELECTRON BEAM WELDING

WITH FIXED GUN



WITH ROTATING GUN



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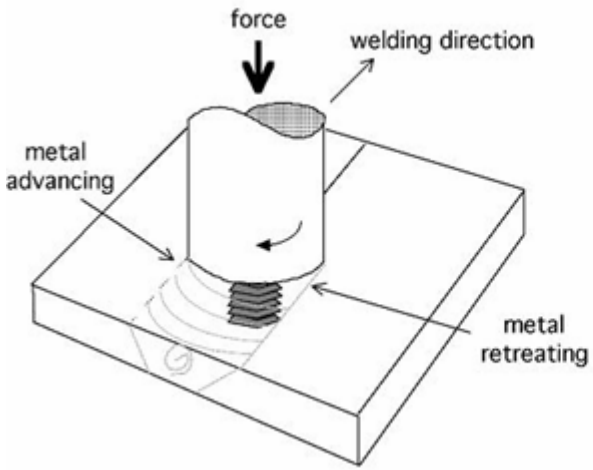
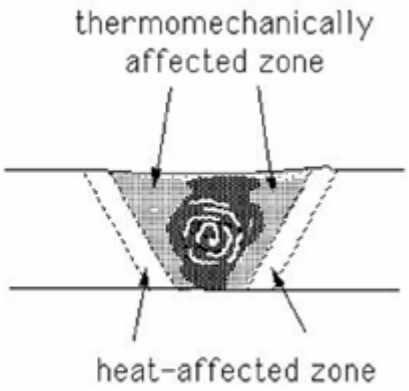
FRICTION WELDING

STIR FRICTION

- Needle rotates between lid and shell until metal melts
- Difficult to apply to carbon steel (needle must be very resistant)





COMPONENT FRICTION

- Lid itself is rotated on shell until metal melts
- Then pressed to join lid and shell
- Risky to apply to components containing nuclear waste

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COMPARISON OF WELDING TECHNIQUES

Welding process	Arc	Friction	Laser	Electron beam
Low thermally affected zone, residual stresses and deformations	--	++	++	++
Low risk of internal defects	--	+	++	++
Adequacy for thick joints	++	-	--	++
Welding speed	--	++	++	+ (1)
Ease of implementation in a hot cell	-	-	++	+
Repair facility	--	--	+	+
Cost effectiveness	++	--	-	--

(1) Including time necessary to reach vacuum in the welding chamber

++ Excellent
+ Good
0 Medium
- Poor
-- Very Poor

⇒ electron beam welding is favored option

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CONCLUDING THOUGHTS

- Unless in the future there is a Government decision to go for a complete reprocessing of SNF, the design of HLW disposal packages basically has to deal with 3 types of waste:
  - Vitrified HLW
  - UOX SNF (of different lengths)
  - MOX SNF
- The Supercontainer disposal package is designed to comply with the long-term safety functions for disposal of HLW / SNF in Boom Clay. In addition, the concept exhibits some important operational and technical feasibility advantages
- Within the Supercontainer, the Overpack is the component with the function to provide the long-term safety requirement to have a period of watertightness that surpasses the thermal phase in the surrounding rock.

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CONCLUDING THOUGHTS

- For UOX SNF, the IOD overpack design is generally more advantageous than the CRD, from a perspective of:
  - technical feasibility (simpler composition)
  - Production costs (estimated overpack production costs are:
    - 4.5 kEUR vitrified HLW
    - 63 kEUR CRD UOX
    - 60.5 kEUR IOD UOX
    - 5 kEUR MOX

  However, its preference over the CRD design relies on a criticality calculation assumption (burn-up credits) that is not yet accepted. Until then, the CRD design remains the preferred design for UOX.

- Future acceptance of burn-up credits could benefit from:
  - latest improvements of direct burn-up measurements
  - refinement of criticality evaluation tools and analysis procedures

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ŠKODA JS a.s.

Workshop on Design and Assessment of Packages for Radioactive Waste

Pavel Růžička, ŠKODA JS, Czech Republic

Hotel Marijke -Petten, the Netherlands

November 21-22, 2006

ŠKODA JS a.s.

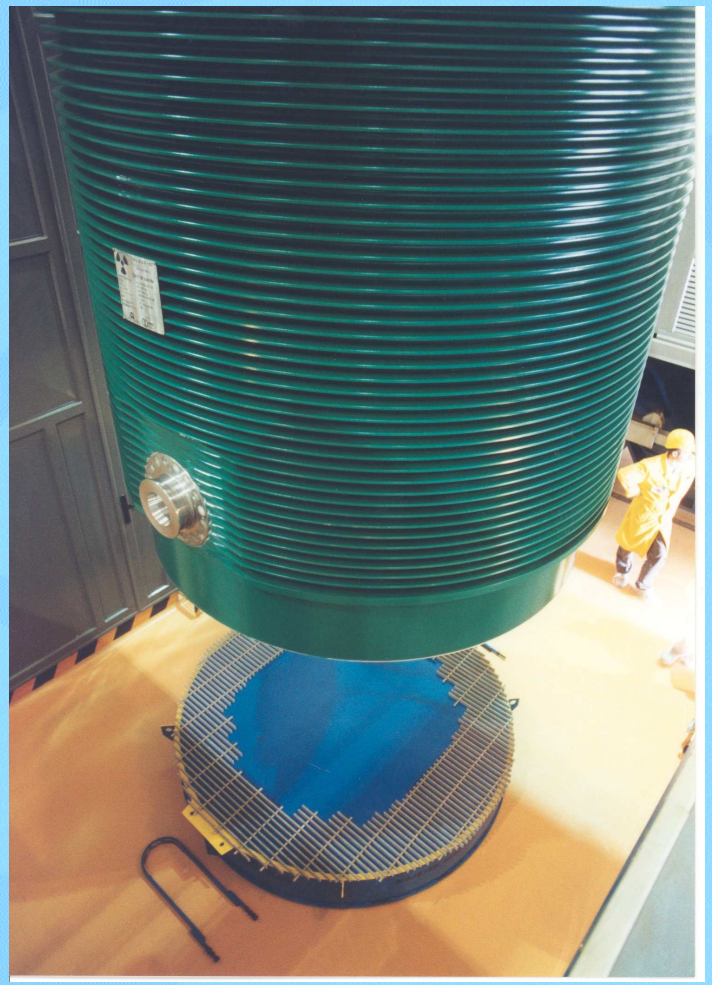
The Storage and Transport System – ŠKODA VPVR/M cask´s system

ŠKODA JS a.s. Company Structure

- **ŠKODA JS a.s. company**
  - **Shareholders‘ structure:**
    - **Member of the OMZ (100% owner) group – OMZ Group is the Russia’s largest heavy machinery**
  - **Currently 760 employees**
  - **Today - Czech leader in supplies for nuclear power plants overall Europe**

Spent Fuel Transport and Storage Casks

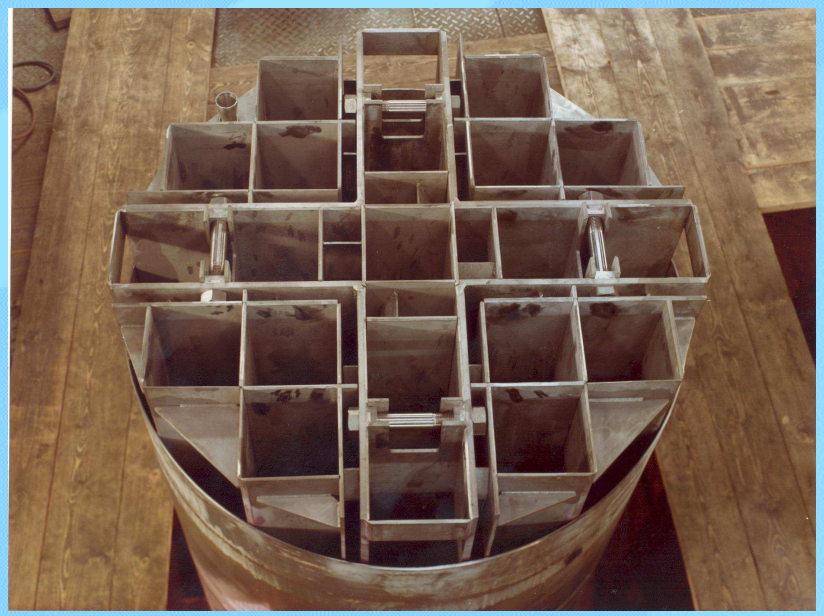
- 236 casks manufactured since 1993
- a long-term agreement with GNS Essen for supplies of CASTOR® and CONSTOR® type of casks

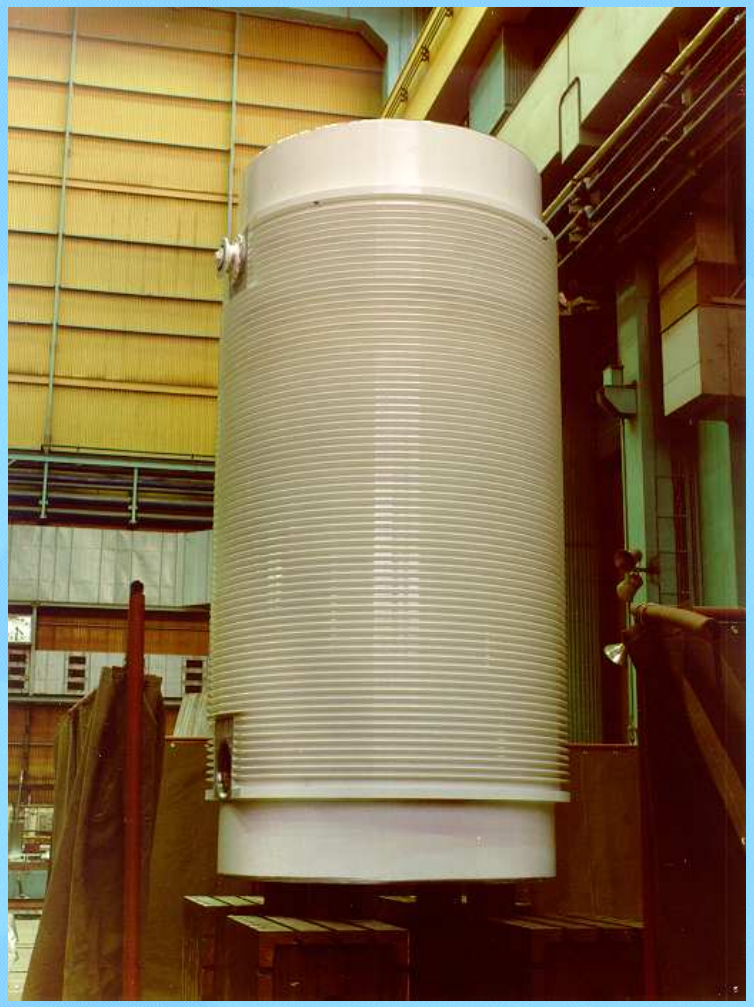


ŠKODA JS a.s.

Spent Fuel Transport & Storage Casks

- **CASTOR® V/21 cask**
  - **4 casks for Virginia Power, USA**





Russian Research Reactor Fuel Return Project (RRRFR)

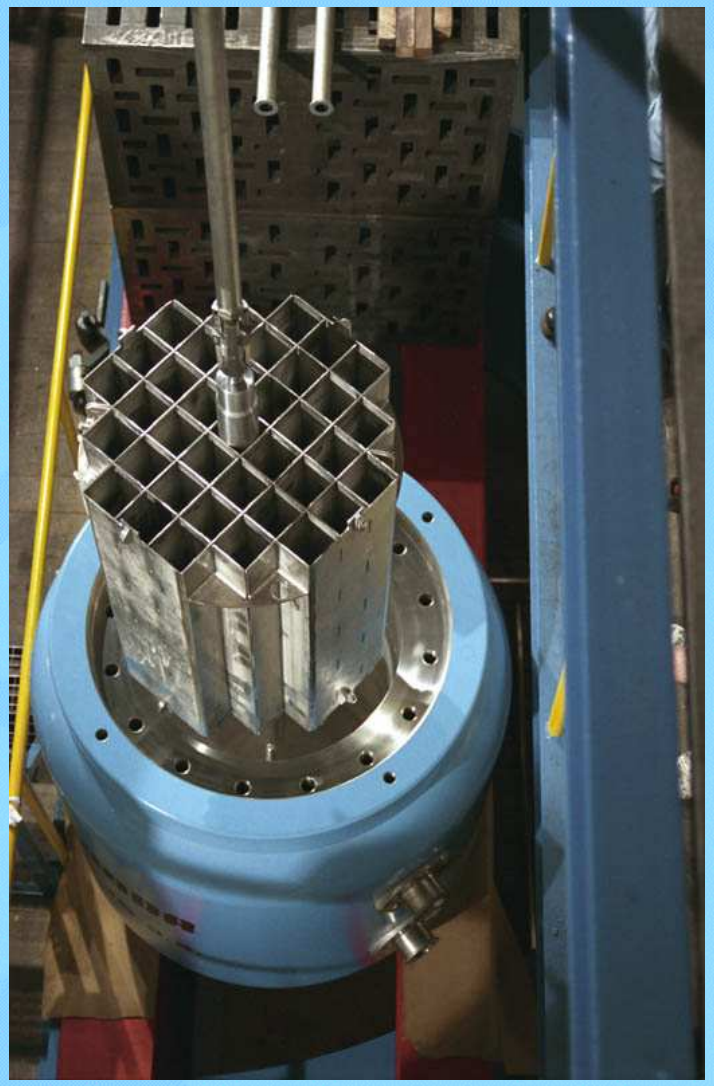
- Goal
  - to eliminate HEU and persuade eligible countries to convert from HEU to LEU
  - to ship Russian supplied HEU spent nuclear fuel from research reactors to Russia for reprocessing
- more than 20 research reactor in 17 countries
- US-Russian Government-to-Government Agreement – May, 2004
- initial planning efforts with the Czech Republic (NRI Rez) in progress
- IAEA, SKODA JS and NRI Rez concluded a tripartite contract with financial support of US Gov/DoE – December, 2005 (financial support DoE via IAEA)

ŠKODA JS a.s.

Spent Fuel Transport & Storage Casks

Type SKODA VPVR/M cask

- for the transport and storage of the research spent nuclear fuel (HEU)
- SKODA JS´s original design of the cask
- Developed in cooperation with NRI Řež near Prague, CZ
- Reference:
  - supplied - 3 casks for NRI Rez
  - in progress - 3 casks for NRI Rez
    10 casks for IAEA



Type SKODA VPVR/M cask - analysis results - shielding

The cask is registered as a package **of type B(U) and S**

Is necessary to transport by **form of exclusive use**

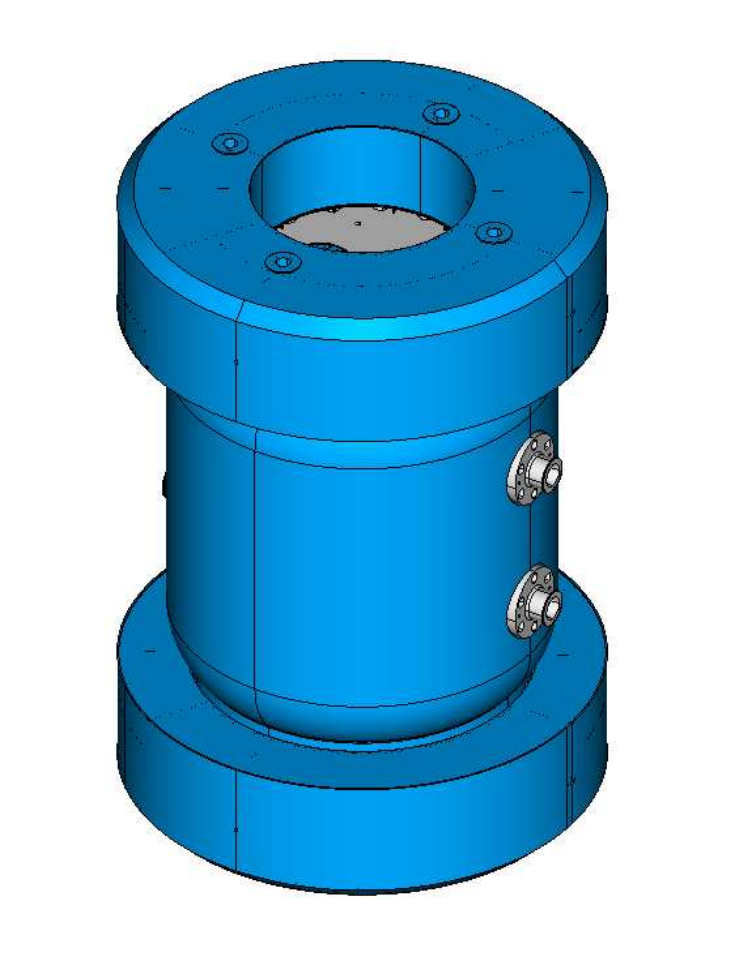
Transport index is **TI = 2,3-7,8 (according to DRE)**

Dose rate equivalent in radial and axial direction are under limit values **- 2 mSv/h**.

Quality assurance Program and applications

Quality Assurance Plan

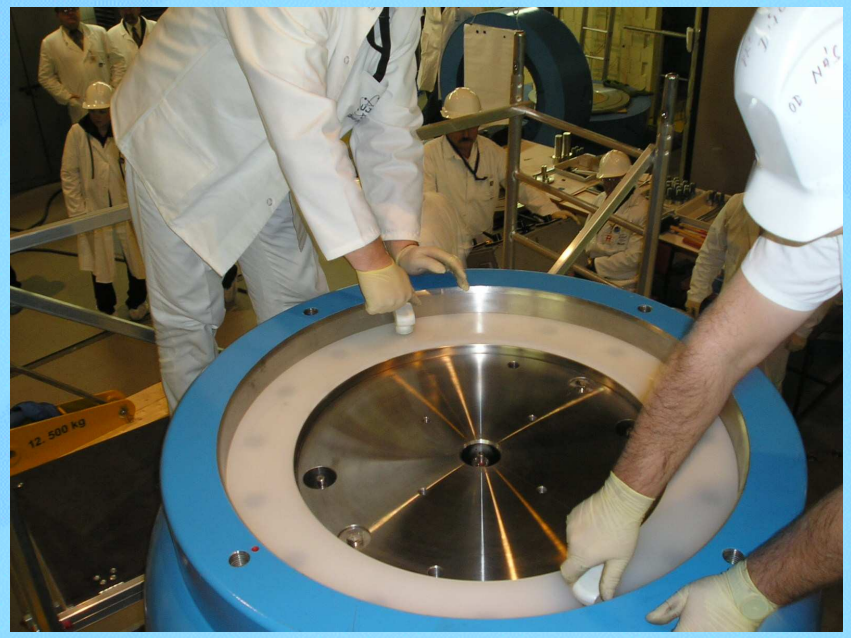
- ŠKODA VPVR/M Cask is Safety Class 2 selected equipment according to the Regulation No. 214/1997 of the SÚJB
- Quality Assurance Plan fulfils the Regulations No. 214/1997 requirements for selected equipment producers including Quality Assurance program implementation, design, procurement, documentation control , distribution and filing, product identification, manufacturing, testing and inspection, calibration, training etc.
- ŠKODA VPVR/M Cask QA Plan was approved by State Office of Nuclear Safety



Quality assurance Program and applications

Conformity Verification and Assessment Manual

- Verification and assessment of product conformity
- Organizational system of verification and assessment of product conformity
- Nameplate Marking of package Inspection
- Supervision of working processes
- Verification of design data
- Operating checks periodicity of package



SKODA VPVR/M cask verification steps

1st step – „Dry Run“ performed at SKODA JS´s premises – March 2005

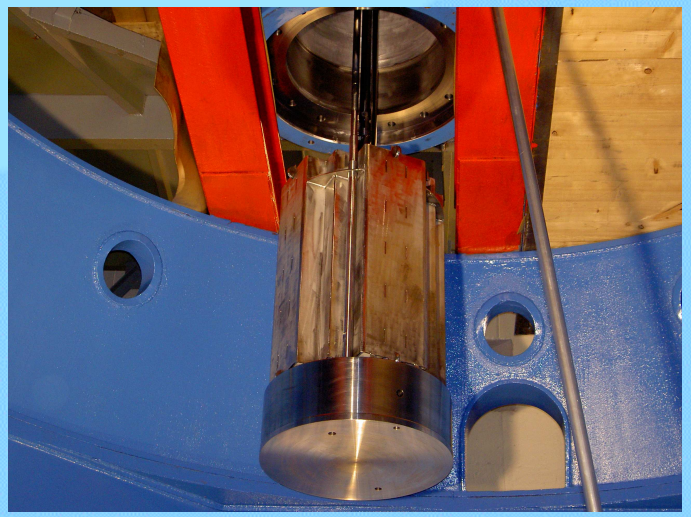
Demonstration of the following:

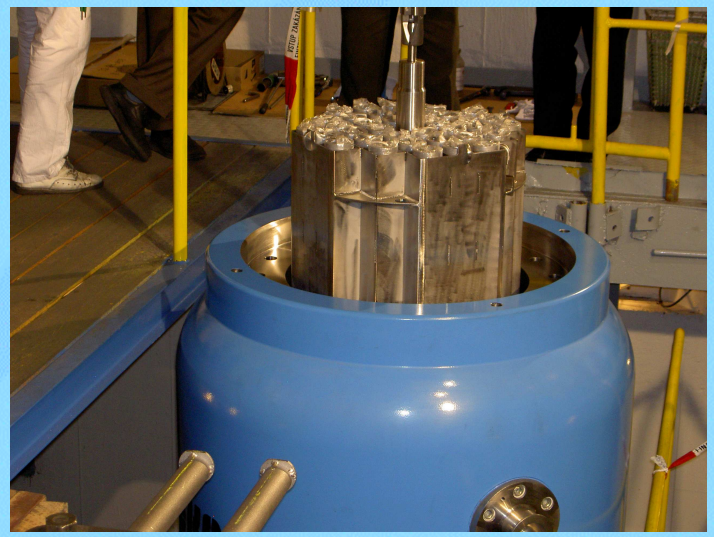
Handling from the top

Handling from the bottom

Basket drop

Tightness



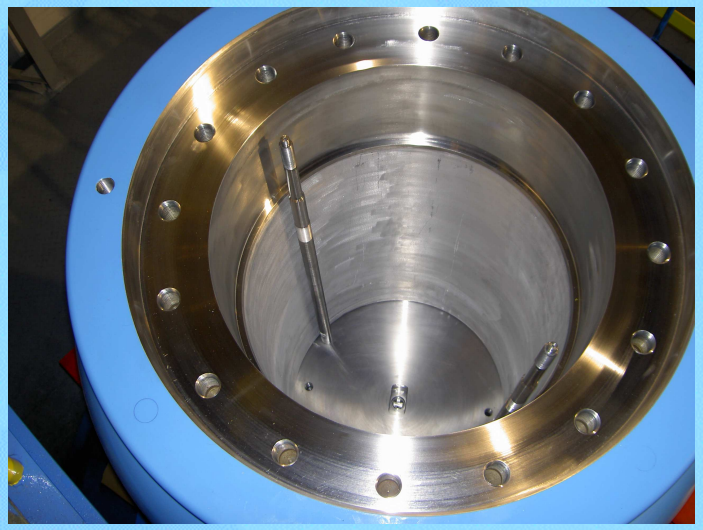


SKODA VPVR/M cask verification steps

2nd step – „Wet Run“ performed at NRI Řež – November 2005

During this step was checked-up the following:

- Manipulations with the VPVR/M cask at the building of the reactor LVR-15
- Manipulations in the high-level waste storage



SKODA VPVR/M cask verification steps

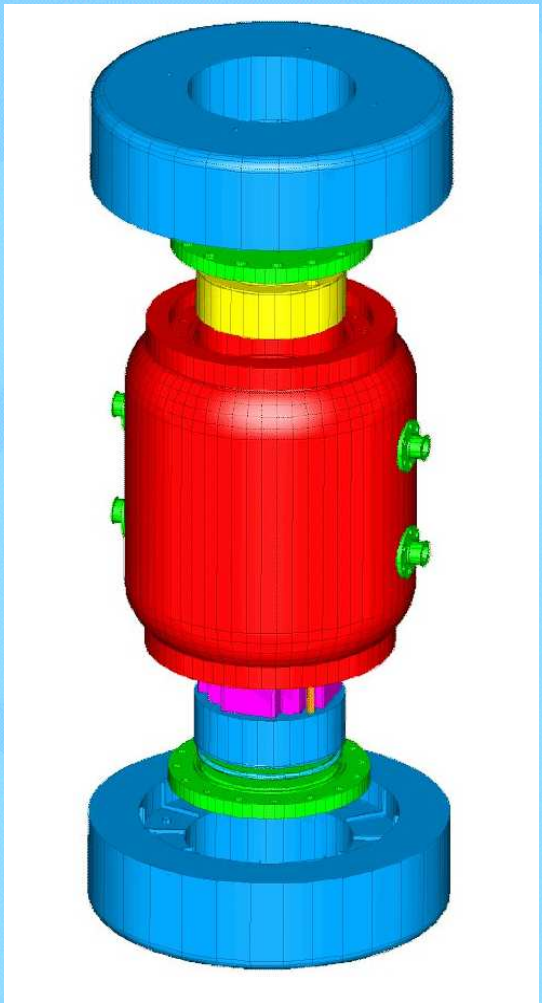
3rd step – „Verification Run“ performed at Mayak plant – June 2006

- to verify the receiving processes, manipulation, handling within the Mayak plant

General information – basic data

Basic components of the ŠKODA VPVR/M

- body of the cask
- inner basket
- upper secondary lid
- bottom primary lid
- impact limiters



Internal arrangement

- 36 fuel assemblies of type IRTM, EK-10, WWR-M

Weights with fuel

- with impact limiters	**12 400 kg**
- without impact limiters	**11 150 kg**
- Weight of 36 Fas	**450 kg**

ŠKODA JS a.s.

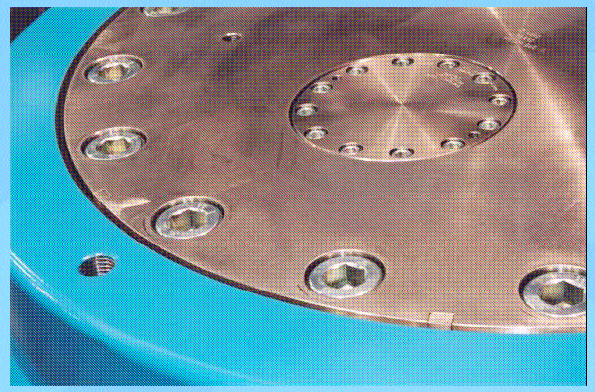
Spent Fuel Transport & Storage Casks

General information

Dimensions

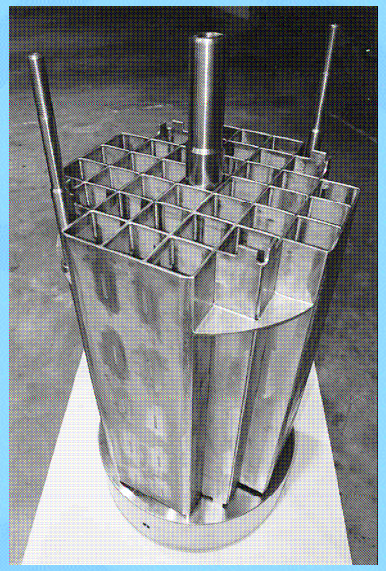
External:

- height (with impact limiters) 2155 mm
- height (without impact limiters) 1505 mm
- diameter 1200 mm
- cask wall thickness 300 mm



Main materials:

- Body cast steel
- Prim. lids stainless steel
- Sec. lids carbon steel
- Shock absorbers carb. steel, wood



Way of transport

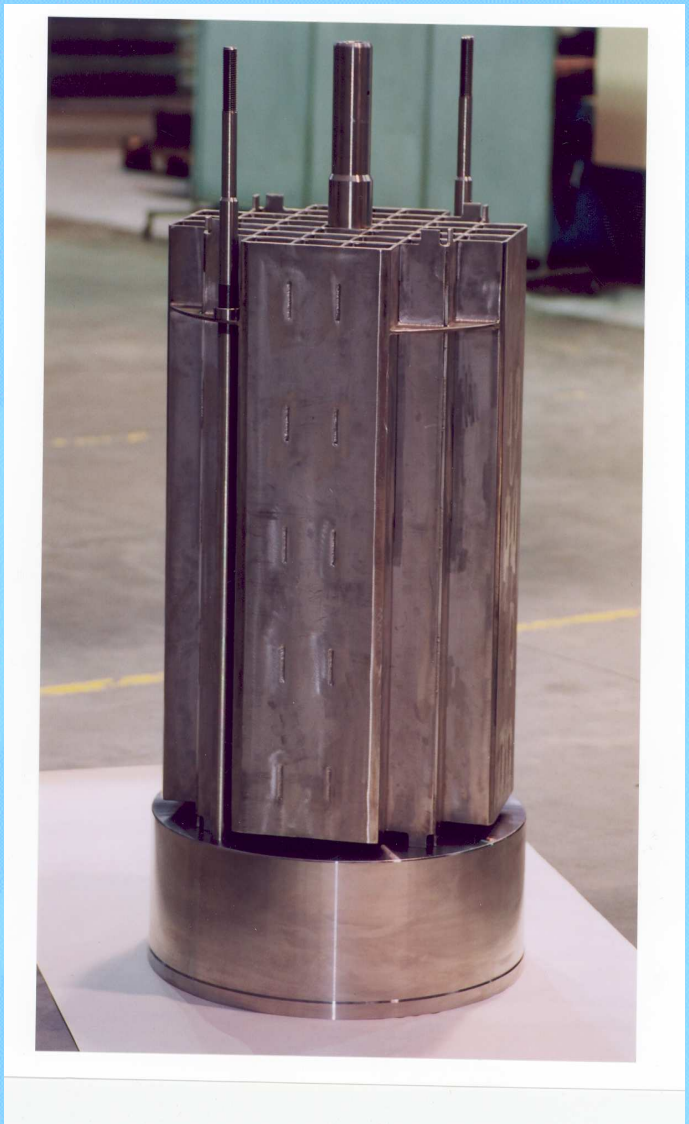
- automobile transport
- for railway transport
- for river and see transport

ŠKODA JS a.s.

Spent Fuel Transport & Storage Casks

Capacity of the basket:

- 36 pcs fuel assemblies - EK-10, S-36, VVR-M, VVR-M2, VVR-M5, VVR-M7, VVR- (S) M, IRT-2M, IRT-3M, TVR-S
- total activity max. $3{,}93.10^{15}$Bq
- heating capacity max. 450W
- maximum allowable weight of U^{235} in one loaded assembly - max. 500 g

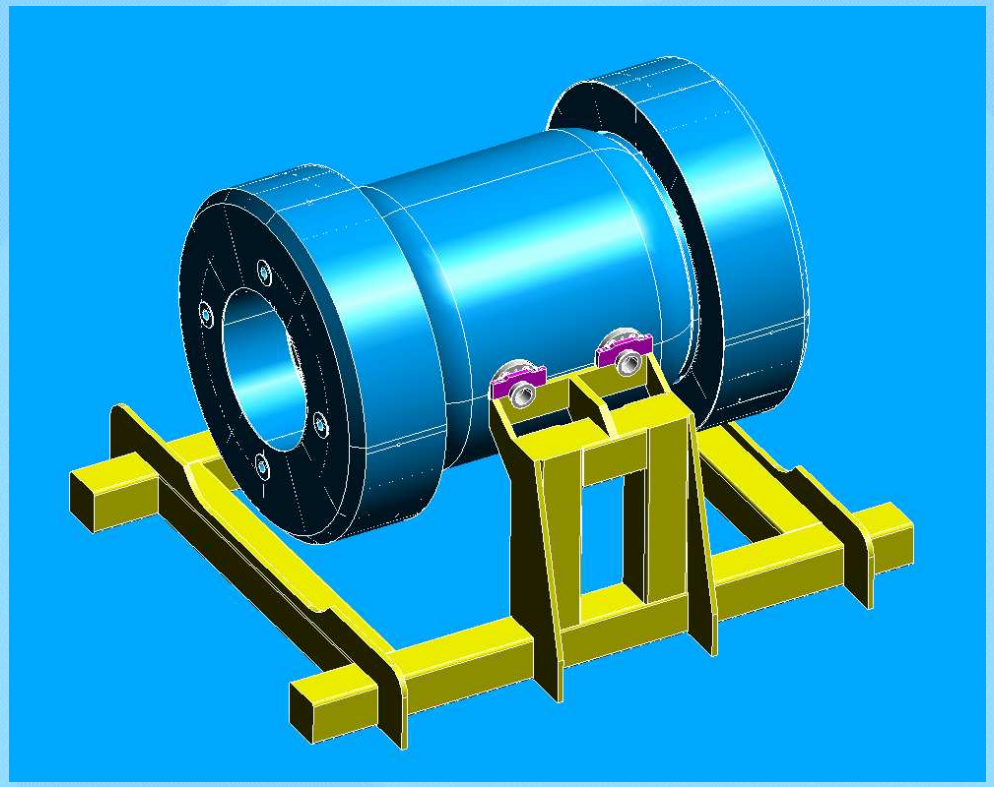


ŠKODA JS a.s.

