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2006 Workshop on Design and Assessment of Packages for Radioactive Waste

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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.
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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 shearography. 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.
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 differs 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 inspection 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"


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


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


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

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

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


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

[21] Conclusions Working Group Drop Tests

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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Radioactive Waste research at JRC

- ITU: Research on P&T, spent fuel behaviour and characterization
- IE: SAFECASK (6\textsuperscript{th} FP)/SAFEWASTE (7\textsuperscript{th} 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
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

Waste form
- vitrified HLW matrix
- spent fuel rods

Canister
- cast iron
- stainless steel
- copper
- titanium

Geology
- clay
- crystalline rock
- salt dome

Buffer
- bentonite
- cement grouts
- concrete

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.

• 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.

• 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……

- 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

Deterministic

Probabilistic

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

21\textsuperscript{th}-22\textsuperscript{nd} 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 collaboration;
  – 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
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 Consideration:
  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:

- 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)

- 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)

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

- Release Fraction - impact energy, oxidation from UO$_2$ to U$_3$O$_8$, high-burnup
- Leak Path Factor – HEPA efficiency, stack height, building leakage

10 μm [3.9x10$^{-4}$ in]
1 J/g [1.1x10$^2$ 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/UO₂ interaction

100 MPa [14.5 KSI]
Operational Safety (Preclosure)
PCSA Exercise Results

**Onsite Dose/PWR Assembly Damaged (mrem)**

- **Assembly Drop/Collision**
- **SNF Oxidation/Pulverization**

Downwind Distance from an Elevated Release (m)

---

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, NaNO₃ and NaNO₃
• 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

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

Ahn, Pan and others (2006)

1 A/cm² [0.5 A/in²]
Waste Isolation (Postclosure)
Stress Corrosion Cracking

• Unlikely to happen

• Long-term effects: threshold potential model or $K_{\text{ISCC}}$ 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$_3$/$\text{HCO}_3$], [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)

- 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 [3.6x10^{-5} lb/in^3]
In-Package Chemistry

- 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

DOE (2002)
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


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

Long-Term Integrity of Overpack Final Closure

Carbon steel overpack (JNC’s H12 Report)
Wall thickness
Corrosion allowance : 40 mm
Pressure resistance : 110 mm
Radiation shielding : 150 mm

1,000 years containment
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

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

Assumption 1: Crack growth due to earthquake is negligible

Assumption 2: Mechanical strength is maintained

Critical crack length: 56 mm

Max. tolerable flaw size: $a_{\text{max}}$ mm

Quantification limit: 2 mm

Time: $t_1$: Lifetime = 1,000 years

At the corrosion allowance layer:
1. Corrosion form → uniform corrosion
2. No susceptibility to SCC
3. No neutron radiation embrittlement

At the mechanical strength layer:
4. Fracture toughness of carbon steel (precise value)
5. Mechanical buffering property of buffer material

An appropriate safety factor $f_s$
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 evaluation 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 °C, synthetic sea water, -650mV(SCE), 100 h)*

- Preferential corrosion / TIG weld specimen

---

**Surface appearance after immersion test**

- Local deposition
- Continuous deposition
**Preferential corrosion**

(1) **Chemical composition**

<table>
<thead>
<tr>
<th>Materials</th>
<th>C</th>
<th>Si</th>
<th>Mn</th>
<th>P</th>
<th>S</th>
<th>Cu</th>
<th>Cr</th>
<th>Mo</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>TIG</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Metal</td>
<td>0.15</td>
<td>0.19</td>
<td>0.36</td>
<td>0.006</td>
<td>0.002</td>
<td>0.01</td>
<td>0.05</td>
<td>0.01</td>
</tr>
<tr>
<td>Weld Metal</td>
<td>0.11</td>
<td>0.67</td>
<td>1.29</td>
<td>0.009</td>
<td>0.011</td>
<td>0.25</td>
<td>0.02</td>
<td>&lt;0.01</td>
</tr>
<tr>
<td>Filler Metal</td>
<td>0.11</td>
<td>0.72</td>
<td>1.41</td>
<td>0.013</td>
<td>0.013</td>
<td>0.24</td>
<td>0.04</td>
<td>0.01</td>
</tr>
<tr>
<td><strong>EBW</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Metal</td>
<td>0.12</td>
<td>0.25</td>
<td>0.65</td>
<td>0.012</td>
<td>0.004</td>
<td>0.05</td>
<td>0.11</td>
<td>0.02</td>
</tr>
<tr>
<td>Weld Metal</td>
<td>0.11</td>
<td>0.25</td>
<td>0.7</td>
<td>0.011</td>
<td>0.003</td>
<td>0.05</td>
<td>0.11</td>
<td>0.02</td>
</tr>
</tbody>
</table>

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


---

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

1. 1M NaHCO₃ + 0.5M Na₂CO₃ & Silicon Oil
2. 80°C
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)
**SCC Susceptibility**

Base Metal \(\geq\) Bond (Base Metal) \(\geq\) HAZ (Base Metal) \(\geq\) Weld Metal \(\geq\) 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.: 45,000 MWD/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/cm²)
   - 3-D Monte-Carlo Code, MCNP-4C2
   - Neutron energy: >1 MeV
   - (α, n) and spontaneous fission neutron

National prediction formula of neutron radiation embrittlement for reactor pressure vessel (Transition of RTNDT)
(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

<table>
<thead>
<tr>
<th>Coolin time (year)</th>
<th>Calculation</th>
<th>Neutron dose deposition (n/cm²)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Lid center/inside</td>
<td>Vitrified waste center /side wall inside</td>
</tr>
<tr>
<td>30</td>
<td>(α, n)</td>
<td>3.114E+13</td>
</tr>
<tr>
<td></td>
<td></td>
<td>8.153E+13</td>
</tr>
<tr>
<td></td>
<td>Spontaneous fission neutron</td>
<td>1.109E+13</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2.896E+13</td>
</tr>
<tr>
<td></td>
<td>(α, n) + s.f.n</td>
<td>4.223E+13</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.105E+14</td>
</tr>
<tr>
<td>50</td>
<td>(α, n)</td>
<td>2.877E+13</td>
</tr>
<tr>
<td></td>
<td></td>
<td>7.539E+13</td>
</tr>
<tr>
<td></td>
<td>Spontaneous fission neutron</td>
<td>6.342E+12</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.657E+13</td>
</tr>
<tr>
<td></td>
<td>(α, n) + s.f.n</td>
<td>3.512E+13</td>
</tr>
<tr>
<td></td>
<td></td>
<td>9.196E+13</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Evaluation point</th>
<th>Neutron dose deposition (n/cm², E ≥ 1 MeV)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lid center/inside</td>
<td>4.223E+13</td>
</tr>
<tr>
<td>Vitrified waste center /side wall inside</td>
<td>1.105E+14</td>
</tr>
<tr>
<td>Availability</td>
<td>1.0 × 10E+17 ~ 1.0 × 10E+20</td>
</tr>
</tbody>
</table>
3. Structural Integrity
Max. Principal Stress & Critical Crack Length at Different Penetration Depth

Body Drop-in Lid Type
(Unit:mm)

Lid Closure weld

190mm Full penetration
80mm Partial penetration
50mm Partial penetration

Max. Principal Stress under (TS+EP+Eq+(RS)) Condition
**Critical crack length under (TS+EP+Eq+(RS)) Condition**

<table>
<thead>
<tr>
<th>Crack Model</th>
<th>Penetration depth (mm)</th>
<th>Full penetration</th>
<th>Partial penetration</th>
</tr>
</thead>
<tbody>
<tr>
<td>Elliptical crack in an infinite body</td>
<td>88</td>
<td>72</td>
<td>59</td>
</tr>
<tr>
<td>Semi-elliptical surface crack</td>
<td>69</td>
<td>56</td>
<td>46</td>
</tr>
</tbody>
</table>

- 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 overpack**

<table>
<thead>
<tr>
<th>Structure NO.</th>
<th>A-1</th>
<th>A-2</th>
<th>B-1</th>
<th>B-2</th>
<th>C-1</th>
<th>C-2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lid Structure</td>
<td>Drop-in lid</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Penetration depth (mm)</td>
<td>190mm full penetration</td>
<td>80mm Partial penetration</td>
<td>50mm Partial penetration</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Welding method</td>
<td>TIG</td>
<td>EBW</td>
<td>TIG</td>
<td>EBW</td>
<td>TIG</td>
<td>EBW</td>
</tr>
<tr>
<td>Welding time (for 1 overpack)</td>
<td>23h.</td>
<td>30min.</td>
<td>10h.</td>
<td>9min.</td>
<td>8.3h.</td>
<td>4min.</td>
</tr>
<tr>
<td>Weld surface</td>
<td>As weld</td>
<td>Machining</td>
<td>As weld</td>
<td>Machining</td>
<td>As weld</td>
<td>Machining</td>
</tr>
</tbody>
</table>

