3-8 October 2010

International
Maritime Organization
4, Albert Embankment
London UK

Looking to the future

PATRAM 2010

PROGRAMME

The 16th International
Symposium on the Packaging
and Transportation of Radioactive
Materials

Hosted by the UK DfT in cooperation with the IAEA, IMO and WNTI.

Department for Transport

IAEA
International Atomic Energy Agency
Atoms for Peace

IMO

WNTI
World Nuclear Transport Institute

3-8 October 2010

International
Maritime Organization
4, Albert Embankment
London UK SE1 7SR

Looking to the future

PATRAM 2010

The 16th International Symposium on the Packaging and Transportation of Radioactive Materials

Welcome to PATRAM 2010 - Looking to the Future

PATRAM, the International Symposium on Packaging and Transport of Radioactive Materials, which takes place every three years, is the foremost event devoted exclusively to this theme. **PATRAM 2010**, the 16th International Symposium in the series, will be held from 3 – 8 October 2010 in London, UK, at the Headquarters of the International Maritime Organization.

It is being hosted by the Department for Transport of the United Kingdom, in cooperation with the International Atomic Energy Agency, the International Maritime Organization and the World Nuclear Transport Institute.

A period of significant growth in industries based on radioactive materials beckons. Nuclear power is expanding to meet the world's needs for secure, sustainable and affordable energy. In the medical field there are increasing demands for supplies of radiopharmaceuticals and sources for diagnosis, treatment and sterilisation, not least in developing countries. Demands for radioactive materials for many industrial applications also are increasing. A reliable transport infrastructure is essential to support these needs.

The dynamic vision for **PATRAM 2010** is Looking to the Future – designed to cast a special light on the new challenges, not only those of growth but also other issues, such as decommissioning, waste handling and security, which the stakeholders in all sectors of radioactive material packaging and transport will have to face to ensure that the infrastructure can be maintained safely, securely and efficiently in a changing world.

PATRAM 2010 will bring together those concerned with regulation, research and engineering, safety, security and communications from international and national governmental and other organisations, as well as the service providers in the several sectors of industry for all modes of transport. The objective is to enable stakeholders to engage and to share their knowledge and experience and to debate topical issues in order to explore how the future challenges can best be met, building on the sound foundations that have assured safety and security for several decades.

PATRAM 2010 promises to be a highly relevant and stimulating gathering of stakeholders committed to the safe, effective and reliable transport of radioactive materials, and we look forward to welcoming you to London in October 2010.

Jeff Hart
UK Department for Transport

Tomihiro Taniguchi
International Atomic Energy Agency

Koji Sekimizu
International Maritime Organization

Lorne Green
World Nuclear Transport Institute

Contents

Organising Committees	06
General Information	08
Registration	08
Badges	08
Catering	08
Dress Code	11
Emergency	11
Gala Dinner	11
Language	11
Meeting Rooms	11
Message Board	11
Proceedings	11
Poster Session	11
Rapporteurs	11
Secretariat	12
Speaker Preparation Room	12
Speakers and Session Chairs Breakfast	12
Telephones	12
Tourist Information Desk	12
Welcome Reception	12
WIFI and Internet Access	12
Floor Plans	13
Sponsors, Exhibitors & Advertisers	14
Speaker Biographies	21
Programme at a glance	25
Daily Programme	
Sunday 03 October	27
Monday 04 October	27
Tuesday 05 October	30
Wednesday 06 October	38
Thursday 07 October	43
Friday 08 October	50
Abstracts Monday	
Concurrent Technical Sessions	52
Abstracts Tuesday	
Concurrent Technical Sessions	65
Concurrent Panel Sessions	78
Concurrent Technical Sessions	85
Abstracts Wednesday	
Concurrent Technical Sessions	94
Concurrent Panel Sessions	107
Abstracts Thursday	
Concurrent Technical Sessions	113
Concurrent Panel & Technical Sessions	126
Concurrent Technical Sessions	131
Abstracts Friday	
Concurrent Technical Sessions	138
Posters Session	144
IMO Security Information	174

Organisation

Steering Group

Jeff Hart,
UK Department for Transport, Chair
Betty Bonnardel-Azzarelli,
World Nuclear Transport Institute (WNTI)
Melvin Draper,
UK Department for Business Enterprise and Regulatory Reform
Lorne Green,
World Nuclear Transport Institute (WNTI)
Malcolm Miller,
Consultant
Irfan Rahim,
International Maritime Organization (IMO)
George Sallit,
UK Department for Transport
Jim Stewart,
International Atomic Energy Agency (IAEA)
William Wilkinson,
Consultant

Project Board: Programme

George Sallit,
UK Department for Transport, Chair
Lorne Green,
World Nuclear Transport Institute (WNTI), Deputy Chair
Charles Carrington,
GE Healthcare
Serge Gorlin,
World Nuclear Association (WNA)
Bernd Lorenz,
Gesellshaft fur Nuklear-Service mbH (GNS)
Pierre Malesys,
AREVA
Makoto Hirose,
Nuclear Fuel Transport Co (NFT)
Frank Nitsche,
German Federal Office for Radiation Protection (BfS)
Toshiari Saegusa,
Central Research Institute of Electric Power Industry (CRIEPI)
Ken Sorenson,
Sandia National Laboratories (SNL)
Jim Stewart,
International Atomic Energy Agency (IAEA)
William Wilkinson,
Consultant

Paper Review Committee

Pierre Malesys,
AREVA, Chair
Clive Young,
Consultant, Co-Chair
Doug Ammerman,
Sandia National Laboratory
Ryoji Asano,
Hitachi Zosen Corporation
Elefteria Askitoglu,
Swiss Federal Nuclear Safety Inspectorate (ENSI)
Steve Bellamy,
Savanah River National Laboratory (SRNL)
Betty Bonnardel-Azzarrelli,
World Nuclear Transport Institute
Yves Chanzy,
AREVA
Marc-Andre Charette,
CAMECO Corporation
Michael Conroy,
US Department of Transportation
Erik Cottens,
Federal Agency for Nuclear Control (FANC)
Malcolm Critchley,
ConverDyn
Sam Darby,
Sellafield Ltd
Bernhard Droste,
Bundesanstalt fur materialforschung und-prufung (BAM)
Earl Easton,
US Nuclear Regulatory Commission (NRC)
Ann-Margareth Erikkson,
International Atomic Energy Agency (IAEA)
Sylvain Faille,
Canadian Nuclear Safety Commission (CNSC)
Christel Fasten,
Bundesamt fur Strahlenschutz (BfS)
Serge Gorlin,
World Nuclear Association (WNA)
Michel Hartenstein,
TN International
Franz Hilbert,
Nuclear Cargo & Service GmbH (NCS)
Makoto Hirose,
Nuclear Fuel Transport Co Ltd, (NFT)
Roland Huggenberg,
GNS Gesellschaft fur Nuklear -Service mbH
Yung Liu,
Argonne National Laboratory (ANL)
Marie-Therese Lizot,
Autorite de Sûreté Nucleaire (ASN)
Carlos Lopez,
Sandia National Laboratory (SNL)
Laurent Milet,
TN International

Organisation : continued

David Miller,
Sandia National Laboratory (SNL)
Arungunram Nandakumar,
Consultant
Tara Neider,
Transnuclear Inc.
Naoteru Odano,
National Maritime Research Institute
Alberto Orsini,
Agenzia nazionale per le nuove tecnologie,
l'energia e lo sviluppo economico sostenibile (ENEA)
Peter Purcell,
International Nuclear Services
Bryan Reeves,
UK Office for Civil Nuclear Security (OCNS)
Johan Rosenblad,
Svensk Karnbranslehantering AB (SKB)
- (Swedish Nuclear Fuel & Waste Management Co)
Toshiari Saegusa,
Central Research Institute of Electric Power Industry (CRIEPI)
Gilles Sert,
Institut de Radioprotection et de sûreté Nucléaire (IRSN)
James Shuler,
US Department of Energy (DOE)
Ken Sorenson,
Sandia National Laboratory (SNL)
Hiroaki Taniuchi,
TNT
Chi-Fung Tso,
ARUP
Gheorghe Vieru,
Institute for Nuclear Research Pitesti
Danny Vince,
UK Department for Transport
Mike Wangler,
US Department of Energy (DOE)
Steve Whittingham,
UK Department for Transport
William Wilkinson,
Consultant
Frank Wille,
Bundesanstalt fur Materialforschung und-prufung (BAM)
Fernando Zamora,
Consejo de Securidad Nuclear (CSN)
Helmuth Zika,
Swedish Radiation Safety Authority (SRSA)

Project Board: Logistics and Accommodation

Betty Bonnardel-Azzarelli,
World Nuclear Transport Institute (WNTI), Chair
Steve Whittingham,
UK Department for Transport, Deputy Chair
Bernhard Droste,
German Federal Institute for Materials Research
and Testing (BAM)
Gerard Holden,
Consultant
Malcolm Miller,
Consultant
Chi-Fung Tso,
ARUP

Industry Liaison Committee

Lorne Green,
World Nuclear Transport Institute (WNTI), Chair
Betty Bonnardel-Azzarelli,
World Nuclear Transport Institute (WNTI)
Charlie Carrington,
GE Healthcare
Serge Gorlin,
World Nuclear Association (WNA)
Richard Ivens,
FORATOM
Felix Killar,
Nuclear Energy Institute (NEI)
Keith Parker,
UK Nuclear Industry Association (NiA)
Ken Sorenson,
Institute of Nuclear Materials Management (INMM)
Steve Whittingham,
UK Department for Transport

Awards Committee

David Miller,
Sandia Laboratories, Chair
Ann-Margareth Erikkson,
International Atomic Energy Agency (IAEA)
Michel Hartenstein,
TN International
Garry Owen,
International Nuclear Services
Ron Pope,
Consultant
Annette Rolle,
Federal Institute for Materials Research and Testing (BAM)
Kohi Shirai,
Central Research Institute of Electric Power Industry (CRIEPI)
Clive Young,
Consultant
Steve Whittingham,
UK Department for Transport
William Wilkinson,
Consultant

General Information

Registration
Participation in PATRAM 2010 is limited to registered delegates.

The full registration includes the following:
- Invitation to attend the Welcome Reception sponsored by the World Nuclear Transport Institute (Sunday 03 October, not available to day registrants)
- Admission to all sessions (Monday 04 to Friday 08 October)
- Access to the PATRAM 2010 Exhibition
- Entry to view the PATRAM 2010 Poster Session (Wednesday 06 October)
- Daily refreshment breaks (this does not include lunch)
- Invitation to attend the PATRAM 2010 Gala Dinner, sponsored by International Nuclear Services (Thursday 07 October, not available to day registrants)
- Delegate materials (delegate bag and contents, Final Programme)
- CD of the PATRAM 2010 Proceedings (available post event, not available to day registrants)

It is also possible to register as a day delegate or an accompanying person.

Registration desks are located in the IMO Main Foyer and are open at these times:

Day	**Opening Times**
Sunday 03 October	1:00pm – 7:30pm
Monday 04 October	7:00am – 6:00pm
Tuesday 05 October	7:00am – 6:00pm
Wednesday 06 October	7:00am – 6:00pm
Thursday 07 October	7:00am – 5:30pm
Friday 08 October	7:00am – 12:00pm

Badges
All delegates are issued with two badges, an IMO security badge to facilitate access into the IMO building and a PATRAM 2010 conference badge which facilitates access in to the PATRAM 2010 conference sessions, Welcome Reception and Gala Dinner. In order to comply with security arrangements for the Symposium, it is essential that delegates and accompanying guests wear their badges at all times.

Please note that lost or misplaced security badges must be reported to the Patram 2010 Onsite Registration desk immediately. A charge will be made to issue replacement badges.

Aoki Awards
The Aoki awards are intended to further develop the PATRAM series of symposia by rewarding those who have contributed most to the meetings and commending those who have presented specific distinguished papers.

PATRAM 2010 participants will participate in choosing authors for the Aoki Awards and forms will be available in each session room to enable audience members to provide feedback on their preferred technical presentation, panel presentation and/or poster display. Session chairs will invite audience members at the beginning and end of their technical or panel sessions to complete a form and either hand it to them, or deposit the completed form in a box in the PATRAM 2010 Secretariat office. The PATRAM 2010 Awards Committee will arrive at its conclusions based on comments received.

The awards will be presented at the Gala Dinner; but as some presentations will follow the Dinner at sessions on Friday 08 October, the Awards Committee will only be able to complete its deliberations thereafter, so that some Awards will be presented at the following PATRAM in 2013.

Catering
Refreshment breaks throughout the Symposium are located on the Delegate Lounge, adjacent to the main exhibition area. Delegates may take lunch within the IMO building in any of the following areas:

Food Outlet	**Location**	**Opening Times**
Restaurant		
Serving selection of hot and cold seasonal dishes	4th Floor, IMO	8:00am – 4:15pm
Coffee Bar		
Serving a selection of beverages, sandwiches and light snacks	4th Floor IMO	8:00am – 4:15pm
Ploughmans Bar		
Serving a selection of boxed lunches to go, containing a drink and light snack	Delegate Lounge	12:30pm – 2:00pm
Express Bar		
Serving selected dishes from the main restaurant at a fixed price	Private Dining Room, 4th Floor, IMO	12:30pm – 2:00pm

General Information : continued

There are also a number of catering establishments located close to the IMO building:

PUB / RESTAURANT:
Tamesis, Albert Embankment,
– 200 metres from the IMO
London, SE1 7TP, UK
(on the river between Vauxhall and Lambeth bridge)
Tel: 0207 582 1066 or 07843 264368
Tamesis, is a converted, loveable, 1933 Dutch barge, now used as a bar-restaurant. Moored between Lambeth and Vauxhall Bridge, Tamesis offers spectacular views of the Houses of Parliament and London Eye. With generous indoor and outdoor space, Tamesis offers a unique, friendly and relaxed environment in which to eat, drink or host a function. Rated by the London Paper as one of London's top 10 floating restaurants and by the Evening Standard as one of the places in London to be seen sipping a cocktail on the terrace.

PUB:
Zeitgeist at the Jolly Gardeners
– 231 metres from the IMO
49-51 Black Prince Road, London SE11 6AB
Tel: 020 7840 0426
Having established themselves as London's first German gastropub, the Rhineland couple in charge of Zeitgeist have wisely decided not to change a thing. With 100 covers and a beer garden at the back, there's plenty of space to wind down and sample something hoppy or grapey. Thirteen draughts and 30 bottled beers include all the big names (Erdinger Weiss, Tannenzäpfle Weizen, Weihenstephan Dunkel et al) and the compact German wine list is bigger on whites – Riesling and Grüner Veltliner – than reds. Hearty dishes appeal to the Northern European urge for sausages & sauerkraut, schnitzels with chips, dumplings, herrings & high-protein grills.

RESTAURANT:
Chino Latino at the Riverbank Park Plaza Hotel
– 285 metres from the IMO
18 Albert Embankment, London SE1 7TJ
Tel: 020 7769 2500
Chino Latino displays a satisfying individuality born of having stand-alone sister restaurants in Nottingham, Leeds and Cologne. The modern pan-Asian menu features sushi, sashimi, dim sum and small plates (anything from seared beef carpaccio with soy, mirin and sesame oil to marinated tofu four ways), alongside deservedly popular bestsellers such as char-grilled sirloin steak on hot rocks with wasabi ponzu sauce, plus newer dishes including tender belly pork with spiced apple chutney. Three-part saké pannacotta is a typically showy dessert, and bento boxes feed the lunchtime crowd. 'Divine food and even better cocktails', notes one admirer, referring to the heady Asian-slanted mixes. Drinkers also have a good choice of oriental beers and international wines.

PUB:
The Rose
– 292 meters from the IMO
35, Albert Embankment, London, SE1 7TL
Tel: 020 7735 3723
The Rose Pub is situated on the River Thames embankment overlooking Parliament. A great opportunity for you to eat good quality modern English and European cuisine at great value is available in the first floor dinning room. And a wine selection to complement. Alternatively the ground floor bar mixes the traditional pub feel with contemporary elements, dispensing a selection of best lagers, ales alongside a complementary selection of classic and signature cocktails.

CAFÉ:
The Garden Café at the Museum of Gardening History
– 314 meters from the IMO
Lambeth Palace Road, London SE1 7LB
Tel: 020 7401 8865
Serves fresh vegetarian cuisine and delicious homebaked cakes. There are 40 seats inside the café and space for another 20 in the courtyard of the 17th century style garden behind the museum, and of course people are welcome to spread out into the rest of the garden wherever they think looks comfortable. The Garden Cafe is closed on the first Monday of each month.

BAR:
Millbank Lounge
– 328 metres from the IMO (across the river)
30 John Islip Street, London SW1P 4DD
Tel: 020 7932 4700
As sanguine as the Bloody Mary that heads the drinks list, the cocktail lounge on the first floor of the City Inn hotel takes no prisoners with its assertive red colour scheme. A recent refurbishment has introduced additional low-slung sofas (red, of course), further opening up the trendy space. During the day, bring your laptop and take advantage of the free Wi-Fi. After dark, when the impressive bar is lit up by scarlet neon, it's a stylish destination for cocktails ranging from classic Manhattans & Mojitos to house concoctions such as wild grass martini (Zubrówka Bison Grass vodka, muddled apple, lemongrass, fresh ginger, brown sugar & freshly pressed apple juice). If cocktails aren't your thing, peruse the mouth-wateringly long selection of whiskies and the concise

General Information : continued

wine list. Bar snacks include salads, sandwiches and a dish of the day, along with ever-popular Millbank Lounge burgers.

RESTAURANT:
City Cafe at the City Inn Hotel
– 328 metres from the IMO (across the river)
30 John Islip Street, London SW1P 4DD
Tel: 020 7932 4600
The City Café scores with its considered cooking, appealing prices and briskly efficient young staff. The kitchen makes the most of seasonal British ingredients and adds a touch of innovation without trying to be too fancy. Local business folk come here at lunchtime for the twice-weekly changing market menus (only £9.95 for two courses), but hotel residents account for most of the evening trade. High-backed banquettes, pillars and curtains suit those looking for some privacy in the spacious dining room, while the covered terrace provides more gregarious alfresco opportunities.

RESTAURANT:
Pizza Express Millbank
– 388 metres from the IMO (across the river)
25 Millbank, London SW1P 4QP
Tel: 020 7976 6214
It's hard to remember a time when Pizza Express wasn't in our lives, although the chain's utter domination of the market is mainly down to its enduring menu of thin-crust pizzas. Top-end chef Theo Randall was recently offered the chance to design his own pizzas for the chain – a good example of the constant efforts to revamp the menu. Elsewhere, a decent selection of salads, antipasti and pasta (cannelloni filled with ricotta & spinach, say) are equally tempting possibilities. Classic Italian desserts and a choice of gelati are there to round things off, while the accessible wine list offers decent drinking at thoroughly affordable prices. Cheery staff and speedy service are bonus points.

RESTAURANT:
Rex Whistler Restaurant at Tate Britain
– 456 metres from the IMO (across the river)
Millbank, London SW1P 4RG
Tel: 020 7887 8825
As you might expect from a restaurant attached to Tate Britain, there's plenty to catch the eye in this visually striking room – not least a gigantic Rex Whistler mural, 'The Expedition in Pursuit of Rare Meats', which recounts an imaginary trek through a strange, surreal landscape. Art lovers, business diners and tourists are also impressed by the expertly wrought food and award-winning wine list. Despite the exotic theme of the mural, most of the kitchen's inspiration comes from close to home with a succession of unfussy, seasonally attuned lunch dishes.

PUB:
The Black Dog
– 478 metres from the IMO
112 Vauxhall Walk, London SE11 5ER
Tel: 020 7735 4440

PUB:
The Riverside
– 849 metres from the IMO
5 St George Wharf, London SW8 2LE
Tel: 020 7735 8129
It's a perfectly agreeable boozer if you want to sit and watch the river roll by with a pint of Young's bitter in your hand; there are also some good things to eat – dressed crab, interesting salads, sausages and mash et al.

SANDWICHES:
Pret a Manger
– 849 metres from the IMO
5 St George Wharf, London SW8 2LE

RESTAURANT:
Brasserie Joël at Park Plaza
– 904 metres from the IMO
First Floor, Park Plaza Hotel, London SE1 7UT
Tel:020 7620 7272
Joël Antunes was a big hitter on the London scene in the 1990s, when he was cooking at Michelin-starred Les Saveurs. He went on to win gongs in the States, but now he's back, fronting a 200-seater in the shiny new Park Plaza on Westminster Bridge.

General Information : continued

Dress Code

The dress code for the Symposium is business attire.

Emergency

The address of the IMO is:
4 Albert Embankment
London
SE1 7SR
United Kingdom
Telephone: +44 (0) 207 735 7611
Fax: +44 (0) 207 587 3210

In the case of emergency, speakers, panellists, poster presenters, chairs and co-chairs should contact the following number:
+44 (0) 7884 246299

All other participants should contact the following number:
+44 (0) 7785 545604

A copy of the IMO Fire and Emergency Brief for Visitors is located at the back of the Final Programme. All PATRAM 2010 participants should familiarise themselves with this important information.

Gala Dinner

Royal Courts of Justice
Strand
London, WC2A 2LL

Sponsored by International Nuclear Services (INS), the PATRAM 2010 Gala Dinner promises to be a night to remember. Held at the Royal Courts of Justice, a venue steeped in history, intrigue and legend, PATRAM 2010 delegates are sure to enjoy the welcome drinks reception, dinner, the Aoki Awards Ceremony and some spectacular entertainment.

Doors to the venue open at 6:45pm and admittance is by invitation only. Please note there are strict security procedures in place and delegates should ensure they allow sufficient time to meet with the venue's security requirements. In addition, delegates are strongly advised not to take any unnecessary bags or luggage to the venue.

Dress Code: Business attire, ladies cocktail dresses

London Underground Stations: Temple, Chancery Lane, St Pauls, Holborn
Buses: Routes 4, 11, 15, 23, 26, 76, 171a

Language

The official language of the Symposium is English. There will be no simultaneous interpretation provided.

Meeting Rooms

A limited number of private meeting rooms are available within the IMO for hire throughout the duration of the Symposium. Each room is arranged boardroom style and can accommodate at least eight guests. Participants wishing to book a meeting room should contact the PATRAM 2010 Secretariat where a diary system will be in operation.
Hire fees shall be payable at the time of booking.

Message Board

Delegates may communicate with each other via the PATRAM 2010 Message Board which is located in the PATRAM 2010 Secretariat Office situated on the Delegate Lounge of the IMO building. Messages should be removed once they have been read. Please note that it will not be possible to interrupt the conference sessions to deliver personal messages unless the organisers consider the communication to be an emergency.

PATRAM 2010 Proceedings

All oral, panel and poster presenters will submit copies of their final papers and presentations for publication in the PATRAM 2010 Proceedings. The Proceedings will be available post event to all full registered delegates. They are not available to day registrants.

Poster Session

The PATRAM 2010 Poster Session will take place from 4:00pm until 6:00pm on Wednesday 06 October. Posters will be displayed in the Delegate Lounge within the main exhibition area and also in the Main Foyer of the IMO on the ground floor.

Poster presenters are invited to display their poster from 8:00 on Wednesday morning and are reminded that they should be removed by 6:30pm on Wednesday evening. A limited amount of fixing materials will be available in the PATRAM 2010 Secretariat Office.

Rapporteurs

Four Rapporteurs will be in attendance throughout the Symposium to record the activities and the outcomes of the Technical, and Panel Sessions. Their report will be delivered during the plenary session at the beginning of each day.

General Information : continued

Secretariat
The PATRAM 2010 Secretariat office is located on the Delegate Lounge adjacent to the Speaker Preparation Room. This should be the first point of call for delegates requiring assistance.

Speaker Preparation Room
The Speaker Preparation Room is located in the Delegate Lounge of the IMO, adjacent to the main exhibition area and the PATRAM 2010 Secretariat office.

Speakers and Session Chairs Breakfast
A daily breakfast meeting for speakers, session chairs and co-chairs will take place from 7:00am to 8:00am in the restaurant on the fourth floor of the IMO building.
The purpose of the meeting is to:
1. facilitate introductions;
2. hear last minute communications on changes;
3. ensure the smooth coordination of your session;
4. collect speakers presentation files and;
5. for speakers, to provide a short 50-word biography for their session chair.

Please note that only the presenter should attend.

Telephones
Public telephones are available throughout the IMO building.

As a courtesy to speakers and audience members, delegates are requested to ensure that all mobile telephones are switched off in the conference sessions.

Tourist Information Desk
A 'Tourist Information Desk' will be available in the Main Foyer of the IMO on Monday 04 October only. The desk will be situated in close proximity to the PATRAM 2010 Registration Desks and will be staffed by an official Blue Badge Guide. The Blue Badge is the highest guiding qualification in Britain and is awarded by the Institute of Tourist Guiding. Delegates and accompanying guests may use the services of the Guide to discover the exciting places to visit whilst in London and book personal or group tours of the City and surrounding areas.

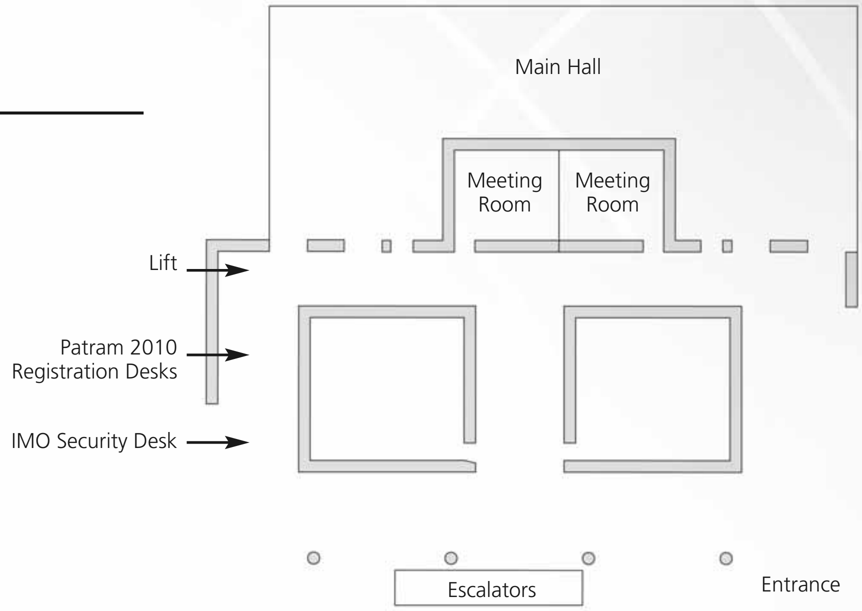
A map of London and a copy of the October *What's On in London* planner is also provided in each delegate bag.

Welcome Reception
The PATRAM 2010 Welcome Reception generously sponsored by the World Nuclear Transport Institute (WNTI) will take place on Sunday 03 October in the Delegate Lounge of the IMO. The Reception heralds the opening of the PATRAM 2010 Symposium and offers delegates and their accompanying guests an opportunity to meet the hosts and sponsors in a relaxed and informal networking environment.
Drinks and a light buffet supper will be served.

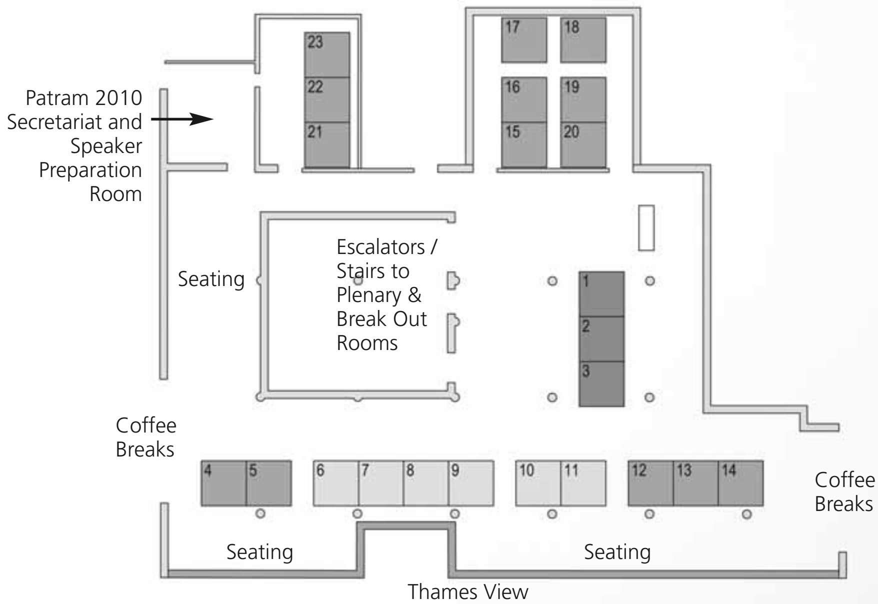
Wifi and Internet Access
Wireless internet access is available throughout the IMO building.