Spent Fuel Transport & Storage Casks

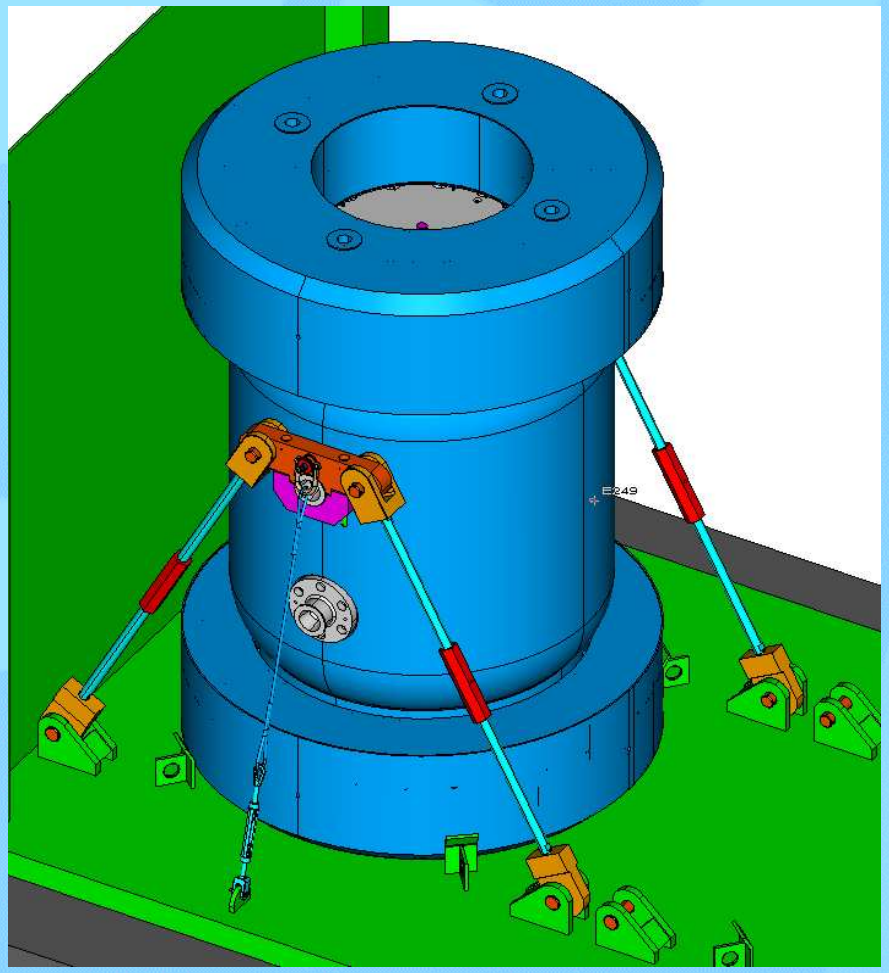
The ancillary equipment :

- Manipulation frame
- Traverse for manipulation by the help of pins and by the help of shock absorber
- Guiding pins for the secondary lid
- Wrench of the central hangings
- Wrench fro the central hangings nut
- Special tubular wrench for the basket rod nuts
- Fixture with the centralizing rods
- Lifting eyes for the secondary lids
- Lifting eyes for the shock absorbers



ŠKODA JS a.s.

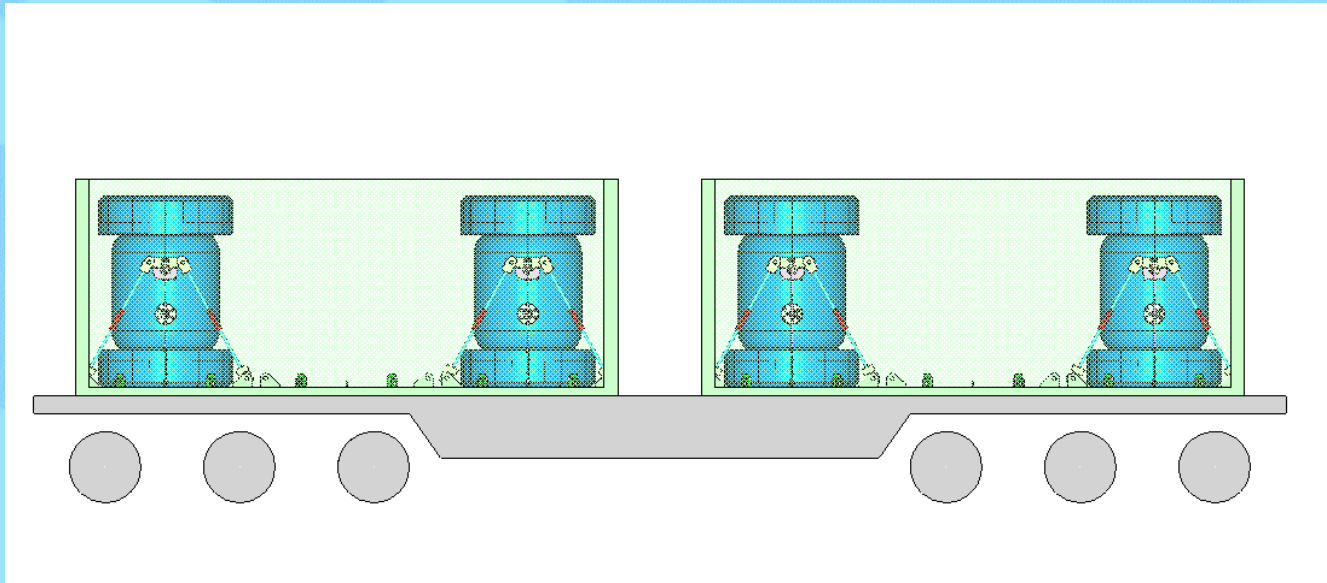
Spent Fuel Transport & Storage Casks



Transport of the casks in ISO container

- 20 ft ISO „Open Hard Top“
- Ability to bear 1 or 2 pcs of the ŠKODA JS ´s cask
- Tie-down system
- Vertical position of the cask

Transport of the casks in ISO container



ŠKODA JS a.s.

Spent Fuel Transport & Storage Casks

ŠKODA VPVR/M cask – package design license status

- March 2005 – Czech Republic - CZ/048/B(U)F-96
- January 2006 – Russian Federation - RUS/3065/B(U)F-96
- June 2006 – Ukraine
- Slovakia – expected within the next two months

Федеральное агентство по атомной энергии

СЕРТИФИКАТ - РАЗРЕШЕНИЕ
НА КОНСТРУКЦИЮ УПАКОВКИ

Транспортный упаковочный комплект SKODA VPVR/M
с ОТВС исследовательских реакторов

RUS/3065/B(U)F-96

Выдан 23 января 2006 г.

Срок действия 23 января 2009 г.

Заместитель руководителя Агентства С.В.Антипов

Москва

ŠKODA JS a.s.

US Ambassador Mr. R. Graber visit to SKODA JS a.s. (November 16, 2006)

ŠKODA JS a.s.

Thank you for your attention

ŠKODA JS a.s.

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Modelling of RBMK-1500 Spent Nuclear Fuel Characteristics and Comparison with Available Data

P. Poskas, A. Smaizys, E. Narkunas

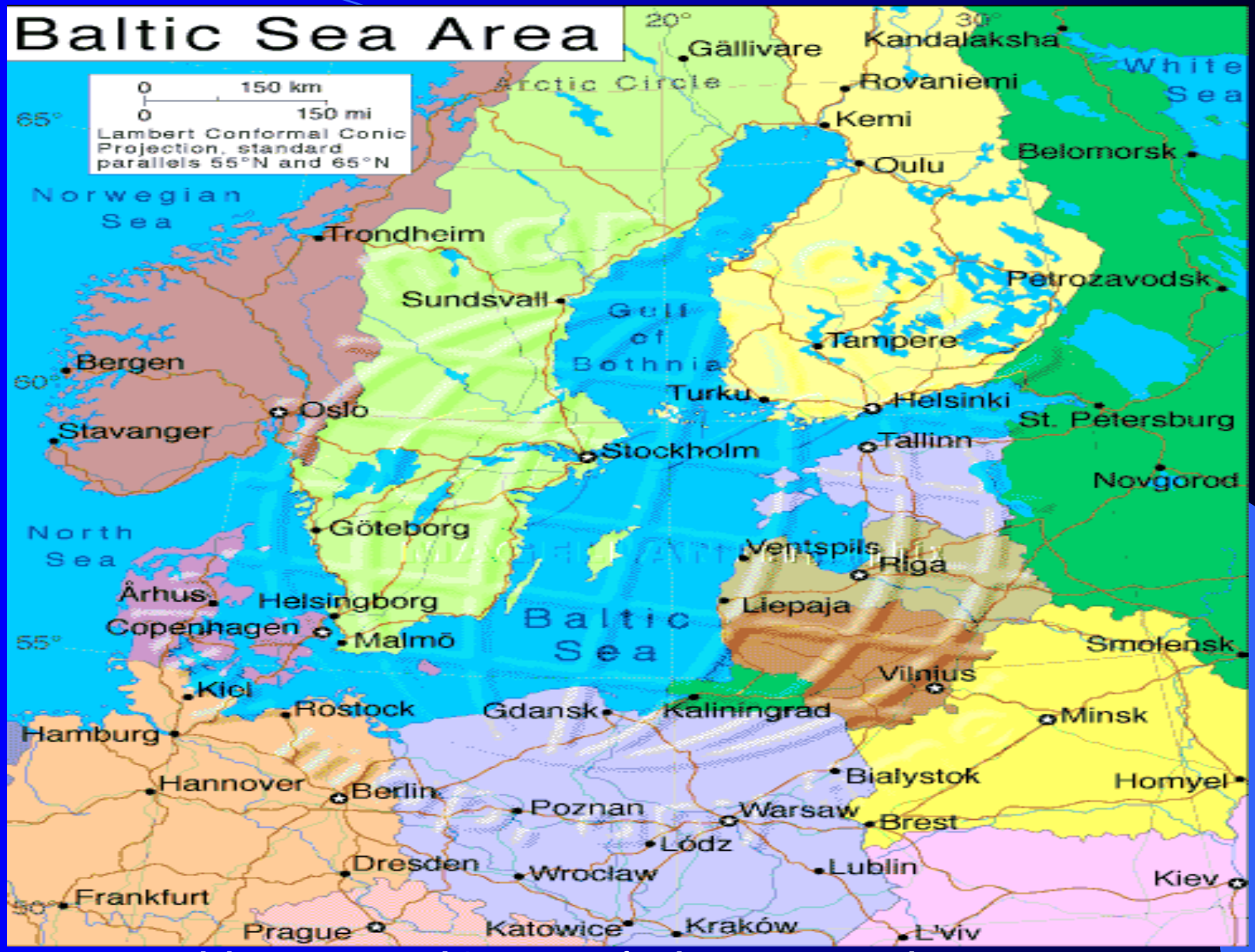
Lithuanian Energy Institute

Nuclear Engineering Laboratory

Workshop on Design and Assessment of Radioactive Waste Packages,
21-22 November 2006 at JRC, Institute for Energy, Petten, The Netherlands

Content

- Introduction
- Interim storage of SNF
- Disposal of SNF
- Determination of RBMK SNF Characteristics
- Calculation Method
- Results
- Conclusions


Baltic Sea Area
0 150 km
0 150 mi
Lambert Conformal Conic Projection, standard parallels 55°N and 65°N
20°
30°
65°
60°
55°
50°
Arctic Circle
Gällivare
Kandalaksha
Rovaniemi
Kemi
Oulu
White Sea
Belomorsk
Norwegian Sea
Trondheim
Petrozavodsk
Sundsvall
Gulf of Bothnia
Tampere
Bergen
Oslo
Turku
Helsinki
St. Petersburg
Stavanger
Tallinn
Stockholm
Novgorod
Göteborg
North Sea
Ventspils
Riga
Århus
Helsingborg
Liepaja
Copenhagen
Malmö
Baltic Sea
Smolensk
Vilnius
Kiel
Rostock
Gdansk
Kaliningrad
Minsk
Hamburg
Bialystok
Homyel
Hannover
Berlin
Poznan
Warsaw
Brest
Lódz
Dresden
Wroclaw
Lublin
Kiev
Frankfurt
Prague
Katowice
Kraków
L'viv

Workshop on Design and Assessment of Radioactive Waste Packages,
21-22 November 2006 at JRC, Institute for Energy, Petten, The Netherlands

Introduction

- There is only one nuclear power plant in Lithuania - Ignalina NPP. It is producing more than 70 % of the total electricity generated in Lithuania;
- Two similar units with RBMK-1500 reactors were installed at Ignalina NPP;

- The INPP reactors were commissioned in December 1983 and August 1987 respectively.
- The original design lifetime was projected to 2010-2015.

Introduction

(continuation)

- The closure of Ignalina NPP will take place in two stages taking into account the conditions of the long-term substantial financial assistance rendered by the European Union, G-7 countries and other states as well as international institutions.
- Unit 1 is already closed down December 31, 2004,
- Unit 2 will be closed in 2009.
- There are no more nuclear facilities in Lithuania.

Introduction

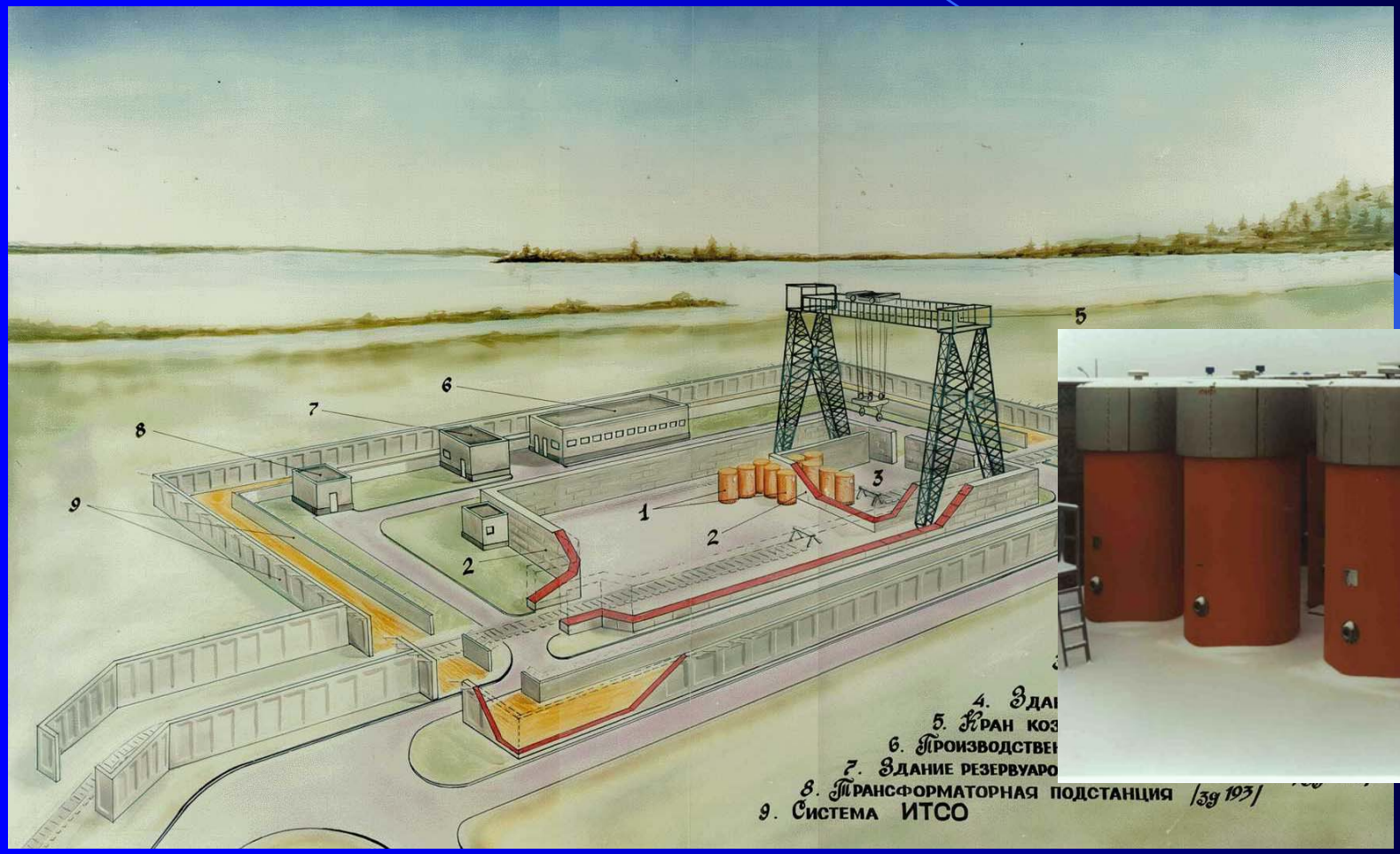
(continuation)

- After final shutdown of INPP Unit 1 in 2004 and Unit 2 in 2009 total amount of spent nuclear fuel will be approximately 22 thousand of fuel assemblies (FA).
- All these assemblies should be stored about 50-100 years and after that disposed of.
- FA unloaded from the reactors are stored in the water pools at least for 5 years.

Interim storage of SNF

- In 1992 it was decided to use dry storage technology for interim storage of SNF at the INPP.
- GNB dual-purpose storage casks have been chosen:
  - Part of them (twenty casks) is ductile cast iron CASTOR RBMK-1500 casks and
  - the remaining ones are metal-concrete CONSTOR RBMK-1500 casks.
- Both type casks are designed to load 102 half-assemblies, which are placed in the basket of special configuration.
- Capacity of existing SNF dry storage facility at the Ignalina NPP is for 80 casks.

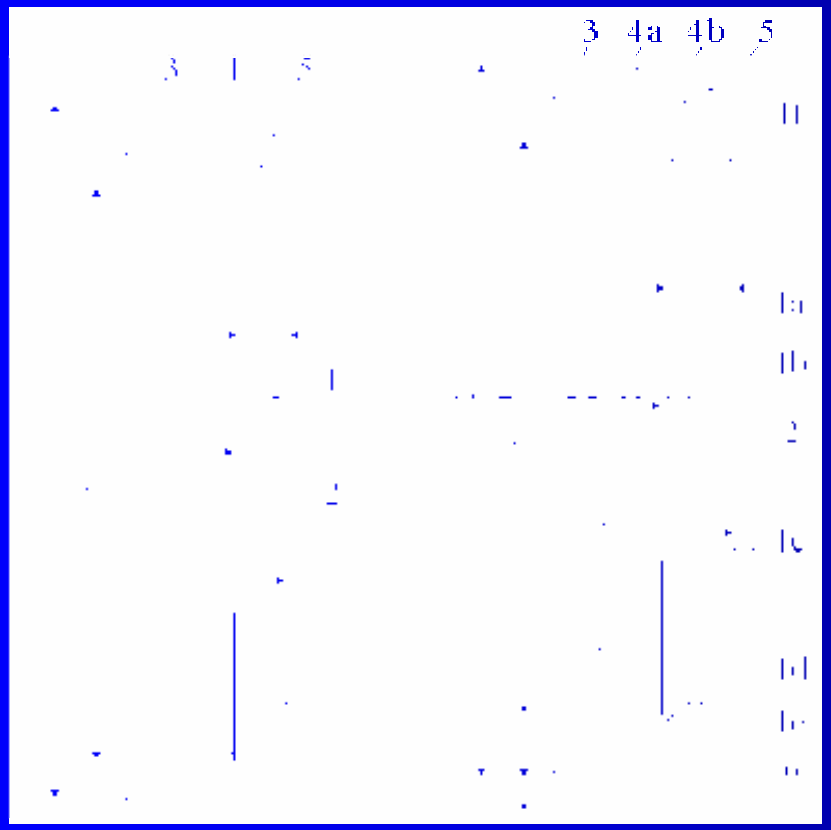
Interim storage of SNF
(continuation)



Interim storage of SNF

(continuation)

Axial cross sections of the casks



CASTOR RBMK-1500 CONSTOR RBMK-1500

1 - cask body (1a, 1b, 1d, 1e, 1f - metal parts; 1c - heavy concrete);

2 - basket;

3 - cask lid;

4 (4a, 4b) - guard plates;

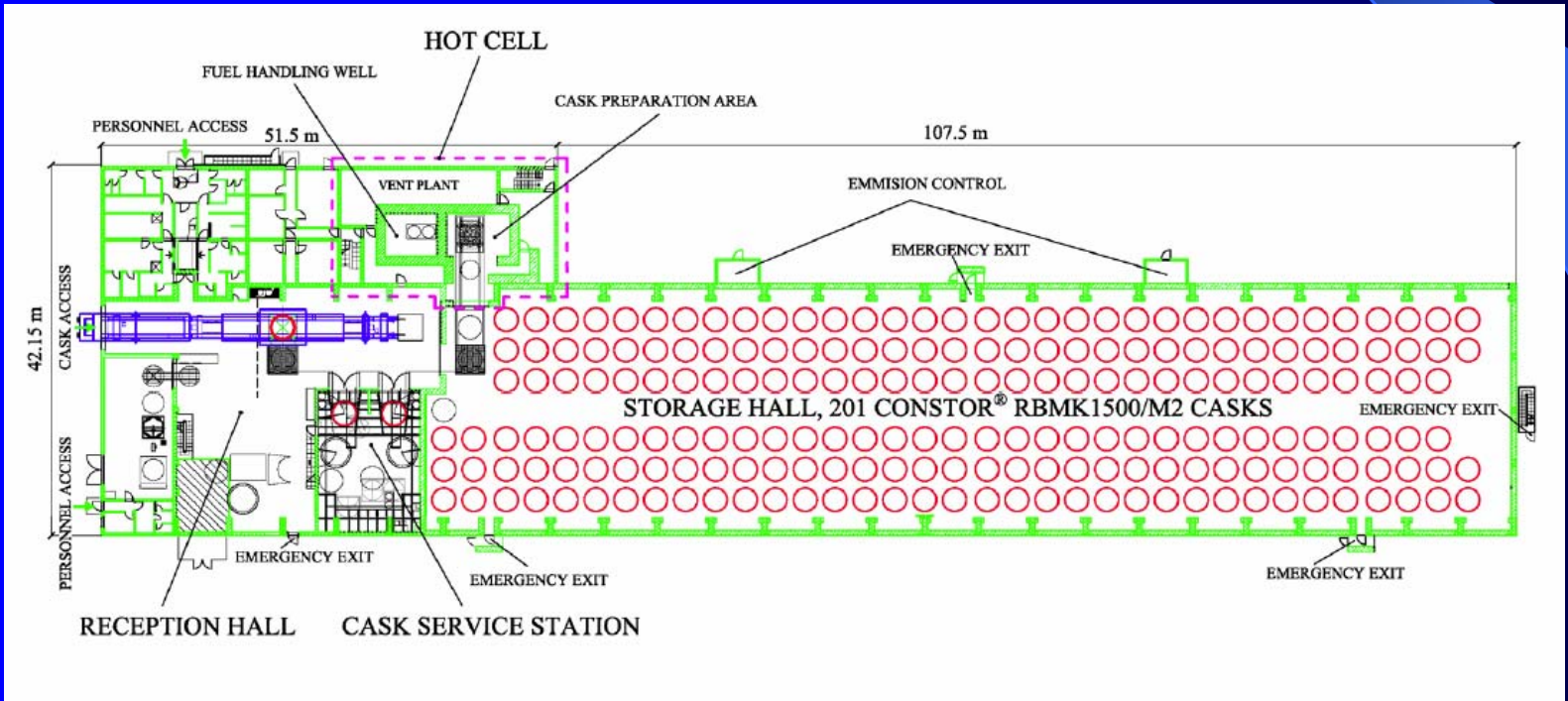
5 - concrete cover;

6 - damper.

Interim storage of SNF

(continuation)

- New storage facility is under implementation to accommodate remaining more than 18 thousand SNF assemblies (202 CONSTOR RBMK M2 casks)



Interim storage of SNF

(continuation)

- New storage facility

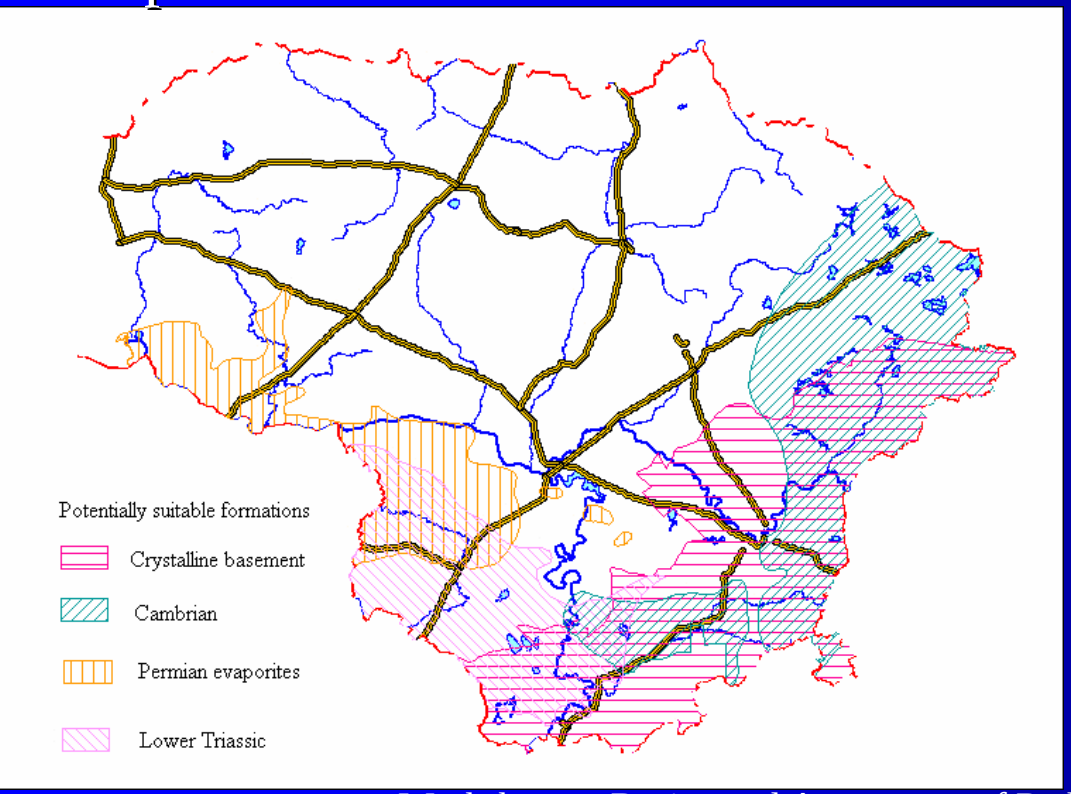
Concept of the main storage building

Concept of the security and administration building (the gate-house) and the vehicle and rail transport inspecion area

Disposal of SNF

(continuation)

Geological formations prioritized as prospective formations for SNF disposal in Lithuania:

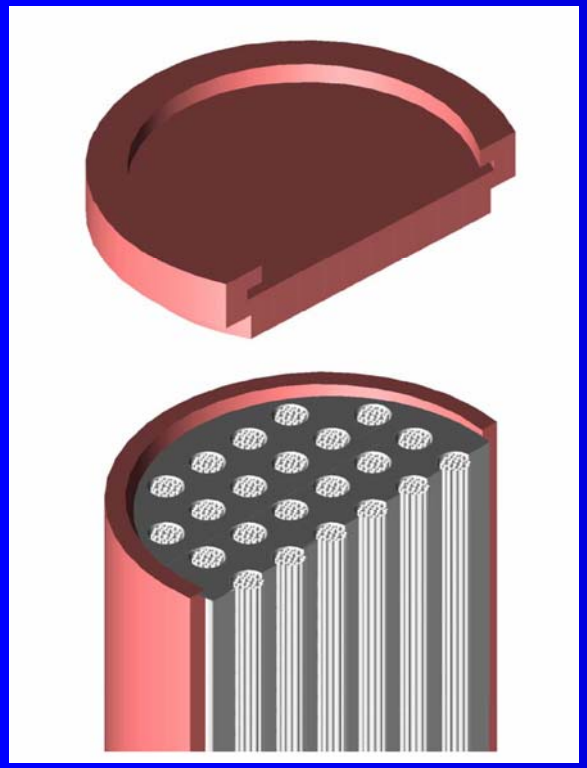


- The crystalline rocks in the southern Lithuania
- The Lower Cambrian Baltic Group clay formation
- The Lower Triassic clay formation

Disposal of SNF

(continuation)

The proposed repository concept in crystalline rocks in Lithuania is based on KBS-3 concept developed by SKB for disposal of the SNF in Sweden.



Disposal canister:

- Copper canister
- Canister insert contains 32 for RBMK-1500 SNF fuel half-assemblies
- Diameter of copper shell -1050 mm, height - 4070 mm, wall thickness of copper canister - 50 mm
- For Lithuania SNF disposal ~1400 canisters would be necessary

Determination of RBMK SNF Characteristics

- Estimations of the nuclide content, concentrations, and radiation characteristics of irradiated SNF are essential in solving the problems related to SNF handling, storage and disposal.
- Experimental investigation of such characteristics is expensive and complicated process, therefore computer codes are widely used.

Determination of RBMK SNF Characteristics

- SCALE computer system codes used for depletion analysis are validated for BWR and PWR fuel.
- For example, the SCALE validation for PWR fuel included comparison of calculated and measured concentrations for 14 actinides and 37 fission and activation products for different enrichments and burnup values.

Determination of RBMK SNF Characteristics

- There are no such comprehensive experimental investigations of the irradiated RBMK fuel that are necessary for validation.
- Nevertheless, the feasibility assessment of SCALE system for RBMK fuel was done performing comparison with available experimental and numerical data for this fuel

Calculation Method

- Characteristics of irradiated RBMK fuel were calculated using SCALE 5 code system.
- It determines nuclide content, activities, neutron and gamma source spectrums, and radiation dose rates for a cask.