**NDE**

1. Surface: CW(DTPT) Detection
2. Surface: CW(DPT) Sizing, ≥2mm
3. Inside(1~10mm) CW(DPT) Sizing, ≥2mm
4. Inside(1~30mm) TOFD Sizing, ≥2~5mm
5. Inside(1~10mm) TOFD Sizing, ≥2mm
6. Inside(1~190mm) TOFD Sizing, ≥2mm

- 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

<table>
<thead>
<tr>
<th>Prerequisite No.</th>
<th>Items</th>
<th>Current Situation</th>
<th>Future work for getting more reliability</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>General Corrosion</td>
<td>・Most Fundamental Property ・Influence of Filler Metal</td>
<td>Effect of filler metal on the corrosion property of the weld metal</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Design of filler metal (chemical composition)</td>
</tr>
<tr>
<td>2</td>
<td>Susceptibility to SCC</td>
<td>・No effect of welding on the susceptibility</td>
<td>More realistic condition</td>
</tr>
<tr>
<td></td>
<td></td>
<td>・Appropriateness of evaluation method(SSRT)</td>
<td>Chemical environment</td>
</tr>
<tr>
<td></td>
<td></td>
<td>・Appropriateness of evaluation time</td>
<td>Loading condition</td>
</tr>
<tr>
<td>3</td>
<td>Neutron Radiation Embrittlement</td>
<td>・Used estimation method for nuclear power plant component (High radiation/shor time)</td>
<td>Prediction method</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Low &amp; long-term radiation</td>
</tr>
<tr>
<td>4</td>
<td>Fracture Toughness</td>
<td>・Used low carbon steel for nuclear power plant component</td>
<td>More realistic estimation</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Selected carbon steel</td>
</tr>
<tr>
<td></td>
<td></td>
<td>・Available data for a geological formation on the estimation of critical crack length</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Preciseness of geological condition</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>External Pressure</td>
</tr>
<tr>
<td></td>
<td></td>
<td>・LFM/NLFM</td>
<td>Fracture mode</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Plastic-elastic fractute mechanics</td>
</tr>
<tr>
<td>5</td>
<td>Mechanical Buffering Property of Buffer Material</td>
<td>・Available data for a buffer material</td>
<td>Maintaining the mechanical buffering property</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>saturation, heat effect, stress condition, chemical reaction etc.</td>
</tr>
<tr>
<td>6</td>
<td>Safety Factor</td>
<td>・Referring from FFS of nuclear power plant component</td>
<td>Estimation basis</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Appropriate value as a disposal container</td>
</tr>
</tbody>
</table>

5. Conclusions

1. Preferential corrosion induced on the weld metal provided 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
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

- An assessment period longer than 100,000 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 100,000 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!

(SKB, TR-04-11, 2004)
FORMATION OF FENNOSKANDIA AND ITS TRAVELLING ON EARTH

- Ice age cycles occurred already during early geological times on earth

- For example, Huron ice age (2.3 m.y. ago) lasted over 200 m. y.

- During the time of dinosaurs (250 – 65 m. y. ago) - Jurassic and Cretaceous 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.

(M. Koivisto, Jääkaudet (Ice ages), 2004)
THE EARTH LIVES THE TIME OF RECURRING ICE AGES

- 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 fossil 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)
• 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 4-5°C and in Eurasia 10-15°C 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

- 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

(M. Koivisto, Jääkaudet (Ice ages), 2004)
ASTRONOMIC CLIMATE CHANGE THEORY OF MILUTIN MILANKOVITCH

- According to theory continental 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 (22,0-24,5°) 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 60,000 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 2°C 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

- The climate evolves in roughly 100,000 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.

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.

(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 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

- 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.

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

(SKB, TR-04-21, 2004)
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

- As an ice sheet advances and retreats, changes occur in temperature, hydrological 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.

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)
DESIGN PREMISES FOR THE CANISTER

- The canister must withstand all known corrosion processes and it is expected to remain intact in the repository for at least 100,000 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.

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

(SKB, TR-04-11, 2004)
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)
- 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

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

(SKi Report 2005: 18)
MECHANICAL DESIGN LOADS OF THE CANISTER

- 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

- 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

**FIGURE 2.** Special local swelling loads postulated during wetting phase. (Posiva, 2006-02)
SPECIAL SWELLING PRESSURE LOADS OF BENTONITE

**FIGURE 3.** Special swelling pressure loads postulated in fully saturated phase. Groundwater hydrostatic pressure can be present simultaneously. (Posiva, 2006-02)
EFFECTS OF ROCK SHEAR MOVEMENTS ON PLASTIC STRAIN IN CANISTER

- 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.

Figure 1.2: Examples of results from a calculation with 20 cm eccentric rock shear (density at water saturation 2,000 kg/m³ 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)
DEFORMATION OF THE CANISTER INSERT IN THE PRESSURE TEST

- 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

- 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)

Figure 2-1. Vertical variation of the main hydrochemical constituents at Olkiluoto. Vertical lines represent smooth changes in groundwater.
<table>
<thead>
<tr>
<th>Constituent</th>
<th>Initial, indisturbed conditions</th>
<th>At closure, infiltration into unsaturated bentonite</th>
<th>After closure and saturation** (i.e. up to 100 years)</th>
<th>After closure up to 10,000 years</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
<td>7.8–8.3</td>
<td>6–8</td>
<td>7–8</td>
<td>7.5–8.3</td>
</tr>
<tr>
<td>Redox</td>
<td>mV</td>
<td>-230 to -280</td>
<td>Oxic to -250</td>
<td>-150 to -250</td>
</tr>
<tr>
<td>DIC</td>
<td>mol/L</td>
<td>(0.7–0.1)·10^-3</td>
<td>(0.1–10)·10^-3</td>
<td>(0.5–10)·10^-3</td>
</tr>
<tr>
<td>Cl^-</td>
<td>mg/L</td>
<td>7,500–15,000</td>
<td>1,000–22,000</td>
<td>1,000–5,000</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>(2.1–4.2)·10^-1</td>
<td>(0.3–6.2)·10^-1</td>
<td>(0.3–1.4)·10^-1</td>
</tr>
<tr>
<td>Na^+</td>
<td>mg/L</td>
<td>2,500–5,000</td>
<td>500–6,500</td>
<td>500–2,000</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>(1.1–2.2)·10^-1</td>
<td>(0.2–2.8)·10^-1</td>
<td>(0.2–0.9)·10^-1</td>
</tr>
<tr>
<td>Ca^{2+}</td>
<td>mg/L</td>
<td>1,500–4,000</td>
<td>100–6,000</td>
<td>100–1,000</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>(0.4–1.0)·10^-1</td>
<td>(0.03–1.5)·10^-1</td>
<td>(0.03–0.2)·10^-1</td>
</tr>
<tr>
<td>Mg^{2+}</td>
<td>mg/L</td>
<td>30–70</td>
<td>10–250</td>
<td>10–250</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>(1.2–2.9)·10^-3</td>
<td>(0.04–1.0)·10^-2</td>
<td>(0.04–1.0)·10^-2</td>
</tr>
<tr>
<td>K^+</td>
<td>mg/L</td>
<td>10–20</td>
<td>5–30</td>
<td>5–30</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>(2.6–5.1)·10^-4</td>
<td>(1.3–7.7)·10^-4</td>
<td>(1.3–7.7)·10^-4</td>
</tr>
<tr>
<td>SO_4^{2-}</td>
<td>mg/L</td>
<td>0–20</td>
<td>0–500</td>
<td>10–500</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>0–0.2·10^-3</td>
<td>0–5.2·10^-3</td>
<td>0–5.2·10^-3</td>
</tr>
<tr>
<td>HS^-</td>
<td>mg/L</td>
<td>0–3</td>
<td>0–3</td>
<td>0–10</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>0.9·10^-4–0</td>
<td>0–0.9·10^-4</td>
<td>0–3.0·10^-4</td>
</tr>
<tr>
<td>NH_4^+</td>
<td>mg/L</td>
<td>&lt;0.05</td>
<td>&lt;0.1, but if marine</td>
<td>&lt;3*</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>&lt;0.03·10^-4</td>
<td>&lt;0.1, but if saline</td>
<td>&lt;5·10^-6,</td>
</tr>
<tr>
<td>CH_4(g)</td>
<td>ml/L</td>
<td>50–400</td>
<td>&lt;0.1, but if saline</td>
<td>&lt;100</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>(0.2–1.8)·10^-2</td>
<td>&lt;0.1, but if saline</td>
<td>&lt;600</td>
</tr>
<tr>
<td>H_2(g)</td>
<td>ml/L</td>
<td>&lt;0.5</td>
<td>&lt;0.5, but if saline</td>
<td>&lt;0.1</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>&lt;2.2·10^-5</td>
<td>&lt;0.5, but if saline</td>
<td>&lt;4.4·10^-6</td>
</tr>
<tr>
<td>DOC#</td>
<td>mgC/L</td>
<td>&lt;2</td>
<td>&lt;2</td>
<td>&lt;10</td>
</tr>
<tr>
<td></td>
<td>mol/L</td>
<td>&lt;1.7·10^-4</td>
<td>&lt;2</td>
<td>&lt;8.5·10^-4</td>
</tr>
<tr>
<td>Microbes</td>
<td>SRB, methanogens</td>
<td>Aerobic bacteria, SRB, IRB, methanogens</td>
<td>SRB, IRB, methanogens</td>
<td></td>
</tr>
</tbody>
</table>

*) 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.