Floor Plans

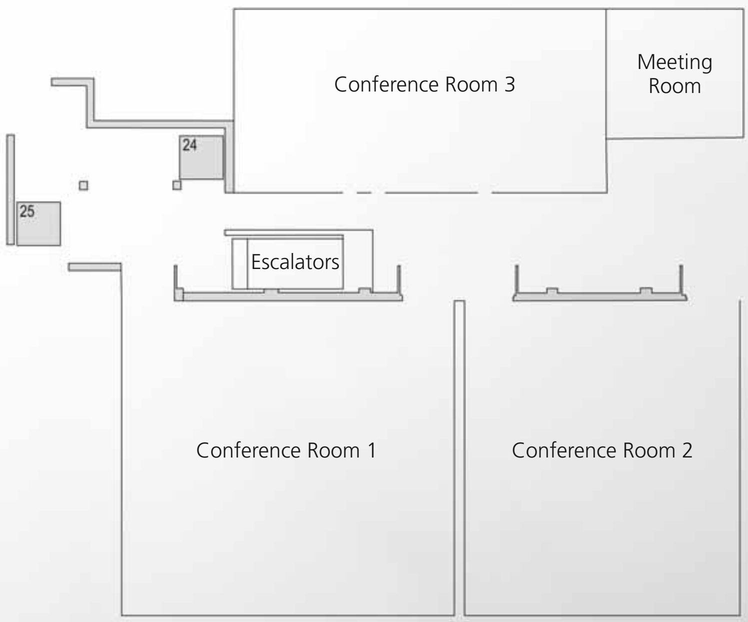
Ground Floor



1st Floor



2nd Floor



Sponsors, Advertisers & Exhibitors

The Organisers would like to gratefully acknowledge the sponsors for their generous support of PATRAM 2010:

International Nuclear Services

PATRAM 2010 Gala Dinner, Delegate Bags, and Pens

World Nuclear Transport Institute

Welcome Reception and Notepads

GNS Gesellschaft für Nuklear-Service mbH

Badges and Lanyards

Areva

Delegate Bags

Advertisers

PacTec Inc

Inside Front Cover

GNS Gesellschaft für Nuklear-Service mbH

Inside Back Cover

Exhibition Opening Times

Day	Times
Sunday 03 October	6:00pm – 7:30pm
Monday 04 October	8:00am – 6:00pm
Tuesday 05 October	8:00am – 6:00pm
Wednesday 06 October	8:00am – 6:00pm
Thursday 07 October	8:00am – 5:40pm
Friday 08 October	8:00am – 12:00pm

Exhibition Stand No	Company Name
1	DAHER CSi
2	Arup
3	GNS Gesellschaft für Nuklear-Service mbH
4	Nuclear Fuel Transport Co. Ltd
5	Ceradyne Inc
6	ROBATEL Industries
7	ONET Technologies UK Gravatom
8	Columbiana Hi Tech LLC
9	General Plastics Manufacturing Company
10, 11	International Nuclear Services
12, 13, 14	AREVA Logistics Business Unit
15	Alcan Extruded Products
16	AttentionIT Limited
17	Croft
18	Westerman Inc
19	Nippon Light Metal Co. Ltd
20	International Source Suppliers and Producers (ISSPA)
21	Quadrant EPP AG
22	European Isotopes Transport Association (EITA)
23	Panalpina World Transport (Panprojects Division)
24	Maney Publishing
25	World Nuclear Transport Institute

Sponsors, Advertisers & Exhibitors : continued

Alcan Extruded Products

Stand 15 GOLD

Aluminium extrusion partner for basket, cask and storage applications. Extrusion technology in various alloys including borated aluminium.

Alcan Extruded Products
Alusingenplatz 1
Singen, 78224
Germany
Tel: +49 7731 80 2536
Contact: Juergen Bieser
Email: juergen.bieser.alcan.com
Web: www.extrudedproducts.com

AREVA Logistics Business Unit

Stands 12, 13 & 14 GOLD

AREVA Logistics BU is present at all stages of the nuclear fuel cycle. It has 45 years of know-how and expertise to its customers' advantage and its mission is to:

- secure material and associated information, from transportation preparation to delivery, through strong risk management
- ensure transport operations for AREVA customers and suppliers as well as for other nuclear operators
- design and manufacture transport packagings
- supply dry storage solutions for electric utilities worldwide

AREVA Logistics Business Unit
1 rue des herons 78180
Montigny Le Bretonneux
France
Tel: + 33 1 34 96 50 00
Contact: Camille Otton
Email: camille.otton@areva.com
Web: www.areva.com

Arup

Stand 2 PLATINUM

Arup is an independent firm of designers, planners, engineers, consultants and technical specialists offering a broad range of professional services across a range of disciplines and sectors. As part of our work in the energy sector, Arup has developed innovative, reliable and cost effective solutions to nuclear energy projects through a combination of management, commercial and technical expertise. We provide clients with a seamless, holistic service through the entire life cycle of a nuclear energy project – development, engineering, procurement, construction and optimised operation – as well as specialist expertise during particular phases of a project as required.

Arup
13 Fitzroy Street
London W1T 4BQ
United Kingdom
Tel: +44 (0)207 755 2193
Contact: Chi-Fung Tso
Email: chi-fung.tso@arup.com
Web: www.arup.com

AttentionIT Limited

Stand 16 GOLD

AttentionIT, Limited is an Information Technology (IT) firm focused on the delivery of high quality, advanced technology, environmental software solutions. We provide database design, software development, device interfaces, and more to support our commitment as a proven environmental application software provider. Our experience includes work with Federal Sites, Commercial and Federal Generators, Processors and Disposition sites. Our environmental solutions are providing a consistent, compliant and secure mechanism for tracking, managing and reporting vital environmental information. Our solutions are also streamlining the approach to waste management via an efficient workflow process methodology that increases efficiency and accelerates the remediation process.

AttentionIT Limited
Newton Place, Suite 101
Birchwood Park
Warrington
Cheshire
WA3 8FW
United Kingdom
Tel: +44 (0)1925 320 070
Contact: Jeanice Pratt
Email: Jeanice@attentionit.com
Web: www.attentionit.com

Sponsors, Advertisers & Exhibitors : continued

Ceradyne

Stand 5 GOLD

Ceradyne nuclear products offer a full range of thermal neutron absorbing materials including BORAL® Composite, BORTEC® MMC, BorAluminum™, BoroBond™, Enriched Boron, and Boron Chemistries. From applications for criticality control in wet and dry used and fresh fuel transportation and storage, to chemical isotopes used for both nuclear waste containment and nuclear power plant neutron radiation control. Established in the early 1970s under Eaglepicher Boron and acquired by Ceradyne, Inc., Ceradyne Boron Products, LLC supplies neutron absorbing and reflecting components in materials utilizing enriched ^{10}Boron and ^{11}Boron isotopes.

Ceradyne Boron Products, LLC
798 Highway 69A
Quapaw, OK 74363
USA
Tel: +1 918-673-2201
Contact: Dennis Manning
Email: inquiry@ceradyneboron.com
Website: www.ceradyneboron.com

Ceradyne Canada, ULC
2702 Boulevard Talbot
Chicoutimi, Quebec G7H 5B1
Canada
Tel: +1 418-693-0227
Contact: Annie Villeneuve
Email: avilleneuve@ceradyne.com
Web: www.ceradyneboron.com

Columbiana Hi Tech LLC

Stand 8 SILVER

Columbiana Hi Tech, LLC (CHT) is a leading manufacturer of front-end radioactive packaging for the nuclear fuel cycle, offering high quality packages for uranium hexafluoride (UF_6), uranium dioxide (UO_2), and reprocessed uranium and fuel bundle shipments. CHT is also a major supplier to the nuclear industry in the fabrication of containers and related products to transport and store spent nuclear fuel, including lead lined dry transfer casks, canister assemblies, and specialized handling and storage equipment. CHT has a Nuclear Regulatory Commission (NRC) approved Quality Assurance Program for the design, manufacture and repair of radioactive packaging.

Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407
USA
Tel: +1 336 852 5679
Contact: Don Olson
Email: dolson@chtnuclear.com
Web: www.chtnuclear.com

Croft Associates Ltd

Stand 17 GOLD

Croft specialise in the design, testing, licensing, supply and operation of packaging for the safe transportation and disposal of radioactive and nuclear materials.

F4 Culham Science Centre,
Abingdon, Oxfordshire, OX14 3DB
United Kingdom
Tel: + 44 (0) 1865 407740
Contact: R A Vaughan
Email: sales@croftltd.com
Web: www.croftltd.com

DAHER CSi

Stand 1 PLATINUM

DAHER is a European integrated equipment and services supplier. DAHER specialises in the Aerospace, Nuclear, Defence and Industry sectors, concentrating on three core activities: manufacturing, services and transport. Founded in 1863, DAHER is an independent international group with sites in 12 countries. DAHER has tripled in size over the last six years to reach an annual turnover of Euros 744 million in 2009. In the Nuclear industry, DAHER specialises in the fuel life cycle for power plants and research reactors. Thanks to the integration of NCS and TLI into the Group, DAHER is today one of the unquestioned leaders for packaging, transportation and logistics over the whole nuclear fuel cycle.

DAHER CSi
Orlytech – Bât 528
1, Alleé Maryse Batisté

Sponsors, Advertisers & Exhibitors : continued

F-91325 Wissous
Cedex
France
Tel: +33 (0) 149 75 98 55
Contact: Johny Ledys
Email: j.ledys@daher.com
Web: www.daher.com

European Isotopes Transport Association (EITA)

Stand 22 GOLD

EITA is a non-profit organisation of road and air transporters and logistics providers, specialized in the transport of radioactive isotopes mainly used for medical purpose or research.
EITA's key objectives are to:

- ensure high quality of handling, packing and transport of radioactive isotopes
- ensure safety, security and traceability during handling and/ or transport
- represent the transport and logistics industry and keep contact with the authorities to obtain harmonisation of the different national, international and European regulations and policies.
- give members of EITA the opportunity to represent their interests and to exchange all legal information and skills to achieve quality within the Association

European Isotopes Transport Association (EITA)
Nederokkerzeelstraat 6
B-1910 Kampenhout
Belgium
Tel: + 32 16 65 08 12
Contact: Kristel Vermeersch
Email: kristel.vermeersch@kvspartners.be
Web: www.eita.org

General Plastics Manufacturing Company

Stand 9 SILVER

General Plastics Manufacturing Company, and our LAST-A-FOAM® FR-3700 crash and fire protection foams are recognized by specifying agencies with nuclear-regulatory authority as being one of the best solutions for the protection of hazardous payloads. LAST-A-FOAM® FR-3700 is used in insulating and isolating radioactive nuclear materials from shock, impact, and fire damage in crash situations. When used as an impact-and-fire insulation liner in transport containers, LAST-A-FOAM® FR-3700 can be engineered to provide the ultimate in protection for hazardous cargo, outperforming wood and other polymeric materials. LAST-A-FOAM® FR-3700 is available in foamed-in-place applications, machined foam parts, and fabricated assemblies.

General Plastics Manufacturing Company
4910 Burlington Way
Tacoma, WA 98409
Tel: +1 253 4735000
Contact: Rick Brown
Email: rick_brown@generalplastics.com
Web: www.generalplastics.com

GNS Gesellschaft für Nuklear-Service mbH

Stand 3 PLATINUM

GNS Gesellschaft für Nuklear-Service mbH, world leading supplier of casks for spent fuel, HLW and ILW, also offers services for management and disposal of spent fuel and all types of radioactive waste. More than one thousand spent fuel casks of the CASTOR® and CONSTOR® type and more than 5500 MOSAIK® casks for intermediate level waste make GNS the supplier of the world's top selling shielded transport and storage casks, which are today used in a number of countries on four continents. Design and supply of treatment facilities, decommissioning services and all kinds of engineering support round off our portfolio.

GNS Gesellschaft für Nuklear-Service mbH
Hollestraße 7A
D-45127 Essen
Germany
Tel: +49 201 1090
Contact: Sabine Ludwig
Email: info@gns.de
Web: www.gns.de

International Nuclear Services

Stand2 10,11 SILVER

International Nuclear Services (INS) has safely transported radioactive cargoes worldwide for over 40 years. Our subsidiary Pacific Nuclear Transport Ltd is the world's most experienced shipper of nuclear materials with a fleet of INF3

Sponsors, Advertisers & Exhibitors : continued

class ships and our sister company Direct Rail Services (DRS) is a national rail freight operator providing a comprehensive range of services. INS and DRS provide a complete transport service to our customers, from the design of nuclear transport packages through to transportation of nuclear materials using our dedicated rail and marine fleet. In addition to specialist transport solutions, we are able to provide a variety of consultancy services. INS and DRS are wholly subsidiary of the UK's Nuclear Decommissioning Authority.

International Nuclear Services Ltd
Hinton House,
Birchwood Park Avenue,
Risley,
Warrington,
Cheshire
WA3 6GR
United Kingdom
Tel: +44 (0)1925 835000
Contact: Nick Bold
Email: info@innuserv.com
Web: www.innuserv.com

International Source Suppliers and Producers (ISSPA)

Stand 20 GOLD

The International Source Suppliers and Producers Association is an association that has been founded by companies that are engaged in the manufacture, production and supply of sealed radioactive sources and/or equipment that contain sealed radioactive sources as an integral component of the radiation processing or treatment system, device, gauge or camera.

International Source Suppliers and Producers (ISSPA)
447 March Road
Ottawa
Ontario
K2K 1X8
Canada
Tel: +1 613 762 0282
Contact: Grant Malkoske
Email: grant.malkoske@mdsinc.com
Web: www.isspa.com

Maney Publishing

Stand 24 SILVER

Maney delivers a personalised service to authors, societies, readers and libraries for the publishing and international dissemination of high quality, peer-reviewed scholarship and research. Maney publishes an impressive collection of highly regarded, peer-reviewed journals covering both niche and general topics in materials science and engineering, Packaging, Transport, Storage & Security of Radioactive Material. *PTSSRM* covers all aspects of the transport of radioactive materials, including regulations, package design, safety analysis, package testing, routine operations and experiences, storage and security, and accidents and emergency planning. It is the only international peer-reviewed journal exclusively devoted to the transport of radioactive materials.

Maney Publishing
1 Carlton House Terrace
London
SW1Y 5AF
United Kingdom
Tel: +44 (0)20 7451 7300
Contacts: Mark Hull/Naomi Asantewa-Sechereh
Email: maney@maney.co.uk
Web: www.maney.co.uk

Nippon Light Metal Co. Ltd

Stand 19 GOLD

Nippon Light Metal is Japan's sole fully integrated manufacturer specialized in wide ranges of operations using aluminium and associated materials, from alumina to chemicals and various fabricated products. MAXUS is the neutron absorber for the nuclear industry which was produced at Nippon Light Metal Niigata plant in Japan.

Nippon Light Metal Co., Ltd
NYK Tennoz Building 2-2-20
Higashi-Shinagawa
Shinagawa
Tokyo, 140-8628
Japan
Tel: +81 3 5461 8309
Contact: Toshiaki Yamazaki
Email: sanada@nikkeikin.co.jp
Web: www.nlmetal.com

Sponsors, Advertisers & Exhibitors : continued

Nuclear Fuel Transport Co. Ltd

Stand 4 **GOLD**

Nuclear Fuel Transport Co., Ltd. is a specialized company for the transportation of nuclear fuel materials, including spent fuel in Japan.

Nuclear Fuel Transport Co., Ltd
1-1-3, Shiba-daimon, Minato-ku
Tokyo, 105-0012
Japan
Tel: +81 3 3438 3242
Contact: Yukari Tanaka
Email: y.tanaka@nft.co.jp
Web: www.nft.co.jp

ONET TECHNOLOGIES UK
Gravatom

ONET Technologies UK Gravatom

Stand 7 **SILVER**

For the past 35 years ONET Technologies has been preparing designs, manufacturing, testing and qualifying transport packages for radioactive materials and handling equipment, including Excepted to Type B(U) Fissile, for a range of companies in the UK and overseas including all the major UK nuclear establishments. We also supply packaging to the radiopharmaceutical industry, NHS, veterinary practices and waste disposers. Our staff includes qualified DGSAs whose knowledge of the various modal regulations is vital in ensuring legal compliance. Our facilities enable us to manufacture and test in-house. All work is performed in accordance with our ISO 9001 accredited QA system.

ONET Technologies UK Gravatom
William Kelvin Building
Claylands Road
Bishops Waltham
Hampshire, SO32 1BH
United Kingdom
Tel: +44 (0)1489 897300
Contact: Gerard Holden
Email: sales@gravatom.com
Website: www.gravatom.com

PANPROJECTS
Panalpina Group

Panalpina World Transport (Panprojects Division)

Stand 23 **GOLD**

Panprojects, a Division of the Panalpina Group, is an independent business division that provides Integrated Logistics Project Management and Turnkey Project Transport Services globally to the Engineering Procurement and Construction industry in the sectors of Energy, Mining, Chemical, Petrochemical, Pipeline Infrastructure, as well as fabricators of Heavy and Over Dimensional equipment and modules. We bring added value to our Clients' with the industry specific expertise, market reach and sourcing strength of the Panalpina Group. We execute complex and challenging projects and missions with our team of project focused specialist assembled and deployed in response to client needs in three specialised areas.

Panalpina World Transport (Panprojects Division)
Suite 5, Panprojects House
Butlers Wharf
London
SE1 2YE
United Kingdom
Tel: +44 (0) 207 234 8880
Contact: John Fludder
Email: John.Fludder@panalpina.com
Web: www.panalpina.com

QUADRANT

Quadrant EPP AG

Stand 21 **GOLD**

Quadrant Engineering Plastic Products is worldwide leading in the production of general and advanced engineering plastics in form of shapes (plates, rods, tubes) for machining. Quadrant also offers customized finished parts plastic solutions for a broad range of industries. Quadrant's borated PE-(U)HMW grades, sold under the Borotron® brand, are used as a medical and industrial shielding material to attenuate and absorb neutron radiation. Borated PE combines the effect of moderation of fast neutrons and absorption of lower energy thermal neutrons. These dimensionally stable plastics are used for medical doors and vaults, hot cells, nuclear storage and transport containers, nuclear waste management, particle accelerators and nuclear detection systems.

Sponsors, Advertisers & Exhibitors : continued

Quadrant EPP AG
Hardstrasse 5
5600 Lenzburg
Switzerland
Tel: +41 80 62 8858 409
Contact: Mr. Stefan Willing
Email: contact@qplas.com
Web: www.quadrantplastics.com

ROBATEL Industries

Stand 6 SILVER

Robatel Industries has been manufacturing transportation casks for radioactive components for over 50 years. Robatel Industries has developed up to 74 type B casks and manufactured over 500 casks units. These casks are designed for different types of materials: spent fuel rods, metallic waste, radioactive liquids, sources.

Robatel Industries
Rue de Genève - BP 203
69741 Genas
Cedex
France
Tel: +33 4 72 22 10 10
Contact: M. Christophe Bruneel
Email: commercial@robatel.fr
Web: www.robatel.fr

Westerman Inc

Stand 18 GOLD

Over the last two decades, Westerman's Companies has emerged as a trusted supplier to many of the world's leading Energy Industries. Westerman's has long been one of the world's largest producers of enriched uranium hexafluoride (UF6) storage and transportation cylinders. Demand for Westerman's products continue to grow in both the US and International Markets. Westerman manufacturing experience combined with established quality assurance processes enable the company to meet stringent NQA-1 standards set by A.S.M.E. and the NRC,10 CFR 50 (Appendix B) and 10 CFR 21 subpart (H) requirements. Westerman's A.S.M.E. Nuclear Certification is "N", "NS", N3, NA, and NPT. WESTERMAN, S COMPANIES WORLD CLASS

Westerman Inc
245 North Broad Street
Bremen, Ohio
43107
USA
Tel: +1 569 4143
Contact: William Moore
Email: bmoore@westermancompanies.com
Web: www.westermancompanies.com

World Nuclear Transport Institute

Stand 25 SILVER

The World Nuclear Transport Institute:

- is a global industrial organisation committed to ensuring that the transport of radioactive materials is conducted safely, efficiently and reliably;
- is dedicated entirely to presenting the industry's view from an international perspective;
- responds to specific issues of concern to its collective membership, through consultations, meetings and workshops;
- consults with governmental and non-governmental bodies to support the work of key international organisations in promoting a safe and harmonised international transport regulatory regime;
- produces both technical and more general information;
- is represented at various international events and develops awareness.

World Nuclear Transport Institute
Remo House 4th Floor
310-312 Regent Street
London
W1B 3AX
United Kingdom
Tel: +44 (0)20 7580 1144
Contact: Lorne Green
Email: wnti@wnti.co.uk
Web: www.wnti.co.uk

Invited Speakers & Rapporteurs Biographies

Bertrand Barré,
Scientific Advisor to Anne Lauvergeon, CEO Areva,

Bertrand Barré is Scientific Advisor to the Chairperson of the AREVA group, and Professor Emeritus of nuclear engineering at the French Institut National des Sciences et Techniques Nucléaires, INSTN.

Born in December 1942, Bertrand Barré joined the French Atomic Energy commission, CEA, in 1967 and has been working ever since, both in France and abroad, for the development of Nuclear Power.

Alternating scientific and managerial positions, Bertrand Barré was notably Nuclear Attaché at the French Embassy in Washington (USA), Director of Engineering in TECHNICATOME (now AREVA-TA), Director of the Nuclear Reactor Directorate of the CEA and Vice-president in charge of R&D in COGEMA (now AREVA-NC).

Bertrand Barré is Past President of the European Nuclear Society (ENS) and of the International Nuclear Societies Council (INSC), and Immediate Past-Chairman of the International Nuclear Energy Academy (INEA).

Bertrand Barré was the first Chairman of SAGNE, the Standing Advisory Group on Nuclear Energy advising the Director General of IAEA, and has been a Member of the EURATOM Scientific & Technical Committee since 1995.

Bernhard Droste
Director, Professor, BAM

Dr Droste is Director and Professor of the BAM Federal Institute for Materials Research and Testing in Berlin, Germany. In 1968 Bernhard Droste began his professional life at the Academy of Applied Sciences, Dortmund, where he studied Material Technology. From 1972 – 1978, Dr Droste's academic career flourished at the Technical University Berlin, where he was awarded a postgraduate qualification, and completed a doctorate in Metal Physics. From 1979 – 1981 Dr Droste investigated the chemical compatibility of tank containers for dangerous goods, and the mechanical design of tank containers.

During the eighties Dr Droste headed the Working Group "Containments of the Storage of Dangerous Goods", where he addressed the safety assessment of LPG tanks and casks for the interim storage of spent fuel. Since 1991 Dr Droste has been Head of Division III.3, the "Safety of Transport Containers": the division that constitutes the German Competent Authority for mechanical and thermal assessment of packages for the transport of radioactive materials. Up until 2007 Dr Droste was also responsible for the safety evaluation of spent fuel and HLW interim storage in dual-purpose casks and design type testing of LLW/MLW disposal containers.

In 2004 Dr Droste coordinated the 14th International Symposium on the Packaging and Transportation of Radioactive Material (PATRAM 2004) in Berlin, Germany; and in 2007 Dr Droste was awarded the PATRAM Aoki Award for long-term contribution. Since 1978 Bernhard Droste has been active in safety science at the BAM Federal Institute for Materials Research and Testing, and he continues to work there to this day. Born in 1950, Bernhard Droste has contributed to more than 170 lecture and poster presentations, and more than 180 publications.

Shamsideen B. Elegba
Director General/CEO, Nigerian Nuclear Regulatory Authority

Born in Nigeria, Professor Shamsideen Babatunde Elegba obtained a first class M.Sc. Degree in Theoretical Nuclear Physics from Kharkov State University, in the former Soviet Union. In 1976 Professor Elegba journeyed to the USA, where he completed a Ph.D. in Theoretical Solid State Physics, at the University of Oregon. With his combined work at both the Ahmadu Bello University in Nigeria and the University of Oregon between 1974 and 1998, Professor Elegba has over 20 years of University research and teaching experience.

From 1986-1991 Professor Elegba was Coordinator and Pioneer Director at one of Nigeria's most advanced research institutions in the area of nuclear science and technology: the Centre for Energy Research and Training (CERT). In 1987 Professor Elegba Installed and operated the first Neutron Generator in West Africa. After winning and successfully packaging a Technical Cooperation project from the IAEA for the donation of a nuclear research reactor to CERT, Professor Elegba successfully licensed the first and only Nuclear Reactor in Nigeria in 2004; consequently the first and only Nuclear Reactor in Nigeria.

Having contributed as author and co-author to over 200 scientific publications and technical reports, Professor Elegba

Invited Speakers & Rapporteurs Biographies : continued

also has wide experience in energy planning and management, in addition to nuclear safety and security. A fellow of the Nigerian Institute of Physics, Professor Elegba is also a member of the American Physical Society, the Society of Radiological Protection, and the Institute of Nuclear Materials Management. In 1986 Professor Elegba joined the Nigerian delegation to the Annual General Conference of the International Atomic Energy Agency (IAEA), in which he is still a member to this date.

A Pioneer Chairperson of the Forum of Nuclear Regulatory bodies in Africa; a member of the African Union Commission Preparatory Committee for the Conference of State Parties to the African Nuclear Weapons Free Zone Treaty A Pioneer; and Director-General of the Nigerian Nuclear Regulatory Authority since 2001, Professor Elegba's contribution to the advancement of nuclear power in Nigeria is unparalleled, and highly commendable. Professor Elegba has been presented with the Award of Excellence by the Nigerian Institute of Physics, and also the Presidential National Productivity Order of Merit Award.

Lorne Green

Secretary General,
WNTI

Lorne Green is Secretary General of the World Nuclear Transport Institute (WNTI) based in London, U.K. For thirty years Mr. Green was an officer in the Canadian Department of Foreign Affairs, with postings in Islamabad, London, NATO Headquarters in Brussels, Belgrade and The Hague. Between 1986-1989, he was Director of Nuclear and Arms Control Policy in the Canadian Department of National Defence. Mr. Green was Charge d'Affaires at the Canadian Embassy in Yugoslavia, and accredited to Bulgaria and Albania, in 1992. In 1996 Mr. Green became the first Director of the Nuclear, Non-Proliferation and Disarmament Implementation Agency within the Canadian Department of Foreign Affairs, with responsibility inter alia for Canada's nuclear co-operation agreements with other countries, Canada's participation in the International Atomic Energy Agency, and implementation in Canada of the Comprehensive Nuclear-Test-Ban Treaty and Chemical Weapons Convention. In 1998 Mr. Green left the Canadian Government to join in the creation of the new World Nuclear Transport Institute.

Mark Jervis

Managing Director,
International Nuclear Services Ltd

Mark Jervis has over 28 years experience in the UK nuclear industry. He commenced his career at Sellafield working on vitrification process development. He has extensive commercial and customer facing experience having held a number of roles associated with development and management of Sellafield's complex nuclear recycling contracts and the associated transport business.

Since 2008, Mark Jervis has been the Managing Director of International Nuclear Services Ltd which is a wholly owned subsidiary of the UK Nuclear Decommissioning Authority. International Nuclear Services is the commercial agency for spent fuel management services in the UK and is the world's most experienced global shipper of nuclear materials. International Nuclear Services operates a fleet of ships dedicated to the transport of nuclear materials on behalf of its subsidiary Pacific Nuclear Transport Ltd and on behalf of the Nuclear Decommissioning Authority.

Mark is also Chairman of the World Nuclear Transport Institute and a Director of Pacific Nuclear Transport Ltd.

Pierre Malesys

Transport and International Relations Specialist
Areva

Pierre MALESYS is a graduated engineer from Ecole Centrale de Paris.

He has more than 26 years of experience in the nuclear field, including more than 24 years in the transport of radioactive material, within the AREVA Group.

He occupied several positions in TN International (a subsidiary of AREVA), where he was – inter alia - in charge of design and licensing of packages, and regulatory issues. He is now with AREVA in charge of international relations in the field of nuclear safety.

Pierre MALESYS is the representative of the International Organization for Standardization (ISO) in the TRANsport Safety Standards Committee (TRANSSC) of the International Atomic Energy Agency (IAEA). He is also a member of the Advisory Committee of the World Nuclear Transport Institute.

Invited Speakers & Rapporteurs Biographies : continued

Boyce M. Mkhize
CEO,
South Africa National Nuclear Regulator

Boyce Mkhize is currently the Chief Executive Officer of the National Nuclear Regulator (NNR), based in Pretoria, South Africa. Boyce is a lawyer by training possessing B.Juris and LLB degrees from the University of Zululand and an admitted Advocate of the High Court of South Africa. He also possesses other qualifications from UNISA and University of Stellenbosch on Business and Management.

Prior to him joining the NNR in February 2010, he was the Registrar and Chief Executive Officer of the Health Professions Council of South Africa (HPCSA) where he served for a period just a little over nine (9) years. While CEO of the HPCSA, Boyce also served as the Secretary-General of the Association for Medical Councils of Africa (AMCOA), a continental body responsible for standardizing professional standards and sharing of information on health regulation. Boyce also served as a member of the International Association of Medical Regulatory Authorities (IAMRA) Constitution and By-Laws Committee as well as its Nominating Committee.