Calculation method

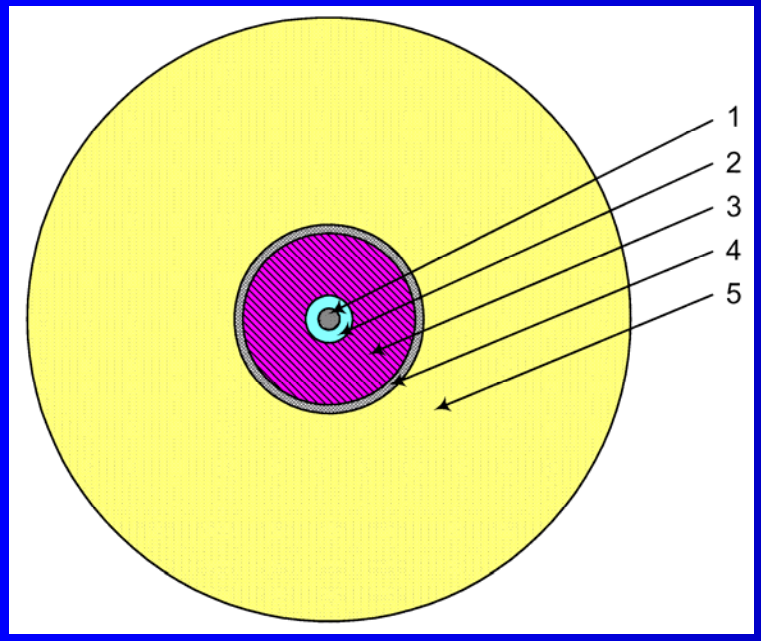
The main input data for SCALE 5 code (SAS2H) are:

- material composition and geometrical parameters of fuel assembly and the fuel channel in the reactor
- material concentrations and temperatures
- reactor power history, irradiation and cooling periods of fuel assembly.

Calculation Method

- For the calculations of irradiated nuclear fuel characteristics the 44GROUPNDF5 library was used.
- This library is developed for use in the analysis of fresh and spent fuel and radioactive waste systems.
- Broad-group boundaries were chosen as a subset of the parent 238GROUPNDF5 boundaries. The resulting boundaries represent 22 fast and 22 thermal energy groups.

Calculation Method

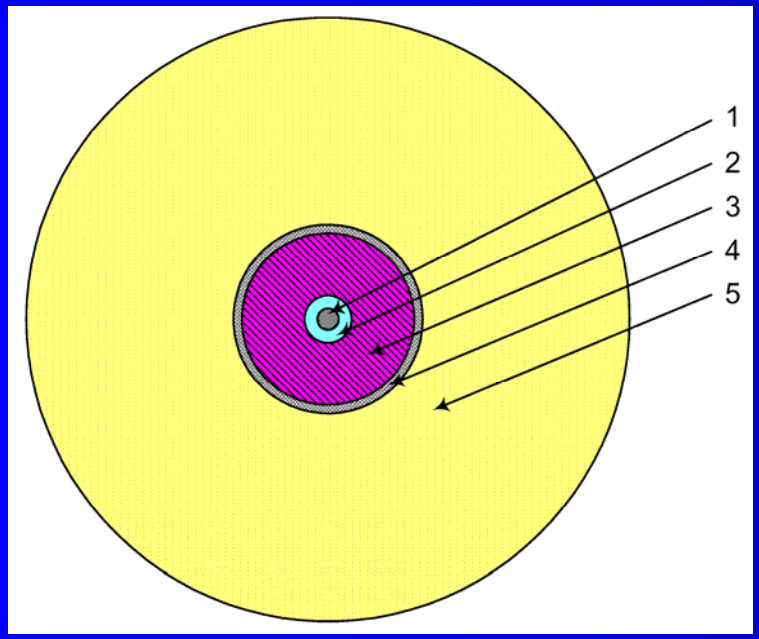


Homogenized FA inside the fuel channel:
1 – carrier rod,
2 – coolant,
3 – homogeneous mixture of UO2 and H2O,
4 – fuel channel tube,
5 – moderator.

The basic assumptions for the modelling of RBMK-1500 fuel assembly irradiation were the following:

- **RBMK-1500 fuel assembly that consists of 18 fuel rods, inside the fuel channel was described as an element of 5 concentric cylinders;**

Calculation Method



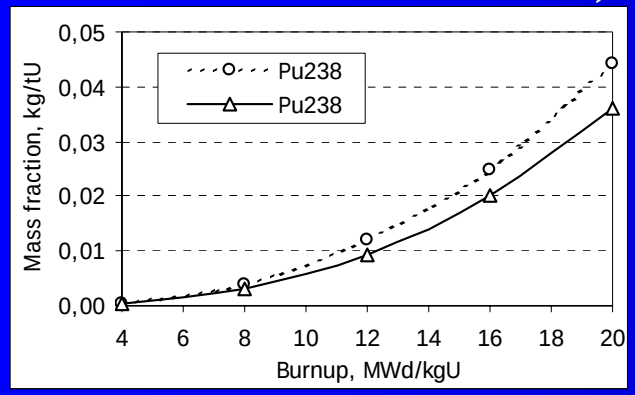
Homogenized FA inside the fuel channel:
1 – carrier rod,
2 – coolant,
3 – homogeneous mixture of UO2 and H2O,
4 – fuel channel tube,
5 – moderator.

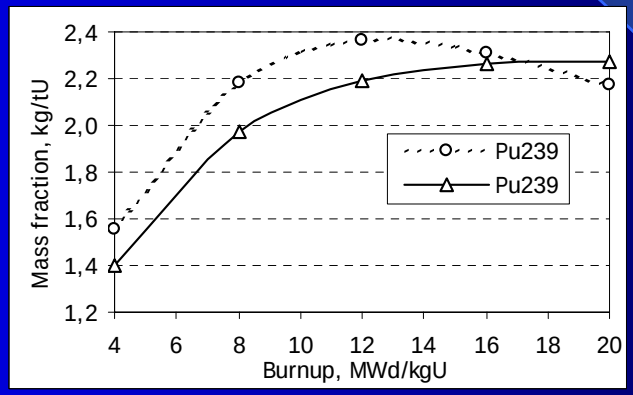
The basic assumptions for the modelling of RBMK-1500 fuel assembly irradiation were the following:

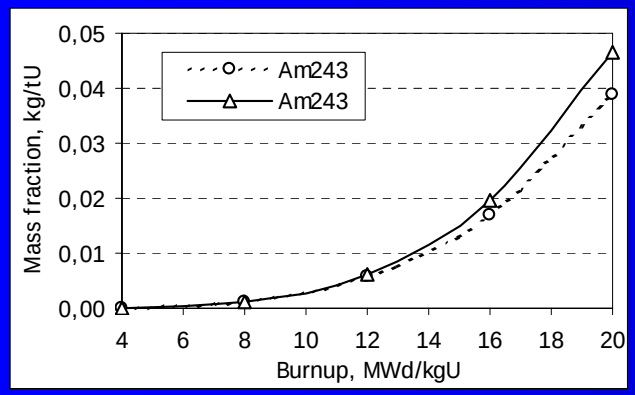
- **235U enrichments: 1.8%, 2.0%;**
- **Burnup varies up to 24 MWd/kgU;**
- **burnup shape of FA assumed as uniform.**

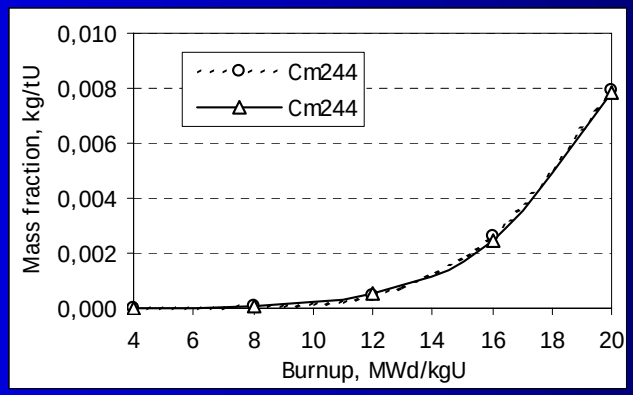
Results (1/5)

Solid line – SCALE results; dotted line – calculations published in "*Zaritskaya T. S., Kruglov A. K., Rudik A. P. The formation of transuranium nuclides in connection with the complex use of VVER and RBMK Nuclear Power Plants // Atomic Energy. 1986. Vol. 46. No. 3. P. 183–185.*")





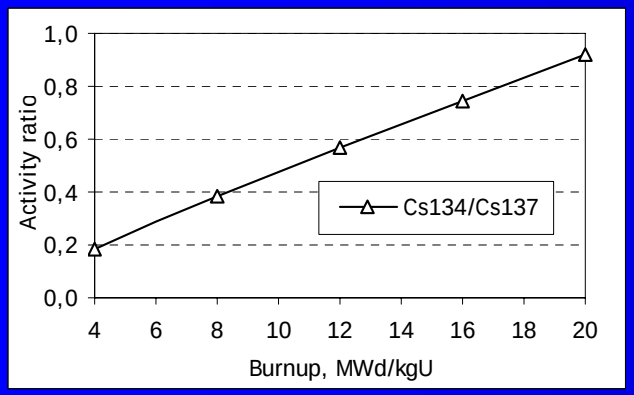


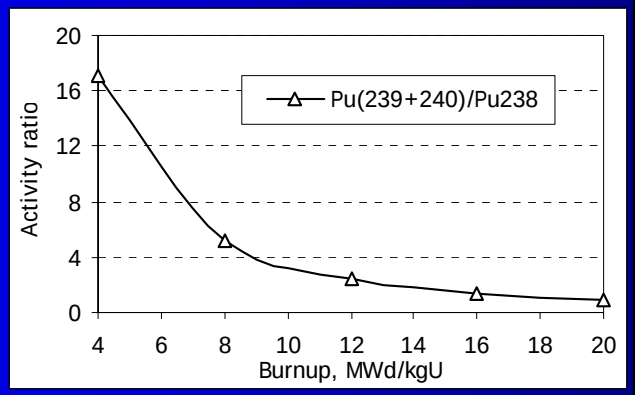


- When burnup varies from 4 to 14 MWd/kgU the difference of results is from 1 to 15%
- When burnup increases from 14 to 20 MWd/kgU the difference is from 15 to 30%.

Results (2/5)

Comparison with the experimental investigations of activities of the plutonium isotopes and fission products in an environmental sample polluted with Chernobyl NPP nuclear waste.



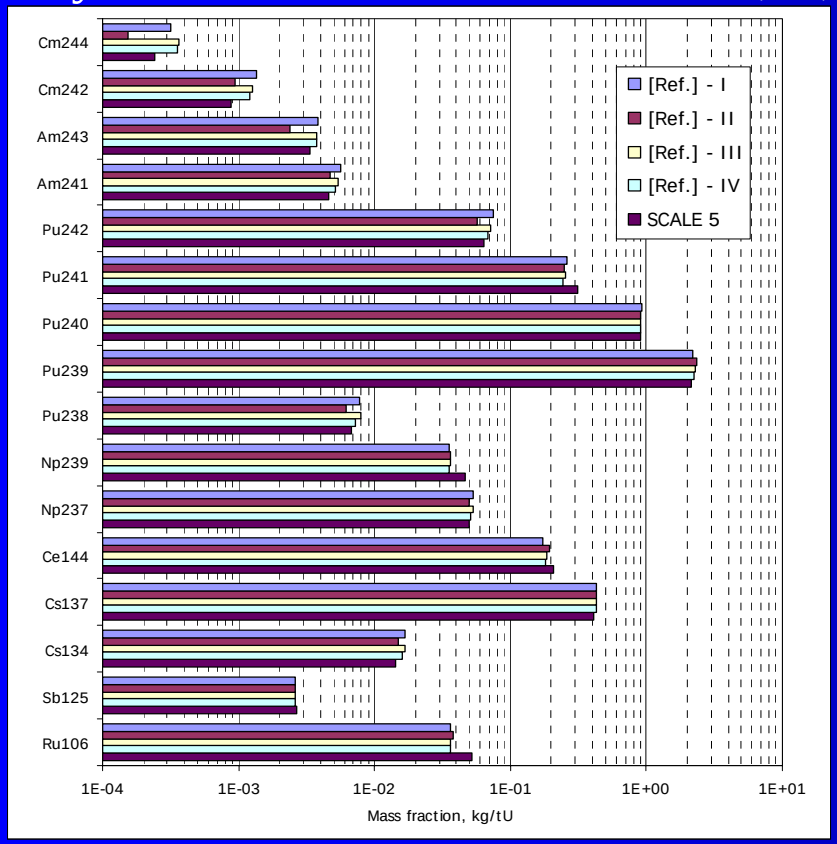


Experimental measurements	SCALE results, corresponding burnup
$(^{239}Pu+^{240}Pu)/^{238}Pu$ =2.6	~11 MWd/kgU
$^{240}Pu/^{239}Pu$=1.25	~9 MWd/kgU
$^{134}Cs/^{137}Cs$=0.53	~11.5 MWd/kgU
$^{144}Ce/^{137}Cs$=15-20	7 – 12 MWd/kgU

According to the Chernobyl NPP reactor loading cartogram average fuel burnup was 10.3 MWd/kgU at the moment of the accident.

Results (3/5)

Comparison of SCALE results with the published mass fractions of different nuclides calculated using different analytical approximations and approaches. Different analytical methods are indicated as I, II, III and IV.



The highest difference between SCALE and analytical methods:

^{244}Cm: 28–53%;

^{242}Cm: 7–36%;

^{243}Am: 10–41%;

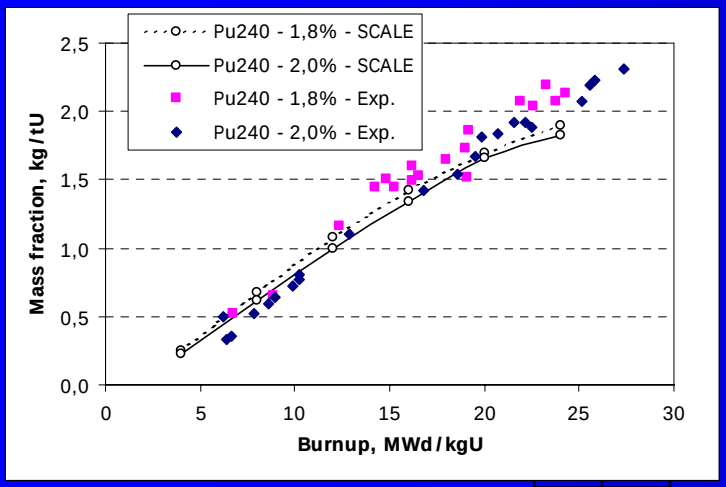
^{241}Pu: 20–28%;

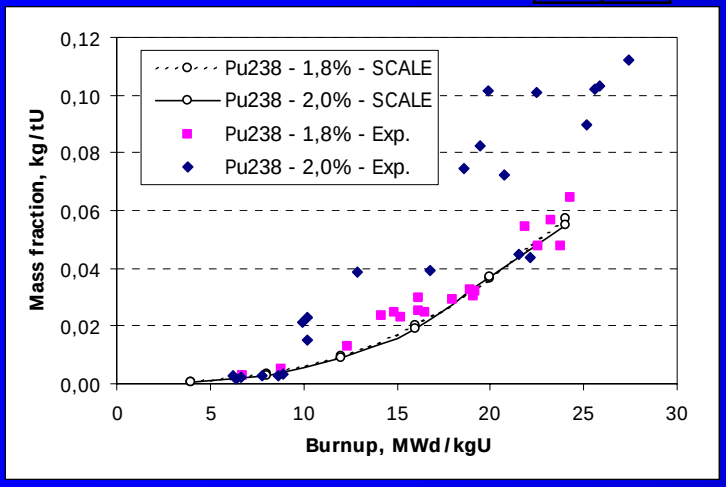
^{239}Np: ~30%;

^{106}Ru: ~40%

For other nuclides: 1 – 15%.

Results (4/5)

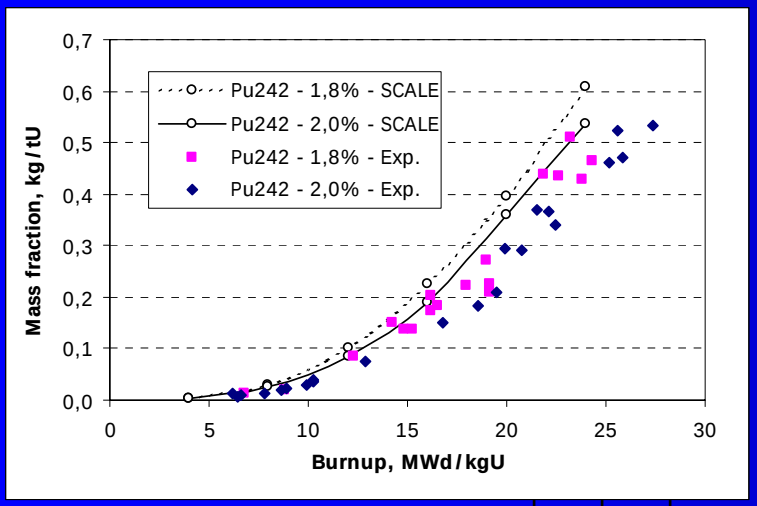


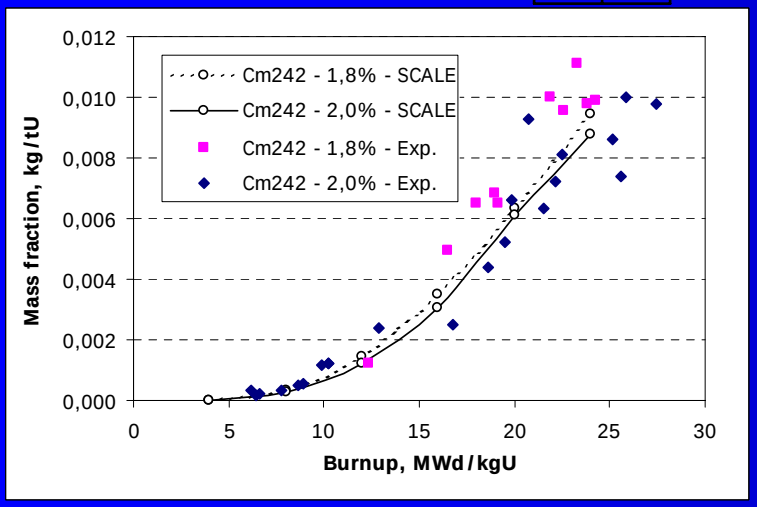


Comparison of SCALE results with experimentally determined concentrations of 238Pu and 240Pu in the RBMK fuel with different burnup and different initial enrichments:

- The accuracy of the experimentally measured 238Pu mass in the samples with 2.0% enrichment is doubtful because.
- Comparison of experimentally measured masses of 240Pu, 242Pu, and 242Cm for 1.8% and 2.0% fuel does not show the significant mass fraction dependence on fuel initial enrichment.

Results (2/5)





Comparison of SCALE results with experimentally determined concentrations of 242Pu, and 242Cm in the RBMK fuel with different burn-up and different initial enrichments:

- The highest difference between SCALE calculations and sample measurements is ~15% for 238Pu (1.8% fuel), ~13% for 240Pu, ~25% for 242Pu, and ~20% for 242Cm.
- The highest differences between calculation results and experimentally measured BWR fuel samples used for the validation of the SCALE depletion codes for 238Pu, 240Pu, 242Pu, 242Cm are ~45%, ~11%, ~43%, ~58%, respectively.

Conclusions

- The comparison of calculated RBMK fuel characteristics using SCALE 5 and published experimental and numerical data obtained using different analytical equations adjusted for RBMK fuel has shown rather good agreement.

- It is reasonable to state that SCALE 5 is quite promising for the determination of the RBMK fuel characteristics, but more comprehensive experimental investigations of the irradiated RBMK fuel characteristics are necessary.

THANK YOU

Workshop on Design and Assessment of Radioactive Waste Packages,
21-22 November 2006 at JRC, Institute for Energy, Petten, The Netherlands

Proficiency Test for Non-Destructive Assay of 220 litres Waste Drums by Gamma Assay Systems

Feasibility study & Design first European Reference Drum

L.P.M. van Velzen (NRG), M. Bruggeman (SCK•CEN),
Ch. Lechner (ARCS), J. Botte (Belgoprocess)

Workshop “Design and Assessment of Packages for Radioactive Waste”,
21and 22 November 2006, Petten, The Netherlands.

Content

- Introduction
  - Aim
    - Principles
      - Questionnaire
        - Response & Results
          - Conclusion

Introduction

- 26th Steering Committee meeting of ENTRAP (2005)
  - NDA waste characterisation laboratories
    - 4th EC Framework project "Round Robin Test"
      - Proficiency test for NDA gamma assay systems
        - Drivers: Operational and post-closure safety, conformation of the waste inventory, sorting and segregation into waste categories, compliance with waste acceptance criteria and QA/QC

Design & Assessment of Packages for Radioactive Waste / *European Reference Drum* / Aim

Aim

Organisation at a regular time interval (e.g. 2 or 3 years) an international proficiency test for NDA gamma assay systems applied for the radiological characterisation of waste.

Principles

- Proficiency test restricted to 220 litre drums
  - No (international) transports of 220 litre drums
    - Every participant an own reference drum
      - Reference drum has to fulfill to all requirements of labs inside the EU, must fit for purpose, must be reusable and will contain only reference sources
        - The costs for the empty drum (once), the reference sources and participation per proficiency test has to be reasonable

Questionnaire

The developed questionnaire contains 6 major sections:

- General *(3)*
- Specifications of empty 220 litre drums *(9)*
- Specifications of waste routinely characterised *(8)*
- Specifications of the radioactive content of the reference drum *(45)*
- Financial aspects *(5)*
- Any other business *(8)*

Design & Assessment of Packages for Radioactive Waste / *European Reference Drum* / Questionnaire / Response

Questionnaire

The questionnaire had been send to about 50 persons in 25 countries of which 8 outside the EU.

Austria, Czech Rep., Turkey, Italy, The Netherlands, Belgium, Japan, South Africa, Slovenia, France, Spain, Finland, Canada, US, United Kingdom, Greece, Germany, EC-Ispra, Ireland, Swiss, Hungary, Romania, Russia, Argentina and Sweden.

Response (country)

~ Returned completely filled in questionnaires *(12)*

~ E-mail contact, but not leading to filled in questionnaires *(5)*

~ No any response received *(8)*

Specifications of empty drum

External dimensions		*Belgium drum*
Average height [mm]	856 ± 38	*(880)*
Average diameter [mm]	606 ± 28	*(606)*
Internal dimensions		
Average height [mm]	835 ± 24	*(835)*
Average diameter [mm]	573 ± 8	*(572)*
Drum material	Steel	*Steel*
Average thickness [mm]	1.1 ± 0.2	*(1)*
Average weight [kg]	17 ± 3.8	*(18)*

Specifications of the waste matrix routinely measured

Average max waste [kg] range	900 – 210
Average min waste [kg] range	350 – 15
Average max waste density [kg/m3]	4000 – 1200
Average min waste density [kg/m3]	1000 – 0
Shielding (Y / N / Blanco)	8 / 6 / 1
Waste matrix varies from	Z = 5 to Z = 14

Specifications of the waste matrix routinely measured

Average max waste [kg] range	900 – 210
Average min waste [kg] range	350 – 15
Average max waste density [kg/m3]	4000 – 1200
Average min waste density [kg/m3]	1000 – 0
Shielding (Y / N / Blanco)	8 / 6 / 1
Waste matrix varies from	Z = 5 to Z = 14

Specifications of the activity content of the waste matrix

Gamma dose rate [µSv/h] range	5000 – 10
Total β/γ activity [Bq] range	3.7e11 – 1e8

Reference nuclides

K-40, Co-57, Co-60, *Se-75, Kr-85*, I-131, Cs-137, Ir-192, Eu-152, NORM, TENORM (Y + Blanco / N) 14 / 1

Depleted Uranium (Y / N / Blanco) 4 / 8 / 3

Accuracy reference sources

Slight preference for “Best accuracy/price ratio” followed by “as accurate as possible”.

Specifications of the activity content of the waste matrix

Gamma dose rate [µSv/h] range	5000 – 10
Total β/γ activity [Bq] range	3.7e11 – 1e8

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Depleted Uranium (Y / N / Blanco) 4 / 8 / 3

Accuracy reference sources

Slight preference for "Best accuracy/price ratio" followed by "as accurate as possible".

Specifications of the activity content of the waste matrix

Reference source distribution

All proposed configurations can be accepted by the majority (Y + Blanco / N) 12 / 3

Measuring time

Min measuring time [ks] range 15 – 0.12

Max measuring time [ks] range 175 – 0.36

Specifications of the activity content of the waste matrix

Reference source distribution

All proposed configurations can be accepted by the majority (Y + Blanco / N) 12 / 3

Measuring time

Min measuring time [ks] range 15 – 0.12

Max measuring time [ks] range 175 – 0.36

Specifications of the activity content of the waste matrix

Allowed physical state of the reference sources

Strong preference for “Solid in certified encapsulation”, and “no liquids”, “no gaseous” and “no toxic/dangerous materials”.

Allowed physical state of the inactive materials

Strong preference for “Solid”, and “no liquids” and “no toxic/dangerous materials”.

Specifications of the activity content of the waste matrix

Allowed physical state of the reference sources

Strong preference for "Solid in certified encapsulation", and "no liquids", "no gaseous" and "no toxic/dangerous materials".

Allowed physical state of the inactive materials

Strong preference for "Solid", and "no liquids" and "no toxic/dangerous materials".

Financial aspects

Reference drum incl. transport [Euro]	2050 ± 1010
Reference sources per proficiency test [Euro]	
2 year interval	1850 ± 1500
3 year interval	2670 ± 2240
Participation per proficiency test [Euro]	
2 year interval	1670 ± 1650
3 year interval	2400 ± 2360

Any other business

Reporting of analyze results [month]	2.6 ± 1.3
Reporting of proficiency test [month]	
Draft report	2.9 ± 0.9
Final report	3.2 ± 0.8
Interest in participation (Y / N / Blanco)	13 / 1 / 1
Assessment of how many institutes:	33
Assessment of really going to participate:	22

Conclusion

- The initiative of the SC of ENTRAP to investigate the need of a new intercomparison was a good idea.
- The number of participants of such an intercomparison will probably larger then expected, based on the response of the questionnaire
- Next phase is the design of the first European Calibration Drum based on the outcome of the questionnaire

End of presentation,

thanks for your attention.

Do you have any questions?

Conceptual Structural Assessment of Polymer Concrete HIC for NPP Waste

Workshop on Design and Assessment of Radioactive Waste Packages

21. Nov. 2006

Dong-Hak Kim, Kyoung-O Nam, Ju-Chan Lee, Kyoung-Sik Bang and Ki-Seog Seo

Spent Fuel Management Technology Research Division

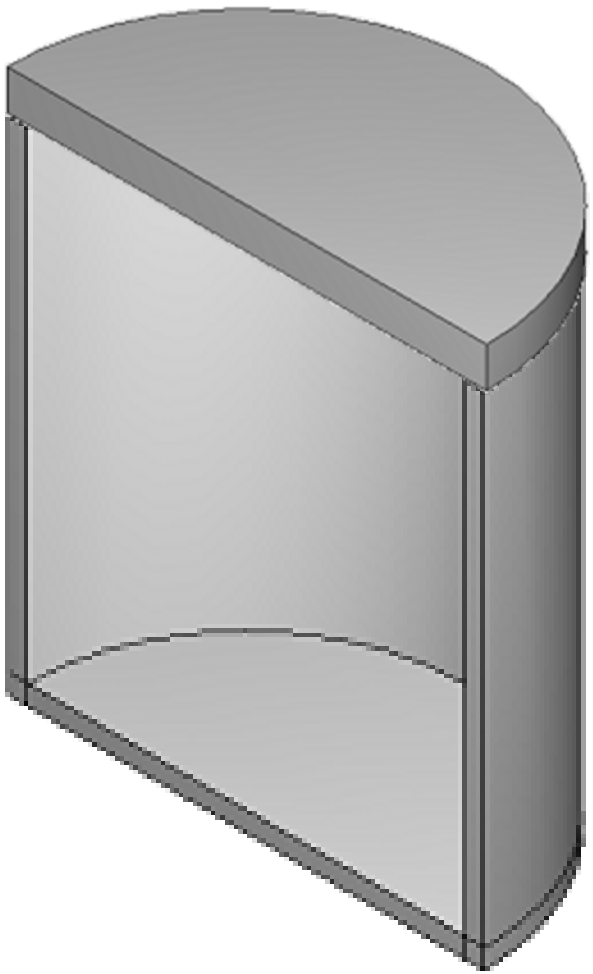
Korea Atomic Energy Research Institute

Contents

Polymer Concrete High Integrity Container

- The dimension is 1,200 mm diameter and 1,260 mm height
- High Integrity Container is constituted by a body and lid
- The thicknesses of the wall and the lid are 50 mm and 100 mm, respectively.
- In this presentation, In accordance with transport conditions, the conceptual structural assessments were conducted.
- The basic model is made from pure polymer concrete.
- The model for the analysis didn't include gripper, O-ring and groove etc.
- Reinforced concept using in-plate steel, reinforced concept covered steel plate and reinforced concept using steel structure were considered.
- Drop, penetration, stacking and handling analysis were conducted.