<table>
<thead>
<tr>
<th>Constituent</th>
<th>Infiltrating groundwater at closure</th>
<th>Pore-water in saturated bentonite (up to 100 years)</th>
<th>Por-water after closure up to 10,000 years</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
<td>6–8</td>
<td>7–9</td>
<td>7–9</td>
</tr>
<tr>
<td>Redox</td>
<td>mV</td>
<td>Oxic to −250</td>
<td>−150 to −250</td>
</tr>
<tr>
<td>DIC</td>
<td>mol/L</td>
<td>(0.02–1.6)⋅10(^{-4})</td>
<td>no estimate</td>
</tr>
<tr>
<td>Cl(^{-})</td>
<td>mol/L</td>
<td>(0.3–6.2)⋅10(^{-1})</td>
<td>(0.3–6.2)⋅10(^{-1})</td>
</tr>
<tr>
<td>Na(^{+})</td>
<td>mol/L</td>
<td>(0.2–2.8)⋅10(^{-1})</td>
<td>(3–5)⋅10(^{-1})</td>
</tr>
<tr>
<td>Ca(^{2+})</td>
<td>mol/L</td>
<td>(0.3–1.5)⋅10(^{-1})</td>
<td>(0.4–4.0)⋅10(^{-2})</td>
</tr>
<tr>
<td>SO(_4^{2-})</td>
<td>mol/L</td>
<td>(0–5.2)⋅10(^{-3})</td>
<td>1.4⋅10(^{-1})</td>
</tr>
<tr>
<td>HS(^{-})</td>
<td>mol/L</td>
<td>(0–0.9)⋅10(^{-4})</td>
<td>(0–3)⋅10(^{-4})</td>
</tr>
<tr>
<td>NH(_4^{+})</td>
<td>mol/L</td>
<td>&lt;5.5⋅10(^{-6})</td>
<td>(0.03–1.7)⋅10(^{-4})</td>
</tr>
<tr>
<td>CH(<em>4)(</em>{(g)})</td>
<td>mL/L</td>
<td>&lt;4.5⋅10(^{-6})</td>
<td>&lt;4.5⋅10(^{-3})</td>
</tr>
</tbody>
</table>

*) 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

- 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.)

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).

(SKB, TR-04-05)
ASSESSMENT OF CORROSION BEHAVIOR OF COPPER CANISTERS

- 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/cm²), 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

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 Cu₂O/CuO equilibrium potential line. The corrosion potential is initially above the Cu₂O/CuO equilibrium potential line but stabilises well below for subsequent stages (source: SKB workshop presentation).
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.
Table 1. Threats to be considered for the evaluation of canister corrosion.

<table>
<thead>
<tr>
<th>Threat</th>
<th>Time frame in EBS evolution of main relevance</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oxygen</td>
<td>Initial phase and possibly glacial</td>
<td>Rapid reaction of oxygen with copper with formation of copper oxides</td>
</tr>
<tr>
<td>Chloride</td>
<td>Throughout</td>
<td>Complexation with copper ions</td>
</tr>
<tr>
<td>Sulphide</td>
<td>Throughout</td>
<td>Reaction with copper by reduction of water and formation of copper sulphides</td>
</tr>
<tr>
<td>Nitrogen compounds</td>
<td>Initial phase</td>
<td>Ammonia and nitrite are along with acetate most important agents facilitating stress corrosion cracking</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Nitric acid is an aggressive reactant formed by radiolysis in the interior of the canister</td>
</tr>
<tr>
<td>Microbes</td>
<td>Throughout</td>
<td>Microbes catalyse otherwise relatively slow inorganic processes, such as consumption of oxygen and reduction of sulphate</td>
</tr>
<tr>
<td>Surface salt</td>
<td>Resaturation phase</td>
<td>Evaporation of water on canister surface leading to accumulation of salt deposits in micro-environments on the canister surface</td>
</tr>
<tr>
<td>Sulphide whiskers</td>
<td>Reducing phase</td>
<td>Rapid formation of whiskers on copper surfaces during pitting corrosion</td>
</tr>
<tr>
<td>General corrosion</td>
<td>Oxygenated phase (by O$_2$)</td>
<td>Uniform thinning of the canister wall</td>
</tr>
<tr>
<td>Pitting corrosion</td>
<td>Initial and resaturation phases</td>
<td>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.</td>
</tr>
<tr>
<td>Stress corrosion cracking</td>
<td>Initial phase and possibly glacial</td>
<td>Cracking at tensile parts of canister wall</td>
</tr>
<tr>
<td>Radiolysis and radiation influenced corrosion</td>
<td>Initial phase</td>
<td>Creation of locally oxidizing conditions due to gamma radiation field</td>
</tr>
<tr>
<td>Galvanic coupling</td>
<td>Early inside canister and after canister failure</td>
<td>Different potentials for iron and copper in presence of water lead to corrosion. Earth currents leading to external corrosion</td>
</tr>
<tr>
<td>Environmentally controlled creep</td>
<td>Throughout</td>
<td>Corrosion might influence creep and result in creep failure</td>
</tr>
</tbody>
</table>

(SKi Report 2006: 11)
Figure 6-14. Mechanism for the general corrosion of copper in compacted buffer material with \( O_2 \)-containing saline groundwater /King 1996b/.
MICROBIAL ACTIVITIES IN REDUCING THE AMOUNT OF OXYGEN IN THE REPOSITORY IN DIFFERENT CLIMATIC 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.

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

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 fossil 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
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”
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
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
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
Guidance on Methods

- Gamma radiography - source and detector
- X-radiography – source and film
- Ultrasonics - probe
- Conventional weight and dimensional checks
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
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
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
37 GBq Colbalt-60 Source
Types of Flaw

- Shrinkage cavities
- Air locks
- Gaps between successive pours
- Blowholes
- Porosity
- Inclusions
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
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
HLW/SF Concept

- HLW or spent nuclear fuel
- Bentonite clay
- Surface facilities of deep repository
- Copper canister with cast iron insert
- Host rock
- Underground portion of deep repository

nirex
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
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
Co-location Concept

Chemical Conditioning
- Alkaline
- Sorbing

Geological Containment
- Low water flow
- Physical stability

Physical Containment
- Container integrity
- Low hydraulic conductivity

Vault
ILW
Cement-based backfill material

Tunnel
HLW or SP
Bentonite buffer

Not to Scale
Packaging process for HLW/SF
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
Standard Waste Packages
(Not to Scale)
Standards

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

Seawater drippers

Vacuum monitoring sensors

40mm
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
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

[Diagram of double wall with labels: Front Wall, Back Wall, Water, Product, Probe]
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 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: $K_{IC} \geq 50 \text{ MPa}\sqrt{\text{m}}$
- **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.

---

Dr.-Ing. B. Droste  Workshop on Design and Assessment of Packages for Radioactive Waste
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.

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.
• 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.
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:

\[
K_{1,\text{appl}} = \max \{ K_{1,\text{appl}}(\bar{x}, t) \}
\]

\[
J_{1,\text{appl}} = \max \{ J_{1,\text{appl}}(\bar{x}, t) \}
\]

- sub-model technique allowed, if quasi-static crack problem
• Measuring of valid fracture toughness values ($K_{lc}$, $K_{ld}$) 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: $J–\Delta a$.