He also served as Chief Legal Advisor and Company Secretary for the then Atomic Energy Corporation, now known as the Nuclear Energy Corporation of South Africa (NECSA). Prior to this he had served in Government Departments in various capacities which included being Chief of Staff in the office of the Minister of Public Service and Administration and Deputy Director for Affirmative Action Policy and Transformation. He also served as a Human Rights Lawyer for the Community Law Centre and later headed the drafting of the first democratic constitution process in the Republic of South Africa, for the Constitutional Assembly. Boyce also serves in Boards and Councils of a few public institutions and private companies in South Africa.

Anita Birgitta Nilsson
Director of Office of Nuclear Security,
International Atomic Energy Agency (IAEA)

As Director of the IAEA Office of Nuclear Security Ms. Nilsson is responsible for the IAEA nuclear security programme, and for the implementation of the Nuclear Security Plan for 2010-2013 which is aimed at protecting against nuclear terrorism.

From 1996 to 2002, in the Department of Safeguards, she was Chair of the Information Review Committee, which assesses evaluations of States' compliance with their safeguards undertakings.

Before joining the IAEA, Ms. Nilsson worked in various managerial and leadership positions at the Swedish Nuclear Power Inspectorate, dealing with non-proliferation, international and national safeguards, bilateral nuclear supply and co-operation. She was in charge of the Swedish nuclear security support to the Newly Independent States and the Baltic States.

Ms. Nilsson is Master of Science and Medical Doctor.

Irfan Rahim
Senior Official
IMO

Mr. Irfan Rahim is the senior official of the International Maritime Organization whose department provides machinery for co-operation among Governments in the field of governmental regulation and practices relating to technical matters affecting the safe and efficient handling and carriage by sea of all types of cargoes, including radioactive materials.

He has represented the Organization at a number of outside meetings such as UNECE, ECOSOC and IAEA and has played a leading role in harmonising the maritime mode specific dangerous goods regulations with those of other modes and in addressing matters associated with delays and denials of shipments of radioactive materials.

Prior to joining the International Maritime Organization in 1998, Mr. Rahim held high level posts in the Asia and the Pacific Region and was responsible for the conduct and direction of a number of projects which had multimodal transport dimension and brought extensive social and economic benefits to the peoples of the region.

Program

Invited Speakers & Rapporteurs Biographies : continued

Toshiari Saegusa

Associate Vice President,
CRIEPI

In 1978, Dr. Toshiari Saegusa graduated from Northwestern University, Chicago, with a PhD in the field of Materials Science and Engineering. In 1981, he started his career with the Central Research Institute of Electric Power Industry (CRIEPI) in Japan. In 2005 he became the Director of the Spent Fuel Transport & Storage Research Project, and the Associate Vice President of CRIEPI. Earlier this year Dr. Saegusa was appointed as the Executive Research Scientist of CRIEPI.

Dr. Saegusa has received awards from The Ministry of Economy, Trade and Industry (METI) for his Contribution to Nuclear Safety, and from the Japanese Atomic Energy Society (AESJ) for the Development of Nuclear Safety Technology.

Dr. Saegusa is the Chairman of the Working Group on the Transport of Nuclear Material at METI, and he chairs the Subgroup on Spent Fuel, Transport and Storage for the Japanese Society of Mechanical Engineers (JSME).

Dr. Saegusa is an extremely valuable member of various industry subgroups and committees. He is a member of the Subcommittee on Nuclear Safety Research of the Nuclear Safety Commission, a member of the Working Group on Storage of Spent Nuclear Fuel at METI, and he is a member of the Subgroups on Spent Fuel Transport and Storage in AESJ, and on NUPACK in ASME.

Christoph Schröder

Senior Administrator, EUROPEAN COMMISSION, Directorate General for Energy, Directorate D – Nuclear Energy, Unit D.2 – Nuclear Energy, Decommissioning, Transport & Waste Management

Christoph Schröder is working as a senior administrator for the Energy Directorate General of the European Commission in Luxembourg. Nowadays he mainly deals with general aspects of the Commission's nuclear energy policy, and in particular with the European Nuclear Energy Forum as well as information and communication matters.

Prior to joining the Commission in 2004, he headed the minister's office and the public relations department of the ministry of economics affairs in a German federal state. He studied economics at the University of Saarland in Germany and the University of Michigan in the US.

Steve Whittingham

HM Superintending Inspector, Head of Compliance Inspection Team (industrial and medical sectors) and RAM Stakeholder Engagement, Dangerous Goods Division, Department for Transport

Steve Whittingham has more than 30 years of experience in the nuclear field, including 24 years in industry concerning the transport of spent nuclear fuel in Europe, MOX fuel to Japan and nuclear wastes in the UK. For the last 6 years he has worked in the UK Competent Authority. He occupied several positions in industry at NTL (a former subsidiary of BNFL) and BNFL where he was responsible throughout his career for the licensing of packages and regulatory issues. He is now responsible for the quality and compliance inspections of the medical and industrial sectors and the transport regulator involvement in the various work streams of the Nuclear Decommissioning Authority in the UK.

Steve Whittingham has represented the UK at the Transport Safety Standards Committee (TRANSSC) and the Denial of Shipment International Steering Committee of the International Atomic Energy Agency (IAEA) and attended the Transport Standing Working Group of the European Commission. He is currently the chairman of the Association of European Competent Authorities.

Clive Young

Consultant

Clive Young is an internationally recognised expert in safety standards for transport of radioactive materials. He worked for the Department for Transport (DfT) of the United Kingdom from 1978 and, in 1996, became Head of the Radioactive Materials Transport Division and Transport Radiological Adviser to the Secretary of State for Transport. In this position he was responsible for carrying out the executive functions of the "Competent Authority" for the transport of radioactive material in the United Kingdom on behalf of the Secretary of State for Transport. He has served as Chairman of the Transport Safety Standards Committee of the International Atomic Energy Agency and Chairman of the Radioactive Material Working Group of the International Maritime Organization. Since retiring from the DfT in 2006, he has continued his activities in radioactive material transport as a consultant. He previously held the position of research engineer at the UK Atomic Energy Authority. Mr. Young earned his B.Sc. in mechanical engineering from the University of Leeds in 1969. He is a member of the Institution of Mechanical Engineers and is a Chartered Engineer.

Programme at a Glance

Sunday 03 October

1:00pm – 7:30pm	Registration Open	IMO Main Foyer
1:00pm – 5:00pm	Exhibitor Set-up	Delegate Lounge
6:00pm – 7:30pm	Welcome Reception and Exhibition Opening	Delegate Lounge

Monday 04 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant IMO
8:00am – 6.00pm	Exhibition Open	Delegate Lounge
9:00am – 10:40am	Welcome Addresses	Main Hall
10:40am – 11:00am	Refreshment Break	Delegate Lounge
11:00am – 12:40pm	Opening Plenary	Main Hall
12:40pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge
4:00pm – 6:00pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3

Tuesday 05 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant IMO
8:00am – 6.00pm	Exhibition Open	Delegate Lounge
8:15am – 8:40am	Morning Plenary	Main Hall
8:40am – 8:50am	Rapporteurs Update	Main Hall
9:00am – 10:40am	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
10:40am – 11:00am	Refreshment Break	Delegate Lounge
11:00am – 12:40pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
12:40pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	Concurrent Panel Sessions	Main Hall & Conference Rooms 1, 2, 3
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge
4:00pm – 6:00pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3

Wednesday 06 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant IMO
8:00am – 6.00pm	Exhibition Open	Delegate Lounge
8:15am – 8:40am	Morning Plenary	Main Hall
8:40am – 8:50am	Rapporteurs Update	Main Hall
9:00am – 10:40am	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
10:40am – 11:00am	Refreshment Break	Delegate Lounge
11:00am – 12:40pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
12:40pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	Concurrent Panel Sessions	Main Hall & Conference Rooms 1, 2, 3
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge
4:00pm – 6:00pm	Poster Session	Delegate Lounge & IMO Main Foyer

Programme at a Glance : continued

Thursday 07 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant IMO
8:00am – 5.40pm	Exhibition Open	Delegate Lounge
8:15am – 8:40am	Morning Plenary	Main Hall
8:40am – 8:50am	Rapporteurs Update	Main Hall
9:00am – 10:40am	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
10:40am – 11:00am	Refreshment Break	Delegate Lounge
11:00am – 1:00pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
1:00pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	Concurrent Panel Sessions and Technical Session	Main Hall & Conference Rooms 1, 2, 3
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge
4:00pm – 5:40pm	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3

Friday 08 October

7:00am – 12:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant IMO
8:00am – 12:00pm	Exhibition Open	Delegate Lounge
8:45am – 9:00am	Rapporteurs Summing-Up	Main Hall
9:00am – 10:40am	Concurrent Technical Sessions	Main Hall & Conference Rooms 1, 2, 3
10:40am – 11:00am	Refreshment Break	Delegate Lounge
11:00am – 12:00pm	Closing Plenary	Main Hall
12:00pm – 12:05pm	Farewell and see you in 2013!	Main Hall

Daily Programme : Sunday 03 - Monday 04 October

Sunday 03 October

1:00pm – 7:30pm	Registration Open	IMO Main Foyer
1:00pm – 5:00pm	Exhibitor Set-up	Delegate Lounge
6:00pm – 7:30pm	Welcome Reception and Ehibition Opening	Delegate Lounge

Monday 04 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant
8:00am – 6:00pm	Exhibition Open	Delegate Lounge
9:00am – 10:40am	**Welcome Addresses**	Main Hall
9:00am	UNITED KINGDOM REPRESENTATIVE	
9:25am	IAEA REPRESENTATIVE	
9:50am	IRFAN RAHIM, IMO REPRESENTATIVE	
10:15am	LORNE GREEN, WNTI REPRESENTATIVE	
10:40am – 11:00pm	Refreshment Break	Delegate Lounge
11:00am – 12:40pm	**Opening Plenary**	Main Hall
	Nuclear Reactor Build Programmes – The Impact on Transport	
	BERTRAND BARRÉ, SCIENTIFIC ADVISOR TO ANNE LAUVERGEON, CEO OF AREVA	
	Developing the Framework for Nuclear in Europe – Perspectives for Transport	
	CHRISTOPH SCHROEDER, SENIOR ADMINISTRATOR, EUROPEAN COMMISSION	
	DOUG WEAVER, DEPUTY DIRECTOR, US NUCLEAR REGULATORY COMMISSION	
	Transport and Storage of Nuclear Materials in Japan	
	TOSHIARI SAEGUSA, CRIEPI	
12:40pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	**Concurrent Technical Sessions**	

T3 – Security (Session 1)

2:00pm – 3:20pm	CHAIR: BRYAN REEVES, CO-CHAIR: OLIVIER LOISEAU	Main Hall
2:00pm	ABSTRACT 213	
	Radioactive and Nuclear Material Transport Security	
	ANN-MARGRETH ERIKSSON EKLUND*, RICHARD RAWL	
2:20pm	ABSTRACT 214	
	The IAEA Assistance and Training Programme for Transport Security	
	RICHARD RAWL, ANN-MARGRETH ERIKSSON EKLUND (PRESENTED BY MARK HAWK)	
2:40pm	ABSTRACT 364	
	Transport Security - an Operational View	
	MATT FOX*, ANDRÉ STASSE	
3:00pm	ABSTRACT 375	
	Protection Of Information: An Essential Component Of Physical Protection Of Nuclear Material Transportation	
	ANDRÉ STASSE*, MATT FOX	

T1 – Spent Fuel Package Designs

2:00pm – 3:40pm	CHAIR: LAURENT MILET, CO-CHAIR: ULRICH ALTER	Conference Room 1
2:00pm	ABSTRACT 165	
	Italian-French Experience in Development, Licensing and Manufacturing of a New Cask for the Transportation of Irradiated Nuclear Fuel from Piemonte Nuclear Sites	
	JOEL BAUDOUIN*, EMILIE BOUYER, THOMAS BRION, HERVÉ RIPERT	

Daily Programme : Monday 04 October : continued

2.20pm	ABSTRACT 230 **Transportation Package for use in Facilities with 25 Ton Crane Capacity** CATHERINE SHELTON*, OLIVIER GANDOU, NICOLAS GUIBERT	
2:40pm	ABSTRACT 209 **Development of a New Dual Purpose Cask** JUSTO GARCIA*, OLIVIER ROULLEAUX-DUGAGE	
3:00pm	ABSTRACT 148 **Development of Type C Packages to Transport Spent Nuclear Fuel from Research Reactors Produced in Russia** LUDMILA BARABENKOVA*, VYACHESLAV SHAPOVALOV, ALEKSANDR MORENKO, VITALY MATVEEV, SERGEY KOMAROV	

T2 – Radiation Protection

2:00pm – 3:40pm	CHAIR: MARIE-THERESE LIZOT, CO-CHAIR: JAMES SHULER	Conference Room 2
2:00pm	ABSTRACT 58 **Measurement of Radiation Level and Surface Contamination for Packages and Conveya Conveyances** ASHOK KAPOOR*, JAMES WILLIAMS, S.Y. CHEN, SUNITA KAMBOJ	
2:20pm	ABSTRACT 225 **A New Solution to Decontaminable and Inspectable Package Handling Features that can be Blocked by a 90° Rotation** FABIEN GIRAULT*, ROBBIE JAMESON	
2:40pm	ABSTRACT 75 **Radiological Safety of Spent Fuel Storage and Transport** CHARLES W. PENNINGTON*	
3:00pm	ABSTRACT 95 **TN International Transportation Procedure for Used Fuel Casks: Transportability Tool** MIKAEL DE BIASI*, STAVROS KITSOS	
3:20pm	ABSTRACTS 277/282 **Utilisation of the Monte Carlo Code 'MCBEND' and the Deterministic Code 'ATTILA' to Assist with the Shielding and Dose Analysis for the Land and Marine Transportation of an International Transport Flask** ANDREW SMITH, ANTHONY CORY	

T4 – Thermal Analysis

2:00pm – 3:20pm	CHAIR: CARLOS LOPEZ, CO-CHAIR: FRANK KOCH	Conference Room 3
2:00pm	ABSTRACT 96 **Computational Fluid Dynamic (CFD) Design and Mock up Test for Heat Removal** OLIVIER BARDON*, JEROME BELLANGER , NASSER ZAHRI	
2:20pm	ABSTRACT 59 **Modelling the Thermal Performance of Cork and Wood in the Thermal Test** CHRIS FRY*	
2:40pm	ABSTRACT 203 **Thermal Shielding of the Shock Absorber is Made of Wood** KYOUNG-SIK BANG*, JU-CHAN LEE, KI-YOUNG KIM, CHUNG-SEOK SEO, KI-SEOG SEO	
3:00pm	ABSTRACT 120 **Behaviour of a Package for Transport of Spent Fuel Assemblies Exposed to Beyond Regulation Fires** BENOIT ECKERT*, GILLES SERT, SARAH FOURGEAUD, IGOR LE BARS	
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge

Daily Programme : Monday 04 October : continued

4:00pm – 6:00pm	**Concurrent Technical Sessions**

T6 – Interfacing with the Public

4:00pm – 6:00pm	CHAIR: RUPERT WILCOX-BAKER, CO-CHAIR: HENRY-JACQUES NEAU	Main Hall
4:00pm	ABSTRACT 155 **Safety Analysis of the Transportation of Radioactive Waste to the Konrad Final Repository - Waste Data Scenarios** ULRICH ALTER*	
4:20pm	ABSTRACT 248 **Safety Analysis of the Transportation of Radioactive Waste to the Konrad Final Repository – Methods and Results** FLORENCE-NATHALIE SENTUC*, WENZEL BRUCHER	
4:40pm	ABSTRACT 237 **Public Acceptability - You Can't Judge a Book by its Cover** LORNE GREEN*	
5:00pm	ABSTRACT 374 **Public Acceptance Approaches Related to Back-end Transport between Europe and Japan** TAKASHI KOMATSU*	
5:20pm	ABSTRACT 257 **Communication and Radioactive Material Transportation** CAMILLE OTTON*, BERNARD MONOT	
5:40pm	ABSTRACT 239 **Communicating the Transport of Radioactive Materials Using New Media** BETTY BONNARDEL-AZZARELLI*	

T5 – Isotopes/Sources Packaging Designs

4:00pm – 5:40pm	CHAIR: ALBERTO ORSINI, CO-CHAIR: PETER LAMBOURNE	Conference Room 1
4:00pm	ABSTRACT 69 **Design and Development of BI-TL-300 Equipment as a Type B (U) Transportation Cask** DHIREN SAHOO*, JOTIRAM MANE, VINAY BHAVE, PIYUSH SRIVASTAV, ANIL KOHLI	
4:20pm	ABSTRACT 40 **Type B Package for the Transport of Large Medical and Industrial Sources** PHILIP NOSS*, DWAINE BROWN	
4:40pm	ABSTRACT 24 **BU-MAN, the new Argentinean Type B(U) Package for the Safe Transport of Radioisotopes** ANA MARIA CASTELLANOS*, EDUARDO ESTEBAN	
5:00pm	ABSTRACT 283 **TN PNS a New Type of Cask** NICOLAS GUIBERT*	
5:20pm	ABSTRACT 192 **Development of the Bulk Tritium Shipping Package** PAUL BLANTON*, PAUL MANN, KURT EBERL	

T16 – Structural Methods

4:00pm – 5:40pm	CHAIR: TBC, CO-CHAIR: ROBERT GRUBB	Conference Room 2
4:00pm	ABSTRACT 124 **Mechanical Design Assessment Approaches of Actual Spent Fuel and HLW Transport Package Designs** BERNHARD DROSTE, FRANK WILLE*, KARSTEN MUELLER, UWE ZENCKER	

Daily Programme : Monday 04 - Tuesday 05 October

4:20pm	ABSTRACT 255
	Acceptability of Dynamic Finite Element Analyses - Material Failure Approach
	ANINDYA SEN*, IAIN DAVIDSON
4:40pm	ABSTRACT 337
	The Effect of Gaps on Response of a Spent Fuel Transportation Package Closure Lid During a Drop Impact
	GORDON BJORKMAN*
5:00pm	ABSTRACT 117
	Numerical Simulation of 9 Meter Drop of a Transport and Storage Cask with Aluminium Impact Limiter
	LINAN QIAO*, UWE ZENCKER, FRANK WILLE, ANDRE MUSOLFF
5.20pm	ABSTRACT 339
	Strain-Based Acceptance Criteria for Spent Fuel Storage and Transportation Containments
	GORDON BJORKMAN*, DOUG AMMERMAN

T8 – Spent Fuel Behaviour

4:00pm – 5:40pm	CHAIR: ROLAND HUEGGENBERG, CO-CHAIR: PETER PURCELL	Conference Room 3
4:00pm	ABSTRACT 100	
	Mechanical Safety Analysis for High-Burnup Spent Fuel Assemblies under Accident Transport Conditions	
	VIKTOR BALLHEIMER*, FRANK WILLE, BERNHARD DROSTE	
4:20pm	ABSTRACT 211	
	Finite Elements Analysis of Inter-Grid Bending Tests on Used Fuel Rods Samples	
	MAURICE DALLONGEVILLE*, ARAVINDA ZEACHANDIRIN, PETER PURCELL, ANTHONY CORY	
4:40pm	ABSTRACT 323	
	Investigation of Spent Fuel Integrity in Dry Storage at Japanese Nuclear Power Plants	
	TAKESHI FUJIMOTO, MASAHIRO YAMAMOTO, MITSUO MATSUMOTO, KATSUHIKO SHIGEMUNE, HIROYUKI MATSUO	
5:00pm	ABSTRACT 137	
	Simplified Thermal Creep Model of an Irradiated Fuel Pin	
	CEDRIC LANGLADE*, MAURICE DALLONGEVILLE	
5:20pm	ABSTRACT 122	
	Development of the Swedish National Database for QA of Spent Nuclear Fuel	
	HENRIC LINDGREN*	

Tuesday 05 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers’ Breakfast	4th Floor Restaurant
8:00am – 6:00pm	Exhibition Open	Delegate Lounge
8:00am – 8:40am	**Morning Plenary**	Main Hall
	Interaction of Package Design Development and Package Design Safety Assessment	
	BERNHARD DROSTE, DIRECTOR & PROFESSOR, BAM	
8:40am – 8:50am	Rapporteurs Update	Main Hall
9:00am – 10:40am	**Concurrent Technical Sessions**	

T10 – High Level Waste

9:00am – 10:40am	CHAIR: SAM DARBY, CO-CHAIR: JURGEN WERLE	Main Hall

Daily Programme : Tuesday 05 October

9:00am	ABSTRACT 259
	European Experience in the First Transports of Universal Canisters Containing Compacted Metallic Waste Coming from Treatment
	FLORIAN DARRAS*, JEAN PASCHAL, DAMIEN SICARD, STEPHANE BEAUVERGER, JEAN-LUC ARNOUX
9:20am	ABSTRACT 402
	Application of the New Flask Type CASTOR HAW28M for the Return of Vitrified Residues from Sellafield, UK to Continental Europe
	MARCO WILMSMEIER*, ANDREW GRAY
9:40am	ABSTRACT 212
	Transportation of Vitrified High Level Wastes from Sellafield to Switzerland
	JUSTO GARCIA*, FRANCOISE GENDREAU, ERIC VICTOR-PUJEBET
10:00am	ABSTRACT 62
	Waste Transport Requirements to the Future Geological Repository
	ALAIN ROULET*, THIBAUD LABALETTE
10:20am	ABSTRACT 52
	Recognizing Interdependencies in the Design of the Nuclear Fuel Cycle and the Transportation of SNF and HLW
	MARK ABKOWITZ*, DANIEL METLAY, NIGEL MOTE

T11 – Security (Session 2)

9:00am – 10:40am	CHAIR: ANN-MARGRETH ERIKSSON, CO-CHAIR: TBC	Conference Room 1
9:00am	ABSTRACT 208	
	Assessment and Approval of Reinforced Protection of Vehicles Used for the Shipment of Sensitive Nuclear Material	
	OLIVIER LOISEAU*, DELPHINE LARRIGNON, BRUNO AUTRUSSON	
9:20am	ABSTRACT 150	
	Real-time Tracking of Nuclear Materials Packages in Transport	
	KUN CHEN*, HANCHUNG TSAI, YUAN SUN, YUNG LIU, JIM SHULER	
9:40am	ABSTRACT 287	
	Global Identification and Monitoring of UF6 Cylinders	
	JESSICA WHITE*, JANIE MCCOWAN, MARK LAUGHTER, MICHAEL WHITAKER	
10:00am	ABSTRACT 141	
	Sabotage of Radioactive Materials in Storage and Transport	
	KEN SORENSON*, ROBERT LUNA , BRUNO AUTRUSSON, OLIVIER LOISEAU, WENZEL BRUECHER, GUNTER PRETZSCH	
10:20am	ABSTRACT 288	
	Two Radioactive Material Transport Security Incidents: Lessons Learned and Questions to Address	
	NORMAN KENT*, PAUL KENT	

T12 – Thermal Studies

9:00 – 10:40	CHAIR: CHRIS FRY, CO-CHAIR: TBC	Conference Room 2
9:00am	ABSTRACT 411	
	Regulatory Fire Test Requirements for Plutonium Air Transport Packages: JP-4 or JP-5 vs. JP-8 Aviation Fuel	
	CARLOS LOPEZ*, VERNON F NICOLETTE	
9:20am	ABSTRACT 410	
	Fire Tests and Analyses of a Rail Cask-Sized Calorimeter	
	CARLOS LOPEZ*, AHTI SOU-ANTTILA, MILES GREINER	

Daily Programme : Tuesday 05 October : continued

9:40am	ABSTRACT 2	
	Transport of UF6 and the Future of Thermal Compliance	
	TIM KORBMACHER*, MARC-ANDRE CHARETTE	
10:00am	ABSTRACT 407	
	Thermo-mechanical Study of Bare 48Y UF6 Containers Exposed to the Regulatory Fire Environment	
	CARLOS LOPEZ*, DOUGLAS J AMMERMAN, MARC-ANDRE CHARETTE, TIM KORBMACHER	
10:20am	ABSTRACT 191	
	Model Development and Computational Analysis of the TUK46 Package with Uranium Hexafluoride (UF6) in Fire Environments	
	VYACHESLAV SHAPOVALOV*, SHOTA POPOV, YURIY POPOV, BORIS BARKANOV, ALEKSANDR MORENKO	

T9 – Burn-up Credit

9:00am – 10:20am	CHAIR: HELMUT KUHL, CO-CHAIR: LUDYVINE JUTIER	Conference Room 3
9:00am	ABSTRACT 327	
	Development of Technical Basis for Burn-up Credit Regulatory Guidance in the United States	
	CECIL PARKS*, JOHN WAGNER, DON MUELLER, IAN GAULD	
9:20am	ABSTRACT 270	
	Representativity Study of the French HTC and FP Experiments for Burn-up Credit Application to the TN 24 E Transport and Storage Cask	
	M. TARDY, C. GARAT*, S. KITSOS, F. RIOU, P. SOUBOUROU, M. LEIN, F. BERNARD, I. DUHAMEL, T. LECLAIRE IVANOVA	
9:40am	ABSTRACT 144	
	Application of Tsunami and Tsurfer for Validation of Burn-Up credit in the Criticality Safety Analysis of a Transport Cask	
	MATTHIAS BEHLER*, ROBERT KILGER, MATTHIAS KIRSCH, MARKUS WAGNER	
10:00am	ABSTRACT 143	
	Burn-up Credit Implementation for Transport and Storage Casks of UO2 Used Fuel Assemblies	
	MARCEL TARDY*, STAVROS KITSOS	
10:40am– 11:00am	Refreshment Break	Delegate Lounge
11:00am – 12:40pm	**Concurrent Technical Sessions**	

T14 – Large Components

11:00am – 12:40pm	CHAIR: RICK BOYLE, CO-CHAIR: HELMUTH ZIKA	Main Hall
11:00am	ABSTRACT 131	
	Transport of Large Components in Germany - Some Experiences and Regulatory Aspects	
	FRANK NITSCHE*, CHRISTEL FASTEN	
11:20am	ABSTRACT 43	
	Transport of Large Nuclear Power Plant Components - Experiences in Mechanical Design Assessment	
	STEFFEN KOMANN*, BERNHARD DROSTE, FRANK WILLE	
11:40am	ABSTRACT 86	
	Qualification of Steam Generators for Shipment with Respect to the Requirements of TS-R-1	
	WILLI SCHIFFER, FRANZ HILBERT (MICHAEL KUEBEL PRESENTING)	
12:00pm	ABSTRACT 223	
	Transportation of Solid Irradiated and Contaminated Non-fuel Radioactive Material in Large Transportation Package	
	MARLIN STOLTZ SR*, JAYANT BONDRE	

Daily Programme : Tuesday 05 October : continued

12:20pm	ABSTRACT 360 **The Transport Of Large Front End Facility Components From Decommissioning Operations** JURGEN WERLE*	

T15 – Front End and Recycled Material

11:00am – 12:40pm	CHAIR: MARC-ANDRE CHARETTE, CO-CHAIR: AL STRATEMEYER	Conference Room 1
11:00am	ABSTRACT 256 **Front-end Transports: Challenges to 2020** PERRINE RUSSIAS*	
11:20am	ABSTRACT 249 **Standard for Uranium Ore Concentrate Transport Drum** FABIEN PERRIN*, MARC DE SAILLY, PASCAL DE BASTIANI, WILLIAM MARTIN	
11:40am	ABSTRACT 103 **Transportation of Reprocessed Enriched Uranium** FRANZ HILBERT*	
12:00pm	ABSTRACT 78 **Future Perspective for MOX Transport Based on Experience in JAEA** TAKAFUMI KITAMURA*, NOBORU TADOKORO, KAN SHIBATA, YUICHIRO OUCHI	
12:20pm	ABSTRACT 170 **Packaging and Transboundary Transport of PuO and MOX Material** FRANCESCO D'ALBERTI*, ROBERTO DONATI, STEPHANIE LUTIQUE, ROBERTO VESPA	

T13 – Criticality Analysis

11:00am – 12:40pm	CHAIR: CECIL PARKS, CO-CHAIR: TBC	Conference Room 2
11:00am	ABSTRACT 272 **Imparting Realism to the Criticality Evaluation of a BWR Fuel Assembly Package** PETER VESCOVI*, TANYA SLOMA	
11:20am	ABSTRACT 189 **Perturbation Analysis for Demonstration of Reactivity in Criticality Safety Analyses** TANYA SLOMA*, PETER VESCOVI	
11:40am	ABSTRACT 351 **Criticality Assessments Using Polyurethane Foam** JAMES LAM*	
12:00pm	ABSTRACT 275 **Transport Criticality Assessment Methodologies the RWMD Spent Fuel Disposal Canister Transport Container** WILLIAM DARBY*	
12:20pm	ABSTRACT 418 **Fissile Exceptions – a General Scheme for Packages Based on CSI Control** NICHOLAS BARTON*, SAM DARBY, DENNIS MENNERDAHL, MICHELE NUTTALL	