Basic model



	Volume (m^3)	Density (ton/m^3)	Weight (kg)
Body	275	2.3	633
Lid	113	2.3	260
Wastes	1,096	1	1,096
Total Weight			1,989

Modified with the round corner with 50 mm

Modified using steel

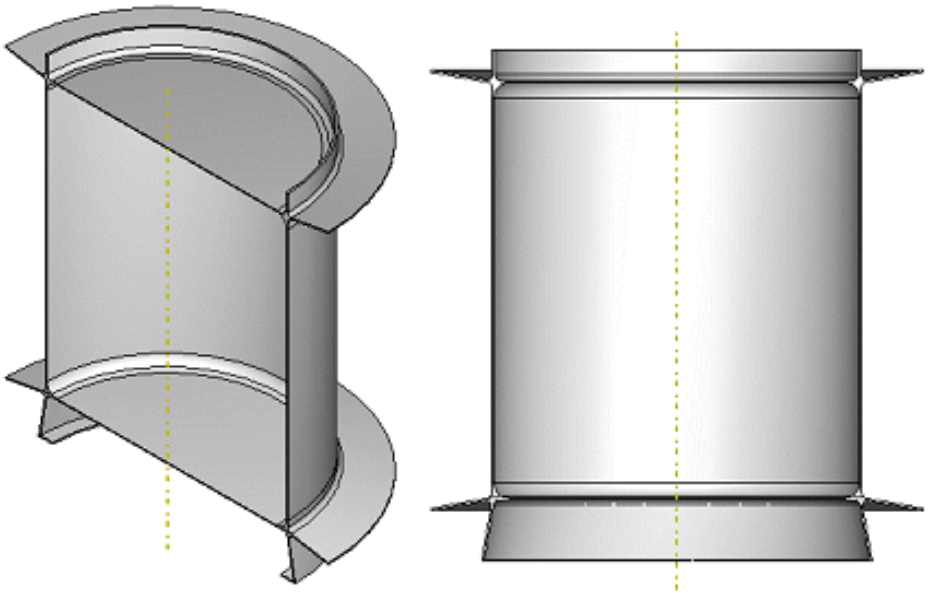
Reinforced concept using in-plate steel

	Volume (m^3)	Density (ton/m^3)	Weight (kg)
Body	221.2	2.3	508.8
Steel (Body)	53.9	7.8	420.6
Lid	100	2.3	230
Steel (Lid)	13.1	7.8	102.2
Wastes	1,096	1	1,096
Total Weight			2,357.6

Thickness of inner and outer shell : 20 mm

Thickness of in-plate steel : 10 mm

Reinforced concept covered with steel plate



Thickness of polymer concrete : 45 mm, 48 mm
Thickness of covering steel plate : 5 mm, 2 mm
Length of steel absorber : 200 mm, 150 mm
Thickness of steel absorber : 7 mm, 2 mm
Inclined angle of steel plate : 10°

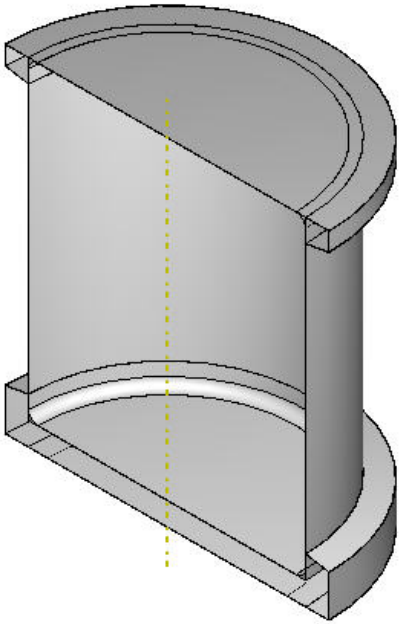
Reinforced concept covered with 5 mm steel plate

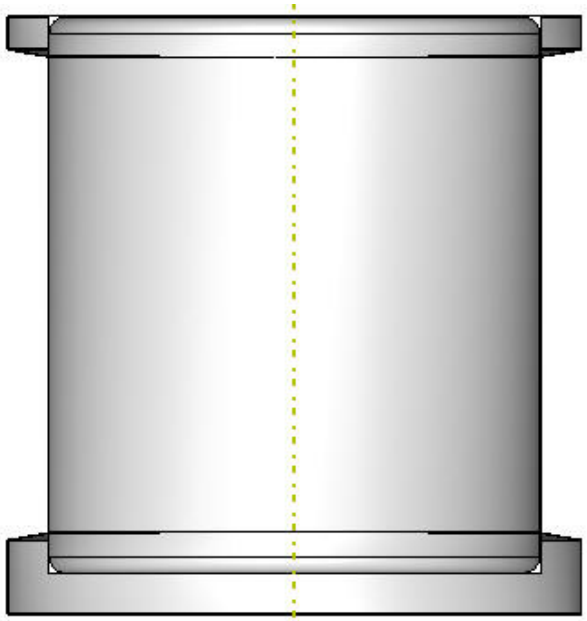
	Volume (m^3)	Density (ton/m^3)	weight (kg)
Body + Lid	348.8	2.3	802.2
Steel Case	65.3	7.8	509.5
Wastes	1,148.7	1	1,148.7
Total Weight			2,460.4

Reinforced concept covered with 2 mm steel plate

	Volume (m^3)	Density (ton/m^3)	weight (kg)
Body + Lid	372	2.3	855.6
Steel Case	22.4	7.8	174.7
Wastes	1,148.7	1	1,148.7
Total Weight			2,179

Reinforced concept using steel structure



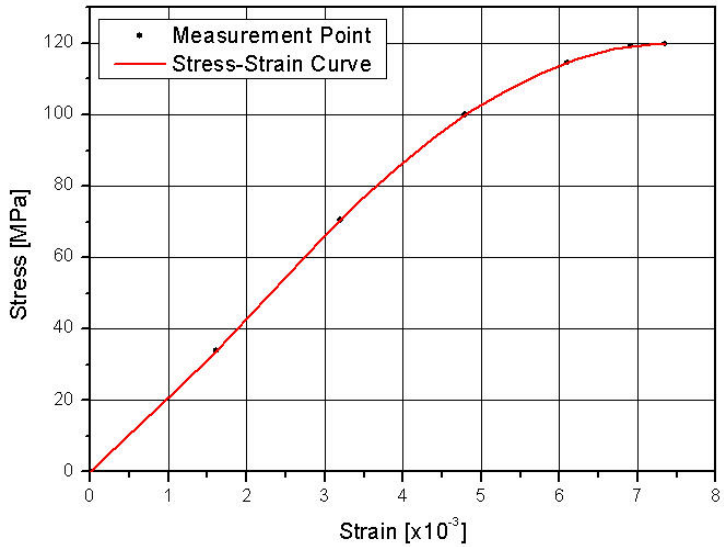


	Volume (m^3)	Density (ton/m^3)	weight (kg)
Body + Lid	372	2.3	855.6
Steel Case	25	7.8	195
Wastes	1,096	1	1,096
Total Weight			2,146.6

Determine the Material Properties

- Properties Test of Polymer Concrete
  - using standard specimen with 30 cm height and 15 cm diameter
  - for compressive strength , tensile strength, Young's modulus, Poisson's ratio
  - determined by average value using three specimen
  - Density was determined using the volume and weight of specimen

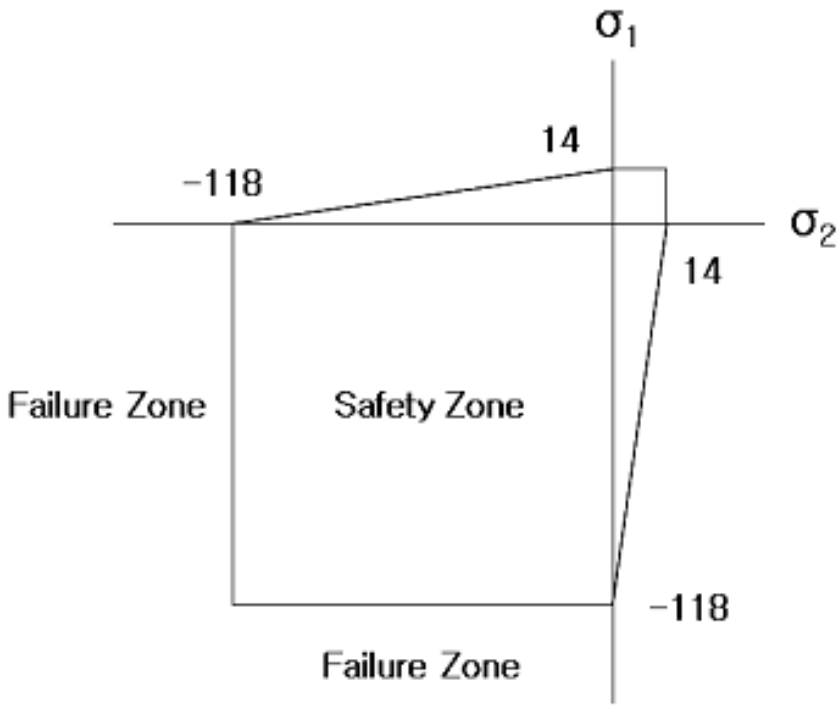
Material Properties	Polymer Concrete
Young's modulus (GPa)	21.764
Poison's ratio	0.16
Compressive strength (MPa)	118
Tensile strength (MPa)	14
density (ton / m^3)	2.3



Compressive stress-strain curve

Determine the Material Properties

- To evaluate failure of polymer concrete we use the failure curve for polymer concrete.
- We determine the failure curve of the polymer concrete using the tensile strength and compressive strength.
- Evaluation is conducted based on the element.
- If the maximum principal stress of element were larger than the tensile strength or the minimum principal stress smaller than the compressive stress, the element would fail.
- Element made by steel was evaluated using von-Mises stress

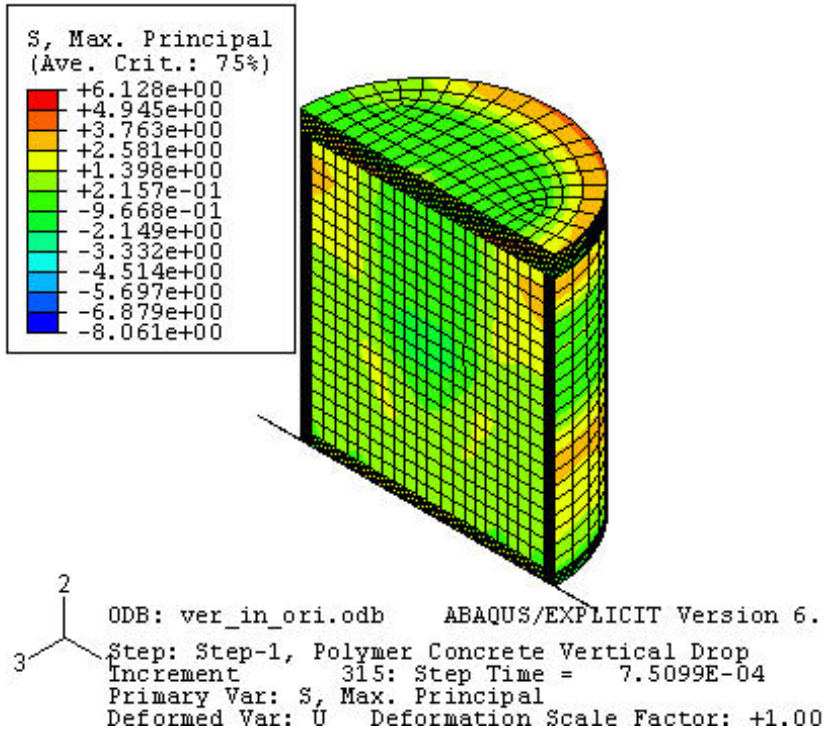


Failure curve for polymer concrete

Results of drop analysis for basic model

- We checked the stress at the center and corner of the impact part
- The maximum principal stress and minimum principal stress in accordance with the location are evaluated
- Location 1 is the inner of the shell
- The last number is the outer of the shell

vertical				
location	center		corner	
	max	min	max	min
1	3.18	-3.7	-1.29	-33.44
2	2.94	-3.7	-0.22	-23.27
3	2.48	-2.72	1.64	-22.44
4	2.27	-3.01	0.57	-21.27
5	1.87	-3.92	-1.85	-20.46
6	1.47	-4.84	-2.95	-19.81



Maximum principal stress contour for vertical drop of a basic model

Results of drop analysis for basic model

horizontal				
location	center		corner	
	max	min	max	min
1	8.23	-19.09	6.22	-117.4
2	9.38	-23.44	10.548	-106.16
3	10.92	-34.96	13.38	-108.48
4	7.72	-52.64	12.99	-115.46
5	-0.69	-87.73	17.85	-155.12
6	-5.5	-112.31	-6.92	-190.65

corner		
location	corner	
	max	min
1	-2.3	-172.74
2	9.7	-307.47
3	29.56	-571.49
4	45.74	-999.14
5	200.79	-1904.97
6	-516.14	-3130.03

Principal stress for a horizontal drop

Failure is occurred at the corner

Principal stress for a corner drop

Failure is occurred at the corner

Results of drop analysis for reinforced model

Reinforced concept using in-plate steel

Vertical				
location	center		corner	
	max	min	max	min
1	-2.73	-8.25	0.38	-2.83
2	-2.26	-8.92	1.97	-39.01
3	-1.79	-9.59	0.06	-36.24
4	41.88		33.98	
5	39.95		229.64	
6	38.03		227.26	
7	-1.23	-11.82	10.8	-240.78
8	-1.2	-11	4.35	-121.17
9	-1.17	-10.19	5.42	-119.21

This part is under safe condition

Failure at the corner is occurred

Principal stress for a vertical drop

Results of drop analysis for reinforced model

horizontal				
location	center		corner	
	max	min	max	min
1	5.38	-15.27	3.06	-12.59
2	9.93	-20.38	10.9	-185.57
3	10.62	-22.09	10.9	-176.42
4	228.88		174.15	
5	162.362		278.14	
6	95.84		305.41	
7	-0.1	-101.15	82.6	-371.26
8	-14.64	-151.15	-41.57	-371.26
9	-29.19	-201.16	-56.72	-386.24

Principal stress for a horizontal drop

Failures is occurred

corner		
location	corner	
	max	min
1	3.4	-11.9
2	-9.76	-126.1
3	-9.76	-161.27
4	136.29	
5	1293.95	
6	1390.9	
7	-1185	-2791.14
8	-476.81	-4264.09
9	-246.21	-4323.68

Principal stress for a corner drop

Failures is occurred

Reinforced concept covered with steel plate

Reinforced model covered with 5 mm steel plate

vertical				
location	center		corner	
	max	min	max	min
1	-0.52	-14.8	-0.04	-4.85
2	-1.8	-7.01	2.35	-2.05
3	0.44	-2.54	0.09	-0.8
4	88.75		25	
5	95.77		29	
6	101.8		36	

Principal stress for a vertical drop

horizontal				
location	center		corner	
	max	min	max	min
1	0.6	-4.1	0.87	-6.7
2	1.1	-4.2	0.9	-5.62
3	0.75	-6.75	-0.5	-25.3
4	20.1		204	
5	21.4		282	
6	22		307	

Principal stress for a horizontal drop

Reinforced concept covered with steel plate

corner		
location	corner	
	max	min
1	-0.6	-19.1
2	1.68	-12.03
3	3.1	-24.38
4	97.45	
5	171.41	
6	197.66	

Principal stress for a corner drop

Structural integrity is maintained

But this concept is heavy.(2.46 ton)

Reinforced concept covered with steel plate

Reinforced model covered with 2 mm steel plate

vertical				
location	center		corner	
	max	min	max	min
1	0.53	0.02	2.06	-1.27
2	1.5	-0.08	2.18	-1.64
3	1.4	-0.23	0.22	-8.38
4	1.14	-0.37	-0.02	-8.72
5	54.58		41.55	
6	54.76		58.1	
7	55.25		104.47	
8	55.75		161.96	

Principal stress for a vertical drop

Horizontal				
location	center		corner	
	max	min	max	min
1	2.32	-2.3	2.54	-9.7
2	2.02	-0.07	4.15	-7.6
3	2.83	-0.3	5.37	-6.1
4	3	-2.5	3.83	-6.3
5	34.6		203	
6	34.5		277	
7	35.5		315	
8	36.1		582	

Principal stress for a horizontal drop

Reinforced concept covered with steel plate

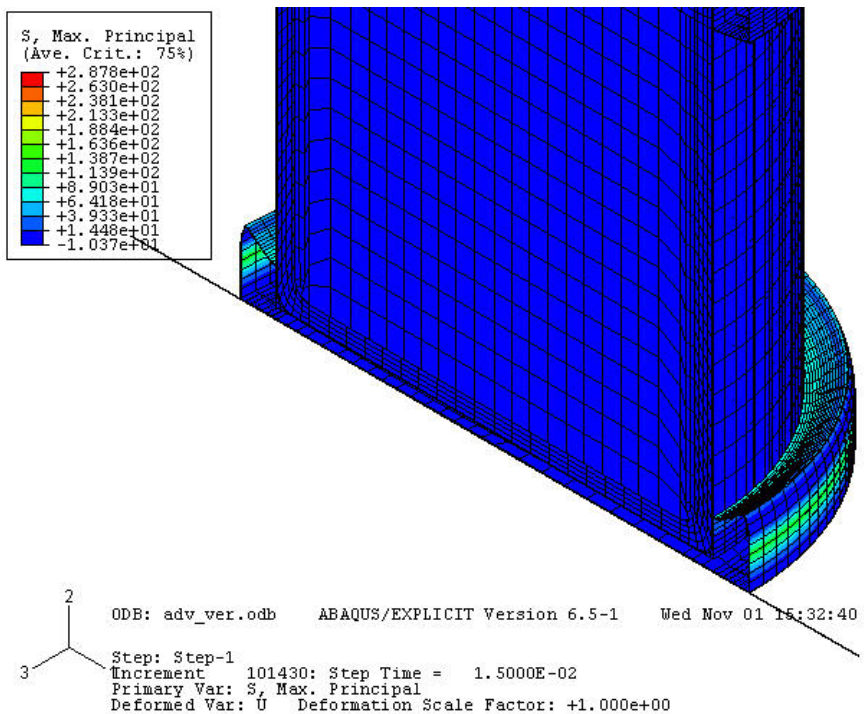
corner		
location	corner	
	max	min
1	0.96	-8.62
2	0.97	-10.9
3	5.4	-15.2
4	5.1	-32.4
5	186	
6	220	
7	316	
8	330	

Principal stress for a corner drop

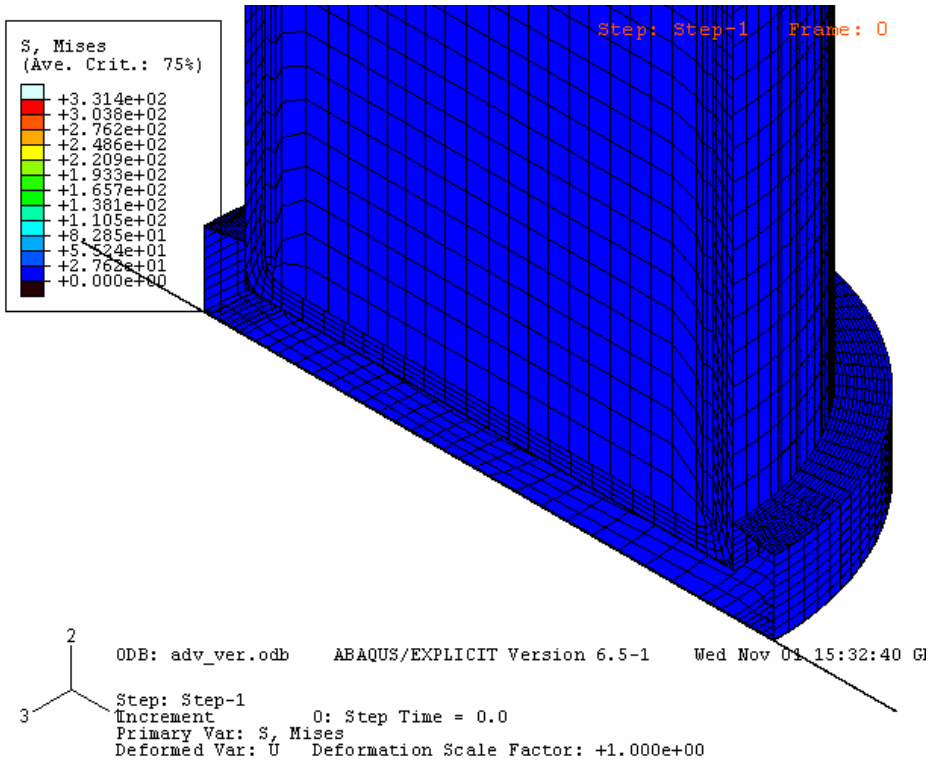
The outer steel is yielded

Reinforced concept using steel structure

- Stresses at the polymer concrete is very low
- The steel structure is absorbing almost of the drop impact energy

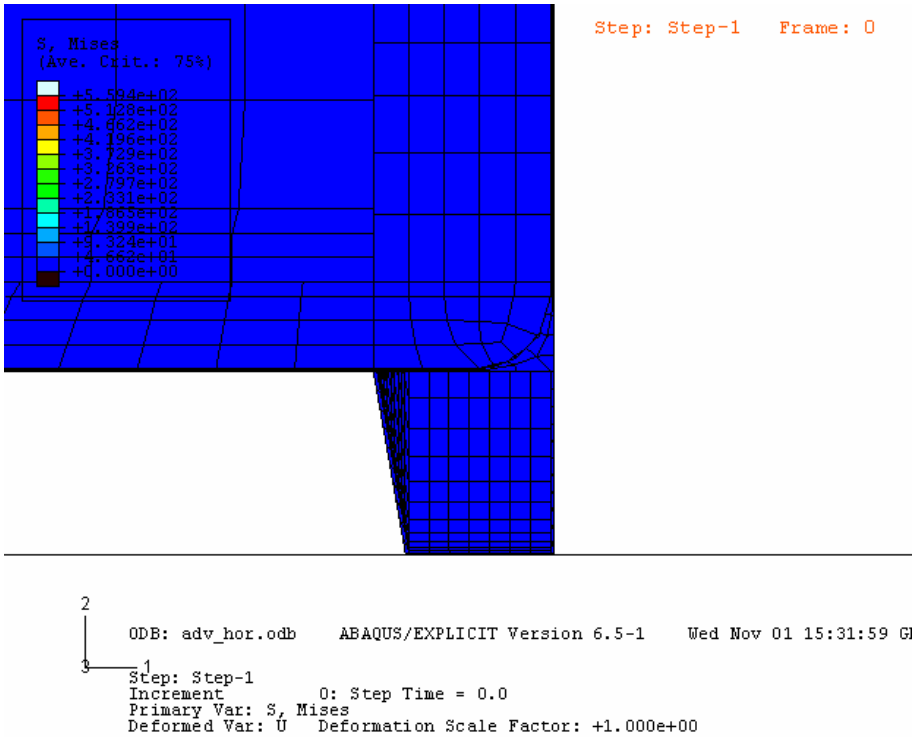


Maximum principal stress

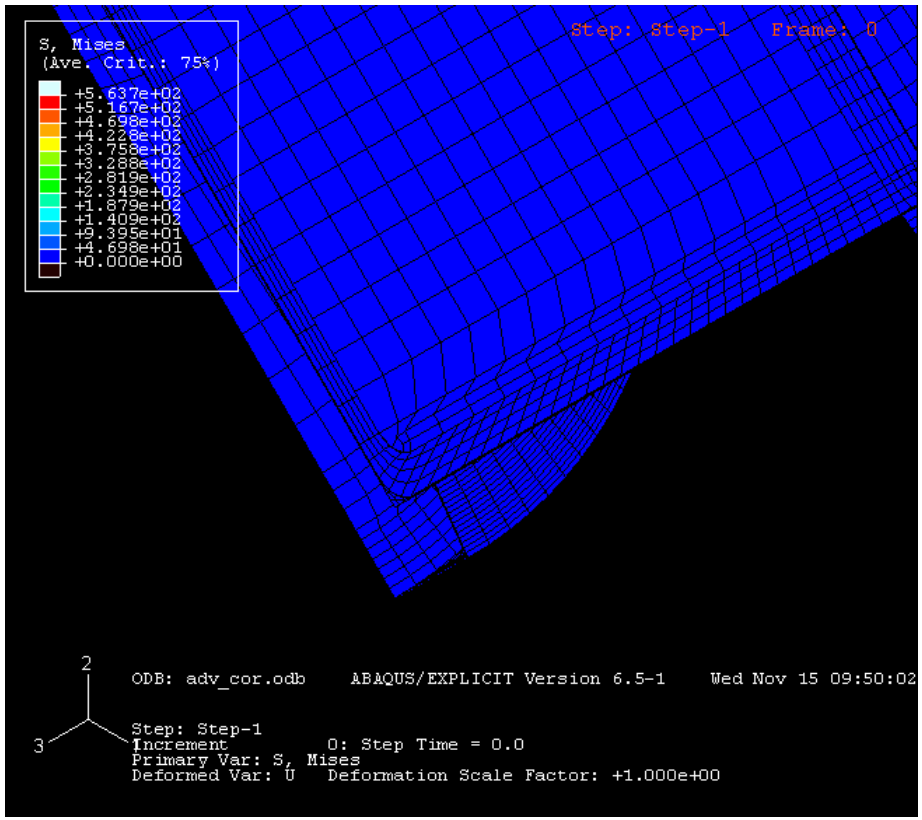


a vertical drop

Reinforced concept using steel structure



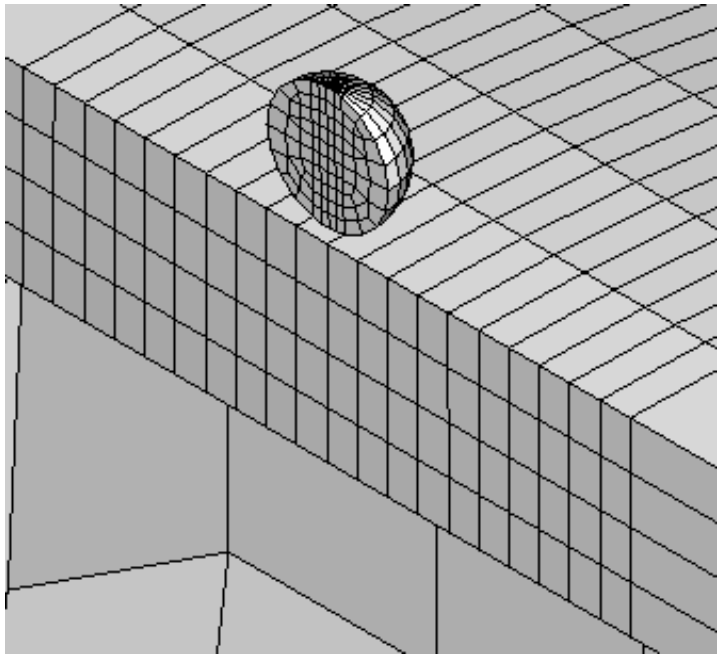
a horizonal drop



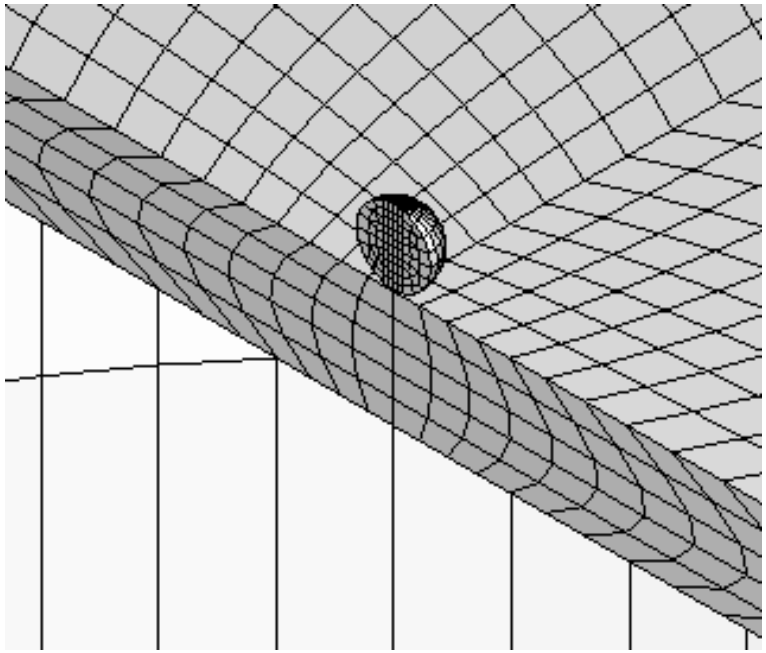
a corner drop

Penetration analysis

- A bar of 3.2 cm in diameter with a hemispherical end and a mass of 6 kg shall be dropped on the center of the weakest part of the specimen
- A side penetration and bottom penetration
- Check the failure of the shell



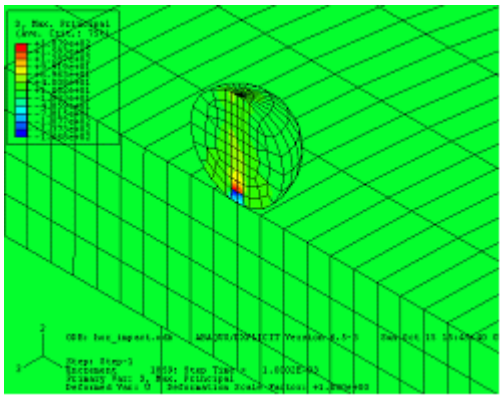
Side penetration



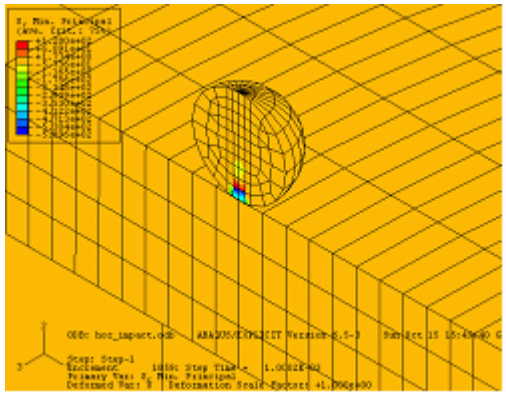
bottom penetration

Result of penetration analysis

Side penetration

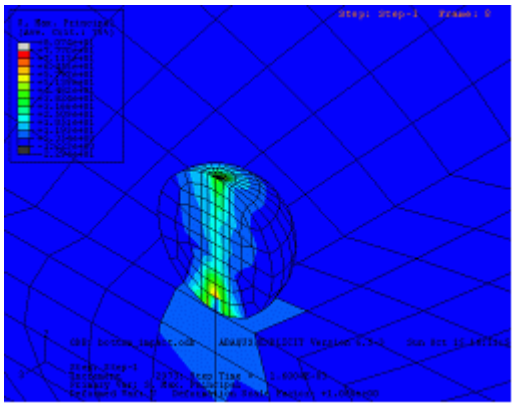


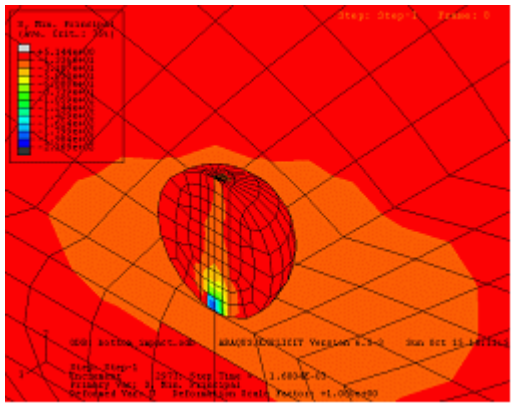
Maximum principal stress



Minimum principal stress

Principal stress (MPa)	center	
	max	min
1	-9.46	-105.3
2	4.1	-30.2
3	4	-2.7
4	2.6	-0.1



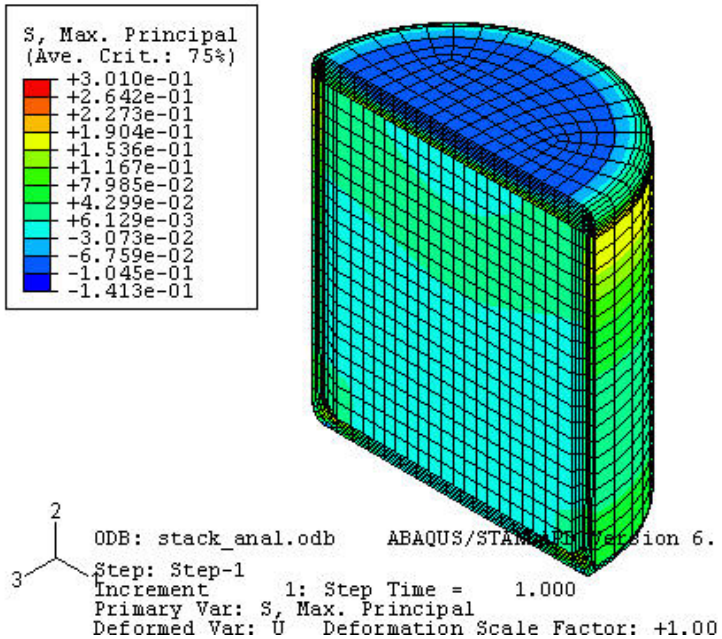


Principal stress (MPa)	center	
	max	min
1	1.3	-53.3
2	1.6	-5.8
3	1.6	-1.1
4	0.9	-1.8
5	0.5	-8.4

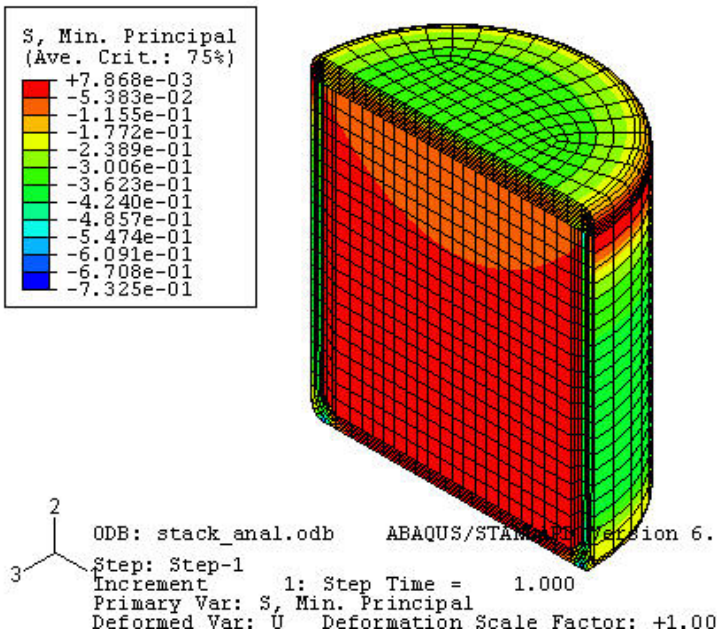
Bottom penetration

The polymer concrete is maintaining the structural integrity

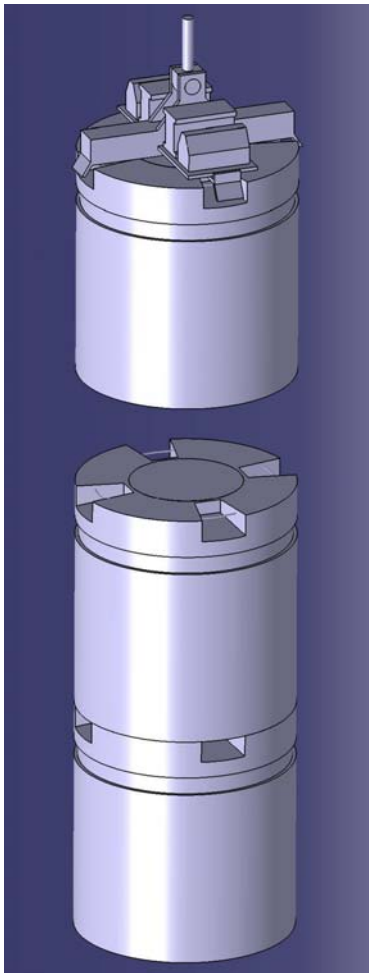
Stacking analysis



Maximum principal stress



Minimum principal stress

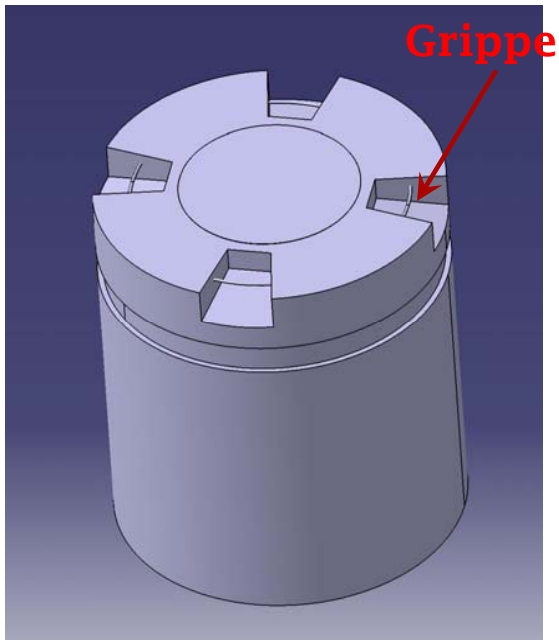


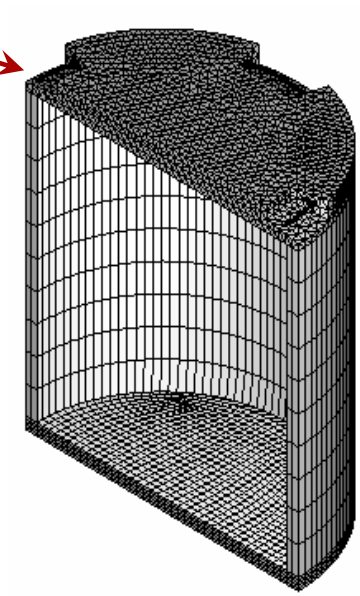
	Lid	Wall	Bottom
Max. Stress (MPa)	0.15	0.15	0.01
Min. Stress (MPa)	-0.3	-0.4	-0.05
evaluation	Safe	Safe	Safe

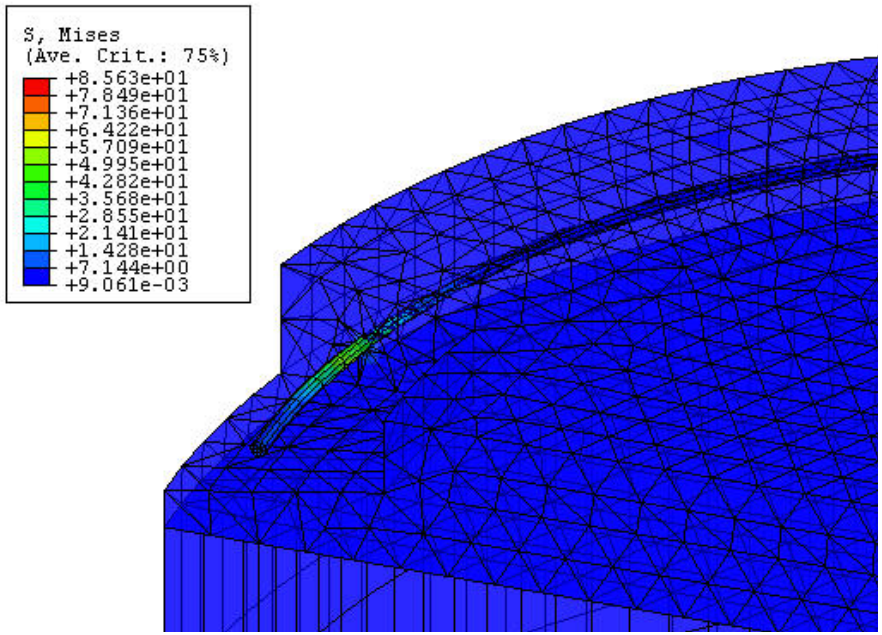
Handling analysis

- Model using the gripper for lifting 3 times weight of total package weight are considered.
- The diameter of gripper are 10 mm, 15 mm, 20 mm.
- At the gripper the maximum stress are lower than the yield stress.