• 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$^{-1}$. 
• 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

Cask Assembly in a Storage Building
Model for Kinematic Analyses

Aircraft Wrackage Load
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**
- ≈80°C
- ≈300°C

**After 20 minutes fire duration**
- >500°C
- ≈300°C
- ≈400°C

**2 hours after end of fire**
- ≈300°C
- ≈350°C
- ≈250°C
BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

**Adressed Objects:**

- **Lid system**
  - top shock absorber
  - secondary lid
  - secondary lid bolts
  - primary lid
  - primary lid bolts
  - cask body

- **Trunnion system**

---

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

**Current Developments:**
Numerical approaches based on local stress limitation concepts

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Dr.-Ing. B. Droste  
Workshop on Design and Assessment of Packages for Radioactive Waste
BAM Guideline for Calculation and Assessment of Lid and Trunnion Systems for RAM Packages (Draft)

Adressed Verifications:

<table>
<thead>
<tr>
<th>Lid system</th>
<th>Trunnion system</th>
</tr>
</thead>
<tbody>
<tr>
<td>• strength of lid bolts</td>
<td>• strength of trunnions</td>
</tr>
<tr>
<td>• compression of seals</td>
<td>• fatigue strength of trunnions</td>
</tr>
<tr>
<td>• strength of lids</td>
<td>• strength of trunnion bolts</td>
</tr>
<tr>
<td>• sliding of lids</td>
<td>• fatigue strength of trunnion bolts</td>
</tr>
</tbody>
</table>

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

local stresses

nominal forces

nominal stresses

nominal loads

von Mises stress

\[ F = \int \int_A \sigma \, dA \quad M = \int \int_A s \cdot \sigma \, dA \]

\[ \sigma_z = \frac{F}{A_s} \quad \sigma_b = \frac{M}{W} \]

\[ \sigma_v = \sqrt{\left(\sigma_z + \sigma_b\right)^2 + 3 \cdot (f_M \cdot \tau_{G,Mon})^2} \]
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`e-s-eye View of BAM`e-s Drop Test Facility in Horstwalde
The new large BAM Drop Test Facility

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

Steel reinforced Concrete Foundation:
14m x 14m x 5m
2,450,000 kg Concrete
103,000 kg Steel Reinforcement

Test Pad of Steel Plates:
10m x 4.5m x 0.22m
77,000 kg
Preparation of the CONSTOR V/TC Cask before Drop Test
Connection of the Measurement Cables to the CONSTOR V/TC Full-Scale Cask
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 Criteria – 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

- Copper canister (Pierce and Draw)
- Copper tube (Extrusion, Forging)
- Welding of Copper bottom
- Copper Lid (Forging)
- Steel channel (Casting)
- Copper canister (Forging)
- Copper Lid
- Acceptance inspection
- Nuclear Fuel
- Welding of Copper lid
- Nodular cast iron insert
- Encapsulation plant

Final Disposal of nuclear fuel canister
Nuclear Fuel Disposal Canister Principle Design in Finland and Schweden

- Copper canister
- Nodular cast iron insert
- Steel channel
Defect types

Pipes, pores, inclusions

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

Detection of a defect

Sizing of a defect

NDT methods

Fracture Mechanics (UT) EN 12680-3

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)**

<table>
<thead>
<tr>
<th>Zone</th>
<th>Height [mm]</th>
<th>Length [mm]</th>
<th>Zone</th>
<th>Height [mm]</th>
<th>Length [mm]</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>37</td>
<td>222</td>
<td>A</td>
<td>53</td>
<td>318</td>
</tr>
<tr>
<td>B1</td>
<td>65</td>
<td>390</td>
<td>B</td>
<td>112</td>
<td>672</td>
</tr>
<tr>
<td>B2</td>
<td>50</td>
<td>300</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>24</td>
<td>144</td>
<td>C</td>
<td>104</td>
<td>624</td>
</tr>
<tr>
<td>D</td>
<td>32</td>
<td>192</td>
<td>D</td>
<td>32</td>
<td>192</td>
</tr>
</tbody>
</table>

- **Volumetric type defects (round holes through whole canister)**

<table>
<thead>
<tr>
<th>Zone</th>
<th>Diameter [mm]</th>
<th>Zone</th>
<th>Diameter [mm]</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>40</td>
<td>A</td>
<td>80</td>
</tr>
<tr>
<td>B1</td>
<td>60</td>
<td>B</td>
<td>100</td>
</tr>
<tr>
<td>B2</td>
<td>20</td>
<td></td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>20</td>
<td>C</td>
<td>100</td>
</tr>
<tr>
<td>D</td>
<td>20</td>
<td>D</td>
<td>20</td>
</tr>
</tbody>
</table>
Crack growth scenario according to JRC
## Acceptance levels EN 12680-3

### Severity level 1

<table>
<thead>
<tr>
<th>Depth range [mm]</th>
<th>Decrease of BWE [dB]</th>
<th>Minimum FBH [mm]</th>
<th>Defect echo S/N [dB]</th>
</tr>
</thead>
<tbody>
<tr>
<td>10 - 20</td>
<td>12</td>
<td>3</td>
<td>6</td>
</tr>
<tr>
<td>20 - 100</td>
<td>20</td>
<td>5</td>
<td>6</td>
</tr>
<tr>
<td>100 - 250</td>
<td>20</td>
<td>8</td>
<td>12</td>
</tr>
<tr>
<td>250 - 500</td>
<td>20</td>
<td>10</td>
<td>12</td>
</tr>
</tbody>
</table>

Areas where no BWE
### Acceptance levels 12680-3

<table>
<thead>
<tr>
<th>Severity level 2</th>
<th>Defect ≥ level 1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Depth zone [mm]</td>
<td>200 -</td>
</tr>
<tr>
<td>% of WT</td>
<td>20</td>
</tr>
<tr>
<td>Largest defect in the surface zone [cm²]</td>
<td>10</td>
</tr>
<tr>
<td>Largest defect in the core zone [m²]</td>
<td>0.015</td>
</tr>
<tr>
<td>Total area of inspected surface %</td>
<td>15 (1.8 m²)</td>
</tr>
</tbody>
</table>
Acceptance criteria EN 12680-3

Severity level 3

Defect ≥ level 1

Depth zone [mm] 200 -

% of WT 25

Largest defect in the surface zone [cm²] 20

Largest defect in the core zone [m²] 0.03

Total area of inspected surface % 20 (2.4 m²)
## Acceptance criteria EN 12680-3

<table>
<thead>
<tr>
<th>Severity level 4</th>
<th>Defect ≥ level 1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Depth zone [mm]</td>
<td>200 -</td>
</tr>
<tr>
<td>% of WT</td>
<td>30</td>
</tr>
<tr>
<td>Largest defect in the surface zone [cm²]</td>
<td>-</td>
</tr>
<tr>
<td>Largest defect in the core zone [m²]</td>
<td>-</td>
</tr>
<tr>
<td>Total area of inspected surface %</td>
<td>30 (3.6 m²)</td>
</tr>
</tbody>
</table>
## Acceptance criteria EN 12680-3

<table>
<thead>
<tr>
<th>Criteria</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Severity level 5</strong></td>
<td>Defect ≥level 1</td>
</tr>
<tr>
<td>Depth zone [mm]</td>
<td>200 -</td>
</tr>
<tr>
<td>% of WT</td>
<td>35</td>
</tr>
<tr>
<td>Largest defect in the surface zone [cm²]</td>
<td>-</td>
</tr>
<tr>
<td>Largest defect in the core zone [m²]</td>
<td>-</td>
</tr>
<tr>
<td>Total area of inspected surface %</td>
<td>40 (4.8 m²)</td>
</tr>
</tbody>
</table>
Inspection system for nodular cast iron insert at SKB Encapsulation laboratory (Rotator)
### Inspection techniques for nodular cast iron insert

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

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 phased array probe construction

UT- PA

n = 16

\[ A = n^2e + g(n-1) \]

\[ A_{\text{active}} \]

\[ W_{\text{passive}} \]
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

Positioning: normal to surface

<table>
<thead>
<tr>
<th>Reflector</th>
<th>dB</th>
</tr>
</thead>
<tbody>
<tr>
<td>SDH</td>
<td>0</td>
</tr>
<tr>
<td>1</td>
<td>-7.3</td>
</tr>
<tr>
<td>2</td>
<td>-6.6</td>
</tr>
<tr>
<td>4</td>
<td>-6.1</td>
</tr>
<tr>
<td>8</td>
<td>-2.7</td>
</tr>
<tr>
<td>10</td>
<td>-2.7</td>
</tr>
</tbody>
</table>

Positioning: perpendicular to surface

<table>
<thead>
<tr>
<th>Reflector</th>
<th>dB</th>
</tr>
</thead>
<tbody>
<tr>
<td>SDH</td>
<td>0</td>
</tr>
<tr>
<td>1</td>
<td>-5.9</td>
</tr>
<tr>
<td>2</td>
<td>-6.2</td>
</tr>
<tr>
<td>4</td>
<td>-9.5</td>
</tr>
<tr>
<td>8</td>
<td>-2.3</td>
</tr>
<tr>
<td>10</td>
<td>-2</td>
</tr>
</tbody>
</table>
Intermediate Zone (Zone B1, B2, B)

PA 2MHz  0° linear scanning  PE, 360° scanning along circumferential direction
1 scanning, 112 elements
Zone between steel channels (Zone C)
Channel zone inspection modelling with Civa (Puls-echo)