T17 – Impact Limiter Materials / Structural Materials

11:00am – 12:20pm	CHAIR: PETER PURCELL, CO-CHAIR: ROBERT VAUGHAN	Conference Room 3
11:00am	ABSTRACT 48 **A Comparison between Mono-wall Body and Multi-wall Body Structures for a Large Scale Metal Cask** RYOJI ASANO*, YOSHIAKI MIYAJI, SHINTARO MIYAZAKI, AKIO NARA, HIROBUMI NUNOME	
11:20am	ABSTRACT 101 **Dynamic Fracture Toughness Tests of Dynamic Loaded Ductile Cast Iron** HANS-PETER WINKLER*, ROLAND HUGGENBERG, ANNETTE LUDWIG, GERHARD PUSCH, PETER TRUBITZ	

Daily Programme : Tuesday 05 October : continued

11:40am	ABSTRACT 93
	Investigation of Availability of Rigid Polyurethane Foam as Shock Absorbing Material for Heavy Cask
	JUN OKADA*, SATOSHI ASHIDA, AKIO OIWA, HIROAKI ARAI, MASAYUKI TANIGAWA
12:00pm	ABSTRACT 152
	Modeling of Polyurethane Foam Thermal Degradation within an Annular Region Subjected to Fire Conditions
	MILES GREINER, JIE LI, SHIU-WING TAM, YUNG LIU, ALLEN SMITH*
12:20pm	ABSTRACT 87
	Novel Reliable Hydrogen Risk Mitigation System for Transportation of Radioactive Materials
	V. ROHR*, M. PARADIS, E. BILLOU, J-M. MERIENNE, D. PINET
12:40am – 2:00pm	Lunch
2:00pm – 3:40pm	**Concurrent Panel Sessions**

P2 – Regulations - A Future Paradigm

2:00pm – 3:40pm	CHAIR: FRANK NITSCHE, CO-CHAIR: SYLVAIN FAILLE	Main Hall
	ABSTRACT 311	
	Compliance Assessment for the Safe Transport of Radioactive Material - Russia Practice and Perspectives	
	VLADIMIR ERSHOV*, GENNADY NOVIKOV	
	ABSTRACT 347	
	Lost and Found - Explanatory, Advisory and Fissile Materials	
	DENNIS MENNERDAHL*	
	ABSTRACT 405	
	How Specific Should Be The Regulations?	
	PIERRE MALESYS*	
	ABSTRACT 322	
	Risks and Regulations in the Transport of Nuclear Material by Sea	
	PHILIP ROCHE*	
	ABSTRACT 126	
	Regulation of the Transport of Radioactive Materials	
	GEORGE SALLIT*	
	ABSTRACT 28	
	Stability of Regulations Versus Confusion	
	FERNANDO ZAMORA*	

P1 – Denial and Delay of Shipments

2:00pm – 3:40pm	CHAIR: TBC, CO-CHAIR: CHRISTEL FASTEN	Conference Room 1
	ABSTRACTS 9/10	
	Denials and Delays on Class 7	
	ANA SOBREIRA, NATHALIA ALBA BRAGA*	
	ABSTRACT 204	
	Economic and Social Consequences of Denial and Delay of Shipments of Radioactive Material	
	MARIO MALLAUPOMA*, NATANAEL BRUNO, ANA SOBREIRA	
	ABSTRACT 160	
	Denial of Shipments	
	GEOFF LEACH*	
	ABSTRACT 383	
	The International Database of Problems Shipping Radioactive Material	
	JIM STEWART*	

Daily Programme : Tuesday 05 October : continued

	ABSTRACT 385 **Regional Networks and their Effectiveness at Combating Problems Shipping Radioactive Material** JIM STEWART*	
	ABSTRACT 404 **Denial - Where are We Now?** JIM STEWART*	
	ABSTRACT 37 **Difficulties in Transporting RAM - What it really means to Some People** STEVE WHITTINGHAM*	

P4 – Quantification of Safety in Transport

2:00pm – 3:40pm	CHAIR: MARC LEBRUN, CO-CHAIR: RUTH WEINER	Conference Room 2
	ABSTRACT 325 **Risk Based Model for Compliance Assurance Inspections for the Non-nuclear Sector** IAIN DAVIDSON*	
	ABSTRACT 113 **Guide for Risk Assessment Studies Required for Transport Infrastructures** FRANCK KALOUSTIAN*, LAURENCE GOZALO, MARIE-THERESE LIZOT, GILLES SERT, CHRISTOPHE GETREY	
	ABSTRACT 330 **A Multi-facet Approach for Evaluating Criticality Risks during Transportation of Commercial Spent Nuclear Fuel** ALBERT MACHIELS*, JOHN KESSLER	
	ABSTRACT 302 **Development of a Web-based Routing Tool for Road Transport of Hazardous Materials** THOMAS MCSWEENEY, JAMES SIMMONS, AUTHUR GREENBERG, WILLIAM QUADE*	
	ABSTRACT 419 **Findings from Non-nuclear Small User Inspections in 2009 / 2010** DAVID ROWE*	

P3 – Crush Testing of Lightweight Packaging

2:00pm – 3:40pm	CHAIR: MATTHEW FELDMAN, CO-CHAIR: MAKOTO HIROSE	Conference Room 3
	ABSTRACT 332 **Crush Testing at Oak Ridge National Laboratory** MATTHEW FELDMAN*	
	ABSTRACT 336 **What constitutes a Valid Crush Test?** GORDON BJORKMAN*	
	ABSTRACT 363 **Historical View and Experiences with the Crush Test for Lightweight Packages** MARKO NEHRIG*, FRANK WILLE, THOMAS QUERCETTI, JORG-PETER MASSLOWSKI, BERNHARD DROSTE	
	ABSTRACT 46 **Historical Background - Early Deliberations on and Assessments of the Need for a Dynamic Crush Test** RONALD POPE*, FRANK WILLE	
	ABSTRACT 21 **Crush Testing of 9977 General Purpose Fissile Packagings** ALLEN SMITH*	
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge

Daily Programme : Tuesday 05 October : continued

4:00pm – 6:00pm	**Concurrent Technical Sessions**

T20 – Long Term Storage Strategies

4:00pm – 6:00pm	CHAIR: YVES CHANZY, CO-CHAIR: TOSHIARI SAEGUSA	Main Hall
4:00pm	ABSTRACT 61	
	Ageing Management for Long Term Interim Storage Casks ANTON ERHARD*, HOLGER VOELZKE	
4:20pm	ABSTRACT 260	
	The BSK3 Concept for Direct Disposal of Spent Fuel in Salt Using Borehole Emplacement Technology STEFAN FOPP*, REINHOLD GRAF, WOLFGANG FILBERT, ROLAND HUEGGENBERG	
4:40pm	ABSTRACT 232	
	Impact on the Transportation Package Design for Transport First and Then Interim Storage Versus Interim Storage First and Transport PETER SHIH*, PRAKASH NARAYANAN	
5:00pm	ABSTRACT 163	
	Confirmation of Maintenance of Function for Transport After Long-term Storage Using Dry metal Dual Purpose Casks TADAYOSHI TAKAHASHI*, MITSUO MATSUMOTO, TAKESHI FUJIMOTO	
5:20pm	ABSTRACT 226	
	Considerations for Transportation Licensing of Used Fuel Already in Interim Dry Storage BONDRE JAYANT*, ROBERT GRUBB	
5:40pm	ABSTRACT 357	
	Advanced Solution for Used Fuel Management FREDERIC PATALAGOITY*, CAMILLE OTTON	

T18 – Waste Management

4:00pm – 6:00pm	CHAIR: DANNY VINCE, CO-CHAIR: MICHAEL CONROY	Conference Room 1
4:00pm	ABSTRACT 422	
	Soft Sided Packaging for Low Level and Hazardous Wastes MIKE SANCHEZ*, STUART BOWE, PAUL MISKIMIN	
4:20pm	ABSTRACT 224	
	Radioactive Waste Inventory Forecasting and Characterization Implications for Packaging and Transport MARC FLYNN*	
4:40pm	ABSTRACT 57	
	Moving a Mountain by Rail! ASHOK KAPOOR*, STEPHEN O'CONNOR, J. RITCHEY, W. RYAN, DONALD METZLER	
5:00pm	ABSTRACT 55	
	Identifying Opportunities for Process Improvements in Addressing Transportation Safety and Compliance Issues JULIA DONKIN*, DANA WILLAFORD	
5:20pm	ABSTRACT 54	
	Radioactive Waste and Fissile Exceptions BRUNO DESNOYERS*	
5:40pm	ABSTRACT 266	
	Optimization of Alpha Contaminated Waste Transportation in France JULIETTE VUONG*	

Daily Programme : Tuesday 05 October : continued

T37 – IP Drum Packages

4:00pm – 6:00pm	CHAIR: MIKE WANGLER, CO-CHAIR: NATANAEL BRUNO	Conference Room 2
4:00pm	ABSTRACT 45	
	Status of US Department of Energy Replacements for the DOT Specification 6M Shipping Containers	
	JEFFREY G. ARBITAL*, DREW WINDER, KENNETH E. SANDERS	
4:20pm	ABSTRACT 27	
	The Transport of Uranium Swarf Immersed in Oil	
	DAVID WINDLEY*	
4:40pm	ABSTRACT 112	
	Packing for Radioactive Waste Transport	
	ALBERTO ORSINI*, RENATO SANTINELLI, NADIA CHERUBINI, SANDRO RIZZO	
5:00pm	ABSTRACT 74	
	UK Low Level Waste Repository - Transport Package Designs Adapting to the Waste Management Hierarchy	
	MARC FLYNN*	
5:20pm	ABSTRACT 16	
	Development of a Specific Activity Distribution Estimation Method for Large Low-Level Radioactive Waste Using Shape Measurement Technique	
	MICHIYA SASAKI*, HARUYUKI OGINO, TAKATOSHI HATTORI	
5:40pm	ABSTRACT 415	
	An Overview of the Development of IP-2 packages ISO Freight Containers in the UK	
	ROBERT VAUGHAN*, R P HOWS	

T19 – Structural Benchmarking

4:00pm – 6:00pm	CHAIR: CHI-FUNG TSO, CO-CHAIR: KOJI SHIRAI	Conference Room 3
4:00pm	ABSTRACTS 26/115	
	From Experiment to an Appropriate Finite Element Model - Safety Assessment for Ductile Cast Iron Casks Demonstrated by Means of IAEA Puncture Drop Test	
	MIKE WEBER*, FRANK WILLE, VIKTOR BALLHEIMER, ANDRE MUSOLFF	
4:20pm	ABSTRACT 231	
	Verification of Computational Models by Comparison of Finite-Element Calculations and Experiments for the Model Cask CASTOR®HAW/TB2	
	WALTER VOELZER, STEPHAN GLUTSCH, RONNY PEREZ-KRETSCHMER, PAVEL VRASTIL	
4:40pm	ABSTRACT 355	
	Validation of Numerical Simulation Method Using a 1/3-Scale Model Drop Test of KN-18 SNF Transport Cask	
	KAP-SUN KIM*, JONG-SOO KIM, KYU-SUP CHOI, IN-SU JEONG	
5:00pm	ABSTRACT 222	
	Benchmarking of Analytical Methods and Analysis Software Used for Transportation Package Drop Analysis	
	RAHEEL HAROON*, PETER SHIH	
5:20pm	ABSTRACT 106	
	Verification of LS-DYNA Finite Element Impact Analysis by Comparison to Test Data and Classic First Principle Calculations	
	ANDREW LANGSTON*, VICTOR SMITH	
5:40pm	ABSTRACT 67	
	Numerical Simulation and Experimental Testing of Brit Lead Cask (BLC)	
	DHIREN SAHOO*, JOTIRAM MANE, VINAY BHAVE, PIYUSH SRIVASTAV, ANIL KOHLI	

Daily Programme : Wednesday 06 October

Wednesday 06 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant
8:00am – 6:00pm	Exhibition Open	Delegate Lounge
8:00am – 8:40am	**Morning Plenary**	Main Hall
	African Perspective SHAMSIDEEN B. ELEGBA, DIRECTOR GENERAL/CEO, NIGERIAN NUCLEAR REGULATORY AUTHORITY	
	The Challenges of Transporting Radioactive Material in South Africa BOYCE M. MKHIZE, CEO, SOUTH AFRICA NATIONAL NUCLEAR REGULATOR	
8:40am – 8:50am	Rapporteurs Update	Main Hall
9:00am – 10:40am	**Concurrent Technical Sessions**	

T23 – Competent Authority Activities

9:00am – 10:40am	CHAIR: FRANZ HILBERT, CO:CHAIR: FERNANDO ZAMORA	Main Hall
9:00am	ABSTRACT 7	
	The Association of European Competent Authorities for the Safe Transport of Radioactive Material STEVE WHITTINGHAM*, LORIS ROSSI	
9:20am	ABSTRACT 171	
	Development and Implementation of the "Joint Canada-United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages" MICHELE SAMPSON*, KARINE GLENN, MICHAEL CONROY	
9:40am	ABSTRACT 91	
	How the UK Competent Authority Has Developed a Risk Based Strategy for Carrying Out Non-Nuclear Small User Inspections MICHAEL TURNER*	
10:00am	ABSTRACT 51	
	Effective Analysis Submissions – A Regulator Perspective JOSEPH OYINLOYE*	
10:20am	ABSTRACT 17	
	Verification of Activity Release Compliance with Regulatory Limits within Spent Fuel Transport Casks Assessment ANNETTE ROLLE*, BERNHARD DROSTE, SVEN SCHUBERT, FRANK WILLE	

T24 – DRY STORAGE ISSUES

09:00am – 10:40am	CHAIR: HOLGER VOELZKE, CO-CHAIR: JAYANT BONDRE	Conference Room 1
9:00am	ABSTRACT 215	
	Meeting the Challenges of International Projects GARCIA JUSTO*	
9:20am	ABSTRACT 15	
	Kozloduy Dry Spent Fuel Storage ANDY ANDREWS*	
9:40am	ABSTRACT 338	
	Influence of ISFSI Design Parameters on the Seismic Response of Dry Storage Casks GORDON BJORKMAN*	
10:00am	ABSTRACT 408	
	Thermal Evaluation of Loading and Drying Operations of a High Capacity Spent Fuel Storage Canister MIKE YAKSH*, CHRISTINE WANG	

Daily Programme : Wednesday 06 October : continued

10:20am	ABSTRACT 217 **Applying Optimization Methods and Stochastic Analysis in Evaluating a Storage Accident** WALTER VOELZER*, ROBERT GARTZ, MATTHIAS HECK, THOMAS SEIDER, MARCO GROSSE	

T22 – Sources and Radiopharmaceuticals

9:00am – 10:40am	CHAIR: PAUL GRAY, CO-CHAIR: TBC	Conference Room 2
9:00am	ABSTRACT 233 **Transport of Radiopharmaceuticals, Cradle to the Patient** CHARLIE CARRINGTON, EUGENIE ROELOFSEN (PRESENTED BY ROB DEKKERS)	
9:20am	ABSTRACT 64 **Existing Practices for Safe Transport of Radioactive Sources in Bangladesh** ABDUS SATTAR MOLLAH*	
9:40am	ABSTRACT 395 **Controlling Sources and the Transport Implications** JIM STEWART*	
10:00am	ABSTRACT 229 **Sustaining Reliable Maritime Shipments of Radioactive and Nuclear Materials** PETER LAMBOURNE*	
10:20am	ABSTRACT 47 **Packaged Material in Foreign Countries** CRISTY ABEYTA*, JAMES MATZKE	

T21 – Shielding Calculations

9:00am – 10:20am	CHAIR: TBC, CO-CHAIR: CATHERINE WEBER-GUEVARA	Conference Room 3
9:00am	ABSTRACT 23 **Comparison of Monte Carlo Codes MCNP and MONACO for Applying to Shielding Calculation of Transport/Storage Cask** HIROAKI TANIUCHI*	
9:20am	ABSTRACT 76 **A Comparison of the TRITON and ORIGEN2 Source Generation Programs** RICK MIGLIORE, PHILIP NOSS*	
9:40am	ABSTRACT 324 **Study of Cross Section Libraries for Shielding Design of Spent Fuel Cask and Cask Storage Facility** TAKUYA TAKAHASHI*	
10:00am	ABSTRACT 228 **Impact of Higher Burn-ups on the Transportation Package Design: Radiation Shielding Perspective** PRAKASH NARAYANAN*	
10:40am – 11:00am	Refreshment Break	Delegate Lounge

11:00 am – 1:00pm	**Concurrent Technical Sessions**	

T27 – Denials of Shipment

11:00am – 1:00pm	CHAIR: STEVE WHITTINGHAM, CO-CHAIR: KASTURI VARLEY	Main Hall
11:00am	ABSTRACT 361 **Maritime Shipments of Radioactive Materials** STEFAN HOEFT*	

Daily Programme : Wednesday 06 October : continued

11:20am	ABSTRACT 190
	Denial of Shipment of Radioactive Material
	PAUL GRAY*, GRANT MALKOSKE
11:40am	ABSTRACT 38
	The Role of National Authorities in Minimizing Denials of Shipments
	NAT BRUNO*, ARANGURAN NANDAKUMAR, MICHAEL WANGLER
12:00pm	ABSTRACT 135
	Experience on the Management of the Regional Network in the Mediterranean Basin for the Denials of Shipments of Radioactive Material
	SANDRO TRIVELLONI*, MARTIN FERNANDO ZAMORA, BERNARD MONOT
12:20pm	ABSTRACT 253
	Nuclear Renaissance, Nuclear Transports : the Communication Challenge: Front end Experience
	BERNARD MONOT*
12:40pm	ABSTRACT 425
	Documentation Requirements for Class 7 Transports
	ROB VAN UFFELEN*

T26 – Non Spent Fuel Package Design

11:00am – 1:00pm	CHAIR: KEN SORENSON, CO-CHAIR: TBC	Conference Room 1
11:00am	ABSTRACT 366	
	Cage Design, Impact Analysis and Experimental Testing of Teletherapy Source Transportation Flask	
	J. V. MANE*, S. SHARMA, V. M. CHAVAN, D. C. KAR, R. G. AGRAWAL	
11:20am	ABSTRACT 73	
	The DN30 Overpack - a New Solution for the Transport of Commercial Grade and Reprocessed Enriched UF6	
	FRANZ HILBERT, WOLFGANG BERGMANN*, FREDERIC NOYON	
11:40am	ABSTRACT 147	
	Recent Approval of the UX-30 as a Type B Package	
	MARK WHITTAKER*	
12:00pm	ABSTRACT 138	
	FCC NG – Transatlantic Design	
	FRANCOIS MARVAUD*, PASCALE FAYE, MICHEL DOUCET	
12:20pm	ABSTRACT 201	
	Outline of Fresh MOX Fuel Transportation in Japan and Development Status of Transportation Cask for LWR	
	NORIHIKO TAMAKI*, AKIRA OE, KAZUNARI ONISHI	
12:40pm	ABSTRACT 32	
	CASTOR® HAW28M – Development and Licensing of a Cask for Transport and Storage of Vitrified High Active Waste Containers	
	ANDRE VOSSNACKE*, RAINER NOERING	

T28 – Impact Testing

11:00am – 1.00pm	CHAIR: KARSTEN MUELLER, CO-CHAIR: TOM DANNER	Conference Room 2
11:00am	ABSTRACT 359	
	Package Testing To Demonstrate Safety with Added Features	
	PIERRE MALESYS*	
11:20am	ABSTRACT 149	
	Certification Testing of the TRUPACT–III Package	
	RICHARD J. SMITH*, PHILIP W. NOSS	

Daily Programme : Wednesday 06 October : continued

11:40am	ABSTRACT 81 **Prototype Test of a New MOX Powder Transport Packaging** YASUHIRO KAWAHARA*, TOKUO TAKE, TAKAFUMI KITAMURA, KAN SHIBATA, YUICHIRO OUCHI	
12:00pm	ABSTRACT 172 **Drop Test Program with the Half-Scale Model CASTOR HAW/TB2** ANDRE MUSOLFF*, THOMAS QUERCETTI, KARSTEN MUELLER, BERNHARD DROSTE, STEFFEN KOMANN	
12:20pm	ABSTRACT 220 **Mechanical Safety Analyses of Cast Iron Containers for the KONRAD Repository** UWE ZENCKER*, MIKE WEBER, LINAN QIAO, BERNHARD DROSTE	
12:40pm	ABSTRACT 373 **Demonstration of the Impact Performance of the Windscale Pile Fuel and Isotope Waste Package** CHI-FUNG TSO*, JOHN CLIFFORD	

T25 – Shielding Materials, Basket Materials

11:00am – 12:40pm	CHAIR: BERNHARD DROSTE, CO-CHAIR: HERVE ISSARD	Conference Room 3
11:00am	ABSTRACT 90 **A New Use for the Vyal-B Neutron Absorbing Resin** GUILLAUME FOUSSARD*	
11:20am	ABSTRACT 173 **Radiation Induced Structural Changes of (U)HMW Polyethylene with Regard to its Application for Radiation Shielding** KERSTIN VON DER EHE*, MATTHIAS JAUNICH, DIETMAR WOLFF, MARTIN BOEHNING, HARALD GOERING	
11:40am	ABSTRACT 136 **Thermal Ageing of Vinylester Neutron Shielding Used in Transport/Storage Casks** FIDELE NIZEYIMANA*, V.BELLENGER , PASCALE ABADIE, HERVE ISSARD	
12:00pm	ABSTRACT 70 **Experimental Study of Heat Removal Ability and Lead Slump of Lead-Type Multi-Wall Cask** SATOSHI ASHIDA*, JUN OKADA, SHINTARO MIYAZAKI, KOJI KITAMURA, DONG HUI MA	
12:20pm	ABSTRACT 262 **The Role of Metamic®-HT – Industry's First Nano-Particle Based Material – in Fuel Basket Design** K.P. SINGH, I. RAMPALL, T. G. HAYNES (PRESENTED BY WILIAM WOODWARD)	
12:40pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	**Concurrent Panel Sessions**	

P5 – Liability and Insurance

2:00pm – 3:40pm	CHAIR: DONNA GOERTZEN, CO-CHAIR: SERGE GORLIN	Main Hall
	ABSTRACT 285 **Nuclear and Third Party Liability Insurance for Nuclear Transport** MIKE PEACH MARK RICHARDS REGIS MAHIEU ANTHONY WETHERALL & NATHALIE HORBACH	

P8 – Public Acceptance

2:00pm – 3:40pm	CHAIR: BERNARD MONOT, CO-CHAIR: LORNE GREEN	Conference Room 1
	ABSTRACT 240 **Communicating the International Transport by Sea of Nuclear Material** RUPERT WILCOX-BAKER*	

Program

Daily Programme : Wednesday 06 October : continued

	ABSTRACT 241
	Public Acceptability for International Sea Shipments of High Level Waste and MOX Fuel
	GAVIN CARTER (PRESENTED BY ALISTAIR BROWN)
	ABSTRACT 271
	Transparency of Operations – Working with Stakeholder Groups
	PAUL HARDING*, HENRY-JACQUES NEAU
	ABSTRACT 384
	A Communication Tool-Kit to Combat Problems Shipping Radioactive Material
	JIM STEWART*
	ABSTRACT 382
	The Benefits of Simple Short and Clear Training for Targeted Audiences
	KASTURI VARLEY*

P9 – Long Term Storage and Transport – Technical Issues

2:00pm – 3:40pm	CHAIR: STEVE BELLAMY, CO-CHAIR: TARA NEIDER	Conference Room 2
	ABSTRACT 34	
	Code Cases of Basket Material for Spent Fuel Transport/Storage Packagings in the Japan Society of Mechanical Engineers	
	MAKOTO HIROSE*, TOSHIARI SAEGUSA, KATSUHIKO SHIGEMUNE	
	ABSTRACT 348	
	How to Transport a Cask Which Has Been Loaded Then Stored for Several Decades?	
	PIERRE MALESYS*	
	ABSTRACT 354	
	A Preliminary Look at Used Nuclear Fuel Transportation Options to a Repository Site in Canada	
	ULF STAHMER*	
	ABSTRACT 157	
	Accelerated Corrosion Testing of Aluminum/Boron Carbide Metal Matrix Composite in Simulated PWR Spent Fuel Pool Solution	
	DAISUKE NAGASAWA*, HIDEKI ISHII, KAZUTO SANADA, VALENTIN ROHR, HERVE ISSARD	

P11 – Management Controls

2:00pm – 3:40pm	CHAIR: MICHEL HARTENSTEIN, CO-CHAIR: TBC	Conference Room 3
	ABSTRACT 234	
	Emergency Response without Borders	
	ALAN BACON*	
	ABSTRACT 3	
	Transport of Radiopharmaceuticals and Labelled Compounds in Cuba	
	ZAYDA HAYDEE AMADOR BALBONA*, SAUL PEREZ PIJUAN, MIRTA BARBARA TORRES BERDEGUEZ, FERNANDO ENRIQUE AYRA PARDO	
	ABSTRACT 161	
	The Role of the Dangerous Goods Safety Adviser and Improving Compliance with the Radioactive Material Road Transport Regulations Amongst Users in the GB Industrial Sector	
	SIMON JAKES*	
	ABSTRACT 273	
	10 CFR PART 71 Quality Assurance and Inspection Experience	
	EARL LOVE (PRESENTED BY ROBERT TEMPS)	
	ABSTRACT 392	
	Applying the Good Practices Identified in IAEA TranSAS Missions	
	JIM STEWART*	

Daily Programme : Wednesday 06 - Thursday 07 October

	ABSTRACT 65	
	Canadian Emergency Response Requirements and Cameco's Experience	
	JOHN ZAIDAN*, MARC-ANDRE CHARETTE	
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge
4:00pm – 6:00pm	**Poster Session**	Delegate Lounge & IMO Foyer

Thursday 07 October

7:00am – 6:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant
8:00am – 5:40pm	Exhibition Open	Delegate Lounge
8:00am – 8:40am	**Morning Plenary**	Main Hall
	Security of the Transport of Radioactive Materials	
	ANITA BIRGITTA NILSSON, DIRECTOR OF OFFICE OF NUCLEAR SECURITY, DEPARTMENT OF NUCLEAR SAFETY AND SECURITY, INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)	
9:00am – 10:40am	**Concurrent Technical Sessions**	

T31 – Regulations and Guidance

9:00 – 10:40	CHAIR: FRANK NITSCHE, CO-CHAIR: GEORGE SALLIT	Main Hall
9:00am	ABSTRACT 397	
	Clear Regulations	
	JIM STEWART*	
9:20am	ABSTRACTS 118/380	
	The Environmental Conditions Experienced by Packages during Routine Transport	
	SARAH FOURGEAUD*, KARIM BEN OUAGHREM, GILLES SERT, IGOR LE BARS, JIM STEWART	
9:40am	ABSTRACT 129	
	Onsite Transport Regulations: How to Adapt International Regulations?	
	LAURENT HANSEL*, YVES CHANZY	
10:00am	ABSTRACT 371	
	TCSC 1086: Good Practice Guide to Drop Testing of Type B Transport Packages	
	CHI-FUNG TSO*, BILL SIEVWRIGHT	
10:20am	ABSTRACT 368	
	The Facilitation Of Criticality Safety Assessments For Fuel Assemblies	
	MICHEL DOUCET (PRESENTED BY SAM DARBY)	

T30 – Emergency Response (Session 1)

9:00am – 10:20am	CHAIR: BETTY BONNARDEL-AZZARELLI, CO-CHAIR: GILLES SERT	Conference Room 1
9:00am	ABSTRACT 56	
	Transport Emergency Preparedness – Lessons learned from WNTI	
	MARC FLYNN*	
9:20am	ABSTRACT 314	
	RASAFE: Meeting the Industry Needs for Transport Emergency Arrangements	
	TERENCE KELLY*, JONATHAN HARRISON, GARETH DAVIES, ANTHONY WETHERALL	
9:40am	ABSTRACT 281	
	DSTL RADSAFE Exercise	
	BRIAN CORBETT*, WILLIAM BLANCHARD	
10:00am	ABSTRACT 166	
	Technical Basis for Transport of Radioactive Materials Emergency Planning	
	SANDRO TRIVELLONI*, LUCIANO BOLOGNA, GIORGIO PALMIERI, ANTONIO SANTILLI, PAOLO ZEPPA	

Daily Programme : Thursday 07 October : continued

T29 – Package Design and Strategies (Session 1)