Steel Ring Diameter (mm)	10	15	20
Max. Stress at the gripper (MPa)	85.6	33	24
Results	Safe	Safe	Safe







Conclusion

- In this presentation, the mechanical properties of the polymer concrete were determined and the conceptual safety analysis of polymer concrete HIC was conducted
- The failure curve of the polymer concrete was determined using uni-axial tensile and compress test.
- The basic design using pure polymer concrete is failed by 1.2 m drop analysis
- The reinforced concept are applied using steel
- Reinforced concept using in-plate steel, reinforced concept covered steel plate and reinforced concept using steel structure were considered.
- Reinforced concept using steel structure is not failed under the drop condition
- For the penetration, stacking and handling conditions, the polymer concrete high integrity container is safe.

Comparison of the Behaviour of a Finned and an Unfinned Surface during a Drop Test of a Cask onto a Steel Bar

R. Hueggenberg, H-P. Winkler, J. Holtorf

GNS Gesellschaft für Nuklear-Service mbH

Contents

1. Purpose of the Test

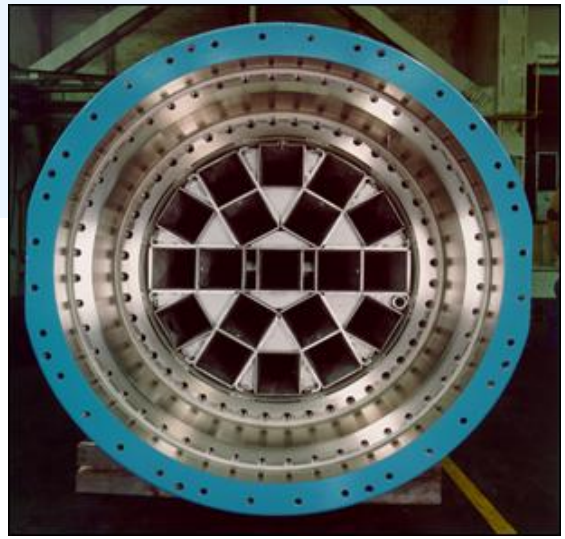
- Based on the requirements of transport regulations, a drop from a height of 1 m onto a steel bar must be assessed during the structural analysis.
- A high strain occurs in the cask body due to the locally induced load force from the bar
- For the impact of the bar onto the finned surface, no sure quantifiable strain with the aid of numerical methods
  - of the complex geometry
  - of the plastic/dynamic material failure
  - and of sensitive impact conditions
- Experimental Investigation of the Effectiveness of the Fins during the Drop Test
  - Comparison of at least two drop tests
  - A) with fins and B) without fins
  - Determination of the induced force of the bar and the strain

2. Requirements

- According to IAEA §726, the 1m-drop onto a steel bar is one part of a sequence of two structural/mechanical tests followed by a thermal test.
- The sequence and boundary conditions (drop orientations) of the two mechanical tests is to be selected in such a way that they cause the maximum damage.
    - For example, the functionability of a penetration protection is to be maintained after the 9 m drop
- The boundary conditions for the test with the 1 m drop onto a steel bar are defined in §727(b).
- According to IAEA §637 and §664, a temperature range of -40 °C to 70°C is to be considered. In the process, no credit may be taken from the thermal power of the contents according to IAEA-Advisory §664.3.

3. Example of a Reference Cask – the CASTOR® V/19

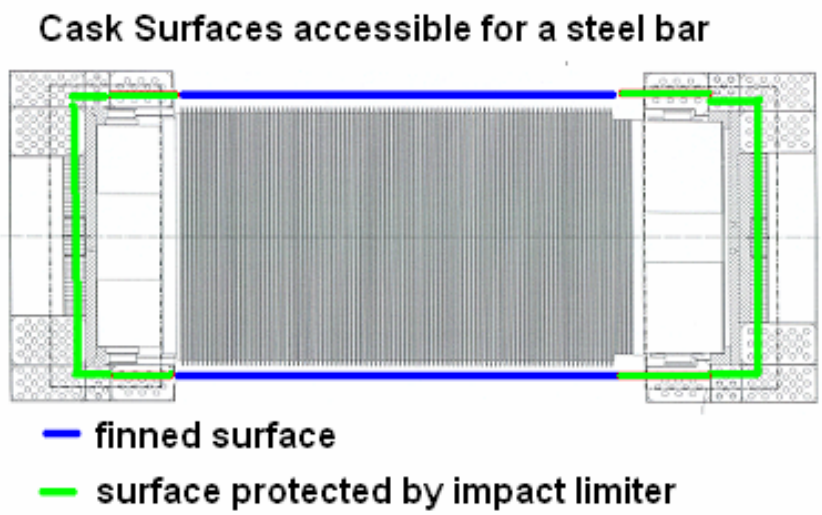
Cask Body Material:	Metal Cask with Ductile Cast Iron Body
Total Height:	5862 mm
Outer Diameter:	2436 mm
Cask Weight Loaded:	125.6 t max.
Loading:	19 F/A from which 4 MOX-F/A
PWR-Fuel Assemblies:	16x16-20 and 18x18-24
Max. Initial Enrichment:	4.45 wt % U-235
Max. Average Burnup:	65 GWd/MT HM
Max. Heat Load:	39 kW
Total Activity:	5.5·E+17 Bq

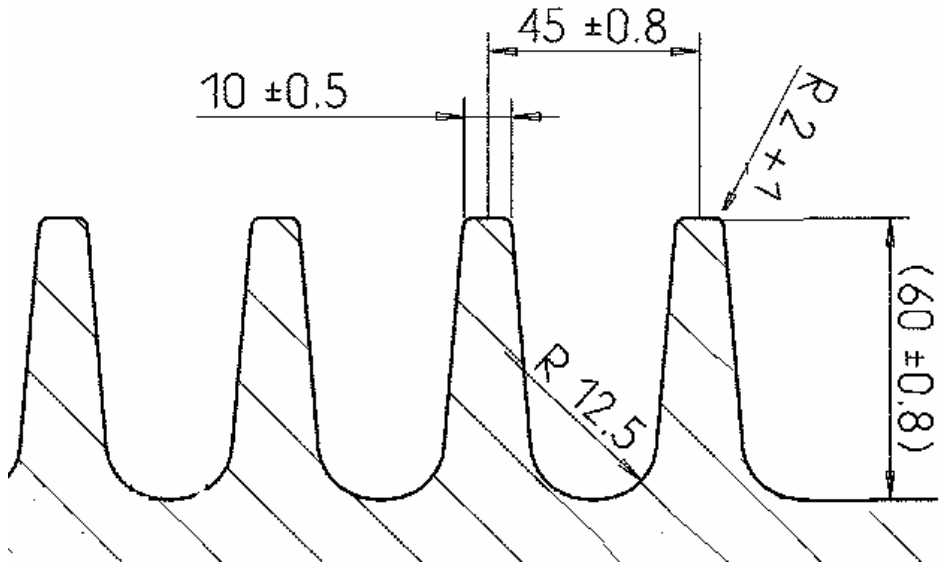


In April 2004, the 100th CASTOR® V/19 was delivered to its Customer.

3. Reference Cask

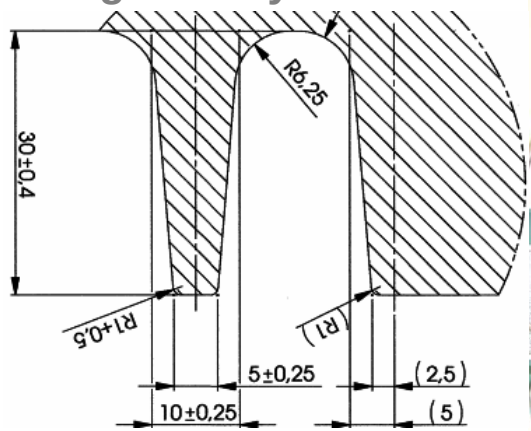
- Due to the requirements for a 1m drop test, the impact location has to be chosen where the maximum damage of the cask will occur.
- Areas underneath the impact limiter are protected by inserts which are designed for this purpose
- Due to the heat load the cask surface area is covered with fins which can not be protected by other components
- Currently the fins were conservatively not taken into consideration for the structural analysis and the stresses inside the cask were overestimated.





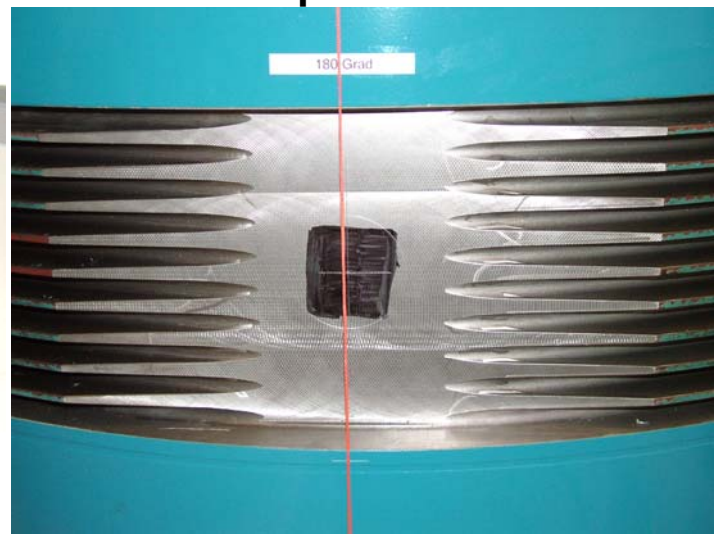
4. Test Cask

- Remachining of the prototype cask made in 1982
  - 9 radial fins around the axial center of gravity
  - Fin geometry as in CASTOR® V/19 (Scale 1:2)



  - Mass for Drop Test : 17310 kg
- 1. Test R1
  - Flattening of the finned area down to the base at 180° -> unfinned impact location
- 2. Test R2
  - Finned impact location at 0 °

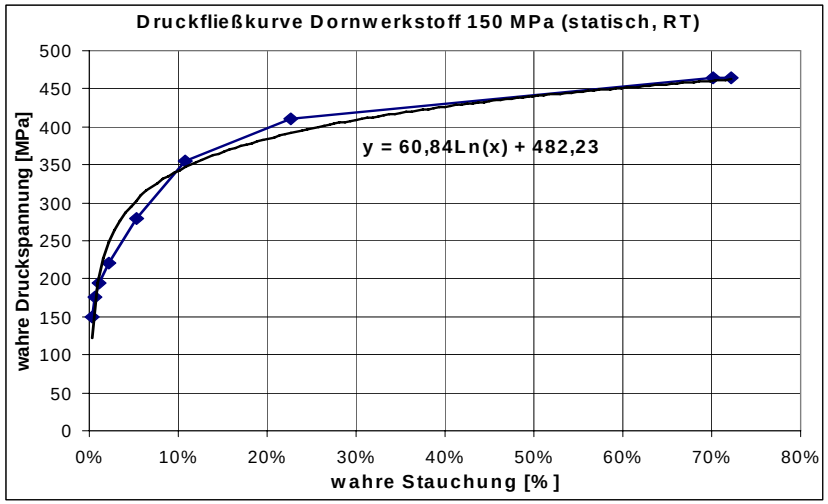
Test R1 – Impact location

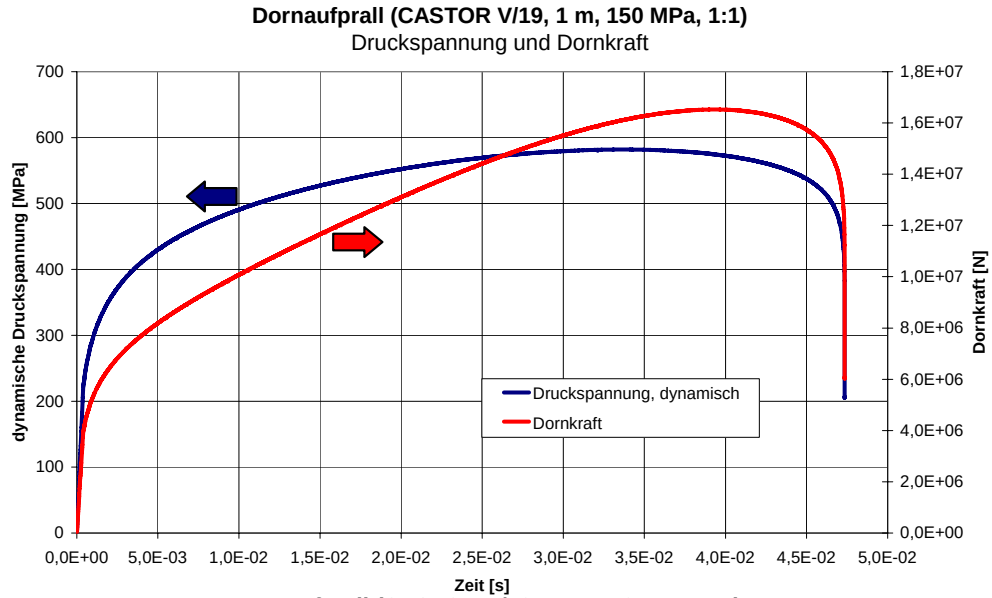


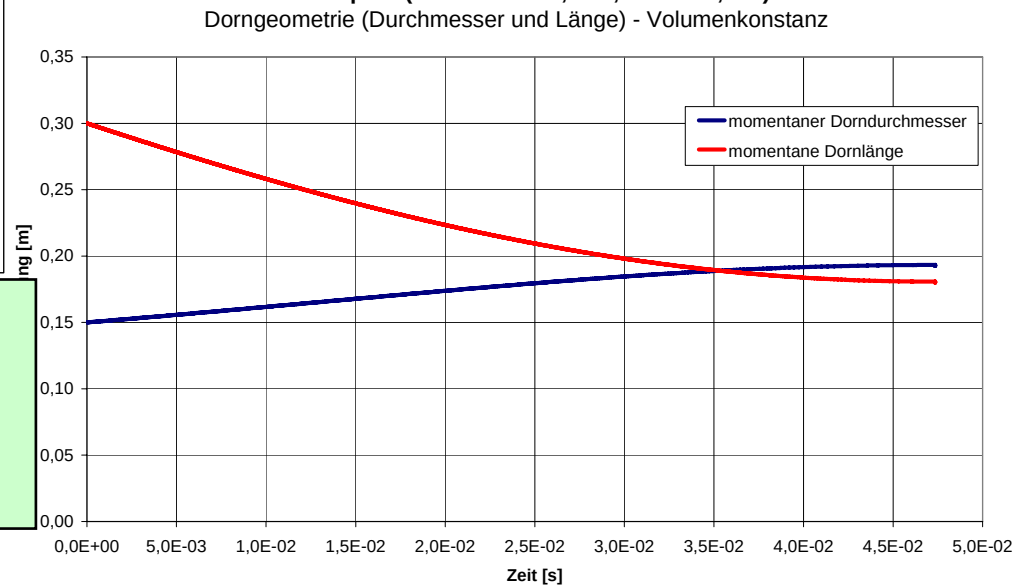
Test R2 – Finned impact location

5. Steel Bar Deformation

- Estimation of the Behaviour during the Impact of the Steel Bar
  - Dropping energy = Deformation energy of the bar
  - Unyielding cask body, fins neglected
  - One dimensional numerical calculation with EXCEL
  - Bar with 150 MPa (compression limit), simplified dynamic flow curve







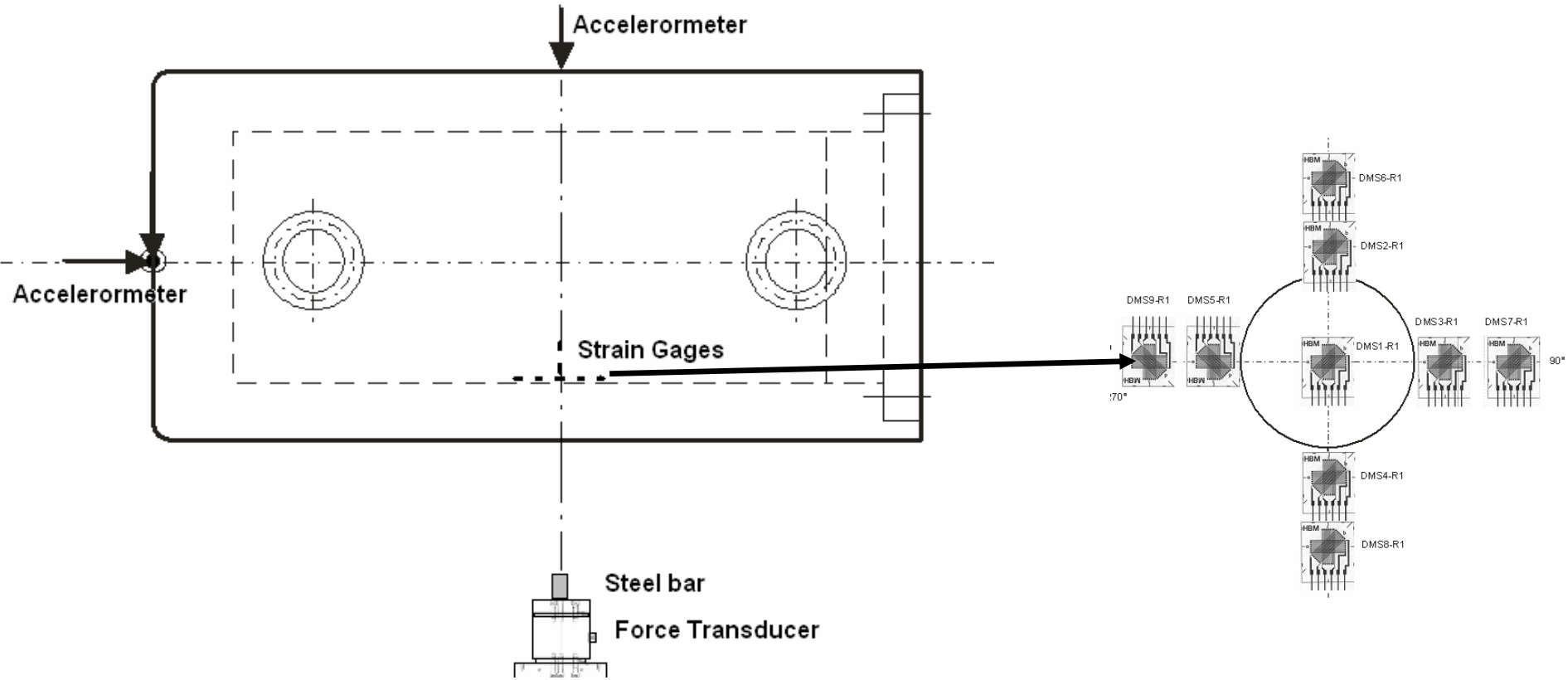
- **Impact duration: 47,5 ms**
- **Compression path: 119,7 mm**
- **Max. penetration force: 1,66E7 N**
- **Max. compressive stress: 582 MPa**

6. Measurements during drop test

- Test temperature: ambient appr. 20 °C
- Measurement of the axial penetration force
  - By means of transducer between bar and substrate
- Measurement of the strains on the cask cavity wall
  - By means of strain gages (rosettes) at a minimum of 5 measuring locations
  - 1 measuring location in the axis through the bar and center of gravity
  - 2 measuring locations at approx. 100 mm axially next to the axis of impact
  - 2 measuring locations at approx. 100 mm tangentially next to the axis of impact
- Measurement of the cask deceleration
  - 1 accelerometer in the axis of impact
- Visual recording of the deformation process over time
  - By means of high-speed camera
  - Path/Time measurement by means of extensometer
- Dimensional Check of the Steel Bar and Impact Area before and after testing

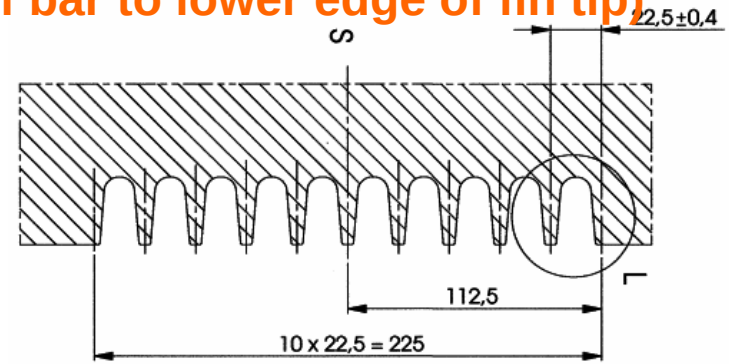
6. Locations of Measurement Devices

- The load force of the steel bar acting onto the cask body was measured with a force transducer
- Strain inside the cask cavity opposite to the impact area was measured with 9 multi-axial strain gages



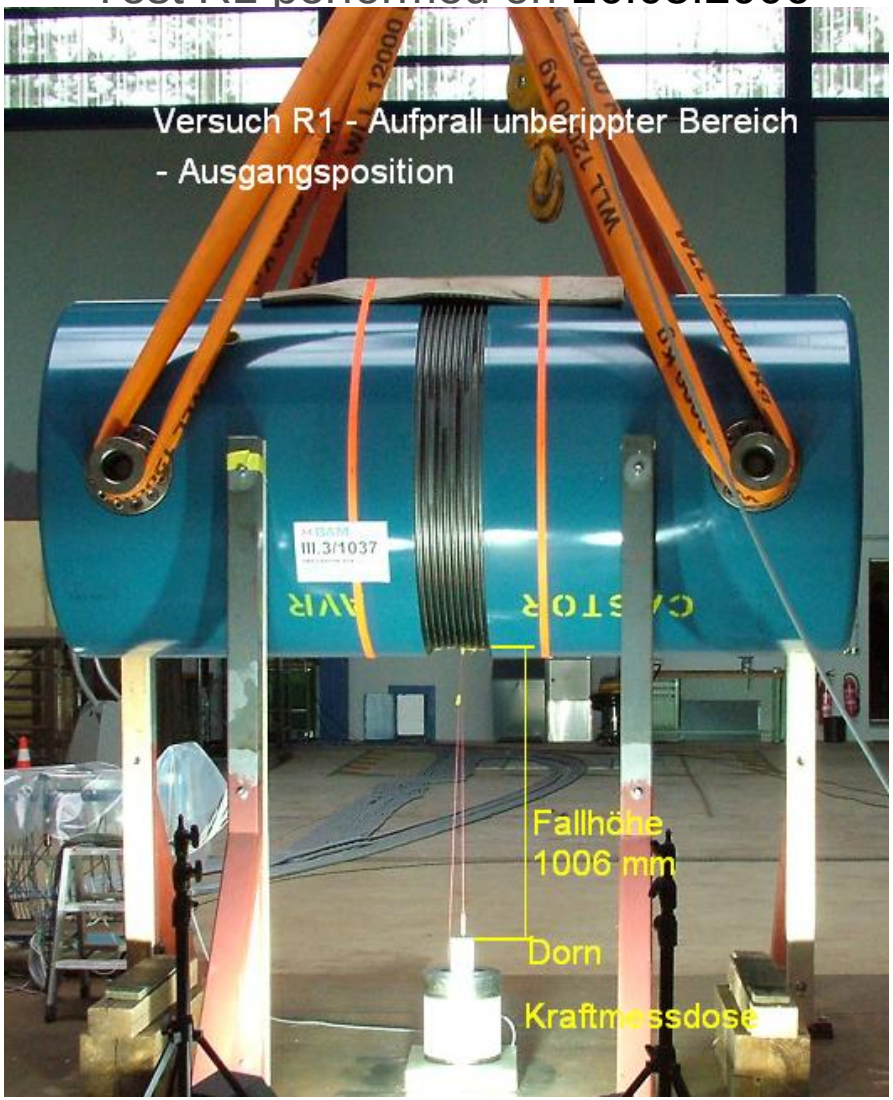
7. Test Performance

- **Test R1**
  - BAM-Test no. III.3/1037 was performed on **16.05.2006**
  - Drop height: 1,006 m measured from top of bar to lower edge of flat area
  - Cask temperature : ambient temperature
  - Ideal horizontal drop position
  - Impact area of bar at 0 °: no fins, but flattened area
  - Axis of bar in the cask‘s center of gravity
- **Test R2**
  - BAM-Test no. III.3/1038 was performed on **17.05.2006**
  - Drop height: 1,006 m **(measured from top of bar to lower edge of fin tip)**
  - Cask temperature : ambient temperature
  - Ideal horizontal drop position
  - Impact area at **180 °: Fins**
  - Axis of bar in the cask‘s center of gravity