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 25° - 70° 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

Corner echo

Tip echo

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, Acoustic 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 qualification 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 instructions 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.
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
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
PART 1

INTRODUCTION (CONTEXT)
BELGIUM: MAIN NUCLEAR SITES

- Doel
  4 PWR units
  2550 MWe

- Tihange
  3 PWR units
  2950 MWe

- Chooz
  25% share of 2900 MW
  no waste

- Fleurus
  (IRE)

- Mol-Dessel
  (SCK.CEN)

- nuclear power plant site
- research institute
WASTE INVENTORY: 3 CATEGORIES

- Low level
- Medium level
- High level

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

Long half-life (30 years or more):
- Low level: B
- Medium level: B
- High level: C
WASTE INVENTORY: 2 SCENARIOS

Resumption of SNF reprocessing

Cat. A (70,400 m³)
Cat. B (8600 m³)
Cat. C (2100 m³)

4,000 vitrified HLW canisters
6,500 hulls & ends canisters (= cat.B)

No further SNF reprocessing

Cat. C (4700 m³)
Cat. B (8600 m³)
Cat. A (69.900 m³)

10,000 spent fuel assemblies
400 vitrified HLW canisters
800 hulls & ends canisters (= cat.B)
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
## SNF INVENTORY DETAILS

<table>
<thead>
<tr>
<th>FUEL MATRIX</th>
<th>FUEL DIMENSIONS</th>
<th>NPP of origin</th>
<th>INVENTORY (# of assemblies)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>GRID</td>
<td>ACTIVE FUEL LENGTH</td>
<td></td>
</tr>
<tr>
<td>UOX</td>
<td>14 X 14</td>
<td>8 ft</td>
<td>Doel 1 and 2</td>
</tr>
<tr>
<td></td>
<td>15 x 15</td>
<td>12 ft</td>
<td>Tihange 1</td>
</tr>
<tr>
<td></td>
<td>17 x 17</td>
<td>12 ft</td>
<td>Doel 3, Tihange 2</td>
</tr>
<tr>
<td></td>
<td>17 x 17</td>
<td>14 ft</td>
<td>Doel 4, Tihange 3</td>
</tr>
<tr>
<td>MOX</td>
<td>17 x 17</td>
<td>12 ft</td>
<td>Doel 3, Tihange 2</td>
</tr>
</tbody>
</table>

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

   ⇒ 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.
REFERENCE SITE AND HOST ROCK

The Netherlands

THICKNESS
- > 120m
- 100-120m
- 50-100m
- 0-50m

- depth base Boom Clay (TAW)
- outcrop Boom Clay

© MOL / DESSEL

ONDRAF/NIRAS

2006-OCT-18
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
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
PART 2

DISPOSAL CONCEPT FOR HLW / SNF
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
SAFIR 2 REPOSITORY CONCEPT
SAFIR 2 SNF DISPOSAL CONCEPT
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
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
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

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

\(\text{(1) OPC} = \text{Ordinary Portland Cement}\)
SUPERCONTAINER (OPC variant)
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
SUPERCONTAINER DESIGN for Vitrified HLW

Characteristics of the Supercontainer for vitrified HLW

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Outer diameter</td>
<td>2.000 m</td>
</tr>
<tr>
<td>Outer length</td>
<td>4.000 m</td>
</tr>
<tr>
<td>Approximate weight</td>
<td>30 t</td>
</tr>
<tr>
<td>Number of primary waste packs per Supercontainer</td>
<td>2</td>
</tr>
</tbody>
</table>
SNF SUPERCONTAINER DESIGN for SNF
(with CRD overpack for 14’ fuel)

Characteristics of the Supercontainer for SNF (CRD overpack)

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Outer diameter</td>
<td>2.100 m</td>
</tr>
<tr>
<td>Outer length</td>
<td>6.100 m</td>
</tr>
<tr>
<td>Approximate weight</td>
<td>60 t</td>
</tr>
<tr>
<td>Number of primary waste packs per Supercontainer</td>
<td>4</td>
</tr>
</tbody>
</table>
PART 3

DESIGN AND FABRICATION OF THE OVERPACK
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
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
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
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:

<table>
<thead>
<tr>
<th>Characteristics of the overpack for vitrified HLW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shell thickness (m)</td>
</tr>
<tr>
<td>Outer diameter (m)</td>
</tr>
<tr>
<td>Outer length (m)</td>
</tr>
<tr>
<td>number of primary waste packs</td>
</tr>
</tbody>
</table>
DESIGN OF OVERPACK for Vitrified HLW
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)
DESIGN OF CRD OVERPACK for UOX SNF

Characteristics of the CRD overpack for 14 ft UOX SNF

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shell thickness (m)</td>
<td>0.030 m</td>
</tr>
<tr>
<td>Outer diameter (m)</td>
<td>0.863 m</td>
</tr>
<tr>
<td>Outer length (m)</td>
<td>5.096 m</td>
</tr>
<tr>
<td>number of primary waste packs</td>
<td>4</td>
</tr>
</tbody>
</table>
DESIGN OF IOD OVERPACK for UOX SNF

Characteristics of the IOD overpack for 14 ft UOX SNF

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shell thickness (m)</td>
<td>0.030 m</td>
</tr>
<tr>
<td>Outer diameter (m)</td>
<td>0.948 m</td>
</tr>
<tr>
<td>Outer length (m)</td>
<td>5.005 m</td>
</tr>
<tr>
<td>number of primary waste packs</td>
<td>4</td>
</tr>
</tbody>
</table>
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

<table>
<thead>
<tr>
<th></th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shell thickness (m)</td>
<td>0.030 m</td>
</tr>
<tr>
<td>Outer diameter (m)</td>
<td>0.389 m</td>
</tr>
<tr>
<td>Outer length (m)</td>
<td>4.261 m</td>
</tr>
<tr>
<td>number of primary waste packs</td>
<td>1</td>
</tr>
</tbody>
</table>
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
ELECTRON BEAM WELDING

WITH FIXED GUN

WITH ROTATING GUN
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
## COMPARISON OF WELDING TECHNIQUES

<table>
<thead>
<tr>
<th></th>
<th>Arc</th>
<th>Friction</th>
<th>Laser</th>
<th>Electron beam</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Low thermally affected zone, residual stresses and deformations</strong></td>
<td>--</td>
<td>++</td>
<td>++</td>
<td>++</td>
</tr>
<tr>
<td><strong>Low risk of internal defects</strong></td>
<td>--</td>
<td>+</td>
<td>++</td>
<td>++</td>
</tr>
<tr>
<td><strong>Adequacy for thick joints</strong></td>
<td>++</td>
<td>-</td>
<td>--</td>
<td>++</td>
</tr>
<tr>
<td><strong>Welding speed</strong></td>
<td>--</td>
<td>++</td>
<td>++</td>
<td>+ (1)</td>
</tr>
<tr>
<td><strong>Ease of implementation in a hot cell</strong></td>
<td>-</td>
<td>-</td>
<td>++</td>
<td>+</td>
</tr>
<tr>
<td><strong>Repair facility</strong></td>
<td>--</td>
<td>--</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td><strong>Cost effectiveness</strong></td>
<td>++</td>
<td>--</td>
<td>-</td>
<td>--</td>
</tr>
</tbody>
</table>

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

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

⇒ electron beam welding is favored option
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.
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
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
The Storage and Transport System – ŠKODA VPVR/M cask’s system
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
- **CASTOR® V/21 cask**
  - 4 casks for Virginia Power, USA
Spent Fuel Transport & Storage Casks

- **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
  - 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)
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

1<sup>st</sup> 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
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
Capacity of the basket:

- total activity \( \text{max. } 3.93 \times 10^{15} \text{Bq} \)
- heating capacity \( \text{max. } 450 \text{W} \)
- maximum allowable weight of \( \text{U}^{235} \) in one loaded assembly - \( \text{max. } 500 \text{g} \)
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 for 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
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 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
Spent Fuel Transport & Storage Casks

US Ambassador Mr. R. Graber visit to SKODA JS a.s. (November 16, 2006)
Thank you for your attention
Contact info:

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316 06 Plzen
Czech Republic

Tel.: +420-378 042 242
Fax: +420-378 042 530
e-mail: info@skoda-js.cz
www.skoda-js.cz
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
Content

- Introduction
- Interim storage of SNF
- Disposal of SNF
- Determination of RBMK SNF Characteristics
- Calculation Method
- Results
- Conclusions
Workshop on Design and Assessment of Radioactive Waste Packages,
21-22 November 2006 at JRC, Institute for Energy, Petten, The Netherlands
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

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

- 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 inspection 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
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.
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

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;

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


- 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.