9:00am – 10:40am	CHAIR: MALCOLM MILLER, CO-CHAIR: HEINZ GEISER	Conference Room 2
9:00am	ABSTRACT 105	
	Considerations in Developing a New Fissile Transport Package	
	TIM GLEED-OWEN*	
9:20am	ABSTRACT 417	
	Multiple Barriers: Application to Package Design for Used Fuel Elements	
	STEPHANE BRUT*	
9:40am	ABSTRACT 134	
	Description of Fuel Integrity Project Methodology Principles	
	MAURICE DALLONGEVILLE*, ARAVINDA ZEACHANDIRIN, PETER PURCELL, ANTHONY CORY	
10:00am	ABSTRACT 369	
	Transportation Implications of a Closed Fuel Cycle	
	RUTH WEINER*, KEN SORENSON, MATTHEW DENNIS, SAMUEL BAYS, MILES GREINER	
10:20am	ABSTRACT 162	
	Current Practise and Experience of Shipping Bulk Powders and How this is Relevant to the Transport of Uranium Ore Concentrates	
	MARC-ANDRE CHARETTE*, AL STRATEMEYER, GUY KARRER	

T32 – Seal Behaviour

9:00 – 10:20	CHAIR: BILL SIEVWRIGHT, CO-CHAIR: HANS-PETER WINKLER	Conference Room 3
9:00am	ABSTRACT 25	
	Influence of Mechanical Vibration in Transport on Leak-Tightness of Metal Gasket in Transport/Storage Cask for Spent Nuclear Fuel	
	TOSHIARI SAEGUSA*, KOJI SHIRAI, HIROFUMI TAKEDA, MASUMI WATARU, KOSUKE NAMBA	
9:20am	ABSTRACT 41	
	The Influence of Thermal Expansion on Package Tightness during Fire Test	
	FRANK KOCH*, JENS STERTHAUS, CLAUS BLETZER	
9:40am	ABSTRACT 108	
	Evaluation of Sealing Performance of Metal Cask Subjected to Vertical and Horizontal im-pact Load due to Aircraft Engine Crash	
	KOJI SHIRAI*, TOSHIARI SAEGUSA, KOSUKE NAMBA	
10:00am	ABSTRACT 8	
	Non Competent Authority Approved Packages – Methods for Leak Testing	
	GERRY HOLDEN*, MARC FLYNN	
10:40am – 10:00am	Refreshment Break	Delegate Lounge
11:00am – 1.00pm	**Concurrent Technical Sessions**	

T39 – Radiation Protection Issues

11:00am – 12:40pm	CHAIR: ASHOK KAPOOR, CO-CHAIR: FLORENTIN LANGE	Main Hall
11:00am	ABSTRACT 396	
	The Interface between the IAEA Basic Safety Standards for Radiation Protection and the IAEA Transport Regulations	
	JIM STEWART	
11:20am	ABSTRACT 130	
	The Revision of the New International Basic Safety Standards and its Effect on the IAEA Regulations for the Safe Transport of Radioactive Material	
	CHRISTEL FASTEN, FRANK NITSCHE	

Daily Programme : Thursday 07 October : continued

11:40am	ABSTRACT 168 **Review of Methodologies and Development of Software to Calculate A1 and A2 and Exemption Values** TIBERIO CABIANCA*, KELLY JONES, MIKE HARVEY, TRACEY ANDERSON, IAIN BROWN
12:00pm	ABSTRACT 251 **Review of Material Requirements of the IAEA Transport Regulations for LSA-II and LSA-III** WENZEL BRUCHER*, UWE BUTTNER, FLORENTIN LANGE
12:20pm	ABSTRACTS 390/391 **The Results of a Coordinated Research Project into the Surface Contamination of Packages / The Practical Application of the Results of a Coordinated Research Project into the Surface Contamination of Packages** YONGHANG ZHAO*
12:40pm	ABSTRACT 399 **Safety of Transport of Naturally Occurring Radioactive Material** KASTURI VARLEY*, ULRIC SCHWELA, TIBERIO CABIANCA

T36 – Structural Analysis

11:00am – 1:00pm	CHAIR: GORDON BJORKMAN, CO-CHAIR: UWE ZENCKER	Conference Room 1
11:00am	ABSTRACT 303 **Numerical Analysis on Ship-Ship Collision Resistance in Design of ‘KAIEI-MARU’ Classified as INF 3 Ship** AKIHIRO YASUDA*	
11:20am	ABSTRACT 372 **Analyses to Demonstrate the Structural Performance of the KN18 in Hypothetical Drop Accident Scenarios** CHI-FUNG TSO*, KAP-SUN KIM, JONG-SOO KIM, KYU-SUP CHOI	
11:40am	ABSTRACT 409 **Structural Evaluation of a Shielded Transfer Cask System for Intra Plant Spent Fuel Transfer** MIKE YAKSH*, MARC GRISWOLD	
12:00pm	ABSTRACT 92 **Mechanical Assessment Criteria of Spent Fuel Assemblies Basket Design** CHRISTIAN KUSCHKE*, VIKTOR BALLHEIMER, FRANK WILLE, STEFFEN KOMANN	
12:20pm	ABSTRACT 205 **Analysis Methodology and Assessment Criteria for Bolted Trunnion Systems of Type B Packages for Radioactive Materials** JENS STERTHAUS*, VIKTOR BALLHEIMER, FRANK WILLE	
12:40pm	ABSTRACT 133 **Numerical Simulation of Dynamic Deformation of Air Transport Package in High-Speed Accidental Impact** ALEXANDER RYABOV*, VLADIMIR ROMANOV, SERGEY KUKANOV, VALENTIN SPIRIDONOV, DENIS DYANOV	

T7 – Fissile Exceptions

11:00am – 1:00pm	CHAIR: DENNIS MENNERDAHL, CO-CHAIR: HIROAKI TANIUCHI	Conference Room 2
11:00am	ABSTRACTS 388/389 **Influence on Transport of Fissile Material by Proposed Changes to TS-R-1** YONGHANG ZHAO*, JIM STEWART	
11:20am	ABSTRACT 328 **Overview of Proposed Modifications for Exceptions to the Requirements for Transport of Fissile Material** CECIL PARKS*, NICHOLAS BARTON, SAM DARBY, BRUNO DESNOYERS, MAKOTO HIROSE	

Daily Programme : Thursday 07 October : continued

11:40am	ABSTRACT 127	
	Changes in the Transport of Fissile Material Resulting From the Latest Proposed Revision of the IAEA Transport Regulations	
	INGO REICHE*, FRANK NITSCHE	
12:00pm	ABSTRACT 403	
	Competent Authority Approved Fissile Exceptions - one Regulator's View	
	NICHOLAS BARTON*	
12:20pm	ABSTRACT 365	
	Why Considering CH2 Moderation for Excepted Fissile Material?	
	IZASKUN ORTIZ DE ECHEVARRIA DIEZ*, LUDYVINE JUTIER, STEPHANE EVO	
12:40pm	ABSTRACT 188	
	Bases for the General Licenses for Fissile Material and Exemptions from Classification as Fissile Material in 10 CFR Part 71	
	JEREMY SMITH*, ANDREW BARTO, CECIL PARKS	

T40 – Characterisation of Energy Absorbers

11:00am – 12:40pm	CHAIR: PETER SHIH, CO-CHAIR: WALTER VOELZER	Conference Room 3
11:00am	ABSTRACT 153	
	Evaluation of Influence of Temperature Below 80°C and Strain Rate on Compressive Property of Wood for Shock Absorber	
	KOJI SHIRAI,* KOSUKE NAMBA, YOSHIYUKI FUJITA	
11:20am	ABSTRACT 175	
	Contribution to Further Development of Simulation Methods for Impact Limiting Materials and Structures - a Report on the Situation from the German Quest-Project	
	EGBERT SCHOPPHOFF*, ROGER VALLENTIN, MANFRED STEEGMANNS, ROLAND HUEGGENBERG	
11:40am	ABSTRACT 367	
	Waste Container Drop Tests onto a Concrete Target	
	THOMAS QUERCETTI*, ANDRE MUSOLFF, BERNHARD DROSTE, NAKAGAMI MOTONORI, KYOSUKE FUJISAWA	
12:00pm	ABSTRACT 114	
	Effect of Dynamic Loading on Compressional Behaviour of Damping Concrete	
	EVA KASPAREK*, ROBERT SCHEIDEMANN, UWE ZENCKER, DIETMAR WOLFF, HOLGER VOELZKE	
12:20pm	ABSTRACT 123	
	A Material Testing Program to Characterize the Concrete Behavior under Static and Dynamic Loads	
	JOE MAGALLANES, RUBENS MARTINEZ, ALOYSE NESER*, DIETMAR SCHREIBER, UWE ZENCKER, MIKE WEBER	
12:40pm – 2:00pm	Lunch Break	
2:00pm – 3:40pm	**Concurrent Panel Sessions and Technical Session**	

P7 – Long Term Storage and Transport – Regulatory Issues

2:00pm – 3:40pm	CHAIR: JIM STEWART, CO-CHAIR: DOUGLAS AMMERMAN	Main Hall
	ABSTRACT 139	
	Long Term Storage of Used Nuclear Fuel	
	KEN SORENSON, HOLGER VOELZKE, TOSHIARI SAEGUSA, MIKE WATERS	

P6 – Security Issues

2:00pm – 3:40pm	CHAIR: ANN-MARGRETH ERIKSSON EKLUND, CO-CHAIR: YUNG LIU	Conference Room 1

Daily Programme : Thursday 07 October : continued

ABSTRACT 299
Developing a Memorandum of Understanding Regarding Transportation Security in the United States
JOHN AHERNE*, RICK BOYLE, AL TARDIFF

ABSTRACT 245
Securing the Transport of Nuclear or Radioactive Material
BRUNO AUTRUSSON*, OLIVIER LOISEAU, PIERRE FUNK

ABSTRACTS 297/298
Transportation Security Rulemaking Activities at the US Nuclear Regulatory Commission
RICHARD CORREIA, MARK SHAFFER, MICHAEL LAYTON, ADELAIDE GIANTELLI*

ABSTRACT 213
Radioactive and Nuclear Material Transport Security
ANN-MARGRETH ERIKSSON EKLUND*, RICHARD RAWL

ABSTRACT 214
The IAEA Assistance and Training Programme for Transport Security
RICHARD RAWL, ANN-MARGRETH ERIKSSON EKLUND (PRESENTED BY MARK HAWK)

ABSTRACT 193
Report on Radio Frequency Identification 2010 Category I Vault Testing Program
RICHARD KOENIG*, TERENCE WILLONER, HANCHUNG TSAI, YUNG LIU, DANIEL LEDUC

P10 – Emerging Regulatory Issues

2:00pm – 3:40pm CHAIR: STEVE WHITTINGHAM, CO-CHAIR: EARL EASTON — Conference Room 2

ABSTRACT 326
A Discussion on the Secure Stowage of Packages
IAIN DAVIDSON*

ABSTRACT 39
Use of Vehicle Radiation Portal Monitors and Transport Regulations in Canada
SYLVAIN FAILLE*

ABSTRACT 413
Designing Tie Down Systems for Heavy RAM Packages - Should Revised Design Criteria Apply?
PETER PURCELL*

ABSTRACT 200
Consideration on Safety Requirement for Large Component Transport with Q System
HIROSHI SUZUKI*, HIROMITSU MOCHIDUKI, MAKOTO HIROSHI, MANABU URAGAMI, MASANORI ARITOMI

ABSTRACT 36
Transport of Abnormal Indivisible Radioactive Loads
DANNY VINCE*

T45 – Finite Element Modelling – ASME

2:00pm – 3:40pm CHAIR: FRANK WILLE, CO-CHAIR: MIKE YAKSH — Conference Room 3

ABSTRACT 400
Flat Plate Puncture Test Convergence Study
DOUGLAS AMMERMAN*, MIKE YAKSH, CHI-FUNG TSO, DAVID MOLITORIS, SPENCER SNOW

ABSTRACT 377
Propped Cantilever Mesh Convergence Study Using Hexahedral Elements
CHI-FUNG TSO*, DAVID MOLITORIS, SNOW SPENCER, DOUG AMMERMAN

ABSTRACTS 335/340
Mesh Convergence Studies for Thin Shell Elements, Developed by the ASME Task Group on Computational Modeling
GORDON S. BJORKMAN*, DAVID P. MOLITORIS

Daily Programme : Thursday 07 October : continued

	ABSTRACT 353	
	Use of Computational Modeling Software for Evaluation of Structural Integrity	
	JASON PIOTTER*	
	ABSTRACT 219	
	Finite Element Mesh Design of a Cylindrical Cask under Puncture Drop Test Conditions	
	UWE ZENCKER*, MIKE WEBER, FRANK WILLE	
3:40pm – 4:00pm	Refreshment Break	Delegate Lounge
4:00pm – 5:40pm	**Concurrent Technical Sessions**	

T35 – Transport Systems

4:00pm – 5:40pm	CHAIR: JUSTO GARCIA, CO-CHAIR: TBC	Main Hall
4:00pm	ABSTRACT 362	
	Land Transport Issues for the Industry	
	DONNA GOERTZEN*	
4:20pm	ABSTRACT 292	
	The Safe, Secure, and Efficient Transportation for Shipping Urania ("Yellow Cake") From the Republic of Kazakhstan to Western Europe	
	AARON WIENER*	
4:40pm	ABSTRACT 306	
	Designing, Building and Delivering a Modern Approach to Consigning Radioactive Materials	
	MARTIN PORTER*, SONYA GRATTAN, ANGELA PARKER	
5:00pm	ABSTRACT 98	
	A New Information System for Transportation	
	PHILIPPE BAGONNEAU*	
5:20pm	ABSTRACT 116	
	Approach for Safe Transport of the Sample Including Nuclear Material	
	KEIICHI MORITA*, TADAHIKO YAMASHITA, DAISUKE TOGURI	

T34 - Emergency Response (Session 2)

4:00pm – 5:40pm	CHAIR: NICHOLAS BARTON, CO-CHAIR: VERONIQUE BAYLAC	Conference Room 1
4:00pm	ABSTRACT 264	
	The Macarthur Maze and Newhall Pass Fires and their Implications for Spent Fuel Transport	
	EARL EASTON*, CHRIS BAJWA	
4:20pm	ABSTRACT 119	
	Lessons from Transport Events Involving Radioactive Materials Occurred in France between 1999 and 2009	
	LAURE CARENINI*, GILLES SERT, MARIE-THERESE LIZOT, CLAIRE SAURON	
4:40pm	ABSTRACT 376	
	Reviewing the Impact of the Revised INES Manual on Transport Activities	
	GARRY OWEN*	
5:00pm	ABSTRACT 398	
	A Transport Risk Assessment Package	
	KASTURI VARLEY*, ARUNGUNRAM NAGARAJAN NANDAKUMAR, CLIFFORD JÃRNRY	
5:20pm	ABSTRACT 53	
	Hazardous Materials Commodity Flow Survey	
	WILLIAM SPURGEON*	

Daily Programme : Thursday 07 October : continued

T33 – Package Design and Strategies (Session 2)

4:00pm – 5:40pm	CHAIR: SANDRO TRIVELLONI, CO-CHAIR: VLADIMIR ERSHOV	Conference Room 2
4:00pm	ABSTRACT 104	
	Innovation: Ahead of the Pack(aging)	
	MICHEL HARTENSTEIN*, CELINE FONTANET, HERVE ISSARD	
4:20pm	ABSTRACT 274	
	Developments of New Radioactive Transport Packages of Type B within the Current EMBAL Plan in CEA	
	EMMANUEL RIGAUT*, ALAIN JOUDON, SEBASTIEN CLAVERIE-FORGUE, THOMAS CUVILLIER	
4:40pm	ABSTRACT 254	
	Transport: A Most Sensitive Link for the Nuclear Industry	
	MARC LEBRUN*, PASCAL CHOLLET	
5:00pm	ABSTRACT 370	
	Transportation Scenarios for Risk Analysis	
	RUTH WEINER*	
5:20pm	ABSTRACT 178	
	Risk Management in the Design, Licensing and Fabrication	
	CHARLES TEMUS, RICHARD J. SMITH*	

T38 – Spent Fuel Transport

4:00pm – 5:20pm	CHAIR: ANTONY CORY, CO-CHAIR: TAKAFUMI KITAMURA	Conference Room 3
4:00pm	ABSTRACT 293	
	Italian-French Experience in the Transportation of INF for Reprocessing	
	FERNANDA DI GASBARRO, JEAN PASCHAL, DAMIEN SICARD, ROBERT VESPA, ROBERTO DONATI, GIANRICO LOMBARDI*, MARTINE VALLETTE-FONTAINE	
4:20pm	ABSTRACT 121	
	Return of the Fuel from the German Compact Sodium-Cooled Nuclear Facility KNK II with the CASTOR® KNK	
	ROGER VALLENTIN*, IRIS GRAFFUNDER, OLIVER PATZOLD, DIETMAR BRAUER	
4:40pm	ABSTRACT 350	
	Transport of Irradiated Fuel Pins	
	XAVIER BAIRIOT*, FABIEN LABERGRI	
5:00pm	ABSTRACT 296	
	Air Shipment of Spent Nuclear Fuel from Romania to Russia	
	IGOR BOLSHINSKY, KEN ALLEN*, LUCIAN BIRO, ALEXANDER BUCHELNIKOV	

6:45pm – 11:30pm	**PATRAM 2010 Gala Dinner**	Royal Courts of Justice

Daily Programme : Friday 08 October

Friday 08 October

7:00am – 12:00pm	Registration Open	IMO Main Foyer
7:00am – 8:00am	Speakers' Breakfast	4th Floor Restaurant
8:00am – 12:00pm	Exhibition Open	Delegate Lounge
8:45am – 9:00am	**Rapporteurs' Summing Up**	Main Hall
9:00am – 10:40am	**Concurrent Technical Sessions**	

T42 – Marine Transport

9:00am – 10:20am	CHAIR: NAOTERU ODANO, CO-CHAIR: ALASTAIR BROWN	Main Hall
9:00am	ABSTRACT 381	
	Responding to a Maritime Emergency Involving Radioactive Material	
	KASTURI VARLEY*	
9:20am	ABSTRACT 145	
	Planning, Licensing, Modifying, and Using a Russian Vessel for Shipping Spent Nuclear Fuel by Sea in Support of the DOE RRRFR Program	
	MICHAEL TYACKE*, IGOR BOLSHINKSY, SERGEY NALETOV, WLODZIMIER TOMCZAK	
9:40am	ABSTRACT 261	
	The Logistic of the MOX Transport from France to Japan	
	PATRICE FORTIER*, YLAN TOUBOL	
10:00am	ABSTRACT 349	
	The R 74 Packaging : the Lightweight Packaging for HLW	
	XAVIER BAIRIOT*, FABIEN LABERGRI	

T43 – Transport Experience

9:00am – 10:40am	CHAIR: ERIK WELLEMANN, CO-CHAIR: DAMIEN SICARD	Conference Room 1
9:00am	ABSTRACT 66	
	Loss of Control Incidents in Transportation: Challenges and Opportunities	
	SHAMSIDEEN ELEGBA*, NASIRUDEEN BELLO	
9:20am	ABSTRACT 146	
	Safe and Secure Life Cycle Management of Radioactive Sources	
	GRANT MALKOSKE*, PAUL GRAY	
9:40am	ABSTRACT 235	
	Anomolies and Challenges of the IAEA Regulations that Effect the Transportation of Radiopharmaceuticals	
	CHARLIE CARRINGTON*, EUGENIE ROELOFSEN	
10:00am	ABSTRACT 406	
	Radiation Level Changes at RAM Package Surfaces	
	ERICH OPPERMAN*, MARK HAWK, RONALD NATALI, ASHOK KAPOOR	
10:20am	ABSTRACT 386	
	The Results of a Coordinated Research Project into the Severity of Air Accidents	
	JIM STEWART*	

T44 – Manufacturing

9:00am – 10:40am	CHAIR: RYOJI ASANO, CO-CHAIR: STEPHANE COMPERE	Conference Room 2
9:00am	ABSTRACT 14	
	Ignalina NPP Defuelling and Dry Fuel Storage Systems	
	GARETH WATKINS*	
9:20am	ABSTRACT 97	
	Manufacturing of a New Transport Cask for MOX and UO2 Fuel	
	RIPERT HERVE*	

Daily Programme : Friday 08 October : continued

9:40am	ABSTRACT 242	
	CASTOR HAW28M - Fabrication and Cold Trials of a Cask for Transport and Storage of Vitrified High Active Waste Containers	
	ANDRE VOSSNACKE*, THOMAS HORN	
10:00am	ABSTRACT 268	
	Performance and Restrictions of Non-destructive Testing (NDT) Within the Quality Surveillance During Manufacturing of Type B- Packages	
	UVE GUENTHER*, MANFRED DR. BADEN, STEFFEN KOMANN, THILO NITZ	
10:20am	ABSTRACT 412	
	Radioactive Packaging Spares Management	
	DAVID MCWILLIAM*, GEOFF ROBINSON	

T41 – Seals, SNF Encapsulation

9:00am – 10:20am	CHAIR: TBC, CO-CHAIR: SARAH FOURGEAUD	Conference Room 3
9:00am	ABSTRACT 151	
	Alternative Frequency for Periodic Leak Rate Testing	
	SHIU-WING TAM*, HANCHUNG TSAI, YUNG LIU, JIM SHULER, YUNG LIU	
9:20am	ABSTRACT 169	
	Understanding the Low Temperature Properties of Rubber Seals	
	MATTHIAS JAUNICH*, KERSTIN VON DER EHE, DIETMAR WOLFF, HOLGER VOELZKE, WOLFGANG STARK	
9:40am	ABSTRACT 334	
	Long-Term Mechanical Behaviour of Rubber O-Rings for MOX Fresh Fuel Transport Packages by Experiment and FEM Numerical Simulation	
	AKIHIRO MATSUDA*	
10:00am	ABSTRACT 71	
	Encapsulation of Fuel Rods for Transport	
	SIMON STANKE*, FRANZ HILBERT	
10:40am – 11:00pm	Refreshment Break	Delegate Lounge

11.00am – 12:00pm	**Morning Plenary**	Main Hall
11:00am	**Regulators View of the Future**	
	JIM STEWART, HEAD, TRANSPORT SAFETY UNIT, IAEA	
11.20am	**Looking to the Future – An Industry Perspective**	
	PIERRE MALESYS, WNTI	
11:40am	**UK Industry's View of the Future**	
	MARK JERVIS, MANAGING DIRECTOR, INS ON BEHALF OF THE UK NDA	

12:00pm – 12:05pm	**Closing and Farewell**	Main Hall

Program

Abstracts – Monday 04 October 2010

T3 - SECURITY (SESSION 1)

2:00PM – 3:40PM – TECHNICAL SESSION – MAIN HALL
CHAIR: BRYAN REEVES, CO-CHAIR: OLIVIER LOISEAU

ABSTRACT 213

Radioactive and Nuclear Material Transport Security

ANN-MARGRETH ERIKSSON EKLUND*, RICHARD RAWL

Transport of radioactive and nuclear material is highly regulated and the transport safety regulations have been in effect for decades. Transport security recommendations for radioactive material were published in 2008 by the International Atomic Energy Agency (IAEA) as an implementing guide, "Security in the Transport of Radioactive Material", and are just now being implemented in many countries. On the other hand, nuclear material transport security has been governed for many years on the basis of a binding international convention, the "Convention for the Physical Protection of Nuclear Material", and its supporting document "The Physical Protection of Nuclear Material and Nuclear Facilities" INFCIRC/225, Revision 4 (corrected).

Experience in implementing the radioactive material transport security recommendations has been gained by countries as they make decisions on specific security provisions to require, provide training to their regulatory staff and licensees, and begin reviewing and approving transport security plans. This experience has led to the development of practical approaches that minimize impacts as the recommendations are put into practice.

The nuclear material transport security requirements contained in INFCIRC/225 are being revised to update them and to incorporate requirements based on the recent amendments made to the Convention. This revision will include development of a new recommendations document within the Nuclear Security Series of documents.

The interface between the nuclear and radioactive material transport security documents is important in order to ensure that appropriate security measures, based on both the nuclear and radioactive properties of the material being transported, are defined and implemented.

This paper will provide up to date information on the development of the IAEA transport security documents and will present information on implementation of the radioactive material transport security recommendations. It will explain how the documents interface with each another and provide examples of how they should both be used in defining transport security requirements for shipments.

ABSTRACT 214

The IAEA Assistance and Training Programme for Transport Security

RICHARD RAW, ANN-MARGRETH ERIKSSON EKLUND
(PRESENTED BY MARK HAWK)

The IAEA Office of Nuclear Security is working cooperatively with the U.S. Department of Energy's Global Threat Reduction Initiative, European Union and Australia to provide transport security assistance to countries throughout the world.

Assistance is available to countries in reviewing and upgrading their transport security programs at all levels

- National level (regulatory and other government agencies)
- Operator level (shippers and carriers)

Assistance is directed at implementing a consistent level of security throughout the life cycle of radioactive material (same level of security during transport as when in a fixed facility) Upgrade assistance can include:

- Expert advisory missions to provide advice and guidance
- Training courses for regulatory, governmental and industry personnel
- Transport security awareness
- Detailed training on designing and implementing transport security programs
- Planning to identify and prioritize needs (developing security approaches and plans)
- Developing model security plans and procedures
- Equipment (vehicles, packages, command and control equipment, etc.)

Country visits are now being scheduled to initiate transport security cooperative activities. A training course has been developed to assist countries in developing and implementing transport security programs. The training course has been given as a national training course (three times) and as a Regional training course (three times). The course addresses recommended security provisions for the transport of all radioactive material.

The course does not address additional physical protection provisions that may be needed for the transport of nuclear material subject to requirements that emanate from the Convention on the Physical Protection of Nuclear Material (CPPNM) and its amendment.

The Goal of the Training Course is to illustrate the need for adequate security during the transport of radioactive material, how to define levels of security with appropriate security measures, and how to effectively implement transport security programs.

ABSTRACT 364

Transport Security – an Operational View

MATT FOX*, ANDRÉ STASSE

Industry takes its responsibility seriously in transporting radioactive materials both safely and securely around the world every day, and has an outstanding record on both counts over several decades.

International radioactive materials transport security standards are contained in international regulations and conventions such as the International Atomic Energy Agency (IAEA) Convention for the Physical Protection of Nuclear Material, its supporting document, the Physical Protection of Nuclear Material and Nuclear Facilities, INFCIRC/225, Revision 4 as corrected, the 2008 IAEA Security Series

Abstracts – Monday 04 October 2010 : continued

"Security in the Transport of Radioactive Material", the International Maritime Organization's "International Ship and Port Facility Security Code" (ISPS Code) and the security requirements for High Consequence Dangerous Goods laid out in the United Nations Model Regulations known as the "Orange Book". National requirements supplement this international framework. It is well understood by the transport industry that security requirements are a matter for individual States and industry must work within these requirements.

A heightened concern in recent years about the consequences of malicious acts/terrorism has been reflected in the increased emphasis on transport security, for all classes of dangerous goods. This concern in turn raises the question of harmonisation of security standards from one jurisdiction to another worldwide in the interest of avoiding possible differences of interpretation and differing standards between jurisdictions

The World Nuclear Transport Institute (WNTI) has established a Transport Security Industry Working Group to evaluate based on industry experiences where there is common ground between jurisdictions, assessing the implications on operations of security measures, and considering the possibility of developing an industry best practices document related to transport security.

This paper will examine some of the transport security challenges for industry in more depth, their operational impact and the role of the industry through the WNTI Transport Security Industry Working Group to ensure the safe, cost-effective and secure transport of radioactive materials.

ABSTRACT 375

Protection of Information: An Essential Component of Physical Protection of Nuclear Material Transportation

ANDRÉ STASSE*, MATT FOX

We must be transparent in our core activity which is an industrial field directly linked to electricity production. This activity is strategic for all countries involved in nuclear electricity programs. Thus, the transportation of nuclear material from one country to the next is not only a typical element of the industry but a strategic element as well.

One of our main preoccupations in the transportation of nuclear material is to ensure the highest level of physical protection because nuclear material is, in one hand, firsly, a material which presents a risk of proliferation, and in the other hand this material is a dangerous good. To assure this protection, it is necessary to restrict information in order to reduce risks. In so doing, the industry is sometimes accused of being too opaque.

It is the same regulation to protect nuclear material into nuclear sites than during the transportation phase. Two types of regulations are applicable within the perimeter of nuclear sites - those being safety and physical protection. During the transportation phase, three types of regulations must be enforced: safety, physical protection and, in addition, international regulations for the transport of dangerous goods which mix safety and security requirements according to the vulnerability in public domain and gives international rules to apply. These rules include protection of the information in nuclear material transport organisation.

T1 - SPENT FUEL PACKAGE DESIGNS

2:00PM – 3:40PM – TECHNICAL SESSION – CONFERENCE ROOM 1
CHAIR: LAURENT MILET, CO-CHAIR: ULRICH ALTER

ABSTRACT 165

Italian-French Experience in Development, Licensing and Manufacturing of a New Cask for the Transportation of Irradiated Nuclear Fuel from Piemonte Nuclear Sites

JOEL BAUDOUIN*, EMILIE BOUYER, THOMAS BRION, HERVÉ RIPERT

SOGIN the Italian state Company founded in 1999 to manage the closure of the nuclear fuel cycle and the decommissioning of the Italian nuclear power plants and research centers, signed, in April 2007, a contract with AREVA for transport and reprocessing of approximately 235 tons of irradiated nuclear fuel stored on Caorso, Trino and Avogadro sites. The international transport activities, coordination and cask provision were entrusted to TN International (AREVA group).