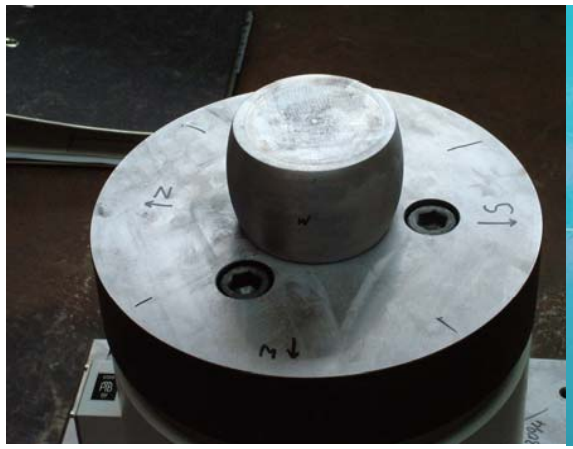


7. Test R1 – Impact onto Unfinned Region

- Test R1 performed on 16.05.2006



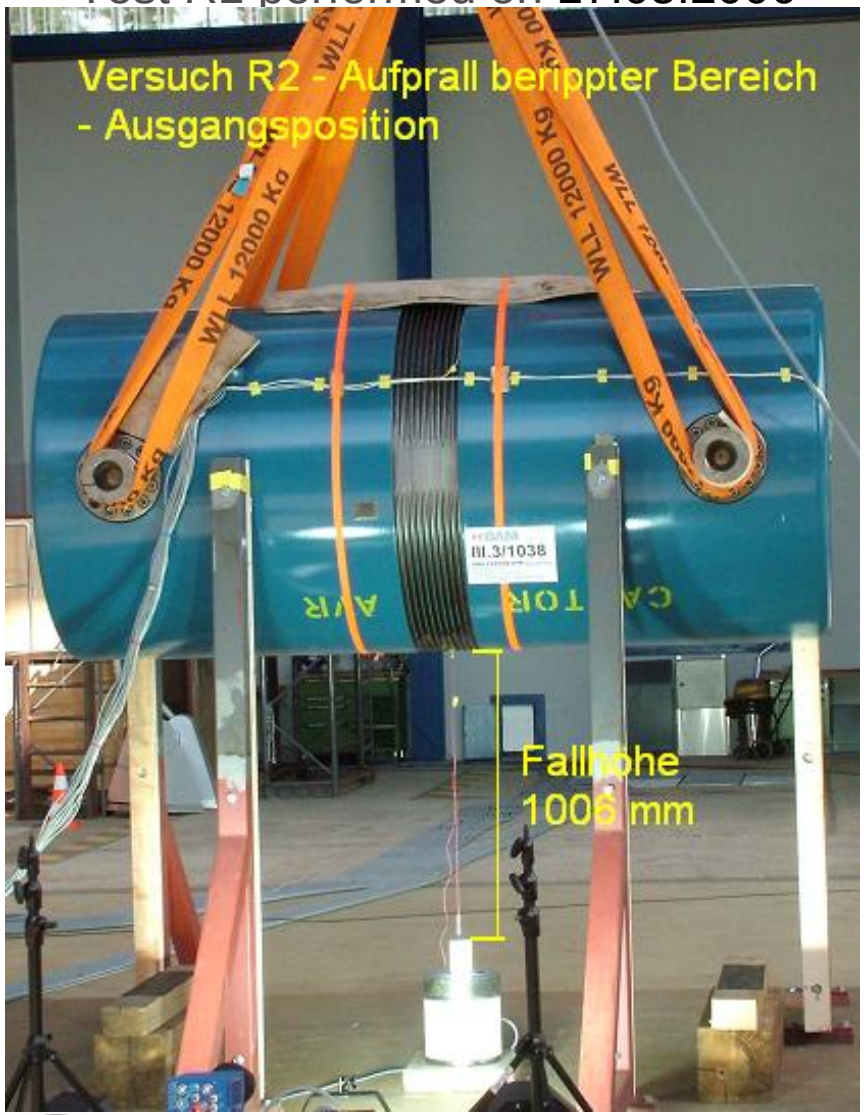
Deformed Steel Bar

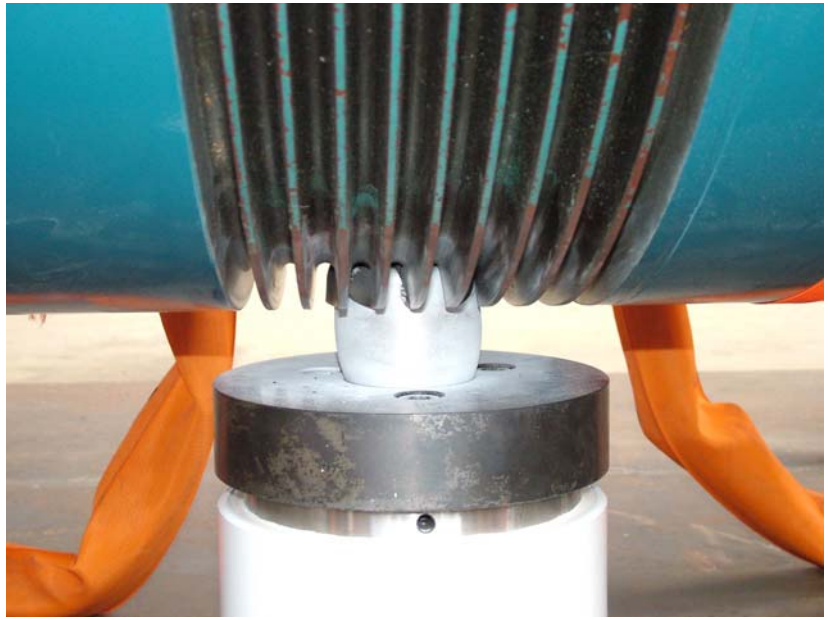


Impact Area at the Cask

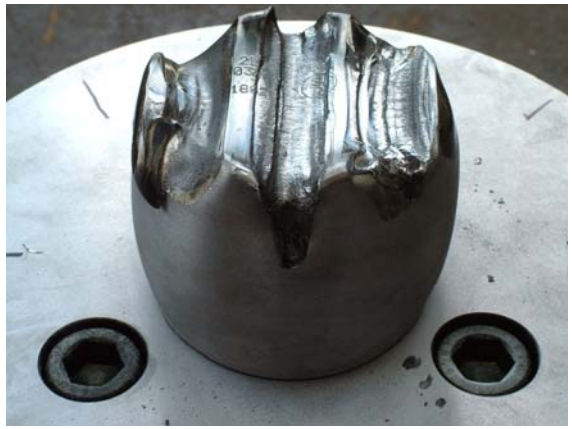
7. Test R2 – onto Finned Area

- Test R1 performed on 17.05.2006





Deformed Steel Bar



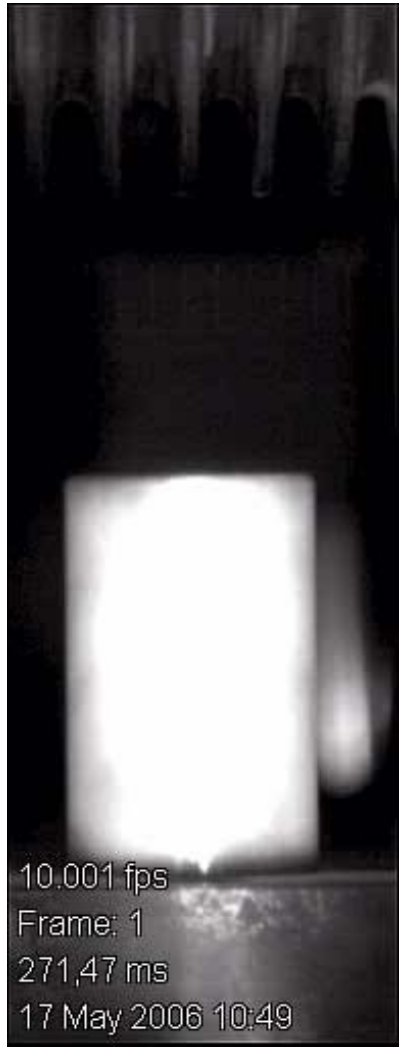
Impact Area at the Cask

7. Comparison of the steel bar deformations

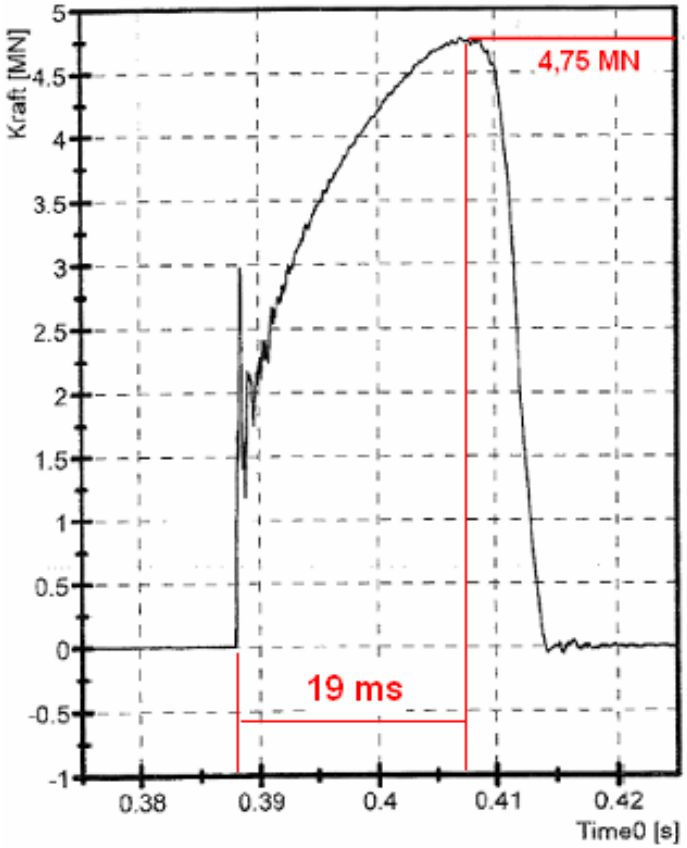
- High-speed video of the steel bar deformation

Test R1 – unfinned surface

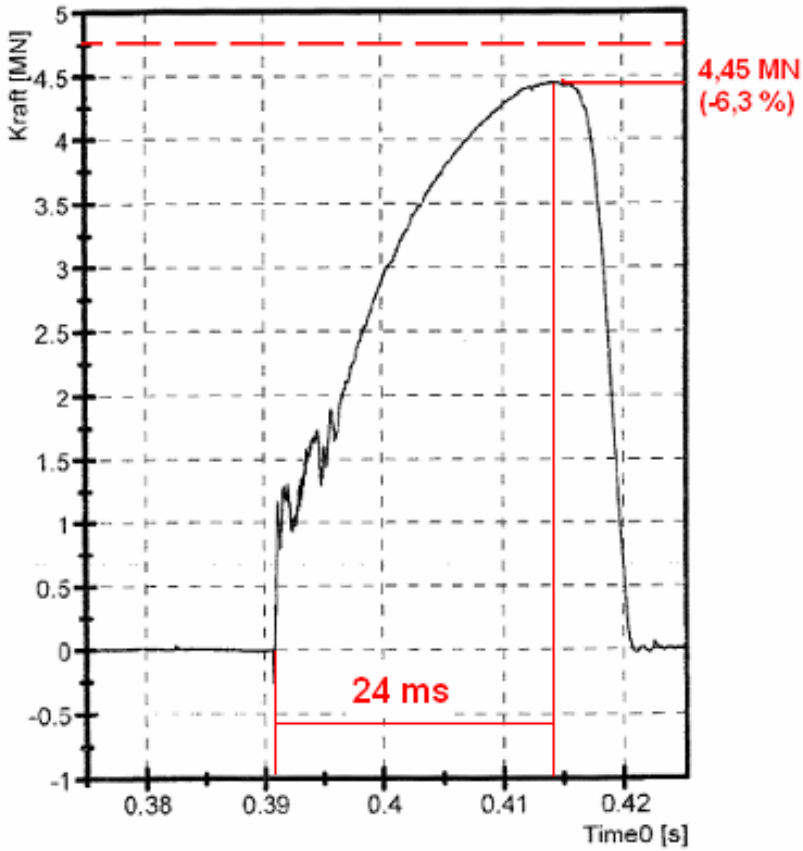
Test R2 – finned surface



8. Results of the Tests – Force at the Steel Bar



Fallversuch R1
ohne Rippen
Kraft-Zeit-Diagramm

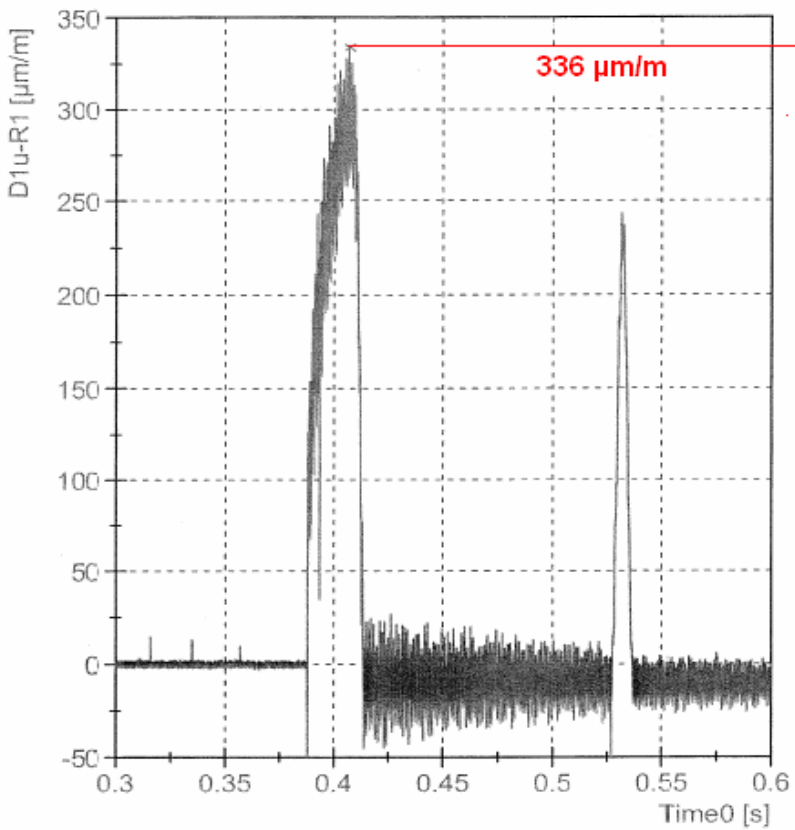


Fallversuch R2
mit Rippen
Kraft-Zeit-Diagramm

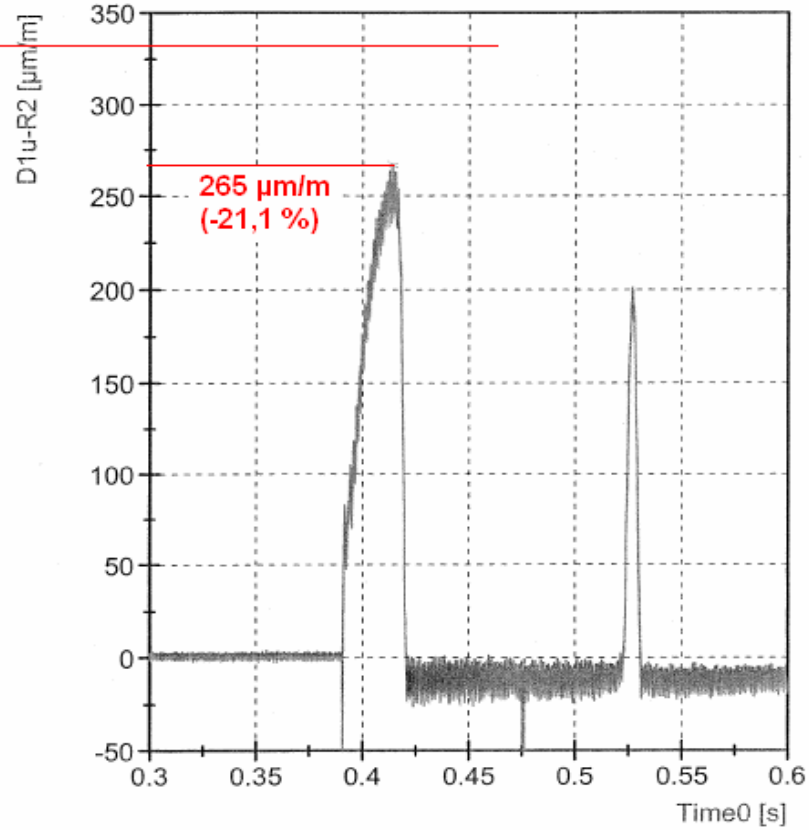
Effect of the Fins

* Penetration force was 4,45 MN lower (6,3 %)
* Impact duration was 5 ms (from 19 to 24 ms) longer

8. Results of the Tests – Strain at the Cask



Fallversuch R1
ohne Rippen



Fallversuch R2
mit Rippen

Effect of the Fins

* max. strain is sígnificantly lower by 265 µm/m (approx. 20 %)
* in both cases there were elastic strains of 63 and 51 MPa

8. Conclusions

- **2 cask drop tests from approx. 1 m height onto a steel bar were performed under comparable boundary conditions**
- **In the first test onto the unfinned area, the bar was essentially deformed plastically, while only insignificant deformations were found on the cask.**
- **In the second test onto the finned area, the fins penetrated into the steel bar on the one side, and on the other side the fins bent and broke in the impact area nearly down to the base of the fins.**
- **Quantitative effect of the fins**
  - Penetrating force lower by 4,45 MN (6,3 %)
  - Impact duration 5 ms longer (from 19 to 24 ms)
  - Max. strain is significantly lower (265 µm/m, approx. 20 %)
  - In both cases there was elastic strain of 63 and 51 MPa
  - More even and therefore flatter strain distribution
- **Altogether the strain level was lowered by approx. 20 %**

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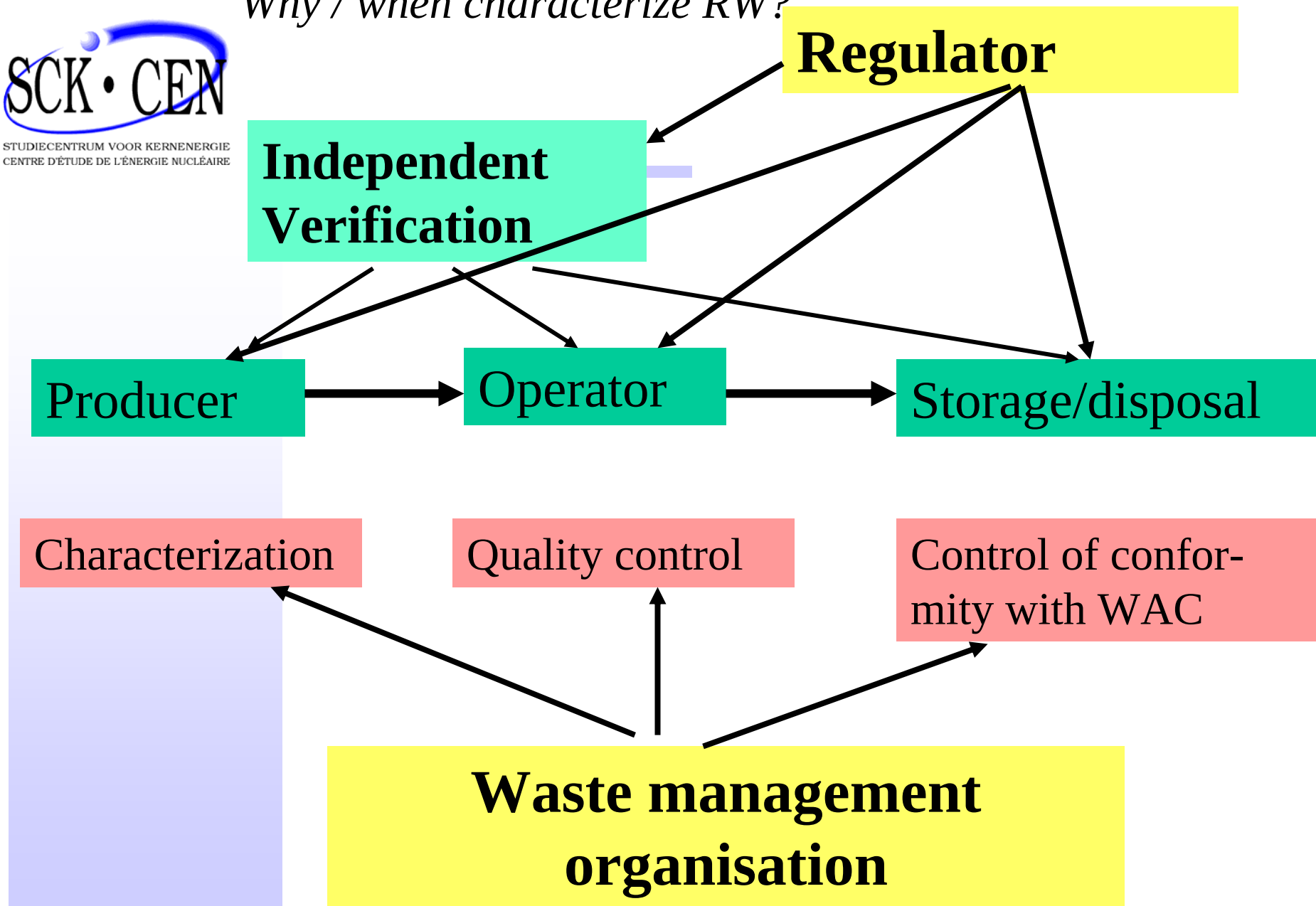
EN-TRAP

The European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages

Pierre Van Iseghem

SCK•CEN

Why / when characterize RW?



How to characterize conditioned radioactive waste?

- Inventory, distribution of radioactivity
  - non-destructive assay
  - destructive radiochemical analysis
  - calculation (scaling factors, ORIGEN code)
- Other properties: chemical, physical, mechanical, thermal, biological

IAEA reports offer good overviews, e.g.:

- *TecRep Series No 383 (Characterization of radioactive waste forms and packages), 1997*
- *Strategy of radioactive waste characterization (to be published)*

EN-TRAP was created in 1992, with the following objectives

- To promote European collaboration in the area of regulatory quality checking of radioactive waste packages
- This is to be achieved a.o. through
  - Information exchange
  - Identification of R&D requirements
  - Joint evaluation of test methods and proficiency testing
  - Co-ordination of national and international standardisation of test methods
  - Provision of training services
  - Promotion of the availability of testing and analytical services

EN-TRAP

Organisation of EN-TRAP

THE EUROPEAN NETWORK OF TESTING FACILITIES FOR QUALITY CHECKING OF RADIOACTIVE WASTE PACKAGES

secretary

Steering Committee

WGA

WGB

WGC

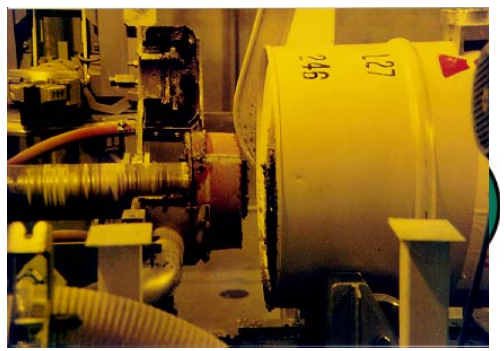
WGD

Non-destructive Methods

Destructive Methods

QA and QC

ILW / HLW



The members of EN-TRAP

CEA Cadarache
CIEMAT Madrid
ENEA Saluggia
ENRESA Madrid
FZ Jülich
FZ Seibersdorf
JRC Ispra
NRG Arnhem
NNC Winfrith
SCK●CEN Mol
TU München
VTT Espoo

+ associated laboratories

EN-TRAP is operating as follows

- Steering Committee and Working Groups (they can create Task Groups)
- Full members, Regulatory bodies,
  Associated members
- SC meets twice a year. A chairman is elected for a one year period.
  WG's meet once a year (average)
- EC assumes the technical secretary of the Steering Committee

www.en-trap.eu

Scientific Output of EN-TRAP

- Synopsis of gamma scanning systems (1998)
- Destructive analyses for the quality checking of radioactive waste packages (2001)
- Synopsis of neutron assay systems (2002)
- Non-destructive analyses for the quality checking of radioactive waste packages – Workshop Proceedings (2002)
- Leaching methods for conditioned radioactive waste (2004)

EN-TRAP today (1 of 2)

- ENTRAP has been successful
  - Discussion forum, technical visits
  - Various R&D proposals were developed
  - Many bi- or multilateral cooperations generated
  - Important scientific output
- Smaller output for the
  - implementation of regulatory checking
  - harmonisation of procedures

EN-TRAP today (2 of 2)

- We prepare the extension with the new EU countries
- Quality control / quality checking is presently a lower priority at EC level
  - The analytical techniques achieved state-of-the-art
  - Round robin tests have been performed
- Possible actions for the future
  - Free release, ILW/HLW
  - Harmonisation of tests (liaison “A” member of ISO)
  - Specific problem wastes
  - Networking outside the EU
  - Training

WG A on Non-Destructive Methods - Objectives

- Evaluate analytical data
- Improve the quality, efficiency and cost-effectiveness of NDT methods
- Exchange information on new developments
- Promote harmonisation of measurement procedures (e.g. determination of uncertainty)

WG A - R&D projects (EC FWP) initiated

- Improvement of Passive and Active Neutron Assay Techniques for the Characterization of Radioactive Waste Packages (EUR 19121)
- Optimization of Gamma Assay Techniques for the Standard Quality Checking of Radioactive Waste Packages and Samples (EUR 19127)
- Round Robin Test for Non-Destructive Assays of 220 Liters Radioactive Waste Packages (EUR 19779)
- Quality Control of Nuclear Waste Packages with a Compton Suppression and Ge-telescope Detection System
- Project on NDA of Large waste containers

The round robin campaign on NDA of LLW packages (EC project)

- 17 drums were selected
- Drums were classified
  - Homogeneous/non-homogeneous
  - Uniform/non-uniform activity distribution
  - Apparent density (low/high) <1g/cm^3<
  - Internal shielding
- 3 drums contain fissile material
- Kind of matrix: cement, bitumen

The NDA systems were classified as follows

C1: assay method assumes homogeneous matrix and uniform activity distributions for the whole drum;

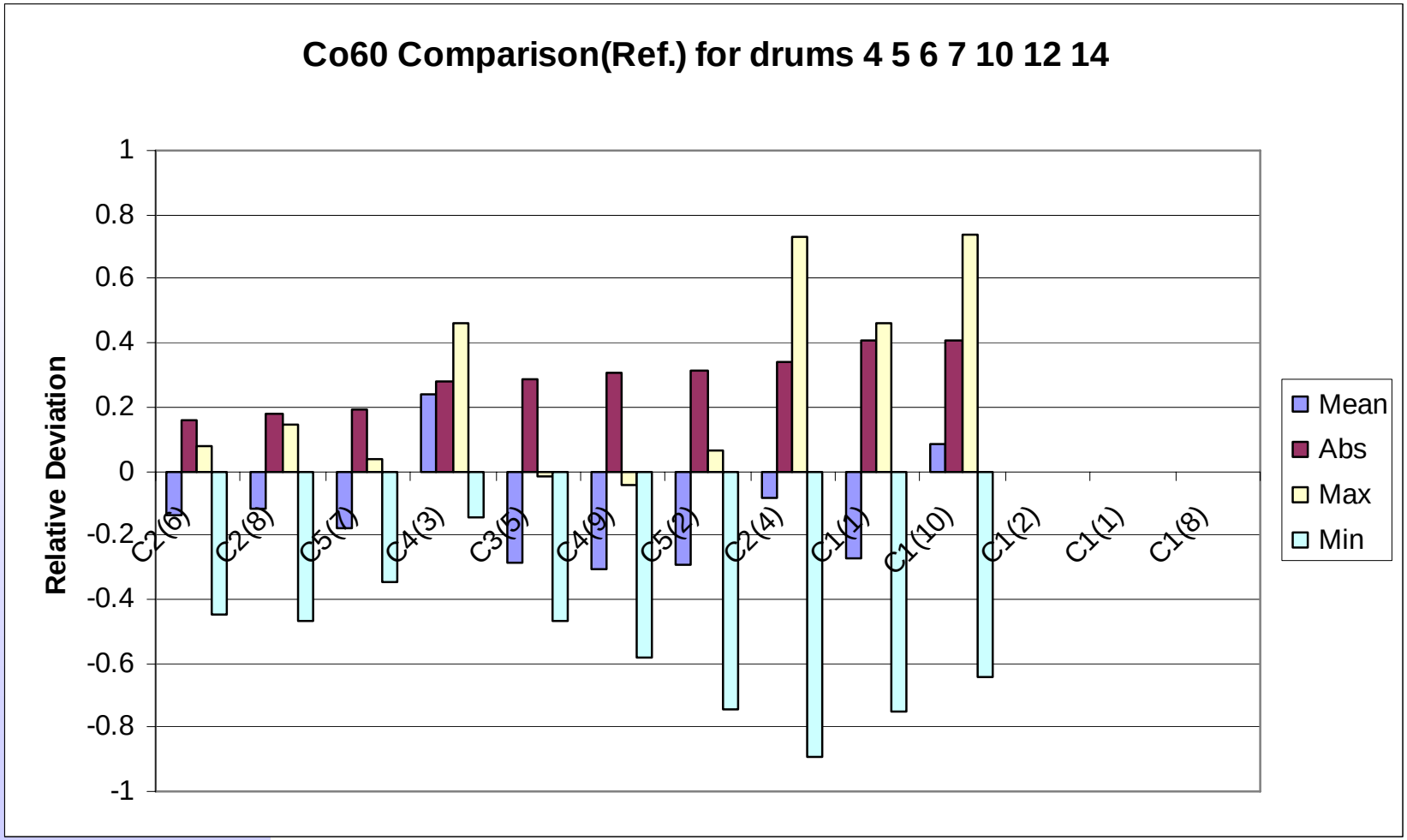
C2: assay method assumes homogeneous matrix for the whole drum and a uniform activity distribution within segments of the drum;

C3: assay method assumes homogeneous matrix for the whole drum and a uniform activity distribution within concentric rings in the drum;

C4: assay method assumes homogeneous matrix and uniform activity distribution only within segments of the drum;

C5: assay method uses complete spatial information on matrix and activity distribution.

Example of results of the NDA-RRT (7 drums)



WG B on Destructive Chemical and Radiochemical Techniques

Main Objectives

- ✓ Build and maintain an up-to-date list of chemical and radiochemical methods and physical techniques.
- ✓ Stimulate R&D programmes in the participating laboratories.
- ✓ Determine the reliability of measurements via detailed discussion and inter-laboratory measurement programmes
- ✓ Consider sampling problems and the representative and stability of the samples.
- ✓ Stimulate the co-operation and the exchange of information.
- ✓ Provide assistance for the harmonisation of methods used by laboratories.

WGB on Destructive Chemical and Radiochemical Techniques

R&D Projects Developed in the Framework of EU Programmes

- ✓ Evaluation and Standardisation of Fast Analytical Techniques for Destructive Radwaste Control. 1994-1998.
- ✓ Qualification of Radiochemical Routines for the Decontamination of Alpha-emitting Nuclides in Conditioned Radioactive Waste. 1994-1998 (EUR 19111).
- ✓ Development and Automation of Chemical Analytical Procedures for Determination of Non-Gamma Emitting Radionuclides in Radioactive Waste (DACAPO). 1998-2002.
- ✓ Interlaboratory Radiochemical Analysis Comparison on a Primary Waste Flux (INTERLAB). 1998-2002 (EUR 20616).

INTERLAB project (EC) – Objectives

- To determine the accuracy and reliability of the different analytical methods applied in the different participating laboratories belonging to “European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages”,
- To compare and validate the analytical methods used at present,
- To detect discrepancies and shortcomings in routine analysis, which will help to identify whether a separation or measurement method need further improvement.

INTERLAB – Radionuclides analyzed

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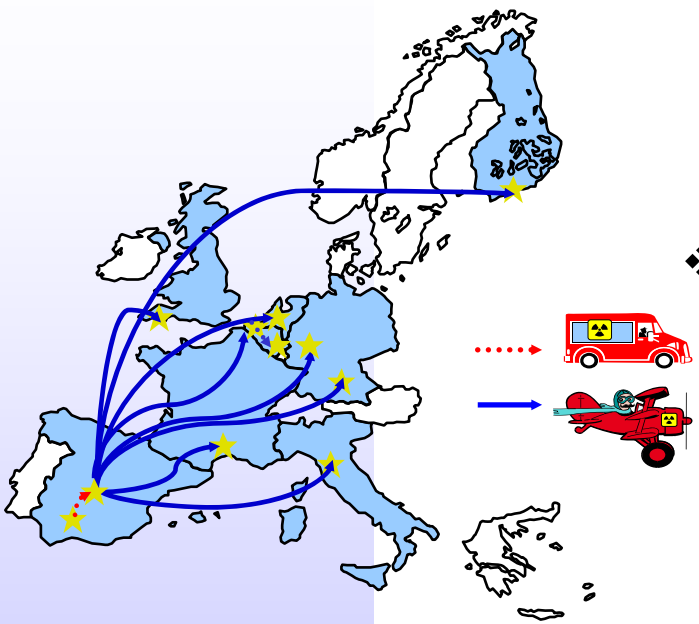
NUCLIDE	CIEMAT	ENRESA EL CABRIL	CEA(1)	SCK·CEN	NRG	FZJ	BELGO-PROCESS	HELSINKI UNIVERSITY	NNC	TUM	ENEA
^{3}H	X	X	X	X	X	X	X	X*	X	X	X
^{14}C	X	X	X	X		X			X	X	
^{55}Fe	X	X	X	X	X	X	X	X	X	X	X
^{63}Ni	X	X	X	X	X	X	X	X		X	X
$^{89/90}Sr$	X	X**	X	X**	X	X	X	X		X	X
^{99}Tc	X	X	X			X			X	X	X
^{129}I	X	X	X			X	X		X		
^{238}Pu	X	X	X	X	X	X	X	X	X	X	X
$^{239/40}Pu$	X	X	X	X	X	X	X	X	X	X	X
^{241}Pu	X	X	X	X	X		X (2)	X	X	X	X
^{241}Am	X	X	X	X	X	X	X	X	X	X	X
^{242}Cm	X	X	X	X (2)	X	X		X	X	X	X (2)
^{244}Cm	X	X	X	X	X	X		X	X	X	X
^{234}U	X		X	X	X	X	X	X	X	X	X
^{238}U	X		X	X	X	X	X	X	X	X	X

* Only from evaporator concentrate; possibilities for determination depend on sample composition.