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

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

- $^{244}\text{Cm}$: 28–53%;
- $^{239}\text{Np}$: ~30%;
- $^{242}\text{Cm}$: 7–36%;
- $^{106}\text{Ru}$: ~40%;
- $^{243}\text{Am}$: 10–41%;
- For other nuclides: 1–15%.
- $^{241}\text{Pu}$: 20–28%;
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.
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)

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

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
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.
Design & Assessment of Packages for Radioactive Waste / European Reference Drum / Principles

**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
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)
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**

<table>
<thead>
<tr>
<th></th>
<th>Belgium drum</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average height [mm]</td>
<td>856 ± 38</td>
</tr>
<tr>
<td>Average diameter [mm]</td>
<td>606 ± 28</td>
</tr>
</tbody>
</table>

**Internal dimensions**

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Average height [mm]</td>
<td>835 ± 24</td>
</tr>
<tr>
<td>Average diameter [mm]</td>
<td>573 ± 8</td>
</tr>
</tbody>
</table>

**Drum material**

<table>
<thead>
<tr>
<th></th>
<th>Steel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average thickness [mm]</td>
<td>1.1 ± 0.2</td>
</tr>
<tr>
<td>Average weight [kg]</td>
<td>17 ± 3.8</td>
</tr>
</tbody>
</table>

**Average thickness [mm] (880)**

**Average weight [kg] (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

<table>
<thead>
<tr>
<th>Specification</th>
<th>Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average max waste [kg] range</td>
<td>900 – 210</td>
</tr>
<tr>
<td>Average min waste [kg] range</td>
<td>350 – 15</td>
</tr>
<tr>
<td>Average max waste density [kg/m³]</td>
<td>4000 – 1200</td>
</tr>
<tr>
<td>Average min waste density [kg/m³]</td>
<td>1000 – 0</td>
</tr>
<tr>
<td>Shielding (Y / N / Blanco)</td>
<td>8 / 6 / 1</td>
</tr>
<tr>
<td>Waste matrix varies from</td>
<td>Z = 5 to Z = 14</td>
</tr>
</tbody>
</table>
Specifications of the activity content of the waste matrix

Gamma dose rate $[\mu\text{Sv/h}]$ range $5000 - 10$

Total $\beta/\gamma$ activity $[\text{Bq}]$ range $3.7 \times 10^{11} - 1 \times 10^8$

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 $[\mu Sv/h]$ range 5000 – 10
Total $\beta/\gamma$ 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

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

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


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

I  Polymer Concrete High Integrity Container

II  Determine the Material Properties

III Preliminary Analysis

1  Drop analysis

2  Penetration analysis

3  Stacking analysis

4  Handling analysis

IV Conclusion
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

<table>
<thead>
<tr>
<th></th>
<th>Volume (m³)</th>
<th>Density (ton/m³)</th>
<th>Weight (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Body</td>
<td>275</td>
<td>2.3</td>
<td>633</td>
</tr>
<tr>
<td>Lid</td>
<td>113</td>
<td>2.3</td>
<td>260</td>
</tr>
<tr>
<td>Wastes</td>
<td>1,096</td>
<td>1</td>
<td>1,096</td>
</tr>
<tr>
<td>Total Weight</td>
<td></td>
<td></td>
<td>1,989</td>
</tr>
</tbody>
</table>

Modified with the round corner with 50 mm
Modified using steel
## Reinforced concept using in-plate steel

<table>
<thead>
<tr>
<th></th>
<th>Volume (m³)</th>
<th>Density (ton/m³)</th>
<th>Weight (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Body</td>
<td>221.2</td>
<td>2.3</td>
<td>508.8</td>
</tr>
<tr>
<td>Steel (Body)</td>
<td>53.9</td>
<td>7.8</td>
<td>420.6</td>
</tr>
<tr>
<td>Lid</td>
<td>100</td>
<td>2.3</td>
<td>230</td>
</tr>
<tr>
<td>Steel (Lid)</td>
<td>13.1</td>
<td>7.8</td>
<td>102.2</td>
</tr>
<tr>
<td>Wastes</td>
<td>1,096</td>
<td>1</td>
<td>1,096</td>
</tr>
<tr>
<td><strong>Total Weight</strong></td>
<td></td>
<td></td>
<td><strong>2,357.6</strong></td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th></th>
<th>Volume (m³)</th>
<th>Density (ton/m³)</th>
<th>weight (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Body + Lid</td>
<td>348.8</td>
<td>2.3</td>
<td>802.2</td>
</tr>
<tr>
<td>Steel Case</td>
<td>65.3</td>
<td>7.8</td>
<td>509.5</td>
</tr>
<tr>
<td>Wastes</td>
<td>1,148.7</td>
<td>1</td>
<td>1,148.7</td>
</tr>
<tr>
<td>Total Weight</td>
<td></td>
<td></td>
<td>2,460.4</td>
</tr>
</tbody>
</table>

### Reinforced concept covered with 2 mm steel plate

<table>
<thead>
<tr>
<th></th>
<th>Volume (m³)</th>
<th>Density (ton/m³)</th>
<th>weight (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Body + Lid</td>
<td>372</td>
<td>2.3</td>
<td>855.6</td>
</tr>
<tr>
<td>Steel Case</td>
<td>22.4</td>
<td>7.8</td>
<td>174.7</td>
</tr>
<tr>
<td>Wastes</td>
<td>1,148.7</td>
<td>1</td>
<td>1,148.7</td>
</tr>
<tr>
<td>Total Weight</td>
<td></td>
<td></td>
<td>2,179</td>
</tr>
</tbody>
</table>
### Reinforced concept using steel structure

<table>
<thead>
<tr>
<th></th>
<th>Volume (m³)</th>
<th>Density (ton/m³)</th>
<th>weight (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Body + Lid</td>
<td>372</td>
<td>2.3</td>
<td>855.6</td>
</tr>
<tr>
<td>Steel Case</td>
<td>25</td>
<td>7.8</td>
<td>195</td>
</tr>
<tr>
<td>Wastes</td>
<td>1,096</td>
<td>1</td>
<td>1,096</td>
</tr>
<tr>
<td><strong>Total Weight</strong></td>
<td><strong>2,146.6</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
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

<table>
<thead>
<tr>
<th>Material Properties</th>
<th>Polymer Concrete</th>
</tr>
</thead>
<tbody>
<tr>
<td>Young’s modulus (GPa)</td>
<td>21.764</td>
</tr>
<tr>
<td>Poison’s ratio</td>
<td>0.16</td>
</tr>
<tr>
<td>Compressive strength (MPa)</td>
<td>118</td>
</tr>
<tr>
<td>Tensile strength (MPa)</td>
<td>14</td>
</tr>
<tr>
<td>density (ton/m³)</td>
<td>2.3</td>
</tr>
</tbody>
</table>
**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.
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

<table>
<thead>
<tr>
<th>location</th>
<th>center</th>
<th>corner</th>
<th>max</th>
<th>min</th>
<th>max</th>
<th>min</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>3.18</td>
<td>3.18</td>
<td>3.18</td>
<td>-3.7</td>
<td>-1.29</td>
<td>-33.44</td>
</tr>
<tr>
<td>2</td>
<td>2.94</td>
<td>2.94</td>
<td>2.94</td>
<td>-3.7</td>
<td>-0.22</td>
<td>-23.27</td>
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<tr>
<td>3</td>
<td>2.48</td>
<td>2.48</td>
<td>2.48</td>
<td>-2.72</td>
<td>1.64</td>
<td>-22.44</td>
</tr>
<tr>
<td>4</td>
<td>2.27</td>
<td>2.27</td>
<td>2.27</td>
<td>-3.01</td>
<td>0.57</td>
<td>-21.27</td>
</tr>
<tr>
<td>5</td>
<td>1.87</td>
<td>1.87</td>
<td>1.87</td>
<td>-3.92</td>
<td>-1.85</td>
<td>-20.46</td>
</tr>
<tr>
<td>6</td>
<td>1.47</td>
<td>1.47</td>
<td>1.47</td>
<td>-4.84</td>
<td>-2.95</td>
<td>-19.81</td>
</tr>
</tbody>
</table>

Maximum principal stress contour for vertical drop of a basic model
## Results of drop analysis for basic model

<table>
<thead>
<tr>
<th>location</th>
<th>center</th>
<th>corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>max</td>
<td>min</td>
</tr>
<tr>
<td></td>
<td>max</td>
<td>min</td>
</tr>
<tr>
<td>1</td>
<td>8.23</td>
<td>-19.09</td>
</tr>
<tr>
<td>2</td>
<td>9.38</td>
<td>-23.44</td>
</tr>
<tr>
<td>3</td>
<td>10.92</td>
<td>-34.96</td>
</tr>
<tr>
<td>4</td>
<td>7.72</td>
<td>-52.64</td>
</tr>
<tr>
<td>5</td>
<td>-0.69</td>
<td>-87.73</td>
</tr>
<tr>
<td>6</td>
<td>-5.5</td>
<td>-112.31</td>
</tr>
</tbody>
</table>