The interface constraints of Trino and Avogadro sites leaded to develop and license a new high capacity transport packaging, the TN117. The transports should start during first half of 2010, which implies a very tight time schedule to achieve the following tasks:

- Design of a new packaging and application for a B(U)F certificate of approval and validation in Italy.
- Manufacturing of two packaging and associated operating tools and transport means.

TN International started the design of the cask in April 2007. The main challenges were to allow the loading of five different types of spent fuel assemblies, including BWR and PWR MOX fuel, in the same basket, and to increase the capacity up to twelve spent fuel assemblies in order to optimize the number of loadings. A B(U)F certificate of approval application has been submitted to French Authorities in June 2008. Nearly two years of expertise were necessary before the granting of the package approval and its validation in Italy.

Due to the time schedule constraints, the manufacturing of two casks has been performed in parallel from the end of 2007 to the first half of 2010, the first cask being available as early as January 2010. This situation implied to manage the design changes made during the licensing.

Handling and transport means, and operating tools have been identified, designed and manufactured in parallel to the cask licensing and manufacturing phases in order to test them as soon as the first cask was ready.

Thanks to the smooth relationship between all parties involved, the optimization of the time schedule was made possible, making this project a major experience and success for the whole actors.

Abstracts – Monday 04 October 2010 : continued

ABSTRACT 230

Transportation Package for use in Facilities with 25 Ton Crane Capacity

CATHERINE SHELTON*, OLIVIER GANDOU, NICOLAS GUIBERT

Transnuclear, Inc. a USA based company belonging to the Logistic Business Unit (BUL) of AREVA is a well known leader in used fuel transportation and storage cask design and manufacturing. Transnuclear has designed and licensed numerous used fuel storage and transportation cask models. However, the used fuel transport market in the United States remains limited as commercial used fuel assemblies are stored onsite due to the lack of either a central repository or reprocessing facility. Consequently, the market shipping needs are related to research reactor fuel shipments, irradiated material shipments in connection with facility decommissioning and irradiated fuel pins for post-irradiation examination. These shipments require smaller transportation packages compared to those of commercial reactor. Additionally, interfacing with loading and unloading facilities designs is complex due to design differences between various facilities. Limited crane capacity is also constraint.

Majority of these types of transport packages have licenses from regulatory authorities outside of USA which cannot be used for domestic shipments. Therefore, to address customer needs for domestic shipments, Transnuclear has developed a 25 ton weight cask design. This design optimizes the currently licensed cask capacity while keeping low the total weight so that it can be used by most research facilities and hot cells. This new cask design is able to handle PWR, BWR, MOX, EPR, TRIGA and various research reactor fuel assembly designs as well as non fissile irradiated contents. This cask features openings at the top and at the bottom, will be loaded and unloaded horizontally and vertically in a hot cell or in a fuel pool, and loaded and unloaded upside down depending upon the facility interface requirements. Transnuclear has worked with the BUL of AREVA in France to ensure that this cask design is also compatible with European facilities. The safety analysis report is under preparation and will be submitted to the US NRC for their review and approval later in 2010 and this transport package is set to perform its first shipment in 2012.

ABSTRACT 209

Development of a New Dual Purpose Cask

JUSTO GARCIA*, OLIVIER ROULLEAUX-DUGAGE

TN International (AREVA group) has proposed for more than 20 years the TN™ 24 cask family which features by forged steel casks used both for the transport and storage of UOX used fuel. Thus more than 20 versions of TN 24 casks have been designed for more than 20 customers in Europe, United States of America and Japan which have ordered more than 300 casks. The PWR or BWR fuel characteristics may have various enrichment value up to 5%, various cooling time down to 2 years and various burn-ups up to 60 000 MWd/tU.

Facing the current international trend towards expanding Used Fuel Interim Dry Storage capabilities with higher performances especially in term of used fuel characteristics and long term storage, TN International decided to launch an extensive innovation process to create the new generation of transport and storage casks.

The TN™ DUO solution is the result of an extensive process to develop innovative and cost effective dual purpose cask.

The purpose of this paper is to present this experience, and furthermore to underline the main advantages of the TN™ DUO dual purpose transport cask.

ABSTRACT 148

Development of Type C Packages to Transport Spent Nuclear Fuel from Research Reactors Produced in Russia

LUDMILA BARABENKOVA*, VYACHESLAV SHAPOVALOV, ALEKSANDR MORENKO, VITALY MATVEEV, SERGEY KOMAROV

Nowadays the problem of international air transportation of SNF is of crucial importance. Air transportation offers the following advantages: absence of transit countries, minimal time of being at the route, minimal vulnerability in terms of physical protection. Technical and regulatory-legislative aspects enable such transportations to be performed.

This can be proved by SNF research reactors air transportations performed in 2009 in Romania and Libya. TUK-19 placed into specialized freight ISO-containers were used for such transportations. The packages were certified as a type B(U) package (total activity did not exceed the accepted value of 3000A2). Air transportations of higher activity require a type C package. According to IAEA Regulations the design of this type of package should meet more strict requirements for strength and tightness. There is not a certified Type C, air transport, cask for spent nuclear fuel anywhere.

RFNC-VNIIEF together with the research-production company the OOO "SOSNY," has developed a concept of a Type (C) package. The design consists of two parts:

- TUK SKODA VPVR/M used for air and land transportation of SNF from research reactors,
- Special damping system, in which TUK SKODA VPVR/M is installed.

The damping system provides reduction of loading onto the TUK SKODA VPVR/M up to the level when the package saves its strength and tightness under emergency conditions (impact at a speed of 90m/s min) and provide nuclear and radiation security according to the Regulations requirements.

Taking into account maximal size-weighted features (length ~3000mm, diameter~2800mm, mass ~30000kg) the package can be transported by An-124 aircraft (up to 4 units) and by IL-76 aircraft (up to 2 units).

The paper describes a Type C package design, principle results of computational studies for package safety under accidental air transportation, as well as the results of the small-scale package prototypes testing for target impact at a speed of 90m/s min

T2 - RADIATION PROTECTION

2:00PM – 3:40PM – TECHNICAL SESSION – CONFERENCE ROOM 2

CHAIR: MARIE-THERESE LIZOT, CO-CHAIR: JAMES SHULER

Abstracts – Monday 04 October 2010 : continued

ABSTRACT 58

Measurement of Radiation Level and Surface Contamination for Packages and Conveyances

ASHOK KAPOOR*, JAMES WILLIAMS, S.Y. CHEN, SUNITA KAMBOJ

This paper describes the development and current status of the new standard "Measurement of Package and Conveyance Radiation Levels and Surface Contamination", approved by the American National Standards Institute (ANSI) in 2007. The purpose of the standard is to minimize variability and therefore, help in demonstrating uniform compliance for contamination and radiation levels with the regulatory limits thus promoting public and occupational health and safety during transportation and the handling of radioactive materials.

The 21-member subcommittee (N14.36) was formed to develop the standard under the procedures of ANSI Accredited Standards Committee N14, "Packaging and Transportation of Radioactive and Non-Nuclear Hazardous Materials." The subcommittee (N14.36) represents radioactive material packaging and transportation industry from the United States and Canada, non-governmental organizations, United States regulatory and government agencies (both Federal and State Governments) including the United States Department of Energy.

In the development of this standard, the subcommittee has taken into consideration the existing operating and administrative procedures, methods, instruments, and processes used in the industry and government. Certain basic general requirements in the standard are applicable to all radioactive material (RAM) packages; however, the risk informed graded approach is considered by the subcommittee in determining package specific requirements in the standard. The contents of this standard include the processes, procedures, equipment, and training required for consistent, reliable, and reproducible measurements of radiation levels and surface contaminations on and near RAM packages and conveyances.

ABSTRACT 225

A New Solution to Decontaminable and Inspectable Package Handling Features that can be Blocked by a 90º Rotation

FABIEN GIRAULT, ROBBIE JAMESON

Nuclear industry uses lot of packages to handle, transport and store radiological materials. The large majority of these packages need to be externally decontaminated and particularly handling features which comes in contact with contaminated handling tools.

The invention addresses ease of decontamination and decontamination control for handling features on nuclear packages. This is a very important feature e.g. for avoiding transfer of contamination and release of contamination into the environment. These operations are difficult to perform in a nuclear environment as they are often undertaken remotely and when not remotely they must at least respect the ALARP principle to limit radiological exposure of workers.

The invention also improves the quality of the package fixation and its ease of inspection. Handling features are indeed an essential part of the nuclear package. They must be exempt of defects due to the catastrophic consequences their failure would have. Quality must be excellent and this invention makes it easier to control. As a consequence this can also reduce maintenance operation which is in line with the ALARP principle.

Some means will also be presented to adapt certain existing handling devices in order to put such invention in application within existing plants and package stores.

Due to all the described advantages such invention enables manufacturing and operational costs saving and failure and contamination risks reduction.

The presentation will provide an overview of the operational and manufacturing difficulties on standard handling features and will show how the invention eases decontamination and implementation on nuclear containers.

ABSTRACT 75

Radiological Safety of Spent Fuel Storage and Transport

CHARLES W. PENNINGTON*

The debate over expanded commercial nuclear power generation in the U.S. focuses on several issues, but the most emotionally gripping topic involves nuclear safety and the threat of radiological harm. A subset of this topic is the prospective safety of spent fuel storage and transportation. The U.S. is backing away from the use of the Yucca Mountain repository, and, with the solution to spent fuel disposition certainly being extended in time and with the prospect of an expanding use of nuclear power, there are some that wish to elevate concern within the public over radiological safety of spent fuel storage and transport.

With the perseverance and prospective expansion of the radiation-fear issue, a study has been performed to assess the credible radiological outcomes of nuclear events that have historically been the centerpieces of nuclear opposition. Analyses of population radiation doses resulting from worst-case, credible events in the U.S. nuclear fuel cycle have been performed. Such events may be very credibly modeled for realistically conservative outcomes using the same accident profile, release patterns, dispersion characteristics, and population exposures as the accident profile and outcomes from the Chernobyl Nuclear Power Plant Unit 4 (CNPP4) accident. Based upon industry and U.S. NRC published studies, the worst-case radiological event for either spent fuel storage or transport would result from a credible sabotage scenario. Using modeling based on the CNPP4 accident, it is shown that worst-case, credible radiological outcomes for both peak exposures and lifetime population doses are less than what would be considered a significant radiological hazard. More importantly, it is demonstrated that these hypothetical outcomes are well below what are actually produced by at least seven non-nuclear industries each year in the U.S., industries that have existed for decades or centuries and whose radiological characteristics are not regulated.

Finally, it is proposed that such information can be used to inform decision-making about commercial nuclear power growth as a key step towards reducing fears and improving knowledge of various stakeholder parties involved in U.S. energy decisions.

Abstracts – Monday 04 October 2010 : continued

ABSTRACT 95

TN International Transportation Procedure for Used Fuel Casks: Transportability Tool

MIKAEL DE BIASI*, STAVROS KITSOS

TN International works in collaboration with nuclear industry operators – such as EDF in France – to manage the transportation of fuel assemblies in casks. In order to optimize the management and evacuation of used fuel from nuclear power plants, TN International has developed the Transportability calculation tool.

The aim of such a tool is to quickly determine the dose equivalent rates and thermal powers for a chosen fuel batch in the suitable cask among those available, and to check whether the transportation is possible according to the regulatory criteria. For each type of cask used, a calculation model is associated in the tool. The addition of a new model in Transportability is systematically carried out with a three-step methodology. First, the cask to be implanted is modelled and dose rate calculations are made using a three-dimensional Monte Carlo code. Then, three dose rate measurement experiments are done around the loaded cask. Finally, the model created in step 1 is tuned according to the experimental measurements while taking into account margins in order to cover uncertainties associated with the statistical calculations and the measurement instruments.

The increase of PWR MOX fuel use in nuclear power plants during the last decade has led TN International to design a new cask called the TN 112. This cask allows the loading of 12 MOX used fuel assemblies per transport, which offers an economical interest for both operator and our firm. Thus, a new model has been added in Transportability to continue optimizing the management of used fuel, and particularly MOX fuel.

The goal of this article is to present the three-step methodology used for the TN 112 cask, notably the calculation procedure with evolution codes ORIGEN-ARP and DARWIN and the 3D Monte Carlo code TRIPOLI-4.3. This article also deals with the global benchmark done around the cask and the comparisons between measured and calculated dose rates.

ABSTRACTS 277/282

Utilisation of the Monte Carlo Code 'MCBEND' and the Deterministic Code 'ATTILA' to Assist with the Shielding and Dose Analysis for the Land and Marine Transportation of an International Transport Flask

ANDREW SMITH, ANTHONY CORY

A shielding and dose uptake assessment is required for the transportation of nuclear fuel to overseas customers. The fuel is contained within transport packages that when transported individually meet the IAEA transport criteria. In addition to dose rate criteria around a transport flask, dose uptake to ship personnel must not exceed the criteria set by the International Atomic Energy Agency (IAEA) of 1milli-Sievert per year for the general public.

Methods of driving down the dose were employed in accordance with the 'As Low As Reasonably Practicable' (ALARP) principle. Ship personnel living onboard the vessel are subject to radiation for the full duration of each day during the voyage. Consequently dose rates in occupied areas are required to be low in order to comply with the stipulated criteria. Methods to drive down dose are applied in line with the ALARP principle to restrict dose uptake.

The Monte-Carlo computer code MCBEND has been used to optimise the shielding to be installed and to determine total neutron, primary and secondary gamma dose rates at key locations around the road vehicle. The use of Monte-Carlo methods in large models such as ships can present potential problems. With ship personnel able to occupy many locations around the ship, dose rates are required in various locations that may require individual acceleration methods in MCBEND. The three-dimensional deterministic code Attila solves the transport equation using a tetrahedral mesh system over all model space, assessing potential problem areas that could be overlooked when selecting dose rate regions. The post-processing tool TecPlot can be used to present two-dimensional or three-dimensional dose rate contours. This can be useful for assessing potential weaknesses around the transport flask and as a visualisation tool for the project.

It was demonstrated that calculated dose rates surrounding the vehicle and dose uptake on board the ship were within the criteria stipulated by the IAEA. With the aid of MCBEND and Attila, dose uptake estimates can be provided with a degree of confidence in addition to two-dimensional contour plots on the vessel.

T4 - THERMAL ANALYSIS

2:00PM – 3:20PM – TECHNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: CARLOS LOPEZ, CO-CHAIR: FRANK KOCH

ABSTRACT 96

Computational Fluid Dynamic (CFD) Design and Mock up Test for Heat Removal

OLIVIER BARDON*, JEROME BELLANGER, NASSER ZAHRI

The increasing of used fuel burn up leads to substantial residual power. To maintain reasonable cooling time, casks with high dissipative capacity need to be developed.

In order to achieve thermal dissipation of about 70 kW in a 12 PWR fuel cask, the outer surface of the cask must be equipped with high efficiency fins surface. In general, heat removal from nuclear casks at high heat load is achieved using various types of fins working in natural convection. Fins' shapes are usually limited by manufacturing considerations. The improvement of automation associated with the electric capacity discharge process for welding pin cylinders led us to examine the thermal performance of using long copper pin cylinders to make a large exchange area at the outer surface of the cask. The main technical interest is that it offers good thermal performance regardless of the general orientation of the cask, which is an important point to be addressed after an accidental drop and a question generally raised by authorities.

In order to get a quick preliminary design of a pin's density and arrangement to reach an objective value of global heat exchange performance, model and tests were conducted with horizontal pins cylinders welded on flat vertical surfaces. Computational Fluid Dynamic preliminary design was a good aid for matching both vertical and horizontal orientation constraints, as well as assessing relatively good predicted values of global heat coefficient.

Abstracts – Monday 04 October 2010 : continued

ABSTRACT 59

Modelling the Thermal Performance of Cork and Wood in the Thermal Test

CHRIS FRY*

Cork and wood are materials which are frequently used in the design of transport packages, both inside transport flasks and in impact limiters or heat shields which are placed on the flasks during transport. Cork and wood are used in transport packages because they readily compress, absorbing energy during an impact and also have a low thermal conductivity, protecting the flask from the heat of a fire.

It is challenging to demonstrate, by testing alone, that a package meets all the thermal requirements of the IAEA Regulations. Nearly all thermal assessments therefore include some modelling. Cork and wood are natural materials and at high temperature their thermal behaviour is complex. A thermal assessment of heat transfer across a cork or wood heat shield, based just on reference values of density, specific heat and thermal conductivity may therefore be subject to considerable error.

This paper addresses the challenge of modelling heat transfer through cork and wood, in a demonstrably pessimistic way, such that the effects such as charring, evaporation and condensation of water and oils, shrinkage and burning can be shown to have been considered and included.

It is concluded that the modelling of heat transfer through cork and wood must be based on experimental data. Examples of thermal tests which include heat transfer through cork and wood are given and the calculation of the corresponding effective thermal conductivities described. How and when these effective thermal conductivities should be used in a thermal model is discussed.

In some situations it is known that the wood inside a transport package or shock absorber may burn, releasing heat. Ways of demonstrating that will not occur and how any heat generation from burning can be included in the thermal model are described.

ABSTRACT 203

Thermal Shielding of the Shock Absorber is Made of Wood

KYOUNG-SIK BANG*, JU-CHAN LEE, KI-YOUNG KIM, CHUNG-SEOK SEO, KI-SEOG SEO

In order to safely transport the radioactive waste arising from the hot test of ACP(Advanced Spent Fuel Conditioning Process) a shipping package is required. Therefore KAERI is developing a shipping package to transport the radioactive waste arising in the ACPF during a hot test. Regulatory requirements for a Type B package are specified in the Korea MOST Act 2008-69, IAEA Safety Standard Series No. TS-R-1 and the US 10 CFR Part 71. These regulatory guidelines classify the hot cell cask as a Type B package, and state that the Type B package for transporting radioactive materials should be able to withstand a test sequence consisting of a 9 m drop onto an unyielding surface, a 1 m drop onto a puncture bar, and a 30 minute fully engulfing fire. This paper discusses the experimental approach used to simulate the response of the hot cell cask to fire in furnace with chamber dimensions of 300 cm(W) x 400 cm(L) x 200 cm(H) by using the 1/2 scale model which damaged by both 9 m drop test and 1 m puncture test.

ABSTRACT 120

Behaviour of a Package for Transport of Spent Fuel Assemblies Exposed to Beyond Regulation Fires

BENOIT ECKERT*, GILLES SERT, SARAH FOURGEAUD, IGOR LE BARS

IRSN is performing a study relative to the thermal behaviour of a new package design for transport of spent fuel assemblies. The aim of this study is to evaluate the behaviour of the package submitted to fires, with durations and temperatures higher than those required in the IAEA regulation TS-R-1 (respectively 30 minutes and 800 Celsius degrees). Its main objective is to provide quantitative data available for safety assessment in emergency situations involving fires. Moreover it can also be used for a cross comparison with the analysis of the thermal behaviour of the package during the IAEA regulatory fire test presented by the applicant in the package design safety report.

This study is based on numerical calculations performed with the code THERMX-PROTEE. The three-dimensional model used represents a quarter of the upper part of the package, where is located the closure system. The thermal behaviour of the resin located in the plug, the trunnions and between the inner and outer shells was modelled considering endothermic reactions of vaporization. During the heating phase of the fire test, the water vapour produced in heated elements is transferred and condensed in the nearby colder elements; the associated thermal transfers can increase very fast the temperature of the latters. The part of the vapour which cannot be condensed when most of the nearby elements reach a temperature above 100 Celsius degrees is evacuated through the holes that are distributed throughout the external envelope of the packaging and closed by fusible plugs. A specific calculation module has been developed to take into account the corresponding energy transfers. This module was qualified by comparison with the results of experimental fire tests.

The calculations performed in the framework of this study cover temperatures of fire between 400 and 1000 Celsius degrees. One of the results of those calculations is the time necessary to reach the maximum allowable temperature of the elastomer gaskets. The aim of this paper is to present the main results of this study.

T6 - INTERFACING WITH THE PUBLIC

4:00PM – 6:00PM – TECHNICAL SESSION – MAIN HALL
CHAIR: RUPERT WILCOX-BAKER,
CO-CHAIR: HENRY-JACQUES NEAU

ABSTRACT 155

Safety Analysis of the Transportation of Radioactive Waste to the Konrad Final Repository – Waste Data Scenarios

ULRICH ALTER*

The Federal Minister of the Environment, Nature Conservation and Reactor Safety (BMU), Bonn, commissioned the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Cologne, to conduct a study that examines the shipment of radioactive waste to the Konrad final re-pository.

Abstracts – Monday 04 October 2010 : continued

In the Federal Republic of Germany Konrad will be the final repository for all types of negligi-ble heat-generating low and medium radioactive waste from nuclear power plants, nuclear facilities, industry, scientific research centers and medicine use of radioactive nuclides.

Within this safety analysis two main topics should be discussed, the assessment of potential radiation exposures from normal (incident-free) transportation, especially in the region of the final repository where all radioactive waste shipments converge, and the assessment of risks from transport accidents.

For the purpose of the study the anticipated waste transport volume and the waste proper-ties, information on transport packagings, activity inventories and dose rates of the waste packages were analysed in detail.

The maximum volume of radioactive waste for the final repository Konrad will be 303 000 cubic meters, 100 000 cubic meters should be shipped within the first ten years, nearly 10 000 cubic meters per year. At least this will be an amount of 50 shipments of standardized shipping units per week. Up to these transport scenarios and arrangements the total weight of one shipping unit (loaded pool pallets and cubical containers) must not exceed 20 metric tonnes.

Three main waste shipment scenarios will be analysed within this study:

- 100 % shipments by rail
- 100 % shipments by road
- 80 % shipments by rail and 20 % shipments by road

For all these shipments by rail or road the basis for transport safety and radiation protection is given in detail by the relevant national and international transport regulations and the Ger-man Radiation Protection Ordinance It was the duty of the Federal Office for Radiation Pro-tection (BfS) to create the "Konrad waste acceptance requirements" for all types of radioac-tive waste with negligible heat-generation in Germany.

ABSTRACT 248

Safety Analysis of the Transportation of Radioactive Waste to the Konrad Final Repository – Methods and Results

FLORENCE-NATHALIE SENTUC*, WENZEL BRUCHER

A transport risk assessment study has been conducted for transport of radioactive waste with negligible heat-generation (low- to medium-level) to the German final repository Konrad. This study is a revision of the former Konrad Transport Study performed by GRS in 1991 implementing updated waste data among other improved methods and assumptions for the purpose of a more realistic approach to risk assessment. According to the results of the revised survey each year approximately 2300 shipping units of low and medium active waste will be transported to the KONRAD site starting with its expected begin of operation in 2014.

The first part of the transport risk assessment study concerns the radiological consequences from routine (incident-free) transportation of radioactive material, i.e. the radiation exposure of transport personnel and the public (expected exposure). Based on the assessed detailed information on transport arrangements and on the average number and radiological characteristics of waste packages the maximum annual effective doses for the representative persons were estimated. The results show predicted doses far below the relevant German annual statutory dose limits.

The risk associated with transport incidents and accidents has been quantified for the area within a radius of 25 km around the repository site. The probabilistic method adopted in this study considers parameters as the frequency and severity of railway or road accidents, characteristics of radioactive waste and transport packagings and the frequency of atmospheric dispersion conditions. From a large set of parameter combinations the spectrum of potential radiological consequences and of the associated probability of occurrence was assessed. Compared to the former risk analysis of 1991 a revised set of release fractions for mechanical and thermal impact conditions and a modern lagrangian particle dispersion model have been applied. In combination with the updated waste database and revised transport scenarios these changes result in potential radiation exposures in the repository area which are about one order of magnitude lower than predicted in the former study.

ABSTRACT 237

Public Acceptability – You Can't Judge a Book by its Cover

LORNE GREEN*

This paper explores fast-evolving developments in the way the media operates and the consequent opportunities and challenges they pose for encouraging understanding of safe and reliable transport of radioactive materials

Successful communication depends critically on three key factors: the message; the messenger, and the means by which the message is delivered. By analysing contemporary examples of media coverage we can see how the same story can be told in contradictory ways. Contemporary media technologies amplify the challenges of reaching a growing demographic sector that relies heavily on the new media.

Traditional media outlets – written and electronic increasingly have their limits in reaching contemporary audiences; how does industry cope? Ultimately the question that resonates is whether we can really judge a book by its cover? This paper will examine how the industry is covered and the challenges and opportunities presented by new messages, messengers and means of communication.

ABSTRACT 374

Public Acceptance Approaches Related to Back-end Transport between Europe and Japan

TAKASHI KOMATSU*

Plutonium is indispensable for a Fast Breeder Reactor (FBR). Since 1960s Japan has been developing FBRs, thus Japan chose reprocessing. When light water reactors (LWR) were fully introduced in early 1970s, the Japanese utility companies made long-term contracts with UK/French state-owned reprocessing companies for 7000 tons of spent fuel. Japan is far away from the reprocessing site, so some states outside of Europe could be "en-route".

In 1992 plutonium shipment from France to Japan faced 35 "coastal states'" objection. In 1995 fifteen "coastal states" opposed or concerned about the safety of the first high level radioactive waste (HLW) transport from France. In 1999 when MOX fuel (Uranium and plutonium oxide fuel) transport from France and UK started, 8 states advocated 3 regional organizations to issue objections, saying that the transport fatally endangers coastal population and environment.

The voices did not occur spontaneously. Strong agitations and negative campaigns by international anti-nuclear business organizations lead them. In 1992 Greenpeace International 'warned' all the possible routes states and traced the transport comboy by two vessels, releasing hour by hour the location, saying that they are warning coastal people the fatal dangers coming. In 1995, in 1999 and in 2002, the similar spectacular campaigns were observed.

In small island developing 'coastal states' neither nuclear business nor nuclear expert exists. Objective domestic scientific resource is not available for the local media and residents. It must be provided from outside. Japanese utilities understood the necessity to organize continuous public acceptance activities for these 'coastal states'. Once political leaders have ridden on the 'anti-nuc' bandwagon, there are few domestic needs for them to change the position.

In IMO and IAEA the authorities denied the 'Unresolved Safety Issues' claimed by anti-nuclear groups. The 'issues' were solved. But 'evidence of safety ' is rarely carried to coastal residents by their government, whereas ' evidence of danger ' is vocally announced by them. Based on the fact that transport is an essential part of any industry and considering that nuclear generating states is a minority in international society, nuclear industry needs to continue the international public acceptance efforts.

ABSTRACT 257

Communication and Radioactive Material Transportation

CAMILLE OTTON*, BERNARD MONOT

Communication about radioactive material transportation is not simple. It is an activity synonym of fear for many people, it is also one of the only nuclear activity made in the public area. A good communication is thus very important but it has also limits. Communication may be totally in line with:

- regulations,
- requirements,
- laws which have to be followed,
- general organisation of a transport,
- rules applied to transport any type of material,
- how a type of transport is chosen,

in fact, all the elements which help to demonstrate that the radioactive material transportation is well managed.

We may disclose a lot of information, but we have also to keep some of them confidential for very simple and logical reasons. First of all, we are not transporting apples, we are transporting radioactive material, it is a sensitive activity. Disclosing schedules and routes of a transport is thus not possible.

In fact, there are two types of communication:

- the one made on the fly
- the crisis one

"On the fly communication" is the basis, we have time to analyse the facts, to prepare our communication on the material, the transport mean and the cask used. "On the fly communication" builds our library which is essential for the crisis communication.

With some real examples such as MOX transport to Japan or radioactive material transport from France to Russia this article will analyse which type of documents are prepared in terms of communication for a transport.

ABSTRACT 239

Communicating the Transport of Radioactive Materials Using New Media

BETTY BONNARDEL-AZZARELLI*

The recent emergence of new ways of communication grouped under the generic name of "new media" has changed dramatically the access to and sharing of information, as well as the means of individual or collective expression of opinions. It can be witnessed in many instances that traditional text-heavy websites and printed publications are progressively been replaced by image-based documents, videos and bite-size texts on electronic supports. In other instances, social media allowing users to share thoughts and information online are rapidly overtaking the more traditional communication channels, such as newspapers and terrestrial television channels.

While surfing on the web, it is easy to find social networks, posts and blogs from those opposed to radioactive transport, trying to reach a wide and mainly younger audience. It is the responsibility of all those involved in the safe transport of radioactive materials to make available factual, sound and supported information in an accessible form to allow each stakeholder to develop an informed opinion of our sector.