** Only ^{90}Sr

(1) Only resin; (2) Foreseen but finally not performed

INTERLAB – Sampling and distribution



- Representative aliquots of the resin and concentrate samples were homogenised, sorted and prepared in ENRESA-LVCR under CIEMAT supervision.
- The procedure performed in order to test the homogeneity of the prepared samples was a Technical Report of **IUPAC** "The International Harmonised Protocol for the Proficiency Testing of Analytical Laboratories".
- The distribution was performed under the supervision and responsibility of ENRESA-El Cabril.

WG C on Quality Assurance / Quality Control

Objectives

- Identifying the requirements for quality checking
- Reviewing procedures, testing and control methods
- Evaluating uncertainties

Main achievements

- Document on QA/QC of LLW/ILW packages (EUR 19615)
- Discussion and promotion of accreditation / certification of the member laboratories

WG D on ILW / HLW - Needs

- Internationally accepted and demonstrated methods and procedures have to be developed and demonstrated

- A common approach in Quality control and Quality checking, and a common understanding of the techniques and procedures should be achieved

WG D on ILW / HLW - Objectives

- To exchange information and discuss on harmonization of existing characterization systems
- To stimulate and coordinate ongoing R&D on characterization methods and procedures for HLW and ILW
- To standardize test methods that are used in destructive and non-destructive testing facilities

Networking of ENTRAP

- ENTRAP is networking with
  - ISO (“liaison “A” member)
  - ASTM
- ENTRAP is present at international conferences (e.g. RADWAP, ICEM)
- ? networking with the Forum on design and assessment of radioactive waste packages

GERMAN SUPPORT FOR UTILIZATION OF NUCLEAR SUBMARINES IN RUSSIA

B. Droste, V. Noack,
Federal Institute for Materials Research and Testing (BAM), Berlin, Germany

D. Rittscher, H. Schmidt
Energiewerke Nord GmbH (EWN), Lubmin, Germany

Workshop on Design and Assessment of Packages for Radioactive Waste
Bergen, The Netherlands, November 21 – 22, 2006
Organized by Joint Research Centre, Institute for Energy, Petten, NL

Energiewerke Nord GmbH Presentation for

CEG Meeting

11. - 13.10.2006

München

German support for utilization of nuclear submarines in Russia

Progress in project execution

Project Organization and Controlling

- Project Management by **Energiewerke Nord GmbH (EWN)** under Contract of **Federal Ministry of Economics and Labour (BMWA, now BMWi)** (dated 18 November 2003)
- Technical Controlling by **Bundesanstalt für Materialforschung und –prüfung (BAM)** under Contract of BMWi (dated 14 may 2004):
  - Evaluation of the project management and project development with respect to:
    - necessary technical changes
    - technical risks
    - changes in development and measures planned
    - problems in practice
    - cross-cut issues
    - changed issues of planning aims
    - project additions and parameter changes

Subject of the Project

(Agreement between the German Federal Ministry of Economics and Labour and the Russian Ministry for Atomic Energy, dated 9 October 2003)

Article 1: „(1) ...

1. Erection of an onshore long-term interim storage facility for reactor compartments in the Sayda Bay, including respective infrastructure;

2. Optimization of the material and technical situation and of the equipment of Russian companies, in order to accelerate disposal of nuclear submarines;

3. Establishing of conditions for a safe handling of waste products, generated in the disposal of nuclear submarines in the northern region of the Russian Federation;

4. Creation of an ecologically sound status of the environment in the Sayda Bay.“

Aim

Construction of a long-term interim storage facility for 120 reactor compartments at Sayda Bay near Murmansk

- storage platform with rails and drains
- physical protection
- pier for floating dock
- indoor reactor compartments repair hall
- necessary auxiliary buildings
- radiation protection system
- roads and external infrastructure

- Original view of Sayda Bay
- Laying of cornerstone on July 10, 2004
- Start of construction

- Completion of construction site facilities on September 25, 2004

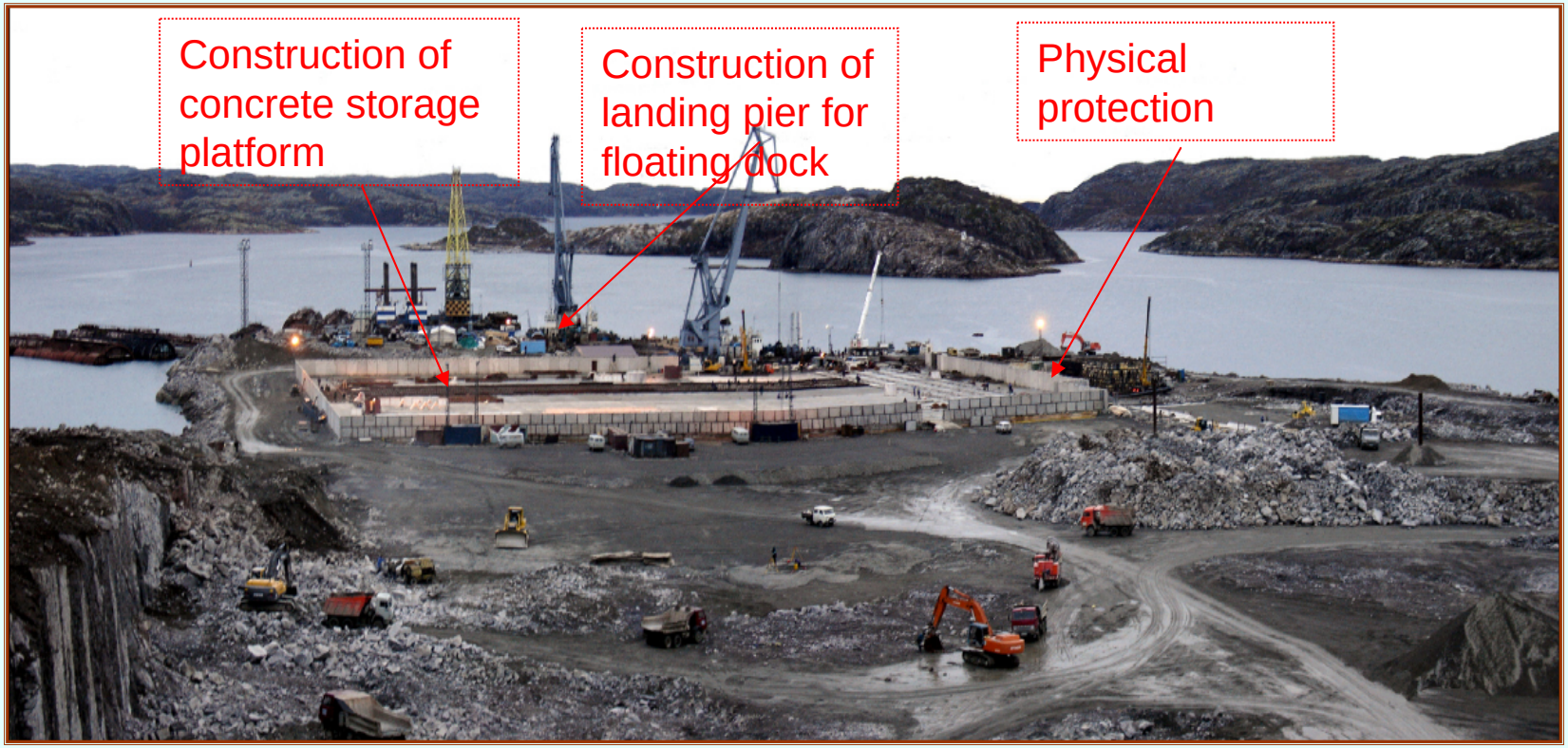
Construction site in February 2005

Preparation of grounds –detonation, stone and surface removal

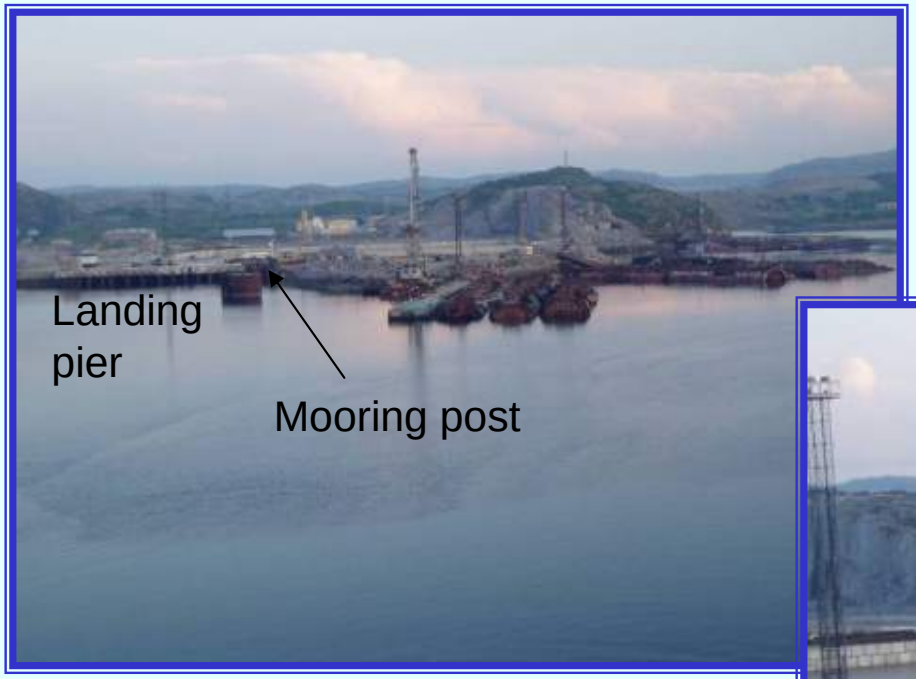
Work on creating a level storage platform

Excavation and removal of parts of western cliff slope

Construction site in November 2005



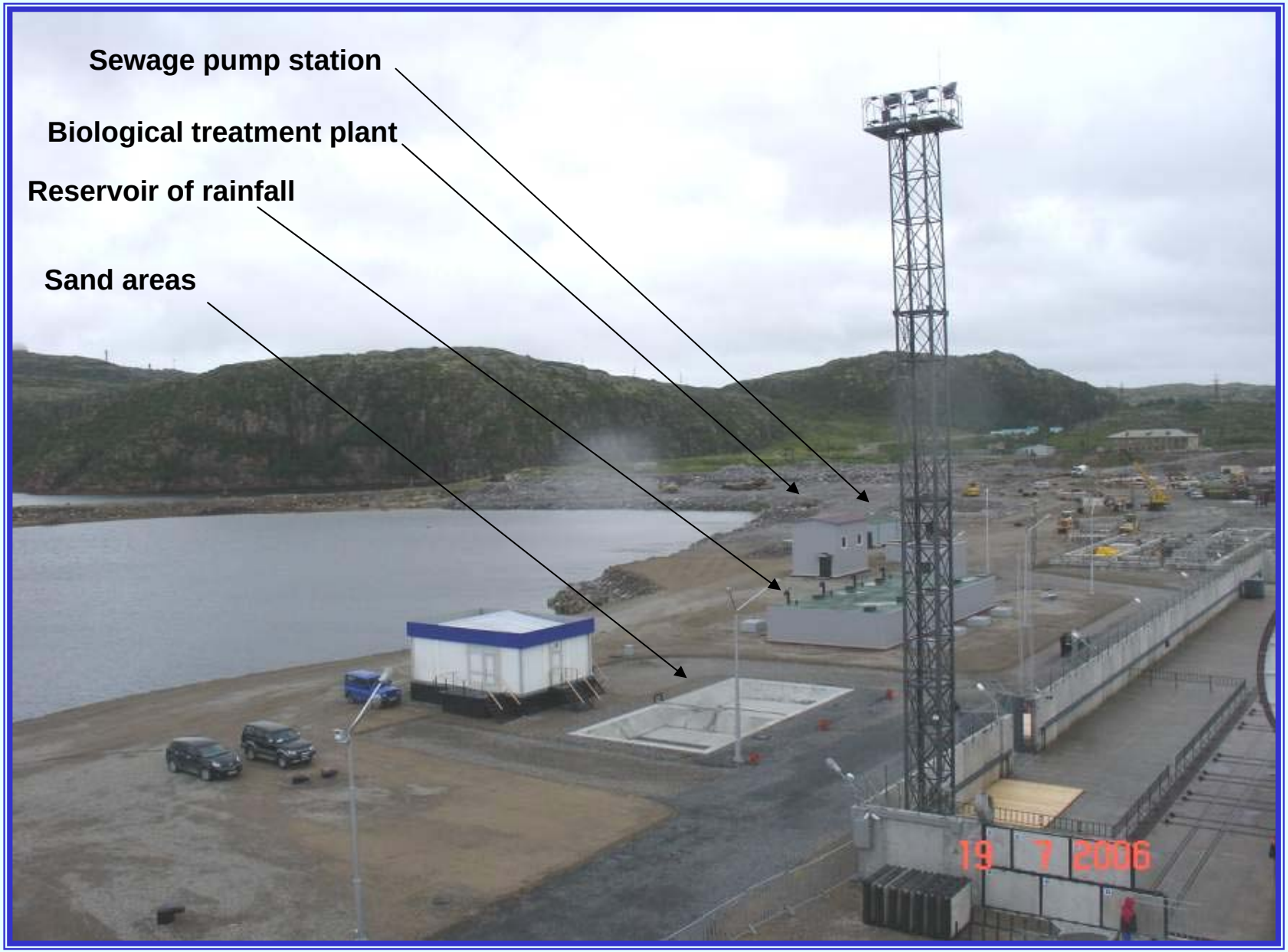
Long-term interim storage facility at the beginning of July 2006



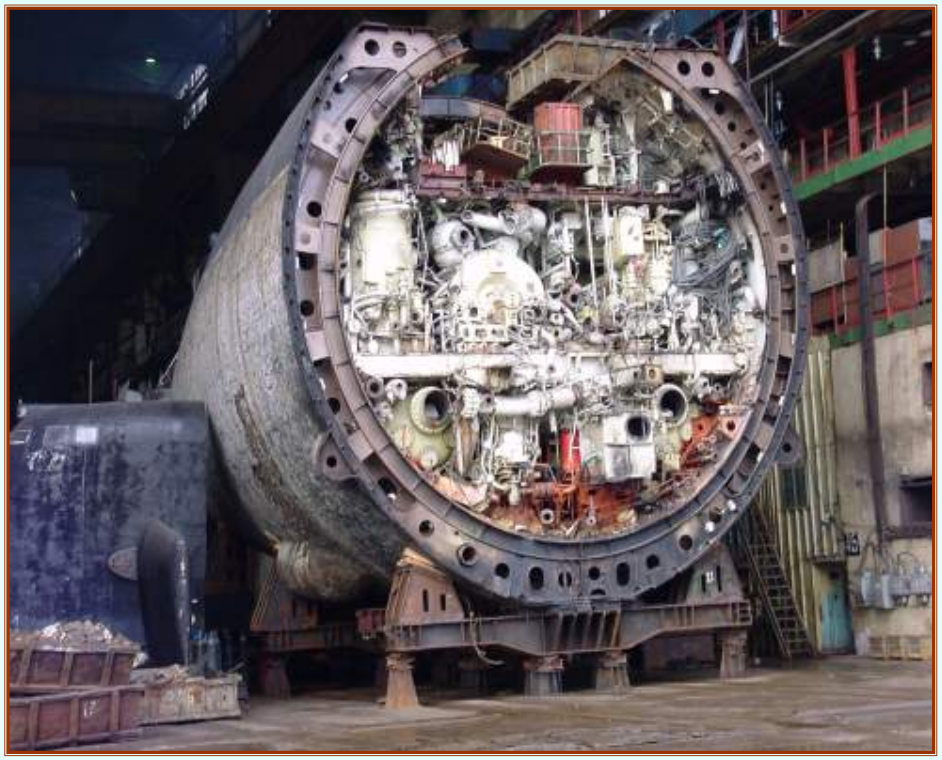
Initial section

19.07.2006 / S33_DSC00474

Auxiliary facilities near the storage plate



Detachment of reactor compartments



SRY „Nerpa“

Testing of German docking-block transport system at Nerpa-Shipyard

SRY „Nerpa“

Forming of reactor compartments

Side supports with wedges

Transport of reactor compartments in the dock PD-42

LTSF – Long term storage facility Sayda

10.07.2004
Cornerstone ceremony

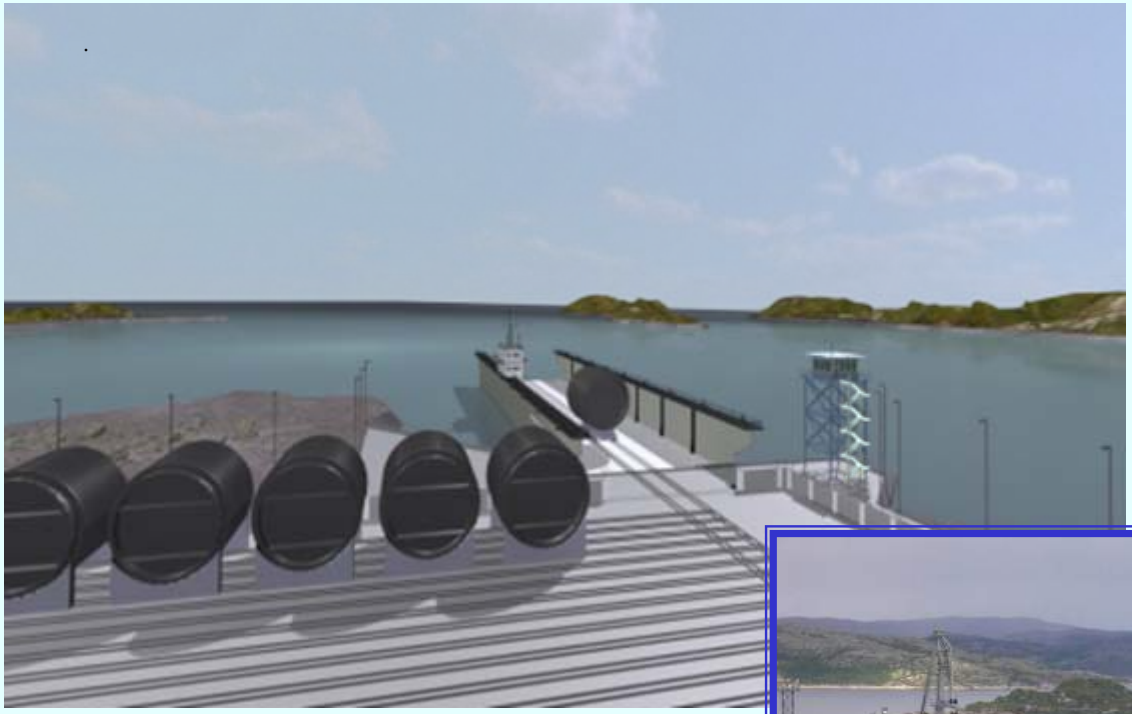
18.07.2006
Commissioning ceremony of the initial section

Former costs brutto: 154 Mio EUR

Former costs LTSF: 73 Mio EUR

30.06.2006

Commissioning ceremony of the initial LTSF section



Plan of the LTSF
in the Sayda Bay

Unloading of dock

Unloading of reactor compartments

Shifting of reactor compartments on storage area of the LTSF

19.07.2006 / S33_DSC00454

Reactor compartments at the final place on storage area of LTSF

19.07.2006 / S33_DSC00449

Initial section of the LTSF in operation

Extension of the project until 2008

2nd construction stage

Functions

Storage of 58 special objects :

2 compartments of the floating technical base „Lepse“

9 compartments of the floating technical bases of the fleet

10 steam-generating plants of icebreakers

3 steam-generating plants of cruisers

30 reactor compartments of submarines without SNF

2 reactor compartments of submarines with SNF

Plan of the LTSF in the Sayda Bay

Plan of the LTSF in the Sayda Bay

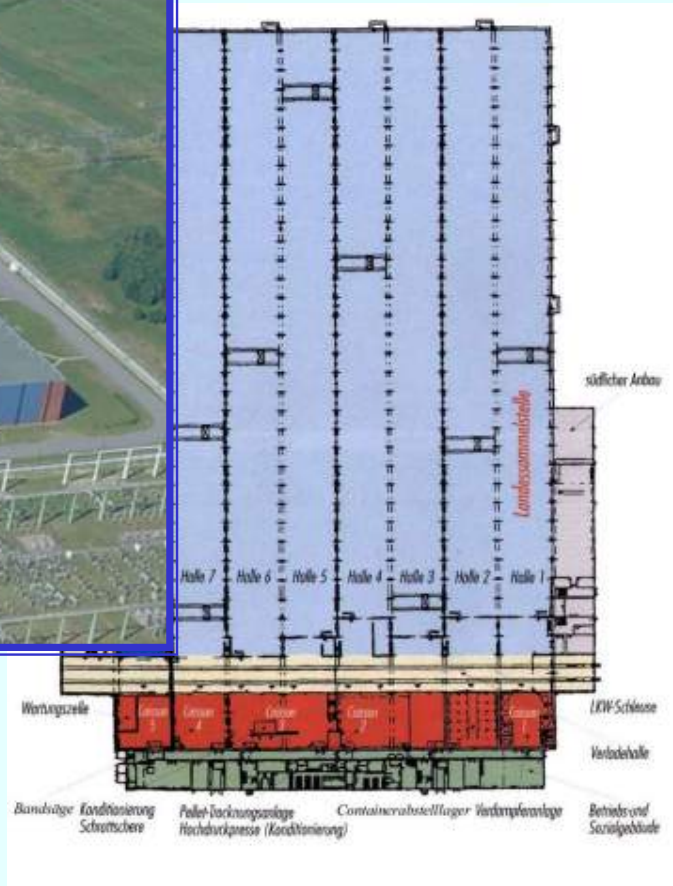
Extension of the project from 2008 to 2013

Radioactive waste storage facility (RWSF) at Sayda Bay

Functions

- Receiving, decontaminating, conditioning and packaging of radioactive waste
- Final measurement of radiation thresholds of radioactive waste
- Storage facility for radioactive waste
- Possibility of definitive dismantling and disposal of reactor compartments and other radioactive waste
- Calculation of the costs about 300 Mio Euro

Example of a storage facility for radioactive waste



Storage facility for radioactive waste „Zwischenlager Nord (ZLN)“ of Energiewerke Nord GmbH, Lubmin, Germany

Example of a storage facility for radioactive waste

Storage hall

Band saw

Storage facility for radioactive waste „Zwischenlager Nord (ZLN)“ of Energiewerke Nord GmbH, Lubmin, Germany

Example of a storage facility for radioactive waste

High-pressure press «FAKIR“

Evaporation plant

Storage facility for radioactive waste „Zwischenlager Nord (ZLN)“ of Energiewerke Nord GmbH, Lubmin, Germany

Example of a storage facility for radioactive waste

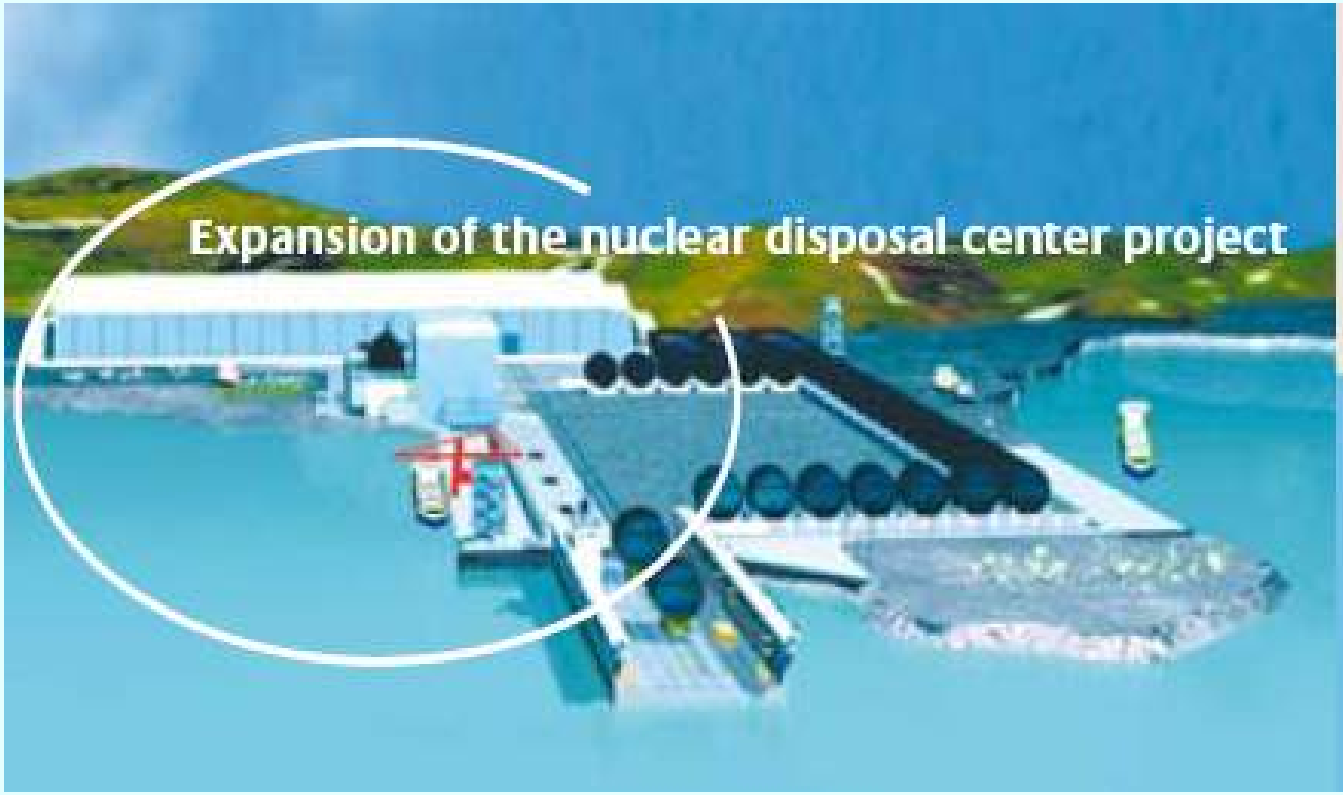
Scrap metal scissors «MARS“

Barrel drying plant

Storage facility for radioactive waste „Zwischenlager Nord (ZLN)“ of Energiewerke Nord GmbH, Lubmin, Germany

RWSF Sayda

Radioactive waste storage facility at Sayda Bay



Calculation of the costs about 300 Mio Euro

Thanks for your attention!

Energiewerke Nord GmbH

Dieter Rittscher (Geschäftsführer)

Holger Schmidt (Projektleiter)

Tel.: +49 3835 44 59 00

E-Mail: holger.schmidt@ewn-gmbh.de

Federal Institute for Materials Research and Testing (BAM)

Bernhard Droste, Volker Noack

Potential Collaboration Work with European Commission

Tae M. Ahn

Division of High-Level Waste Repository Safety

U.S. Nuclear Regulatory Commission

Washington, D. C., U. S. A.

Workshop on Design and Assessment of Packages for Radioactive Waste, Bergen, The Netherlands

November 21 – 22, 2006

Potential Collaboration Work with EC

- Studsvik Cladding Integrity Project (SCIP): hydride effects
- ASTM Round Robin Tests on the Dissolution of Spent Nuclear Fuel (SNF)
- ASTM Workshop “Hydride Re-Orientation in Zirconium Alloys”
- Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA) and International Atomic Energy Agency (IAEA) supported Symposium “Long Term Prediction of Corrosion Damage in Nuclear Waste System”
- Bundesanstalt für Materialforschung und –prüfung (BAM, Germany) and NRC - Canister Drop Test

Potential Collaboration Work with EC (Cont'd.)

- Radioactive Hot Testing of SNF: dissolution, solubility limit, and sorption
- Radioactive Hot Testing of Cladding: swelling of the matrix of spent nuclear fuels, and bonding/chemical reaction of the cladding and the matrix of SNF
- Information Exchange
  - American Society for Testing and Materials (ASTM) and American Society of Mechanical Engineers (ASME) Standards/Guides: long-term materials prediction, and modified pressure vessel codes in welding
  - NRC Guides: Regulatory Guides, Interim Staff Guidance, and Standard Review Plans

Disclaimer

The NRC staff views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed or of the acceptability of a license application for a geological repository at Yucca Mountain.

Ideas for future collaboration

H.Asano, RWMC

Workshop on Design and Assessment of Radioactive Waste Packages
November 21- 22, 2006
Petten, The Netherlands

Influence of welding on welded joint

Welding process

- **Heat-input, Melting, Solidification, Thermal cycle**

↓

- Microstructure
- Chemical composition
- Residual stress
- Weld flaws
- Surface roughness

↓

- Mechanical property
- Corrosion property

Quality Issues

Long-term integrity of welded joint for 1,000 years as same as the base metal

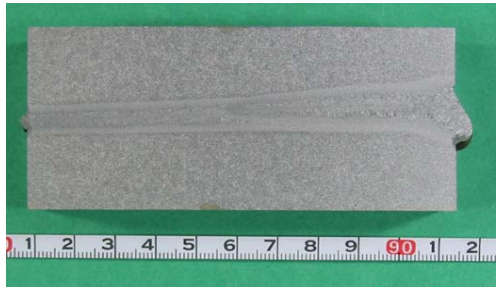
How to obtain ? → Technical

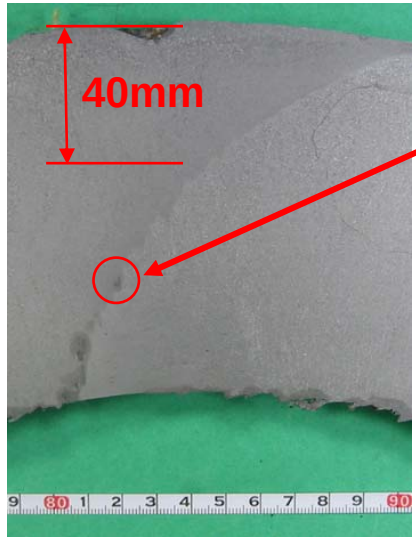
and

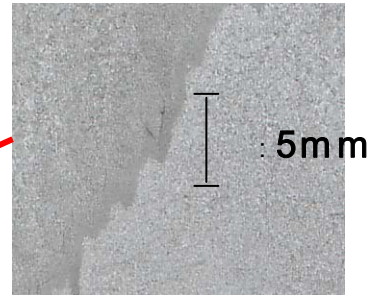
How to explain? → Theoretical

Example : Welding test, EBW (Electron Beam Welding)

100mm full penetration test for carbon steel overpack lid structure







Spike

Surface roughness

Microstructure

Weld flaws

My concern is :

Is this weld joint has a sufficient quality to be assured the long-term integrity?