### Principal stress for a horizontal drop

Failure is occurred at the corner

<table>
<thead>
<tr>
<th>location</th>
<th>corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>max</td>
</tr>
<tr>
<td>1</td>
<td>-2.3</td>
</tr>
<tr>
<td>2</td>
<td>9.7</td>
</tr>
<tr>
<td>3</td>
<td>29.56</td>
</tr>
<tr>
<td>4</td>
<td>45.74</td>
</tr>
<tr>
<td>5</td>
<td>200.79</td>
</tr>
<tr>
<td>6</td>
<td>-516.14</td>
</tr>
</tbody>
</table>

### 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

<table>
<thead>
<tr>
<th>Location</th>
<th>Vertical</th>
<th>Center</th>
<th>Corner</th>
<th>Center</th>
<th>Corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>max</td>
<td>min</td>
<td>max</td>
<td>min</td>
</tr>
<tr>
<td>1</td>
<td></td>
<td>-2.73</td>
<td>-8.25</td>
<td>0.38</td>
<td>-2.83</td>
</tr>
<tr>
<td>2</td>
<td></td>
<td>-2.26</td>
<td>-8.92</td>
<td>1.97</td>
<td>-39.01</td>
</tr>
<tr>
<td>3</td>
<td></td>
<td>-1.79</td>
<td>-9.59</td>
<td>0.06</td>
<td>-36.24</td>
</tr>
<tr>
<td>4</td>
<td></td>
<td>41.88</td>
<td>33.98</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5</td>
<td></td>
<td>39.95</td>
<td>229.64</td>
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</tr>
<tr>
<td>6</td>
<td></td>
<td>38.03</td>
<td>227.26</td>
<td></td>
<td></td>
</tr>
<tr>
<td>7</td>
<td></td>
<td>-1.23</td>
<td>-11.82</td>
<td>10.8</td>
<td>-240.78</td>
</tr>
<tr>
<td>8</td>
<td></td>
<td>-1.2</td>
<td>-11</td>
<td>4.35</td>
<td>-121.17</td>
</tr>
<tr>
<td>9</td>
<td></td>
<td>-1.17</td>
<td>-10.19</td>
<td>5.42</td>
<td>-119.21</td>
</tr>
</tbody>
</table>

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

#### Principal stress for a horizontal drop

<table>
<thead>
<tr>
<th>Location</th>
<th>Center</th>
<th>Corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Max</td>
<td>Min</td>
</tr>
<tr>
<td>1</td>
<td>5.38</td>
<td>-15.27</td>
</tr>
<tr>
<td>2</td>
<td>9.93</td>
<td>-20.38</td>
</tr>
<tr>
<td>3</td>
<td>10.62</td>
<td>-22.09</td>
</tr>
<tr>
<td>4</td>
<td>228.88</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>162.362</td>
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</tr>
<tr>
<td>6</td>
<td>95.84</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>-0.1</td>
<td>-101.15</td>
</tr>
<tr>
<td>8</td>
<td>-14.64</td>
<td>-151.15</td>
</tr>
<tr>
<td>9</td>
<td>-29.19</td>
<td>-201.16</td>
</tr>
</tbody>
</table>

#### Principal stress for a corner drop

<table>
<thead>
<tr>
<th>Location</th>
<th>Corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Max</td>
</tr>
<tr>
<td>1</td>
<td>3.4</td>
</tr>
<tr>
<td>2</td>
<td>-9.76</td>
</tr>
<tr>
<td>3</td>
<td>-9.76</td>
</tr>
<tr>
<td>4</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td></td>
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<td></td>
</tr>
<tr>
<td>7</td>
<td>-1185</td>
</tr>
<tr>
<td>8</td>
<td>-476.81</td>
</tr>
<tr>
<td>9</td>
<td>-246.21</td>
</tr>
</tbody>
</table>

Failures is occurred
## Reinforced concept covered with steel plate

### Reinforced model covered with 5 mm steel plate

<table>
<thead>
<tr>
<th>location</th>
<th>vertical</th>
<th>corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>center</td>
<td>corner</td>
</tr>
<tr>
<td></td>
<td>max</td>
<td>min</td>
</tr>
<tr>
<td>1</td>
<td>-0.52</td>
<td>-14.8</td>
</tr>
<tr>
<td>2</td>
<td>-1.8</td>
<td>-7.01</td>
</tr>
<tr>
<td>3</td>
<td>0.44</td>
<td>-2.54</td>
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<tr>
<td>4</td>
<td>88.75</td>
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<td>5</td>
<td>95.77</td>
<td>29</td>
</tr>
<tr>
<td>6</td>
<td>101.8</td>
<td>36</td>
</tr>
</tbody>
</table>

### Principal stress for a vertical drop

<table>
<thead>
<tr>
<th>location</th>
<th>horizontal</th>
<th>corner</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>center</td>
<td>corner</td>
</tr>
<tr>
<td></td>
<td>max</td>
<td>min</td>
</tr>
<tr>
<td>1</td>
<td>0.6</td>
<td>-4.1</td>
</tr>
<tr>
<td>2</td>
<td>1.1</td>
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<td>-6.75</td>
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<tr>
<td>4</td>
<td>20.1</td>
<td>204</td>
</tr>
<tr>
<td>5</td>
<td>21.4</td>
<td>282</td>
</tr>
<tr>
<td>6</td>
<td>22</td>
<td>307</td>
</tr>
</tbody>
</table>

### Principal stress for a horizontal drop
Reinforced concept covered with steel plate

<table>
<thead>
<tr>
<th>location</th>
<th>corner</th>
<th>max</th>
<th>min</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>corner</td>
<td>-0.6</td>
<td>-19.1</td>
</tr>
<tr>
<td>2</td>
<td></td>
<td>1.68</td>
<td>-12.03</td>
</tr>
<tr>
<td>3</td>
<td></td>
<td>3.1</td>
<td>-24.38</td>
</tr>
<tr>
<td>4</td>
<td></td>
<td>97.45</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td></td>
<td>171.41</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td></td>
<td>197.66</td>
<td></td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Location</th>
<th>Vertical</th>
<th></th>
<th></th>
<th></th>
<th></th>
<th>Horizontal</th>
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<td>corner max</td>
<td>corner min</td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
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<td>0.02</td>
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**Principal stress for a vertical drop**

**Principal stress for a horizontal drop**
### Reinforced concept covered with steel plate

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<thead>
<tr>
<th>location</th>
<th>corner</th>
<th>max</th>
<th>min</th>
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<tr>
<td>8</td>
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</tbody>
</table>

The outer steel is yielded.

---

Principal stress for a corner drop

Korea Atomic Energy Research Institute 17

Spent Fuel Management Technology Research Division
Stresses at the polymer concrete is very low
The steel structure is absorbing almost of the drop impact energy
Reinforced concept using steel structure

- a horizontal 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
Result of penetration analysis

<table>
<thead>
<tr>
<th>Principal stress (MPa)</th>
<th>center</th>
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<tr>
<td></td>
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<td>1</td>
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<tr>
<td>4</td>
<td>2.6</td>
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Maximum principal stress

Minimum principal stress

The polymer concrete is maintaining the structural integrity.
### Stacking analysis

#### Maximum principal stress

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<tr>
<th></th>
<th>Lid</th>
<th>Wall</th>
<th>Bottom</th>
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</thead>
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<td>Max. Stress (MPa)</td>
<td>0.15</td>
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<td>0.01</td>
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<td>Min. Stress (MPa)</td>
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<td>Evaluation</td>
<td>Safe</td>
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#### Minimum principal stress

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<th>Lid</th>
<th>Wall</th>
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<tbody>
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<td>Min. Stress (MPa)</td>
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<td>-0.05</td>
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<tr>
<td>Evaluation</td>
<td>Safe</td>
<td>Safe</td>
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</table>
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.