After briefly reminding what constitutes new media, the paper will discuss how they are affecting communications on the safe transport of radioactive materials. In order to provide a case study for the purpose of this paper, the experience of the World Nuclear Transport Institute (WNTI) will be exposed. The WNTI has developed a communications strategy which integrates the use of new media while adapting and modernising the traditional ways of communication, to propose a comprehensive set of information means aimed at a wide and diffuse audience of industry's stakeholders.

T5 - ISOTOPES/SOURCES PACKAGING DESIGNS

4:00PM – 5:40PM – TECHNICAL SESSION – CONFERENCE ROOM 1

CHAIR: ALBERTO ORSINI, CO-CHAIR: PETER LAMBOURNE

ABSTRACT 69

Design and Development of BI-TL-300 Equipment as a Type B (U) Transportation Cask

DHIREN SAHOO*, JOTIRAM MANE, VINAY BHAVE, PIYUSH SRIVASTAV, ANIL KOHLI

BI-TL-300 equipment, designed to irradiate blood and its components to prevent Graft Versus Host Disease in immune deficient patient, is a Type B (U) transportation cask. This is designed as an innovative package consisting of main body and outer enclosure as its impact limiter. The package is made compact in its overall size and mass by suitably placing tungsten and lead as shielding material. The blood bag moves in an elliptical bend pipe by gravity. The design of the irradiator ensures dose uniformity ratio within the specified band. The package is designed to meet the national and international

Abstracts – Monday 04 October 2010 : continued

regulations of radioactive material transportation by road and sea and it conforms to the requirements of "Type B (U)" Package as specified therein.

In this paper design features of the package are presented along with the demonstration of accident condition test by numerical simulation. The structural integrity and performance optimization under 9m drop is carried out by numerical simulation using commercial F.E.M software PAM-CRASH and it is meeting the requirements. Performance under thermal test assessed by numerical simulation is also presented.

ABSTRACT 40

Type B Package for the Transport of Large Medical and Industrial Sources

PHILIP NOSS*, DWAINE BROWN

AREVA Federal Services LLC, under contract to the Los Alamos National Laboratory's Offsite Source Recovery Project, is developing a new Type B(U)-96 package for the transport of unwanted or abandoned gamma and neutron sealed sources. The sources were used primarily in medical or industrial devices, and are of domestic (USA) origin. To promote public safety and mitigate the possibility of loss or misuse, the Offsite Source Recovery Project is recovering and managing sources worldwide.

The package, denoted the LANL-B, is designed to accommodate the sources within internal gamma or neutron shields. The gamma sources are located in the IAEA's Long Term Storage Shield (LTSS), and the neutron sources are located in a newly developed neutron shield. Since the sources are separately shielded, the package does not include any shielding of its own. A particular challenge in the design of the LANL-B has been weight. Since the LTSS shield weighs approximately 5,000 lb, and the total package gross weight must be limited to 10,000 lb, the net weight of the package was limited to 5,000 lb, for an efficiency of 50% (i.e., the payload weight is 50% of the gross weight of the package). This required implementation of a light-weight bell-jar concept, in which the containment takes the form of a vertical bell which is bolted to a base. A single impact limiter is used on the bottom, to protect the elastomer seals and bolted joint. A top-end impact is mitigated by the deformation of a torispherically-shaped head. Impacts in various orientations on the bottom end are mitigated by a cylindrical, polyurethane foam-filled impact limiter. Internally, energy is absorbed using honeycomb blocks at each end, which fill the torispherical head volumes.

Since many of the radioactive sources are considered to be in normal form, the LANL-B package offers leak-tight containment using an elastomer seal at the joint between the bell and the base, as well as on the single vent port. Leak testing prior to transport may be either using helium mass spectrometry or the pressure-rise concept.

ABSTRACT 24

BU-MAN, the new Argentinean Type B(U) Package for the Safe Transport of Radioisotopes

ANA MARIA CASTELLANOS*, EDUARDO ESTEBAN

With the objective of satisfying the needs of safe transport both domestic and foreign of radioisotopes produced by CNEA facilities that are needed in medicine and industry and taking into account reglatory requirements, it was developed a project to manufacture a multipurpose package which can contain radioactive substances both for solid and liquid form.

This paper provides the design information of the transport package denominated BU-MAN and the used design criteria to assure the accomplishment of IAEA Safety Standards Series No. ST-1, Regulations for the Safe Transport of Radioactive Material, which was adopted for our Argentinean Regulatory Body in the AR-10.16.1 Standard.

The radioactive substances that this pakage would contain, are Molibdenum 99 or Iodine 131 in liquid form, or sealed sources of Iridium 192 in solid form.

As this is a B (U) type package, its licensing process implied the demonstration of the fulfilment of general requirements established in the standards, both for normal and accidental conditions of transport and all the additional that are necessary for the transport by air.

The elaboration of a safety report, the manufacture of prototypes of essay, and the essays themselves to demonstrate the fulfillment of the established requisites, are necessary steps in the licensing process. They are briefly described in this job.

At present, we are journeying the last stages of the licensing process, mainly the evaluation of the essays that have been done on the prototypes, according to the standards requirements to start with the first series manufacturing.

So we are close to concluding BU-MAN has good perspectives both for domestic use, and for exportation purposes of our products also for other countries that need an available, safe and licensed package to provide to the society of radioisotopes used in medicine and industry.

ABSTRACT 283

TN PNS a New Type of Cask

NICOLAS GUIBERT*

TN International has developed a new package called TN PNS dedicated to the transport of Primary Neutron Source rods (PNS) containing Californium-252 neutron sources. PNS are necessary to start the new nuclear cores and to calibrate the control devices.

In the framework of the nuclear renaissance, new nuclear power plant will have to be started, including the EPR™ reactors. Transport are done from France where the PNS rods are manufactured (at LEA CERCA) to any nuclear site all over the world, by road, sea, rail and air transport.

Main characteristic of the TN PNS is that it is fully integrated to the PNS rods manufacturing line for better safety and operability performances. Contrarily to most of packages, trunnions are not located on package but on the frame. Lodgements in the packages are ensuring tie-down and tilting functions without any relief on the cask body. Geometry of the holes is designed so that tie-down is resistant to accelerations for air transport (up to 11g in any direction). All the tilting and tie-down system can be set up very easily by hand.

For the loading operations TN PNS can be directly face to manufacturing cask: PNS rods are horizontal loaded and can slide directly from manufacturing cask inside TN PNS. High performance radiation shielding is provided during the transport and loading/unloading operations by using water filling if necessary.

The TN PNS can transport up to 6 PNS rods for the EPR™ reactor or other PWR reactors with a maximum activity 100 GBq in Cf-252.

Abstracts – Monday 04 October 2010 : continued

Rods are introduced into a removable basket which may be changed allowing future rod design or bring additional shielding. That gives the TN PNS a lot of modularity.

Design, licensing and fabrication have been carried out in a very tight schedule. The first PNS delivery was done in January 2009 to start the core of the TOMARI 3 PWR on Hokkaido Island (Japan). Design, licensing, and manufacturing have been achieved in a little more than 1 year. Now this singular package is ready to deliver new EPR™ PNS rods all over the world.

ABSTRACT 192

Development of the Bulk Tritium Shipping Package

PAUL BLANTON*, PAUL MANN, KURT EBERL

A radioactive shipping package for transporting tritium has been developed for use by the National Nuclear Safety Administration as a replacement for the DOE Model UC-609, a tritium package developed and used by the DOE and NRC since the early 1970s. This paper presents the major design features and highlights the improvements made over its predecessor by incorporating new engineered materials and implementing improved testing, handling, and maintenance capabilities, while improving manufacturability. A discussion will be provided demonstrating how the BTSP is in compliance with the regulatory safety requirements of the Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission. The paper further summarizes the results of testing to 10 CFR 71 Normal Conditions of Transport and Hypothetical Accident Conditions events.

T16 - STRUCTURAL METHODS

4:00PM – 6:00PM – TECHNICAL SESSION
– CONFERENCE ROOM 2
CHAIR: TBC, CO-CHAIR: ROBERT GRUBB

ABSTRACT 124

Mechanical Design Assessment Approaches of Actual Spent Fuel and HLW Transport Package Designs

BERNHARD DROSTE, FRANK WILLE*, KARSTEN MUELLER, UWE ZENCKER

In recent years BAM finalized the competent authority assessment of the mechanical and thermal package design in several German approval procedures of new spent fuel and HLW package designs. The combination of computational methods and experimental investigations in conjunction with materials and cask components testing is the most common approach of mechanical safety assessment. The methodology in the field of safety analysis including associated assessment criteria and procedures has evolved rapidly during last years. New aspects relating to analysis aspects and assessment methodologies are summarized in this paper.

The design safety analysis has to be based on a clear and comprehensive safety evaluation concept, including defined assessment criteria and constructional safety goals. In general for new package designs the implementation of experimental package drop tests in the approval process should be obligatory. Additionally, pre- and post-test calculations as well as components or material testing could be important. Depending on the individual package construction, the materials used and identified safety margins in the design, it has to be justified whether and to what extend drop tests are necessary. Numerical calculations by means of the Finite-Element method are part of safety analysis concepts of different package design approvals. The calculations are carried out statically or dynamically depending on the particular loading situation (static load or impact) and material behavior (e.g. strain rate dependence). The use of an appropriate small-scale or a full-scale test model determines the extent and depth of the correlated calculations. Exact calculations require an input of realistic material laws which often have to be generated by appropriate material testing.

This paper concentrates on the complex relation between the chosen drop test program with a small scale model and related mechanical Finite-Element analyses for design verification, exemplarily described for a specific new German dual purpose cask design developed for transport of vitrified high-level waste from France to interim storage in Germany.

ABSTRACT 255

Acceptability of Dynamic Finite Element Analyses – Material Failure Approach

ANINDYA SEN*, IAIN DAVIDSON

IAEA [701] details how compliance with the regulatory tests may be achieved. The UK Competent Authority has always been open minded to new calculation procedures and has encouraged the use of Finite Element Analysis where appropriate. In the last few years there has been a marked increase in the complexity of some analyses, which support applications, with attempts made to model material failure in a conservative manner. We have noticed a range of failure theories and modelling methods that may, or may not be acceptable. This paper attempts to clarify our expectations with regard to modelling failure.

In elastic analyses, the main acceptance criterion is to check that the calculated stresses, e.g. Von Mises or Tresca do not exceed the lower bound yield stress/es for the material/s. If the stresses exceed the yield limit/s, the determining criterion shifts from stress to strain; especially under multi-axial load conditions. Elastic-plastic analysis using finite element approach, solicits, true stress-strain curve for the material/s to be incorporated. The most relevant material property for comparison now is the available engineering uniaxial ductility.

It is a global measure of failure, whereas, the local strains in the vicinity of the failure zone could be much higher. For a static analysis, the enumerated strains are available at the end of the analysis, whereas, in a dynamic analysis, the set period for analysis should be "long" enough to ensure that the response has been captured correctly at the end of the analysis. In most of the engineering applications, the state of stress is generally multi-axial, thus affecting the available ductility, which decreases exponentially under predominant tensile multi-axial stress state. The elastic-plastic assessments need to show that the final accumulated equivalent plastic strains are below the available multi-axial ductility for the respective material/s at the relevant temperature/s.

Extensive FE analyses related to regulatory requirements will be done. Comments will be made on mesh quality, input parameters and post-processing diagnostics in an attempt to guide the analyst to a fit-for-purpose FE model. Finally, a quantifiable and recommendable margin when comparing the strains as stated above will be suggested.

ABSTRACT 337

The Effect of Gaps on Response of a Spent Fuel Transportation Package Closure Lid During a Drop Impact

GORDON BJORKMAN*

During an impact event, gaps between the various components of a spent fuel transportation package may create secondary impacts that result in higher dynamic loads than would have occurred if the gaps had not been present. A condition of particular interest is the gap that may exist between the fuel assemblies and package closure lid and the effect this gap may have on amplifying the response of the closure lid during an impact.

Through the use of a simple dynamic model this paper investigates the effect of a secondary impact due to a gap between the package internals and the package closure lid on the response of the closure lid during a 30 foot end drop. The dynamic model consists of five components (parameters): (1) The mass of the internals traveling at the impact velocity for a 30 foot drop (44.4 ft/sec), (2) the gap between the internals and package lid, (3) the package lid, assumed to be a simply supported circular plate, (4) the modal mass of the lid, and finally, (5) an impact limiter that applies a constant deceleration to the package overpack. In addition, the dynamic model assumes elastic behavior. This is consistent with the Standard Review Plan (NUREG-1617), which recommends that the closure lid bolts and closure lid system within the region of the lid bolts remain elastic in order to demonstrate leak-tightness by finite element analysis.

The response results are presented in terms of the Dynamic Load Factor (DLF) for the closure lid. Response is shown to be a nonlinear function of the impact limiter deceleration, gap size and closure lid diameter and thickness. These results provide valuable insights into the parameters that effect response and show the conditions under which gaps may be of sufficient size to significantly influence response.

ABSTRACT 117

Numerical Simulation of 9 Meter Drop of a Transport and Storage Cask with Aluminium Impact Limiter

LINAN QIAO*, UWE ZENCKER, FRANK WILLE, ANDRE MUSOLFF

The drop test of casks from 9 meter height onto an IAEA target is one of the most important proofs for the transport safety of radioactive materials. In this case the casks are usually equipped with impact limiters to reduce the dynamic load on the cask by absorbing a major part of the kinetic energy.

Some casks are equipped with aluminium rings as mantle impact limiters to absorb the kinetic energy at the horizontal drop. Compared to other materials, aluminium has many advantages; it is homogeneous, relative robust and has high capacity for energy absorption.

The stress-strain relation of aluminium is needed in numerical simulations of the drop test scenario and should be given as a function of constant strain rate and constant temperature. Because of the heating of the specimen during the compression test at high strain rates, it is very difficult to get suitable isothermal stress-strain relations especially for large strains. Therefore, a stress-strain relation measured at a very low strain rate (quasi-static case) is isothermal whereas at very high (dynamic) strain rates adiabatic conditions prevail.

The mechanical deformation of an impact limiter at short times is a coupled thermo-mechanical dynamic process. The numerical simulation of this process needs much effort. Sometimes the real conditions during a drop test may be simplified as adiabatic. Nevertheless, isothermal stress-strain relations are needed as input parameters for popular numerical material models.

In this study an elastic-incremental plastic material model with strain rate hardening acc. to Cowper-Symonds is used for the development of isothermal as well as adiabatic stress-strain relations of aluminium from the compression test at constant ambient temperature. After that, two different simulation strategies are compared. At first, the drop test is calculated fully coupled, i.e. with isothermal stress-strain relations and possible heat generation in the material. Then the drop test is recalculated in a very simplified manner with adiabatic stress-strain relations from the compression test in an isothermal simulation. Both calculation strategies show similar results in the investigated load scenario. The limitations of the simplified approach are discussed.

ABSTRACT 339

Strain-Based Acceptance Criteria for Spent Fuel Storage and Transportation Containments

GORDON BJORKMAN*, DOUG AMMERMAN

Modern finite element codes used in the design of nuclear material transportation and storage casks can readily calculate the response of the packages beyond the elastic regime. Hypothetical accidents considered for transport packages include a 9-meter free drop onto an essentially unyielding target and a 1-meter free fall onto a 30-cm diameter puncture spike. For storage casks, accident conditions can include drops, tip-over, and aircraft impact. All of these accident events are energy-limited rather than load-limited, as is typically the case for boilers and pressure vessels. Therefore, it makes sense to have analysis acceptance criteria that are more closely related to absorbed energy than to applied load. Strain-based acceptance criteria are the best way to meet this objective.

As computational tools have improved cask vendors' ability to perform non-linear impact analysis, the need for a code-based method to interpret the results of this type of analysis has increased, and in 2006 the NRC encouraged ASME to develop strain-based criteria for energy-limited events.

An important aspect of the strain-based criteria is that it can only be applied to a "Quality Model:" where a Quality Model is defined as a model that adheres to the guidance set forth in the ASME Computational Modeling Guidance Document for Explicit Dynamics Software (currently being developed by the Task Group on Computational Modeling for Explicit Dynamics), or has been developed with the use of convergence and sensitivity studies. This paper will briefly discuss the efforts within the ASME, detail the advantages of using strain-based criteria, discuss the problem areas associated with establishing strain-based criteria, provide insights into inelastic analyses as applied to radioactive material transportation and storage casks in general, and discuss the NRC's efforts to incorporate strain-based acceptance criteria into the new revision of Regulatory Guide 7.6.

T8 - SPENT FUEL BEHAVIOUR

4:00PM – 5:40PM – TECHNICAL SESSION – CONFERENCE ROOM 3
CHAIR: ROLAND HUEGGENBERG,
CO-CHAIR: PETER PURCELL

ABSTRACT 100

Mechanical Safety Analysis for High-Burnup Spent Fuel Assemblies under Accident Transport Conditions

VIKTOR BALLHEIMER*, FRANK WILLE, BERNHARD DROSTE

Transport packages for spent fuel have to meet the requirements concerning containment, shielding, and criticality as specified in the IAEA-Regulations for different transport conditions. Physical state of spent fuel and fuel rod cladding as well as geometric configuration of fuel assemblies are, among others, important inputs for the evaluation of correspondent package capabilities under these conditions. The kind, accuracy, and completeness of such information depend upon purpose of the specific problem.

In this paper the mechanical behaviour of spent fuel assemblies under accident conditions of transport will be analysed with regard to assumptions to be used in the criticality safety analysis. In particular the potential rearrangement of the fissile content within the package cavity, including the amount of the fuel released from broken rods has to be properly considered in these assumptions.

In view of the complexity of interactions between the fuel rods of each fuel assembly among themselves as well as between fuel assemblies, basket, and cask body or cask lid, the exact mechanical analysis of such phenomena under drop test conditions is nearly impossible. The application of sophisticated numerical models requests extensive experimental data for model verification, which are in general not available. The gaps in information concerning the material properties of cladding and pellets, especially for the high-burnup fuel, make the analysis more complicated additionally.

In this context a simplified analytical methodology for conservative estimation of fuel rod failures and spent fuel release will be described. This methodology is based on experiences of BAM acting as responsible German authority within safety assessment of packages for transport of spent fuel.

ABSTRACT 211

Finite Elements Analysis of Inter-Grid Bending Tests on Used Fuel Rods Samples

MAURICE DALLONGEVILLE*, ARAVINDA ZEACHANDIRIN, PETER PURCELL, ANTHONY CORY

TN International and International Nuclear Services (INS) started in early 2000s a joint project, the Fuel Integrity Project (FIP), in order to develop a methodology to assess the response of Light Water Reactor (LWR) fuel assemblies (FA) during 9 meters regulatory drops. To this end, several series of mechanical tests were carried out on fresh and used fuel rods samples, including inter-grid bending tests on samples of used fuel rods with average burn-up of 50 GW.d/tU.

In this frame, a preliminary analysis of the commissioning test (test 11.1) results was presented during Patram 2004; in complement the analysis of the whole test series 11 (tests 11.1 to 11.6) is now presented. The used test span matches a typical inter-grid length of LWR FA. The load is applied at mid-span of the fuel rods samples by a pulley wheel. This test series leads to failures starting at a net lateral deflection of about 35 mm at room temperature and 60 mm at 500°C, and with few percent high total elongations.

Calculation of the whole tests series was carried out with the ANSYS code using a shell and brick model. The different mechanical phenomena occurring during the tests were distinguished and the adequate fuel rods material parameters were determined.

The determination of these phenomena by preliminary calculations and the models validation were followed by a sensitivity study of the parameters values in the material constitutive laws to insure a good agreement between the obtained forces / deflection curves and the actual tests curves.

This sensitivity study was all the more efficient and reliable as the effects of each material parameter appeared almost sequentially and cumulatively during the loading of the fuel rods samples.

Even though models improvements might be possible, the guidelines of the retained approach lead to reference maximum elongations at rupture in consistency with literature values for used fuel rods. The methodology to translate this Finite Element Analysis (FEA) of bending test series 11 results to an actual used FA during a 9 m lateral drop test is finally presented.

ABSTRACT 323

Investigation of Spent Fuel Integrity in Dry Storage at Japanese Nuclear Power Plants

TAKESHI FUJIMOTO*, MASAHIRO YAMAMOTO, MITSUO MATSUMOTO, KATSUHIKO SHIGEMUNE, HIROYUKI MATSUO

In Japan, the first interim spent fuel storage facility away-from-reactor (AFR) will start its operation in 2012.

This facility stores BWR / PWR spent fuel assemblies using dry metal dual purpose casks (storage / transport) which will be transported to their destinations after the interim storage for decades. This facility is not equipped with a hot-cell for opening the primary lid of the cask because one of the basic concepts of the facility is a simple operation not to handle a radioactive material directly, that reduces radiation exposure of workers and a risk of contamination troubles.

Abstracts – Monday 04 October 2010 : continued

On the other hand, a visual inspection of spent fuel assemblies is usually carried out before spent fuel transportation in Japan. Although a visual inspection of spent fuel assemblies is not carried out in the interim storage facility as has been discussed, we consider that it is necessary to confirm spent fuel integrity by the same level of confirmation as visual inspection before transportation after the interim storage. For this purpose, we will establish a quality control system of dry metal casks from manufacturing to the end of storage, and for more conservatism, we are continuously investigating spent fuel integrity in dry storage at the nuclear power plants (NPP).

This paper introduces results and future plans of our investigation of spent fuel integrity in dry storage in Japan, as shown below. (The investigation was also aimed at the integrity of metal gasket.)

Results of investigation

Investigation Year	Site of dry storage (Reactor type) Type of dry storage]	Storage period
2000	Fukushima-Daiichi NPP (BWR) [Dry metal cask]	Approx. 5 years
2005	Fukushima-Daiichi NPP (BWR) [Dry metal cask]	Approx. 10 years
2009	Tokai-Daini NPP (BWR) [Dry metal cask]	Approx. 7 years

Investigation items and results

- Internal gas sampling
  Result: Kr-85 was not detected.
- Visual inspection of spent fuel assemblies
  Result: The appearance of fuel assemblies remained the same as observed at the time the storage started.
- Leak tightness test of cask lid seal
  Result: Leak rate satisfied the criterion.
- Visual inspection of primary lid metal gasket and seal surface
  Result: A slight oxidation was observed on the gasket caused by the incomplete water removal.

Future plans

1) Periodically continual investigations at Fukushima-Daiichi NPP and Tokai-Daini NPP
2) Dry storage demonstration test using a test container which can accommodate two PWR spent fuel assemblies, etc.

ABSTRACT 137

Simplified Thermal Creep Model of an Irradiated Fuel Pin

CEDRIC LANGLADE*, MAURICE DALLONGEVILLE

TN International started recently a study of irradiated fuel claddings thermal creep under routine transport conditions to evaluate creep rupture risk of LWR fuel.

Fuel cladding thermal creep consists in local swelling of cladding parts submitted to internal pressure and high temperature. Cladding rupture occurs when a critical strain is reached. Creep rate is influenced by a combination of several parameters: temperature profile, stress level, irradiation and oxidation profiles. The study is focused on the hottest fuel pin during transport.

An axi-symmetric finite elements modelling of a fuel pin is generally used with different temperature profiles, temperature evolutions, irradiation profiles and internal pressures. Thermal creep behaviour includes irradiation defects and hardening recoveries with temperature. Claddings rupture criteria are based on a critical strain depending on irradiation damage and stress. The used creep laws and criteria have been developed by CEA R&D laboratories.

Analytical calculations are not possible because of creep behaviour non-linearity and large number of input parameters. Finite elements calculations (FEA) are very time-consuming and not flexible in practice. Therefore, TN International has developed a simplified FORTRAN calculation model.

The simplified modelling has the same input parameters as the FEA one. Cladding mesh has one element in thickness in order to calculate mean stress. Axial mesh refinement is chosen to give good description of temperature and irradiation axial profiles. The internal pressure is determined by thermodynamics balance. Creep laws and rupture criteria implementations allow switching from one law to another one easily. As for FEA modelling, rupture criterion consists in comparing at each node local creep strain reached to local limit strain. These calculations last only a few seconds versus several hours or days for FEA calculations. This simplified modelling is under validation by comparison with FEA results. It should allow making reliable calculations of rupture risk by thermal creep with large time and means benefits. With appropriate creep laws and rupture criteria, this method can be applied to casks vacuum drying, fire of half an hour during transport, routine transport during from a week to a year and interim storage for up to 40 years.

ABSTRACT 122

Development of the Swedish National Database for QA of Spent Nuclear Fuel

HENRIC LINDGREN*

The Swedish Nuclear Fuel and Waste Management Co (SKB) is responsible for all the back-end issues in the Swedish nuclear industry. SKB operates a central interim storage for spent nuclear fuel (Clab), and is responsible for research, development, construction and operation of a future final repository for spent nuclear fuel (SNF).

Data for all the nuclear fuel assemblies (FA) in Sweden is stored in a national database, administrated by SKB. The national database is used for the administration and quality assurance (QA) of handling, transportation and storage of SNF.

The database is continuously updated and includes data for all FA at the nuclear power plants (NPP), and all FA at the Clab facility. The data includes identification, fuel type, initial and irradiated masses for heavy metals, assembly enrichment and burnup, location (i.e. at NPP or Clab) and date for unloading. The database is an important tool for QA and administration of SNF for SKB. The QA consists of checking that fuel parameters are in accordance with the requirements in the certificate for the transport casks and the Safety Analysis Report (SAR) for the Clab facility prior to transportation from the NPP:s to Clab.

Internal requirements at SKB regarding more fuel data for the long term safety analysis at a final repository, and the fact that SKB plans to apply burn up credit for the Clab facility and later the final repository, requires a new database in order to meet future QA requirements. The paper describes the program for developing and replacing the current database in order to meet future requirements regarding more detailed fuel data, and to use the application for QA of burnup credit. The database will ensure that required fuel data follows the FA from the reactor to the closure of the final repository for SNF.

Abstracts – Tuesday 05 October 2010

T10 - HIGH LEVEL WASTE

9:00AM – 10:40AM – TECHNICAL SESSION – MAIN HALL
CHAIR: SAM DARBY, CO-CHAIR: JURGEN WERLE

ABSTRACT 259

European Experience in the First Transports of Universal Canisters Containing Compacted Metallic Waste Coming from Treatment

FLORIAN DARRAS*, JEAN PASCHAL, DAMIEN SICARD, STEPHANE BEAUVERGER, JEAN-LUC ARNOUX

In the AREVA NC La Hague plant, the structural elements of the used nuclear fuel resulting from the operation of the recycling facilities are compacted in the ACC workshop (ACC) with the aim of reduction in volume of the corresponding waste while achieving the standardization of final residue in a stainless steel Universal Canisters called CSD-C (Conteneur Standard de Déchets Compactés in French or Universal canister of compacted waste), as for the so-called CSD-V (Conteneur Standard de Déchets Vitrifiés in French, or Universal canister of vitrified waste) CSD-C contains a pile of compacted metallic disks.

Within the framework of the recycling contracts between AREVA NC and its European Customers, the return of CSD-C to their country of origin started in 2009 with the Netherlands and will continue on several years, with respect to customers' agreement. Taking the tentative 2010 transport program into account, the significant number of transports to The Netherlands, Switzerland and to Belgium makes TN International European experienced in the transportation of such residue.

Experience on preparation and execution of these transports is now significant. For instance, different types of existing casks were used. Their selection has been tailored to the specific situation of each customer, in order to provide the best economic solution while ensuring the service quality. As far as cask licensing procedures are concerned, various issues were solved allowing the granting of all package approval certificates for the CSD-C content. When necessary, transport equipments were adapted to the different cask designs taking all interface constraints into account. Indeed, lessons were drawn from the cask operations themselves: preparations of the casks, loadings at La Hague plant, road and rail transports, multi-modal transhipments and unloading at the delivery sites.

ABSTRACT 402

Application of the New Flask Type CASTOR HAW28M for the Return of Vitrified Residues from Sellafield, UK to Continental Europe

MARCO WILMSMEIER*, ANDREW GRAY

Highly Active Waste (HAW) in the form of Vitrified Residue (VR) has arisen from the reprocessing of overseas spent fuel at the Thorp plant at Sellafield. The VR product is contained within engineered canisters which must be loaded into a flask to enable their return to overseas customers.

The new CASTOR HAW28M flask will be used for VR returns to continental Europe. On the technical basis, GNS Gesellschaft fuer Nuklear Service mbH (GNS), International Nuclear Services Ltd (INS) and Sellafield Ltd (SL) are working together to ensure a smooth integration of this flask into the overall VR return system. For example, about 20 flasks of this type will have to be returned to the German interim storage facility at Gorleben.

When the project for the integration of the new flask type started its design phase, meetings between the above mentioned companies have taken place continuously to discuss the requirements and boundary conditions for specific equipment and reconstruction of the relevant Sellafield site and port facilities to support handling and transport of the CASTOR HAW28M flasks.

Empty and loaded flasks will have to be shipped by vessel between the UK and continental Europe. The first empty flask shipment is scheduled as a one-off transport in 2012 to accommodate the necessary commissioning of the flask in REF and subsequent cold trials. This transport will also support the cold trial of the transport route and transfer at transhipment locations.