Answer to the question

-To explain the long-term containment capability of the overpack final closure
-The 1,000 years duration is not only an aim but something which is an achievable requirement

Stepwise approach

1. Applicability of welding and NDE methods
2. Evaluation methods for the obtained data
3. Presentation of promising methods for the final closure

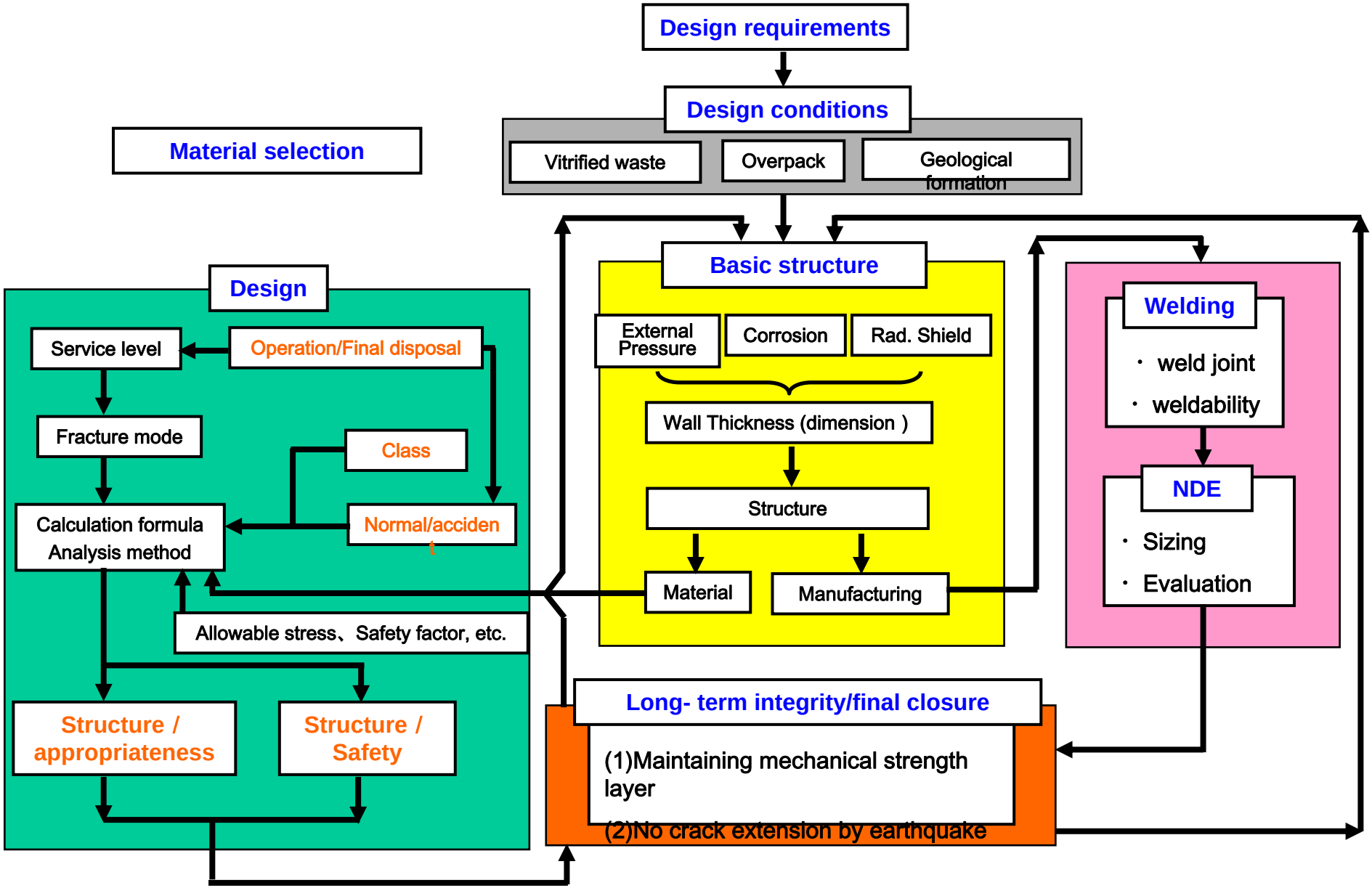
4. Finalization of the final closure method

Authorization = Acceptance criteria

Joining !

Technical issues ⟷ Theoretical issues

Overpack, Design & Assessment Procedure



Idea No.1 : damage tolerance and manufacturing issues

To know the technical issues through collecting data of NDE and welding performance

(1) NDE

a. Detectability & Sizing capabilities of weld flaws

b. UT, RT and others

(2) Welding

a. Weldability

b. Weld joint property

- mechanical property
- corrosion property

Idea No.2 : damage tolerance and manufacturing issues

To think the theoretical issues through collecting R&D data & knowledge

(1) Tolerable flaw size

a. How to define?

b. Size, shape & location

(2) Safety factor to fix the max. tolerable flaw size

a. Critical crack length

b. Safety factor

Idea No.3 : Reliability of long-term prediction of weld joint integrity

To know the property of weld joint and its uncertainty related to assuring the long-term integrity of the weld joint

(1) Mechanical property / Mechanical strength

a. Fracture mode

b. Fracture toughness

(2) Corrosion property / Resistivity / Susceptibility

a. Uniform corrosion

b. Localized corrosion - Preferential corrosion, SCC -

c. Test condition, Acceleration condition

(3) Uncertainty

a. How to assure the uncertainty ?

b. Geological condition – earthquake, fault, other external loading –

c. Effect of surrounding buffer material

Future works need for more reliable evaluation of the closure weld integrity

Prerequisite No.	Items	Current Situation	Future work for getting more reliability	
			View point	Future work
1	General Corrosion	・Most Fundamental Property ・Influence of Filler Metal	Effect of filler metal on the corrosion propery of the weld metal	design of filler metal (chemical composition)
2	Susceptibility to SCC	・No effect of welding on the susceptibility	More realistic condition	Chemical environment ⇒ groundwater composition (low carbonate concentration)
		・Appropriateness of evaluation method(SSRT)		Loading condition⇒ Constant loading
		・appropriateness of evaluation time		Test period(Loading time)
3	Neutron Radiation Embrittlement	・Used estimation method for nuclear power plant component (high radiation / shor time)	Prediction method	Low & long-term radiation
4	Fracture Toughness	・Used low carbon steel for nuclear power plant component	More realistic estimation	Selected carbon steel
		・Available data for a geological formation on the estmation of critical crack length	Preciseness of geological condition	External Pressure
				Earthquake ・Max. Loading ・Fatigue fracture
		・LFM / NLFM	Fracture mode	Plastic-elastic farctute mechanics
5	Mecanical Buffering Property of Buffer Material	・Available data for a buffer material	Maintainng the mechanical buffering property	saturation, heat effect, stress condition, chemical reaction etc.
6	Safety Factor	・referring from FFS of nuclear power plant component	Estimation basis	Appropriate value as a diposal container

END

EUROPEAN COMMISSION
DIRECTORATE-GENERAL
Joint Research Centre

Joint Research Centre

Input to Discussion on collaboration

Workshop on Design and Assessment of Packages for Radioactive Waste

K-F Nilsson

Institute for Energy

Motivations for Collaboration/Forum

- We sit “in the same boat”, application of overall best methods is beneficial for all.
- Consensus on Best-practice procedures increases the public confidence in management of radioactive waste
- Cost saving without impairing safety
- Acceptance of national concepts by “peer-review”

But the situation in different countries depend on

- Public attitude towards nuclear energy
- Amount and type of radioactive waste and spent fuel
- National legislation
- Geography and geology

Future trends that effect the need for collaboration

- Regional repositories and encapsulation plants are now discussed.
- For national long-term solutions a limited number of concepts will serve as reference. Technology transfer to recipient countries from donor countries for these reference cases will be important.
- In geological disposal we now enter the implementation phase (Sweden, Finland, USA and France). The emphasis will shift from basic research to engineering aspects.
- Retrievability is requirement in certain countries and in other countries where it is not a requirement it is now considered as long as it does not impair safety in others
- Long-term storage (planned and un-planned) and higher burn-up will introduce new integrity issues
- Phased concepts with dual purpose casks for transport/storage or storage/disposal.
- Increase of transport of radioactive material and its public acceptance

Potential topics for collaboration (1)

- Manufacturing, quality control and operation
  - Defect tolerance
  - Quality control and NDT
  - Operational safety
  - Demonstration of feasibility
- Long term-degradation
  - Identification and modelling of mechanisms
  - Environmental effects
  - Validation of models

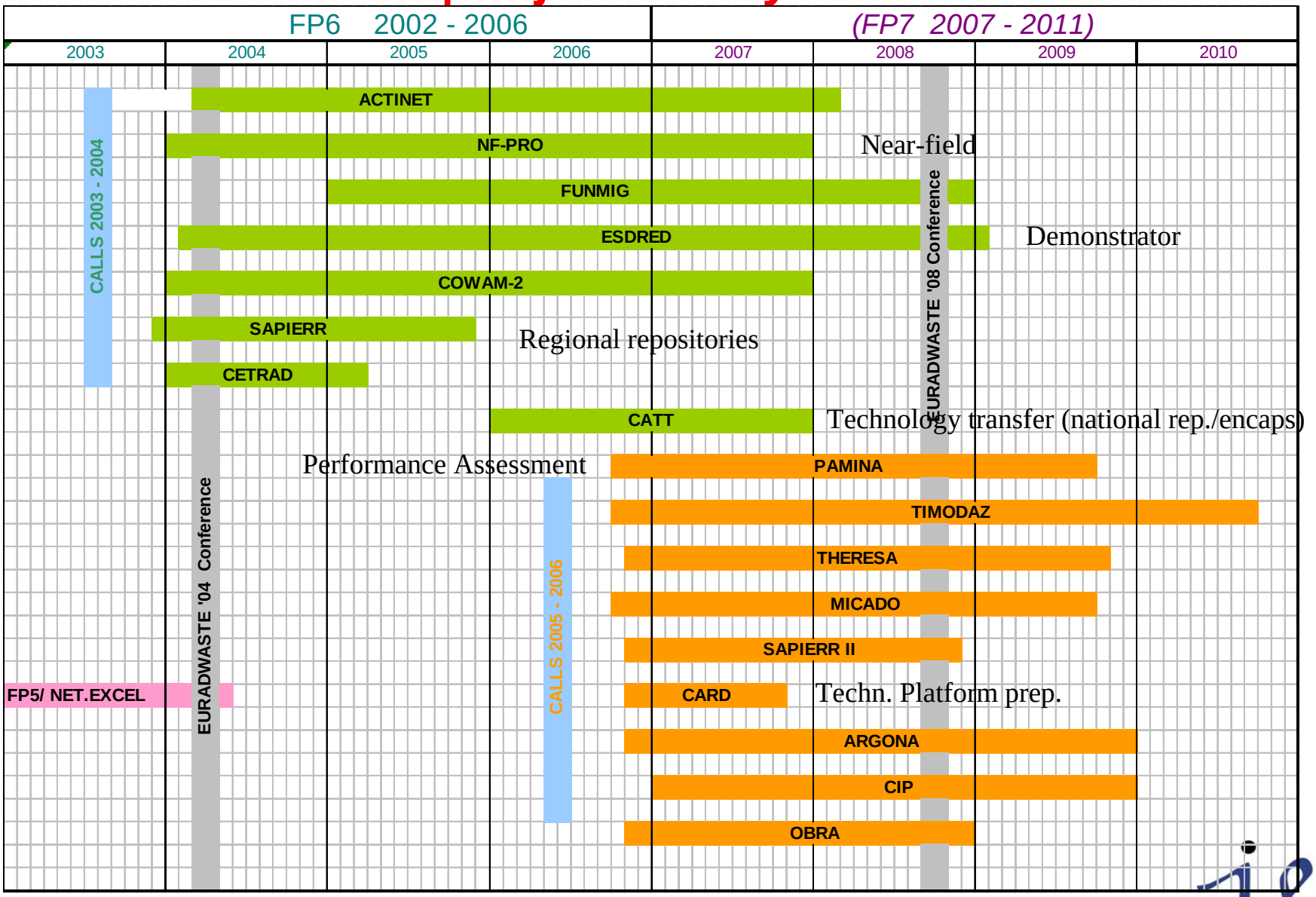
Potential topics for collaboration (2)

- Drop testing with supporting dynamic analysis
  - Transferability full-size casks and mock-ups
  - Modelling procedures
- Integrity and behaviour of spent fuel under long term storage
  - Cladding degradation (e.g. delayed hydride cracking)
  - Dissolution, solubility limit and sorption
- Development of Standard Casks
  - Standard canisters for SNF
  - Overpack families compatible with specific disposal for different canister/drums
  - Design of flexible transport casks

Thoughts on “Forum”

- Exchange of information is a natural part of a Forum but not a sufficient basis.
- Proposals for joint research projects (possibly quite small) supported by key organizations must be the starting point. These should ideally be partially funded by third party.
- Any activity should be co-ordinated with activities in other for a such as IAEA, NEA/OECD,
- JRC-IE is willing to provide the management of such a network.

FP6 projects beyond 2006



FP 7 Calls

- Reference to disposal
- Consideration to outcome of 6th FP Projects
- No duplication of ongoing FP6 Activities
- Smaller projects than in FP6

II.2.1 Activity: Management of Radioactive Waste
Through implementation-oriented research and development, the activities aim to establish a sound scientific and technical basis for demonstrating the technologies and safety of disposal of spent fuel and long-lived radioactive wastes in geological formations, to underpin the development of a common European view on the main issues related to the management and disposal of waste, and to investigate ways of reducing the amount and/or hazard of the waste by partitioning and transmutation or other techniques.

II.2.1.1 Area: Geological disposal
Research and development in the field of geological disposal of high-level and/or long-lived radioactive waste involving engineering studies and demonstration of repository designs, in-situ characterisation of repository host rocks (in both generic and site-specific underground research laboratories), understanding of the repository environment, studies on relevant processes in the near field (waste form and engineered barriers) and far-field (bedrock and pathways to the biosphere), development of robust methodologies for performance and safety assessment and investigation of governance and societal issues related to public acceptance.

Topics that could be related to Waste Packages

Topic: EURATOM-Fission-2007-1-2.1.1.1: Phenomenology and performance assessment

Objectives: To further improve approaches and methods related to a limited number of specific scientific and technical issues of relevance for the safety of geological disposal of spent nuclear fuel and high-level and/or long-lived radioactive waste

- *development of full-scale prototypes of repository sub-systems to allow long-term monitoring and observation of phenomena and validation of numerical simulations;*
- *development of modelling tools for simulation over long periods of time and at scales that cannot be represented through experimentation.*

- *Topic: EURATOM-Fission-2007-1-2.1.1.2: Technology demonstration of safe repository operation*
- *Objectives: To propose and evaluate, through the development of mechanical demonstrators, effective and efficient technical solutions to key issues associated with the operation of a geological repository for radioactive waste, with an emphasis on requirements for licence application.*
- *Scope: Development of one-to-one scale demonstrators and/or pilots in repository situations that integrate operability and safety aspects in order to test, inter alia:*
  - *the safe on-site transport and handling of waste packages, including their transfer from surface to underground (by shaft or ramp) and their emplacement in disposal cells;*

Working Groups

1) What do you see as the most important challenges in the coming 2-10 years in terms of R&D and practical implementation?
2) Which of these areas do you consider suitable for international collaboration?
3) Based on 1) and 2) make a road map for work.

Design and Assessment of Packages for Radioactive Waste
Bergen Nov 21-22

Working group on Long-Term behaviour of Waste Packages.

Summary of the Round Table.

1. A general request from most participants is to get more information and more collaboration. This request is strongly expressed by the smaller countries. Some of them are ready to contribute actively to a common project as a counterpart.

2. A huge lot of valuable information is available but not under a convenient form. There is a need for bibliographies, seminars or training for these who need some coaching. E.g., experiments are underway at ITU

3. The long-term behaviour of the fuel and cladding is a concern for many participants who whish collaboration and exchange of information. Methods to predict the long-term aging are lacking. Aging acceleration techniques as well. This requires a better understanding of the material microstructure.

4. The effect of radiolysis of water in interim storage conditions is also a concern for the storage casks.

5. A few participants would appreciate having a sort of summary of the materials that are candidates for disposal casks (overpacks). They want to know more about the reasons of the choices and whether new candidates could be envisaged.

6. Concern for low level waste packaging is low. Efforts should concentrate on high level waste.

7. The development of a standard disposal cask common to several countries would be appreciated.

8. The renewal of the agreements for interim storage should be prepared on time (around 2025).

9. Attempts to explain the large variety of the available disposal cask designs might help public acceptance.

Recommendations, common needs identified

- Transfer of existing knowledge (for instance on spent fuel corrosion, container material selection and performance) to the audience. One way could be via the running EC projects (they all have a work package on dissemination of knowledge)
- Exchange of information on the issues discussed during the Workshop amongst the partners of the Forum. This may include practical recommendations, *etc*.
- Background/basic information on the methodology to investigate the performance (e.g. chemical durability) of spent fuel, cladding materials and container materials on the long-term (geological disposal timeframes). This would be of benefit for many countries, developed as well as non-developed ones. A good driver might be IAEA (NEA?), through expert meetings (but the time to deliver the output should be acceptable).
- The interim storage of the radioactive waste packages might impose additional requirements to the waste packages, certainly if the interim storage will take longer than planned (up to 300y). The canisters for instance might not be corrosion resistant for that long. There's a need to discuss possible requirements due to the interim storage.

- Specific problems proposed within the Forum might better be tackled via specific projects – proposals to the EC? The ageing and microstructure evolution of fuel cladding materials is such an issue.

Conclusions Groups Discussion Manufacturing, Quality Control and Non-Destructive Testing

Rapporteur: Ian Norris TWI

All participants were asked to give their views on the most important challenges for manufacturing of packages for radioactive waste over the coming 2-10 year period.
A number of ideas were raised and discussed and the following 5 areas were defined as being of importance and as topics in which some form of information sharing and collaboration would bring benefit.

1) Non destructive testing
It was felt that there was a strong need for application of NDT to critical fabricated structures but that there was little knowledge or experience in some of the participating countries of the techniques available, their areas of application, capabilities and limitations. Sharing of knowledge of these aspects and also of the influence of human factors on the interpretation of signals from the various techniques was thought to be of value. There is the potential for a valuable round robin exercise on testing of known parts containing known defects in different organisations and in different countries to understand the effect of different methodologies and operating practices on accuracy of defect identification and sizing. The issue of NDT of cast iron using ultrasonic techniques was raised as a current industrial problem

2) Application of fracture mechanics techniques to waste packages
It was felt that the correct application of fracture mechanics techniques to waste packages to establish allowable flaw sizes and defect significance was important but that the approach was difficult to apply across all package types because of the vast differences in materials used and their defect tolerance. E.g. the application of fracture mechanics analysis for cast iron packages would be vastly different to the approach required for copper canisters. There may be value in dividing the 'users' in to groups with similar acceptance criteria so that they can work as small groups with a common need to apply the approach for their particular package types.

3) Cast iron vessels
It was stated that cast iron was accepted for intermediate waste storage in Germany but that the approach to assessment of flaws was not necessarily optimum for the application. Flaws in cast iron are generally volumetric but are treated as being sharp cracks in current analyses. Also, the assessment of the significance of defects was often made on the basis of performance in fatigue whereas the packages were generally not subject to fatigue loading. There was felt to be great benefit in sharing knowledge on improvements to cast iron casting quality, evaluation of castings by NDT and application of fracture mechanics to cast iron parts. Currently fracture toughness testing of cast iron is not standardised and equally reliable across different testing facilities. A round robin testing activity between different testing facilities would help to identify issues and raise the knowledge level amongst users and testers of cast iron components

4) Long term performance of welded structures
The long term performance of structures containing welds was felt to be any area in which more information and sharing of knowledge would be beneficial. As examples, there was concern that the very fine grain size achieved in friction stir welds in copper may give rise to local creep and to strain concentration in small regions in the weld. In steels, preferential corrosion in welds has been seen in trials in Japan. Techniques for modelling the long term effects and for carrying out reliable accelerated testing were felt to be required and were a valid topic for collaboration. The influence of residual stress on joint performance long term was also a concern and the investigation of residual stress mitigation techniques could be useful.

5) Commonality of technical standards
Fabricators use manufacturing standards when producing structures and it was felt that there would be benefit in sharing the format of standards to be adopted for components designed for long term storage to ensure that all important aspects of fabrication for such critical parts were taken in to account by manufacturers regardless of their country of origin and previous experience and practices.

It was stressed that this did not mean sharing the details of those standards but only the framework to allow others to write their own specific standards to an agreed format. This was felt to be useful because the default standards that would be likely to be used if specific standards did not exist would be pressure vessel standards which are probably highly conservative for long term storage packages.

I hope that this summary is clear but please let me know if you require any further information.

Best regards,

Ian

Impact Accidents

- Transport stage is well defined
  - Cumulative testing
  - unyielding target
  - defined performance criteria
  - design solution with impact and thermal protection
  - licensed design, non-active materials that are not aged.

Impact Accidents

- Storage / Repository situation
  - Bare waste packages
  - Development of real target
  - Consideration of package on package

- Areas for collaboration include:
  - Modelling
  - Testing

Impact Accidents

Development of Reference Cases

- Real target
- Thin shell package
- Thick shell package
- Drop scenario

- Finite Element Modelling
  - For testing instrumentation and prediction
  - Require knowledge of properties for target reference case
- Drop Testing
- Further modelling
  - Require knowledge of properties for waste package reference case, materials and components such as bolts
- Radiological model and consequence analysis
  - Dose conversion
  - Release fraction

EUROPEAN COMMISSION
DIRECTORATE-GENERAL JRC
JOINT RESEARCH CENTRE
Institute for Energy
Nuclear Safety Unit

Petten, June 22, 2006
D(06) 15412

European Forum for Best-Practices in the Design and Assessment of Packages for Radioactive Waste

Concept Document June 2006

1. RATIONALE & BACKGROUND

The safe long-term management of radioactive waste is considered by the public to be one of the most important issues for the future use of nuclear energy. In the European Union it is the responsibility of the member states to ensure that radioactive waste management solutions are implemented, but is clearly of mutual benefit that stakeholders in different countries work together towards finding the best solutions. Waste packages for storage, transport and final disposal are an important and integral part in the radioactive waste management system. While there has been considerable funding of research on treatment of nuclear waste at European level, up to now the majority of attention has been directed to design of repositories for geological disposal and to partitioning and transmutation. The setting up of a body on design and assessment of waste packages is therefore intended to meet a need in this area.

In 2005 the JRC contacted a number of organizations to explore the interest for such a body. Based on these discussions JRC made a preliminary proposal for a Forum or network type organisation [1]. Initially this was meant as a European initiative; however, there has been a strong interest from US organizations as well as Japan.

A meeting to discuss this proposal in more detail was organized in October 2005 in conjunction with the Workshop on Design and Assessment of Radioactive Waste Packages. The participating organizations expressed a strong support for the JRC's initiative [3], but there was consensus that it should not duplicate work done elsewhere (for instance within IAEA or OECD/NEA). The meeting also identified a number of potential research areas for the Forum.

This document is an updated proposal for the Forum, including the main points raised at the October meeting.

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2. OBJECTIVES

The Forum has two overall objectives:

To promote consensus on assessment methods/requirements/procedures.

- Sharing of results from national R&D in support of national programmes.
- Develop common views on basic safety criteria, design-basis scenarios, procedures and methods (analysis, qualification and testing).
- Organise open events (seminars, conferences) to ensure wider use and acceptance of the technology.
- Qualification of evaluation tools and techniques.
- Exploration of best practices used in design and assessment for waste packaging.

To undertake joint research projects, either self-funded or with EC support:

- Identify topics for R&D and establish a R&D priority programme. These topics will form the basis for project proposals.
- Undertake collaborative R&D projects, e.g. validation benchmarks based on large-scale benchmark tests, round-robin of materials testing techniques, etc.

The work in the Forum will be based on technical and scientific assessments. Hence there will be no direct link to future European directives or any particular commercial products. Activities in other groups, in particular IAEA, NEA and DG-RTD funded programmes, need to be considered. Some interaction could be beneficial to establishing consensus, but duplication should be avoided.

3. PARTICIPATION/MEMBERS

The Forum should involve different stakeholders, such as:

- national waste management organizations,
- industry,
- research organizations,
- national safety authorities.

In general, while it is essential that the most advanced organisations (so-called “key players”) be involved, we will also strive to have broad participation from the EU member states with less advanced concepts.

There has to be a win-win situation for all involved organizations. This is particularly important for the most advanced organizations, which have perhaps more to give and less to learn. A number of member states are presently in a very hectic and advanced stage, including license applications for particular engineering designs. The technical experts

are therefore very busy and it may be difficult to devote the time to collaboration. Nevertheless we believe that each organization could benefit directly from the technical output. Another advantage for "key-players" is that by working together may provide a "European acceptance" for the more mature concepts.

4. TECHNICAL SCOPE

The Forum should cover development of containers for packaging wastes for surface storage, handling and transport as well as those for ultimate geological disposal. The basic requirements and the fundamental research may differ for the different types of waste packages. For geological disposal the requirements can be matched to time scales that need to be taken into account, along with the evolving waste form and environment. For surface storage, handling and transport containers more localised and short-term criteria constitute key requirements.

The proposed Forum on waste package design can fill a gap by considering relevant issues and performance criteria in an integrated way, covering transport and storage as well as disposal. The Forum should contribute to guidance for assessment, design and manufacture of waste packages. Design considerations should be linked to performance requirements. It could assess applicable codes & standards in terms of rationale, limitations and background, but should not take a detailed prescriptive approach.

There is already a Network for Quality checking of Waste Packages, "European Network of Testing facilities for the quality checking of RAdioactive waste Packages", EN-TRAP, (www.en-trap.org/corpo.htm). A link with this Network should be explored.

5. IPR

It is recognised that commercial/competitive aspects and intellectual properties rights are very important issues. For storage, handling, and transport commercial products are available and for the geological disposal, one could expect that the current concepts which are nearing first implementation may also become commercial. The commercial propriety aspects need to be taken into account.

6. FUNDING

Participation will be initially on a self-funded and contribution-in-kind basis. No funding at a European level is available at present, but could become available in future EU framework programmes. The possibilities for such funding would increase by establishing this forum.

7. ORGANISATION AND COORDINATION

The JRC is willing to co-ordinate the Forum (in addition to contributing technically to various working groups and projects). We propose an organisational model based on established experience in the nuclear safety area, whereby the group (Forum) has a formal partnership agreement, which the participants sign. This sets out the working rules and technical scope. The main features typically are:

- A coordination committee is formed with clearly defined participation and voting rights.

- There is an obligation for each organisation to provide a minimum contribution in-kind, which may take a variety of forms e.g. R&D, data, reports meeting participation, etc. etc.

- The JRC acts as administrative coordinator.

- A working plan should be prepared and approved on an annual basis.

- Definition of the rights and obligations concerning the exchange and dissemination of information and data.

8. PROPOSED WORK PROGRAMME

The technical areas were discussed at the October meeting. The proposal below is therefore mainly based on the discussion and recommendations from that meeting.

The scope should not be too wide. At a first instance emphasis should be on design and fabrication aspects, and possibly also on long term degradation. Welding, NDE performance and flaw acceptance criteria are potential cross-cutting areas of significant interest. Sharing data and experience are clearly important aspects, but the group would need to have common R&D issues where work is done jointly.

Table 1 below presents technical areas that were identified at the October meeting. The areas have been separated vertical and horizontal activities. The vertical are three thematic areas related to disposal, storage and transport operations. The horizontal areas are more specific assessment areas. The crosses, X, indicates a direct connection between horizontal and vertical activities, the parenthesis, (X), indicates a weaker link and the sign, –, no direct connection.

Table 1 Matirx of operational and generic technical assessment areas discussed for collaboration.

	Repository operation (handling, remote, welding, monitoring…..) Package closure operation, Condition monitoring	Storage operation (handling, remote, monitoring, higher burn-up…..) Condition, monitoring Initial characterization	Transport operation (handling, remote, monitoring ..) Acceptance criteria Initial characterization
NDE and welding for fabrication & closure	**X**	**X**	**X**
Flaw assessment	**X**	**X**	**X**
Long-term degradation processes a) Microstructural changes b) corrosion	**X**	**(X)**	**-**
Waste/SNF characterization and degradation	**X**	**X**	**X**
Benchmark-type experiments	-	X	**X**

8.1. Priority areas for technical work

Following the guidelines that the scope should not be too wide and the discussion on the specific technical areas we propose the following priority order:

(1) The Forum has *NDT, damage tolerance and manufacturing issues* as first priority area for technical work. This would involve NDE & welding for manufacturing purposes together with damage tolerance methods. Round-robin testing to explore different NDT techniques for different waste packages components could be a very useful exercise. This could then be combined with a parallel activity on damage tolerance for defects. Such a damage tolerance activity would include different defect analysis techniques as well as some small component tests to verify the predictions. The work should cover aspects related to uncertainty. The details of such a program would of course depend strongly on the components, the materials and the loads or degradation mechanisms. This area has also the advantage that there are no ongoing activities in other international fora.

(2) The long-term degradation is perhaps the most characteristic problem for waste package components and would be the second priority area. A very basic issue is to identify the very slow degradation processes that may jeopardize the waste package integrity. Risk-informed and performance based approaches to capture long-term uncertainties are important aspects that need to be linked to safety assessments. Another fundamental issue is how to perform accelerated tests. Progress in this area requires modelling of the processes at the microscale in combination with selected experiments. It is also an area that requires basic research and there should therefore be a broad interest for collaboration. It is also directly linked to manufacturing aspect, for instance how does a certain heat treatment of a material affect its long-term behaviour? Before embarking on a detailed work programme activities in international for a need to be closely reviewed to avoid duplication. Corrosion is generally considered to be the most important degradation mechanism that needs to be considered in the overall assessment. However, specific corrosion studies are performed in other fora and it should probably not be a focal point here.

(3) Characterization and modelling of the long-term behaviour and degradation of the waste or spent fuel is important since it sets the acceptance requirements for waste packages. Several organizations expressed interest to work in this area. A specific issue is for instance whether and how data for one waste type can be transferred to another. The RMBK fuel was in particular mentioned as one fuel type where knowledge of the long-term performance is not well known. There are projects, for instance within IAEA (SPAR-II project), that deal with these issues. This is an area where another JRC institute, ITU, is a leading organization. Fuel cladding is the first barrier for spent fuel. Its integrity is important for storage and could also be very relevant in case that the spent fuel should be retrieved. There is an on-going NEA project lead by Studsvik on this issue SCIP (Studsvik Cladding Integrity Project). Work in these areas should involve ITU from the JRC side.

(4) JRC-IE has organized a number of very successful projects built around large benchmark tests in the nuclear plant safety area. This is an ideal form for collaboration for large experimental tests. One partner offers to share his test results and other partners contribute by for instance supporting material characterization tests, NDT work, pre-test analysis for optimal test design, micro-structural post-test analysis and modelling. The very expensive drop tests for transport casks would be ideal for such a benchmark exercise but it requires that one or several organizations would offer to share the test results.

8.2. Working System

We propose that the work of the Forum is organised as follows:

a) The focal point is an annual 2-3 day workshop, which would include a ½ day meeting of the Forum coordination committee. These workshops would cover general thematic aspects (1day) as well as the results of the technical activities in the priority areas outlined above. The Workshops would have different targeted areas. The EBS workshops organized by OECD/NEA on "Integration of Engineered Barrier System in the Safety Case" could serve as a good example. In addition to the Annual workshop dedicated project meeting would be organized for the technical projects according to the need.

b) Ad hoc project groups would be established to address specific R&D issues. These would have a nominated responsible. The work-plan and schedule would be approved by the coordination committee.

9. NEXT STEPS

We propose that a two-day meeting of the forum is organized in the late autumn here in Petten (or elsewhere if some organization is keen to host it). This would be composed of 1 day meeting for the forum and one day of technical presentations a "mini workshop".

The objective would be:

- To formally kick-off the "forum" and establish Terms-of-Reference and programme,
- Have technical presentations and discussions for the topics listed above. Emphasis on Topic (1) and (2) but (3) and (4) also possible. We would like to have about ten 30 minute presentations.
- Prepare proposals for technical projects based on the list above.

As a date we would propose November 21 – 22. As an alternative date we have December 6-7. We have funding left from last year's workshop which we can use. This funding is ear-marked for "new member states and candidate countries". Thus some restrictions apply. We will be able to fund the participation from these countries, the organization of the meeting and cover the cost for some participants from other countries.

A detailed Agenda or more defined work plan will be established in the coming months, but for this we will need the active input from you.

We would like that you before July 15:

- Let us know whether you or someone else from your organization can attend the meeting. If the proposed date is not suitable, please indicte if other dates this week or the week before or after would suit you better.

- We would welcome very much concrete ideas for the technical projects.

- Let us know if there are other organizations or persons that we should involve.

We will return to you in August as regards possible presentations

10. REFERENCES

[1] European Forum/Network for Best-Practise in Design and Assessment of Radioactive Waste Packages, D(05) 6497, K-F Nilsson and N. Taylor, JRC-Petten, march 2005.

[2] Proceedings Workshop on Design and Assessment of Radioactive Waste Packages, EU 22138EN, edited by K-F Nilsson, Petten, The Netherlands, October 2005.

[3] Minutes of the Design & Assessment of Radioactive Waste Packages Workshop Follow-Up Meeting 7th October 2005, Petten, The Netherlands (K-F Nilsson and N. Taylor) D(05)30978