<table>
<thead>
<tr>
<th>Steel Ring Diameter (mm)</th>
<th>10</th>
<th>15</th>
<th>20</th>
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<tr>
<td>Max. Stress at the gripper (MPa)</td>
<td>85.6</td>
<td>33</td>
<td>24</td>
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</table>

| Results | Safe | Safe | Safe |

The diameter of gripper are 10 mm, 15 mm, 20 mm. At the gripper the maximum stress are lower than the yield stress.
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
2. Requirements
3. Reference Cask
4. Test Cask
5. Steel Bar Deformation
6. Measurements
7. Test Performance
8. Test Results
9. Conclusions
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 cannot 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°
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

Versuch R1 - Aufprall unberippter Bereich
- Ausgangsposition

Deformed Steel Bar    Impact Area at the Cask

Fallhöhe 1006 mm
Dorn
Kraftmessdose
7. Test R2 – onto Finned Area

Test R1 performed on 17.05.2006

Versuch R2 - Aufprall berippter Bereich
- Ausgangsposition

Fallhöhe 1005 mm

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

Effect of the Fins
* Penetration force was 4.45 MN lower (6.3 %)
* Impact duration was 5 ms (from 19 to 24 ms) longer
Effect of the Fins
* max. strain is significantly 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 %
EN-TRAP

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

Pierre Van Iseghem

SCK•CEN

Workshop on Radioactive Waste Packages, Petten 21-22 November, 2006
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
Organisation of EN-TRAP

Steering Committee

- WGA: Non-destructive Methods
- WGB: Destructive Methods
- WGC: QA and QC
- WGD: ILW / HLW

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

Co60 Comparison(Ref.) for drums 4 5 6 7 10 12 14
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.
R&D Projects Developed in the Framework of EU Programmes

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

** Only \(^{90}\)Sr

(1) Only resin; (2) Foreseen but finally not performed
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
• 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
German support for utilization of nuclear submarines in Russia

Progress in project execution
Project Organization and Controlling


• 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


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

Work on creating a level storage platform

Preparation of grounds—detonation, stone and surface removal

Excavation and removal of parts of western cliff slope
Construction site in November 2005

- Construction of concrete storage platform
- Construction of landing pier for floating dock
- Physical protection
Long-term interim storage facility at the beginning of July 2006

Landing pier

Mooring post
Initial section
Auxiliary facilities near the storage plate

- Sewage pump station
- Biological treatment plant
- Reservoir of rainfall
- Sand areas
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

30.06.2006
Former costs LTSF: 73 Mio EUR
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
Reactor compartments at the final place on storage area of LTSF
Initial section of the LTSF in operation
LTSF Sayda

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
RWSF Sayda

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!

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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

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

How to obtain? and How to explain?

- Technical
- Theoretical

Quality Issues
Example: Welding test, EBW (Electron Beam Welding)

100mm full penetration test for carbon steel overpack lid structure

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

Design requirements

Design conditions

Material selection

Vitrified waste
Overpack
Geological formation

Basic structure

External Pressure
Corrosion
Rad. Shield

Wall Thickness (dimension)

Structure

Material
Manufacturing

Long-term integrity/final closure

(1) Maintaining mechanical strength layer
(2) No crack extension by earthquake

Service level
Operation/Final disposal
Fracture mode
Class
Calculation formula
Analysis method

Welding
- weld joint
- weldability

NDE
- Sizing
- Evaluation

Structure / appropriateness
Structure / Safety

Allowable stress, Safety factor, etc.
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

<table>
<thead>
<tr>
<th>Prerequisite No.</th>
<th>Items</th>
<th>Current Situation</th>
<th>Future work for getting more reliability</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>View point</td>
</tr>
<tr>
<td>1</td>
<td>General Corrosion</td>
<td></td>
<td>Future work</td>
</tr>
<tr>
<td></td>
<td>Most Fundamental Property</td>
<td></td>
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<tr>
<td></td>
<td>Influence of Filler Metal</td>
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<tr>
<td></td>
<td>• Effect of filler metal on</td>
<td></td>
<td></td>
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<td></td>
<td>the corrosion property</td>
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<tr>
<td></td>
<td>of the weld metal</td>
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<tr>
<td></td>
<td>• Design of filler metal</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>(chemical composition)</td>
<td></td>
<td></td>
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<tr>
<td>2</td>
<td>Susceptibility to SCC</td>
<td></td>
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<tr>
<td></td>
<td>• No effect of welding on the</td>
<td></td>
<td>Chemical environment</td>
</tr>
<tr>
<td></td>
<td>susceptibility</td>
<td></td>
<td>• Groundwater composition</td>
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<tr>
<td></td>
<td>• Appropriateness of evaluation</td>
<td></td>
<td>• Low carbonate concentration</td>
</tr>
<tr>
<td></td>
<td>method (SSRT)</td>
<td></td>
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<tr>
<td></td>
<td>• Appropriateness of evaluation</td>
<td></td>
<td>Loading condition</td>
</tr>
<tr>
<td></td>
<td>time</td>
<td></td>
<td>• Constant loading</td>
</tr>
<tr>
<td></td>
<td>• Loading condition</td>
<td></td>
<td>Test period (Loading time)</td>
</tr>
<tr>
<td>3</td>
<td>Neutron Radiation Embrittlement</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>• Used estimation method for</td>
<td></td>
<td>Prediction method</td>
</tr>
<tr>
<td></td>
<td>nuclear power plant component</td>
<td></td>
<td>Low &amp; long-term radiation</td>
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<tr>
<td></td>
<td>(high radiation / short time)</td>
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<td>4</td>
<td>Fracture Toughness</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>• Used low carbon steel for</td>
<td></td>
<td>Selected carbon steel</td>
</tr>
<tr>
<td></td>
<td>nuclear power plant component</td>
<td></td>
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<tr>
<td></td>
<td>• Available data for a geological formation on the estimation of critical crack length</td>
<td>More realistic estimation</td>
<td></td>
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<tr>
<td></td>
<td>• LFM / NLFM</td>
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<tr>
<td></td>
<td>• Fracture mode</td>
<td></td>
<td>External Pressure</td>
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<td></td>
<td>• Earthquake</td>
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<td></td>
<td>• Max. Loading</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>• Fatigue fracture</td>
<td></td>
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<tr>
<td>5</td>
<td>Mechanical Buffering Property</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>of Buffer Material</td>
<td></td>
<td>Maintaining the mechanical buffering property</td>
</tr>
<tr>
<td></td>
<td>Property of Buffer Material</td>
<td></td>
<td>• Saturation, heat effect, stress condition, chemical reaction</td>
</tr>
<tr>
<td></td>
<td>• Available data for a buffer</td>
<td></td>
<td>• Etc.</td>
</tr>
<tr>
<td></td>
<td>material</td>
<td></td>
<td></td>
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<tr>
<td>6</td>
<td>Safety Factor</td>
<td></td>
<td>Estimation basis</td>
</tr>
<tr>
<td></td>
<td>• Referring from FFS of nuclear</td>
<td></td>
<td>Appropriate value as a disposal container</td>
</tr>
<tr>
<td></td>
<td>power plant component</td>
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</table>
END
Input to Discussion on collaboration

Workshop on Design and Assessment of Packages for Radioactive Waste

K-F Nilsson
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 fora such as IAEA, NEA/OECD,
- JRC-IE is willing to provide the management of such a network.
<table>
<thead>
<tr>
<th>Year</th>
<th>FP6 2002 - 2006</th>
<th>(FP7 2007 - 2011)</th>
</tr>
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<tbody>
<tr>
<td>2003</td>
<td>ACTINET</td>
<td>EURADWASTE '08 Conference</td>
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<tr>
<td>2004</td>
<td>NF-PRO</td>
<td>EURADWASTE '08 Conference</td>
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<td>2005</td>
<td>FUNMIG</td>
<td>Demontrator</td>
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<tr>
<td>2006</td>
<td>ESDRED</td>
<td>Regional repositories</td>
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<tr>
<td>2007</td>
<td>COWAM-2</td>
<td>Technology transfer (national rep./encaps)</td>
</tr>
<tr>
<td>2008</td>
<td>SAPIERR</td>
<td>Performance Assessment</td>
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<tr>
<td>2009</td>
<td>CETRAD</td>
<td>PAMINA</td>
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<tr>
<td>2010</td>
<td>CATT</td>
<td>TIMODAZ</td>
</tr>
</tbody>
</table>

**Near-field**

**Regional repositories**

**Technology transfer (national rep./encaps)**

**Performance Assessment**
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


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 wish 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 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 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.
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 Matrix of operational and generic technical assessment areas discussed for collaboration.**

<table>
<thead>
<tr>
<th></th>
<th>Repository operation (handling, remote, welding, monitoring…..) Package closure operation, Condition monitoring</th>
<th>Storage operation (handling, remote, monitoring, higher burn-up…..) Condition, monitoring Initial characterization</th>
<th>Transport operation (handling, remote, monitoring ..) Acceptance criteria Initial characterization</th>
</tr>
</thead>
<tbody>
<tr>
<td>NDE and welding for fabrication &amp; closure</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Flaw assessment</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Long-term degradation processes a) Microstructural changes b) corrosion</td>
<td>X</td>
<td>(X)</td>
<td>-</td>
</tr>
<tr>
<td>Waste/SNF characterization and degradation</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Benchmark-type experiments</td>
<td>-</td>
<td>X</td>
<td>X</td>
</tr>
</tbody>
</table>
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 fora 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 (1 day) 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 indicate 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