Flasks arriving at Sellafield site are transferred via the Flask Marshalling and Storage Area to the Residue Export Facility (REF) where they are prepared for loading with VR canisters. The flasks are then sealed, checked and interim stored before being transported by rail from Sellafield to the port of Barrow. Here they are transferred safely onto a specialist ship in readiness for transport.

This paper will detail the application of the new flask and cover how the new and unique challenges from design and manufacture stages of transport and handling equipment through to the flask interface with Sellafield site and existing transport infrastructure.

ABSTRACT 212

Transportation of Vitrified High Level Wastes from Sellafield to Switzerland

JUSTO GARCIA*, FRANCOISE GENDREAU, ERIC VICTOR-PUJEBET

In the mid nineties, TN International started to design casks for the transport and the storage of vitrified high level wastes conditioned at AREVA La Hague recycling facility: TN 81 and TN 85 casks.

The TN 81 cask is currently licensed in France and in Switzerland and the TN 85 cask is currently licensed in France and in Germany. The first cask (a TN 81) was loaded in June 2004, and up today, 18 casks are stored in Switzerland (7 TN 81) or Germany (11 TN 85).

In order to face up to their obligations to get back the wastes issued of treatment their used fuel, Swiss customers asked TN International to propose a solution for the transportation of their high level wastes produced at the Sellafield site in the United Kingdom. TN International performed some investigations and analysis in order to check the conformance of the TN 81 cask with the new requirements. In particular TN International performed the following studies:

- Assessment of the compatibility of the content with the TN 81 specification,
- Assessment of the compatibility of the TN 81 cask with the interfaces at Sellafield plant,
- Identification of all interfaces modifications and design of specific tools required to receive the TN 81 at the Sellafield site and at the associated port of Barrow,
- Radiological assessment,
- Assessment of the proposed route from Sellafied plant to Zwilag, the central Würenlingen interim storage facility in Switzerland and

Abstracts – Tuesday 05 October 2010 : continued

of the logistical organization of the transport.

The purpose of this paper is to present this experience, and furthermore to underline our know-how and ability to manage the shipment to European Customers of vitrified high level wastes produced in United Kingdom.

ABSTRACT 62

Waste Transport Requirements to the Future Geological Repository

ALAIN ROULET*, THIBAUD LABALETTE

In France, the deep geological repository operations are scheduled to start in 2025, provided that it is authorized after the evaluation in 2015 of the license application and the future Act on reversibility requirements for the disposal. The feasibility of waste transportation from production sites to the repository has to be examined. The repository will handle both HLW and ILW-LL. Some of these wastes have already been transported to their "home" country after reprocessing (CSD-V glass canisters, CSD-C). However, for a large number of these wastes, no transport outside of the facility boundaries has yet been fulfilled.

This paper will recall the 2009 waste inventory of the repository and identify the needs for new transport packages, and the associated transport means. The transport flow to the repository is assumed to last about 100 years, therefore it is necessary to define the local integration of the transports to the future disposal. This will be a key point for the public debate in 2013. On this basis, this paper will also give an overview of the transport logistics, and on the associated infrastructures.

The local transport logistics (rail and/or road) and associated infrastructures shall take into account the siting of the repository and the results of the dialogue with local stakeholders. A sustainable development approach should lead to favor as much as possible railroad transport for the whole journey from the producers'sites. The waste producers are responsible of the transport operations to the disposal. The transport package designs have to be anticipated because they are an input data for repository design. Type B packages are compulsory for HLW and for a large part of ILW-LL. However, on-going studies show that some ILW-LL have a low specific activity (A2/g) and could be transported in industrial package. The inventory includes a large variety of waste that have been produced and conditionned long ago.

As usual, transport is a key factor in nuclear operations. Anticipation of the needs is compulsory in order to avoid difficulties in the future and provide the community with a reliable solution for waste transfer when the repository will be authorized.

ABSTRACT 52

Recognizing Interdependencies in the Design of the Nuclear Fuel Cycle and the Transportation of SNF and HLW

MARK ABKOWITZ*, DANIEL METLAY, NIGEL MOTE

Spent nuclear fuel (SNF) discharged from a nuclear power plant and high-level radioactive waste (HLW) generated during reprocessing are typically stored at their respective sites, often for prolonged periods of time. Eventually, however, both types of waste must be transported off-site, either to an interim storage facility or directly to a deep-geologic repository. This paper considers the interdependencies between nuclear fuel cycle options and the waste management transportation system and argues that both must be addressed as part of an integrated system. Two examples are presented to illustrate why this is important.

The first example draws from experience gained during development of the U.S. Department of Energy's (DOE) program for disposing of HLW and SNF in a repository at Yucca Mountain. In particular, DOE decided in 2005 to adopt the Transportation-Aging-Disposal (TAD) waste package design so as to minimize the handling and re-packaging of SNF at the repository surface facility. However, the fully loaded weight of the TAD design necessitated construction of a 330-mile rail line to the repository site. Further, this decision had significant implications on the loading requirements at the reactor sites as well as modal access options to the line-haul portions of the rail network.

The second example relates to operational changes now under way at nuclear power plants. Currently, burn-ups of 40-50 GWd/MTU are routinely achieved, and burn-ups of over 60 GWd/MTU are envisioned. In parallel with this, the length of time SNF is stored has been increasing, and storage periods of 100 years or more are now foreseen in some countries. While the performance of advanced fuel designs in the reactor is well established, the impact on fuel integrity of storage over such long periods and subsequently in transportation is not known, particularly for high burn-up fuel. As this may have implications for future handling operations, especially if repackaging is required prior to transport, these considerations should be taken into account as new fuel designs and transport packages are developed.

T11 - SECURITY (SESSION 2)

9:00AM – 10:40AM – TECHNICAL SESSION
– CONFERENCE ROOM 1
CHAIR: ANN-MARGRETH ERIKSSON, CO-CHAIR: TBC

ABSTRACT 208

Assessment and Approval of Reinforced Protection of Vehicles Used for the Shipment of Sensitive Nuclear Material

OLIVIER LOISEAU*, DELPHINE LARRIGNON, BRUNO AUTRUSSON

Most sensitive nuclear materials are usually shipped in specific vehicles with a reinforced protection; such vehicles are generally escorted, tracked and watched over from a distant control centre.

The competent authority defines the framework of a validation process starting with the design of the vehicle and ending with the vehicle protection approval. This paper focuses on the approval process of reinforced protection vehicles in France; it aims at showing how such a process may contribute to the security of nuclear material shipments. The paper notably focuses on the responsibilities of the operators, the competent authority and the technical support organizations.

This approval process of the protection of a vehicle allows the authority to ensure that the protection setup is effective and

Abstracts – Tuesday 05 October 2010 : continued

operational in order to protect the cargo from a malicious threat. In such a process, the authority defines the threat and the objectives of protection; the authority may choose, in certain case, to recommend protection devices or solutions; the need for recommendation versus objective definition mostly depends on the environment of the vehicle and the constraints induced. The authority may appeal to a technical support organization for evaluating technical solutions and their implementation. In the French security framework the role played by the technical support body is emphasized by the fact that IRSN houses the national control centre used for tracking of sensitive nuclear shipments. Due to that position, IRSN is acting in the definition of the technical solutions for tracking and securing the material during shipment.

Among all the conditions of approval of a vehicle protection, the paper presents the provisions to be taken by families depending on the sensitivity of the cargo. These provisions take the form of technical specifications to be observed such as embarked systems. The paper also describes the process starting from the vehicle protection design, including the appliance for approval, the technical instruction until the final approval. The paper also shows the approach followed to control the vehicles protection efficiency all along the period of their use.

ABSTRACT 150

Real-time Tracking of Nuclear Materials Packages in Transport

KUN CHEN*, HANCHUNG TSAI, YUAN SUN,
YUNG LIU, JIM SHULER

ARG-US, a real-time tracking system, has been developed to track and monitor the status of nuclear material packages in transport. The ARG-US system includes automatic alert/alarm notifications and is capable of tracing shipments at the detailed package level by using a combination of the following technologies:

- Radiofrequency identification (RFID),
- Global positioning system (GPS),
- Satellite communication,
- Geographic information system (GIS), and
- Secure database and web site.

RFID tags are attached to the packages to monitor the temperature, seal integrity, and shock and to report the data to a reader mounted in the vehicle. At preset intervals, the data collected by the reader and information on the vehicle's location are transmitted to a command center via a Qualcomm OmniTRACS satellite communication unit.

When an alert state is encountered, an alert message is immediately sent to trigger notifications. A dedicated, secure database and web server at the command center manages the data and messages from a single vehicle to multiple convoys. To check the status of an individual package, users have access to an encrypted web page on the server. The system also allows the user to send commands to the vehicle so certain operations, such as resetting the alert state or adjusting the alarm thresholds, can be performed remotely, without involving the driver.

In a transportation incident, a near real-time GIS report of the vehicle's location can be generated from the user terminal to aid the first responders. Several on-the-road tests have been performed to demonstrate the capabilities of the ARG-US tracking and monitoring system. The results of the tests are highlighted in this paper.

An effort is under way to integrate the system with the U.S. Department of Energy's TRANSCOM system, which has an established infrastructure for vehicle tracking and protocols for secure communication. The integrated ARG-US and TRANSCOM system is expected to further enhance safety, security and safeguards of nuclear material packages in transport. The integrated system can also be used for other radioactive materials, such as sources and by-product materials.

ABSTRACT 287

Global Identification and Monitoring of UF6 Cylinders

JESSICA WHITE*, JANIE MCCOWAN,
MARK LAUGHTER, MICHAEL WHITAKER

An expansion of global nuclear material commerce will accompany the forecasted worldwide renaissance in nuclear power. The increased commerce amplifies the risk that uranium hexafluoride (UF6) in cylinders could be mishandled, intentionally stolen, or diverted during transport. While there are typically few problems with UF6 cylinder shipments, the complicated logistics of international truck, rail, and sea transport often result in increased difficulty in locating shipments at any given time and can allow for significant delays in transport and reporting. The recognized threat of undeclared enrichment plants has made UF6 more attractive for attempted diversion. A global system of registering, identifying, and monitoring UF6 cylinders would provide more robust and timely assurance that no UF6 is stolen or diverted during transport.

The U.S. Department of Energy National Nuclear Security Administration's Office of International Regimes and Agreements (NA-243) has formed a multi-laboratory team to focus on universal UF6 cylinder identification and global monitoring. The team produced an overview report discussing the current situation and potential solutions and identified future tasks that fall into three general categories. The first category involves policy initiatives to bring all relevant stakeholders—facility operators and industry, state regulators and government agencies, international inspectorates, and technology developers—together to reach consensus on universal identification and the importance of cylinder monitoring. Such policy-level discussions would include standardization of components, protocols, and procedures. The second category concerns a deeper investigation of the concepts involved, reflected in the research and publication of papers and technical reports. Such background work is merely the "tip of the iceberg," but may facilitate acceptance of a global regime by the stakeholders. The third category involves technology assessments and field trials to enable the development of theoretical concepts into practical approaches. This paper gives an overview of the progress to date and examines the next steps for governments and international agencies (from the policy side) and industry and national laboratories (from the technology and development side).

Abstracts – Tuesday 05 October 2010 : continued

ABSTRACT 141

Sabotage of Radioactive Materials in Storage and Transport

KEN SORENSON*, ROBERT LUNA , BRUNO AUTRUSSON, OLIVIER LOISEAU, WENZEL BRUECHER, GUNTER PRETZSCH

Work related to the assessment of radiological health consequences resulting from a sabotage attack on a used nuclear fuel storage or transport cask has been on-going since the late 1970's. The level of effort in this area has been uneven over these three decades due to policy priorities, funding levels, and programmatic priorities of the countries funding this type of work. In spite of this uneven level of effort, substantial progress has been made in quantifying source term and resultant radiological consequence from a potential sabotage attack on a used fuel cask. This quantification of source term provides substantial justification to consequence assessments that heretofore have had to rely on conservative assumptions in lieu of empirical data.

One constant since the 1990's in addressing this problem has been an international working group whose primary focus has been to develop source term data from experimental simulations of sabotage-types of attacks. This working group, titled; the International Working Group for Sabotage Concerns of Transport and Storage Casks (WGSTSC) is comprised of experts mainly from the U.S., France, and Germany. Technical support has also been provided, on an intermittent basis, from the U.K. and Japan. The WGSTSC has pooled resources and expertise to design and conduct experiments that produce the data needed to perform radiological consequence assessments. In addition, this group has also conducted research on analytic techniques associated with high energy impact devices. The coupling of experimental data with advanced analytic techniques will provide for credible estimates of radiological consequence resulting from a sabotage attack.

This paper will review the history of experiments related to radiological consequences resulting from a sabotage attack on a used fuel cask. The history points to a narrowing of the conservatism typically assigned to these types of analyses due to a lack of data. The paper will also discuss the status of the work of the WGSTSC.

ABSTRACT 288

Two Radioactive Material Transport Security Incidents: Lessons Learned and Questions to Address

NORMAN KENT*, PAUL KENT

While it is true that radioactive material transport security is regulated by the several competent authorities and national, state, and local agencies, and controlled by those who consign the transport, those who manage the transport, those who carry out the transport, and those who receive, it is also true that security-related incidents still happen. Two such incidents recently occurred, involving Westinghouse and other organizations, that demonstrate (1) even "simple" lapses in procedural compliance can result in a significant response effort and (2) a perfectly compliant shipment can cause momentary international agitation.

The first is a successful multi-corporate, multi-agency response to an international shipment of low enriched UF6 that arrived at the receiving site with no tamper-safe seals intact. The second is an equally successful response to an immanent international LEU powder consignment that would transit the United States enroute from Europe to South America, and which received Department of Homeland Security scrutiny as a potential target for diversion of radioactive material. The matter was quickly and efficiently resolved, again with a multi-agency, multi-corporate response, but it left questions unanswered and prompted the need to look at provisions governing such consignments.

The paper intends to review what happened in each case, and discuss lessons learned, questions that remain, and the potential impact on future transport operations.

T12 - THERMAL STUDIES

9:00AM – 10:40AM – TECHNICAL SESSION
– CONFERENCE ROOM 2
CHAIR: CHRIS FRY, CO-CHAIR: TBC

ABSTRACT 411

Regulatory Fire Test Requirements for Plutonium Air Transport Packages: JP-4 or JP-5 vs. JP-8 Aviation Fuel

CARLOS LOPEZ*, VERNON F NICOLETTE

For certification, packages used for the transportation of plutonium by air must survive the hypothetical thermal environment specified in 10CFR71.74(a)(5). This regulation specifies that "the package must be exposed to luminous flames from a pool fire of JP-4 or JP-5 aviation fuel for a period of at least 60 minutes." This regulation was developed when jet propellant (JP) 4 and 5 were the standard jet fuels. However, JP-4 and JP-5 currently are of limited availability in the United States of America. JP-4 is very hard to obtain as it is not used much anymore. JP-5 may be easier to get than JP-4, but only through a military supplier. The purpose of this paper is to illustrate that readily-available JP-8 fuel is a possible substitute for the aforementioned certification test. Comparisons between the properties of the three fuels are given. Results from computer simulations that compared large JP-4 to JP-8 pool fires using Sandia's VULCAN fire model are shown and discussed. Additionally, the Container Analysis Fire (CAFE) code was used to compare the thermal response of a large calorimeter exposed to engulfing fires fueled by these three jet propellants. The paper then recommends JP-8 as an alternate fuel that complies with the thermal environment implied in 10CFR71.74.

ABSTRACT 410

Fire Tests and Analyses of a Rail Cask-Sized Calorimeter

CARLOS LOPEZ*, AHTI SOU-ANTTILA, MILES GREINER

Three large open pool fire experiments involving a calorimeter the size of a spent fuel rail cask were conducted at Sandia National Laboratories' Lurance Canyon Burn Site. These experiments were performed to study the heat transfer between a very large fire and a

large cask-like object. In all of the tests, the calorimeter was located at the center of a 7.93-meter diameter fuel pan, elevated 1 meter above the fuel pool. The relative pool size and positioning of the calorimeter conformed to the required positioning of a package undergoing certification fire testing. Approximately 2000 gallons of JP-8 aviation fuel were used in each test. The first two tests had relatively light winds and lasted 40 minutes, while the third had stronger winds and consumed the fuel in 25 minutes. Wind speed and direction, calorimeter temperature, fire envelop temperature, vertical gas plume speed, and radiant heat flux near the calorimeter were measured at several locations in all tests. Fuel regression rate data was also acquired.

The experimental setup and certain fire characteristics that were observed during the test are described in this paper. Results from three-dimensional fire simulations performed with the Cask Analysis Fire Environment (CAFE) fire code are also presented. Comparisons of the thermal response of the calorimeter as measured in each test to the results obtained from the CAFE simulations are presented and discussed.

ABSTRACT 2

Transport of UF6 and the Future of Thermal Compliance

TIM KORBMACHER*, MARC-ANDRE CHARETTE

The International Atomic Energy Agency (IAEA) Transport Safety Regulations (TS-R-1) require, amongst other, compliance with the standard thermal test for packages designed to contain 0.1 kg or more of uranium hexafluoride (UF6).

This compliance with TS-R-1 requires H(M) or H(U) approvals for packages involved.

The H(U) approvals are currently based on the use of thermal protectors on large UF6 cylinders (mainly 48Y).

The thermal protectors most in use are the so-called Blanket Thermal Protector (BTP) and the Composite Thermal Protector (CTP).

The use of BTPs and CTPs started in early 2005 and more than 4 years of experience is available now. The paper will review this experience.

Following the development and approval of the BTP/CTP, further work on the computer modelling and analysis used in the approval process has been started, in order to improve the precision of the thermal case.

With more refinement in the calculations and additional support of physical testing a demonstration for thermal compliance without additional thermal protectors on standard UF6 cylinders shall be considered. The paper will report on the status of this work.

ABSTRACT 407

Thermo-mechanical Study of Bare 48Y UF6 Containers Exposed to the Regulatory Fire Environment

CARLOS LOPEZ*, DOUGLAS J AMMERMAN, MARC-ANDRE CHARETTE, TIM KORBMACHER

Most of the regulatory agencies world-wide require that containers used for the transportation of natural UF6 and depleted UF6 must survive a fully-engulfing fire environment for 30 minutes as described in 10CFR71 and in TS-R-1. The primary objective of this project is to examine the thermo-mechanical performance of 48Y transportation cylinders when exposed to the regulatory hypothetical fire environment without the thermal protection that is currently used for shipments in those countries where required.

Several studies have been performed in which UF6 cylinders have been analyzed to determine if the thermal protection currently used on UF6 cylinders of type 48Y is necessary for transport. However, none of them could clearly confirm neither the survival nor the failure of the 48Y cylinder when exposed to the regulatory fire environment without the additional thermal protection.

A consortium of five companies that move UF6 is interested in determining if 48Y cylinders can be shipped without the thermal protection that is currently used. Sandia National Laboratories has outlined a comprehensive testing and analysis project to determine if these shipping cylinders are capable of withstanding the regulatory thermal environment without additional thermal protection. Sandia-developed coupled physics codes will be used for the analyses that are planned. A series of destructive and non-destructive tests will be performed to acquire the necessary material and behavior information to benchmark the models and to answer the question about the ability of these containers to survive the fire environment. Both the testing and the analysis phases of this project will consider the state of UF6 under thermal and pressure loads as well as the weakening of the steel container due to the thermal load. Experiments with UF6 are also planned to collect temperature- and pressure-dependent thermophysical properties of this material.

ABSTRACT 191

Model Development and Computational Analysis of the TUK46 Package with Uranium Hexafluoride (UF6) in Fire Environments

VYACHESLAV SHAPOVALOV*, SHOTA POPOV, YURIY POPOV, BORIS BARKANOV, ALEKSANDR MORENKO

Using the elements of nonequilibrium thermodynamics we receive the ratios describing UF6 phase transformations in casks in fire environments. Based on these ratios the algorithms and computational program AJAX-UF6 are developed, that take into account the processes of melting and solidification, boiling, evaporation and condensation.

The process of aggregate transformations results in changes of volumes of solid, liquid and gas phases. Gas phase density and pressure are also changed. The gas phase density is also increased due to liquid UF6 evaporation. The system UF6 + a cask is considered as an integrated system with three areas changed in time, which are correspondent to UF6 different aggregate states.

The AJAX-UF6 program enables us to conduct computational study of TUK-46 with outer diameter of 1.216m. The following variants were discussed: TUK casing thickness of 8mm and 16mm, UF6 mass of 7500 kg and 5000kg. Fire environments were studied as thermal impact in accordance with the IAEA requirements – 800°C during 30 minutes and the degree of fire medium emissivity of 0.9. Based on the computations we obtained temperature and pressure unsteady fields in the structure areas changing in time.

Using the same program we have performed computational analysis of the TUK-48Y model. Experimental data for this model in

Abstracts – Tuesday 05 October 2010 : continued

fire environments similar to the IAEA fire model are available. The computational results are well conformed to experimental results.

It is anticipated that the AJAX-UF6 program can be used to perform computational justification of various casks safety that are used for international and domestic transportations of depleted and enriched UF6 (TUK-46, 48X, 48Y, etc.)

T9 - BURN-UP CREDIT

9.00AM – 10:20AM – TECHNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: HELMUT KUHL, CO-CHAIR: LUDYVINE JUTIER

ABSTRACT 327

Development of Technical Basis for Burnup Credit Regulatory Guidance in the United States

CECIL PARKS*, JOHN WAGNER, DON MUELLER, IAN GAULD

Taking credit for the reduction in reactivity associated with fuel depletion can enable more cost-effective, higher-density storage and transport of spent nuclear fuel (SNF) while maintaining a sufficient subcritical margin to establish an adequate safety basis. In the United States, there has been and continues to be considerable interest in the use of burnup credit as part of the safety basis for SNF systems and this interest has motivated numerous technical studies related to the application of burnup credit for maintaining subcriticality. Responding to industry requests and needs, the U.S. Nuclear Regulatory Commission (NRC) initiated a burnup credit research program, with support from the Oak Ridge National Laboratory (ORNL), to develop regulatory guidance and the supporting technical basis for allowing and expanding the use of burnup credit in pressurized-water reactor SNF storage and transport applications. Considerable progress has been achieved in many key areas in terms of increased understanding of relevant phenomena and issues, availability of relevant information and data, and preparation of updated regulatory guidance. This paper will review the technical studies performed by ORNL for the U.S. NRC burnup credit research program. Examples of topics that have been addressed include:

1) reactivity effects associated with reactor operating characteristics, fuel assembly characteristics, cooling time and assembly misloading;
2) experimental data and approaches for estimating uncertainty in the predicted subcritical margin; of criticality calculations; and
3) operational issues and data related to assembly burnup confirmation.

The objective of this paper is to provide a summary of the recent work and significant accomplishments relative on the technical basis for modifying the U.S. regulatory guidance for transport and storage of SNF.

ABSTRACT 270

Representativity Study of the French HTC and FP Experiments for Burn up Credit Application to the TN 24 E Transport and Storage Cask

M. TARDY, C. GARAT*, S. KITSOS, F. RIOU, P. SOUBOUROU, M. LEIN, F. BERNARD, I. DUHAMEL, T. LECLAIRE IVANOVA

The criticality calculations for a transport and storage cask containing irradiated fuel assemblies associated with the burn-up credit practice needs the validation of the criticality codes and the associated cross-section libraries. One of the requirements for the criticality code validation is to demonstrate the similarity between the selected set of critical experiments and the cask configuration.

A set of "Haut Taux de Combustion" (HTC) and Fission Products (FP) experiments, co-funded by the AREVA group and IRSN, was selected for the validation of the criticality code to be applied on the TN 24E cask loaded with one-cycle irradiated UO2 fuel assemblies (12 GWd/tHM).

Similarity studies require in a first step to identify the neutron-physical parameters of most influence for the cask and the selected critical experiments. The traditional method for assessing the similarity between experiments and the industrial application is to compare spectral parameters (EALF, H/X, Vmoderator/Vfissile, etc..) or reaction rates for uranium, plutonium, and fission products. This method applied to the TN 24E cask, by using the HTC and FP experiments, shows that the TN 24E configuration is bounded in terms of spectral parameters by the selected HTC and FP experiments.

An alternative way to study the similarity is to use sensitivity/uncertainty (S/U) analysis methods. As a further assessment for the TN 24E cask application, its similarity with the selected set of HTC and FP experiments is analysed using the TSUNAMI codes of the SCALE 5.1 package. The TN 24E cask sensitivity profiles obtained with the TSUNAMI-3D module are compared to the selected FP and HTC profiles. It is shown that the profiles of the selected experiments match well those of TN 24E cask configuration. Comparisons of the correlation coefficients (ck) calculated with the TSUNAMI-IP module further demonstrate a good correlation between the selected set of HTC and FP experiments and the TN 24E cask configuration.

The full paper will present the results of the similarity study based upon comparison of both the traditional neutron parameters and the sensitivity data.

ABSTRACT 144

Application of Tsunami and Tsurfer for Validation of Burn-up credit in the Criticality Safety Analysis of a Transport Cask

MATTHIAS BEHLER*, ROBERT KILGER, MATTHIAS KIRSCH, MARKUS WAGNER

In up-to-date criticality safety analysis of spent nuclear fuel modern calculation methods are applied to take into account the reduction of reactivity of nuclear fuel due to the burn-up process. The use of these methods has to be validated by comparison to experimental data and usually this leads to a bias, which has to be considered in the neutron multiplication factor keff. Applying the so-called burn-up credit, this task is complex since any fission products and actinides

considered in the calculation have to be validated by adequate experiments.

However, for typical applications like a spent fuel transport cask there are no free available experimental data which directly match the conditions of an application and include all the fission products typically being used. Thus the user is obliged to validate the fission products separately by choosing experimental data which match the conditions of the application at least partially and include one or more of the fission products of interest. In this case a dedicated analysis tool named TSUNAMI can be used to quantify the similarity of an experiment to the respective application. Since 2009 new tools in the latest version 6 of the American code package SCALE ("Standardized Computer Analyses for Licensing Evaluation") from Oak Ridge National Laboratory have been provided to study and quantify the bias and uncertainty of an application calculation based on the validation calculations of experiments.

We are applying these tools, amongst those especially TSUNAMI and TSURFER, to a generic cask model and study their potential with regard to a possible validation. The experimental data used are taken from the International Criticality Safety Benchmark Evaluation Project (ICSBEP), an internationally supported benchmark database of highest quality. TSURFER is intended to allow for the determination of the bias of a computation even if no experiment exactly matching the application condition is available. Special attention will be drawn to the influence of the fission products on the bias and the reliability of this bias in dependence on the available experiments.

ABSTRACT 143

Burn-up Credit Implementation for Transport and Storage Casks of UO2 Used Fuel Assemblies

MARCEL TARDY*, STAVROS KITSOS

The concept of taking credit for the reduction in the reactivity of nuclear fuel due to burn-up of the fuel, is referred to as « Burn-up Credit » (BUC). Consider the reactivity credit for used fuel offers an improvement of casks performance and an economical interest.

TN International currently uses BUC methodology for the design of casks dedicated to the transport of PWR UOx used fuel assemblies. As long as the fuel enrichment of the PWR fuel assemblies was sufficiently low, a simplified BUC methodology based on the sole consideration of 8 major actinides and the use of a partial burn-up was satisfactory to cover the needs without necessity to design new casks.

Nevertheless, the continuous increase of the fuel enrichment during the last decade has led TN International to continue the investigations on new BUC methodology in order to limit both the increase of the neutron poison content inside the new basket designs and the burn-up constraints attached to the acceptability of the fuel assemblies for transport. The strategy of TN International was then to take benefit of the large reactivity reserves, which might be gained by the consideration of the main fission products (103Rh, 133Cs, 143Nd, 149Sm,152Sm and 155Gd) that make up 50% of the anti-reactivity of all fission products and a more realistic axial profiles of burn-up instead of a uniform axial burn-up profiles .

The "BUC" calculation route for PWR UOx used fuel is based on the connection of the French depletion codes DARWIN 2.1.1 and the French Criticality-Safety Package, CRISTAL V1.0 which are developed by the CEA and the IRSN in collaboration with French nuclear industry. French BUC experimental programs have been separately performed in Cadarache (France) and in Valduc (France) in order to validate respectively DARWIN 2.1.1 depletion code and CRISTAL V1.0 Criticality-Safety code system.

The aim of this article is to present for the transport and storage casks, the new BUC methodology apply at TN International including the calculation procedure and all conservatives associated assumptions

T14 - LARGE COMPONENTS

11:00AM – 12:40PM – TECHNICAL SESSION – MAIN HALL
CHAIR: RICK BOYLE, CO-CHAIR: HELMUTH ZIKA

ABSTRACT 131

Transport of Large Components in Germany – Some Experiences and Regulatory Aspects

FRANK NITSCHE*, CHRISTEL FASTEN

After decommissioning of nuclear facilities it is very often necessary to transport large components such as steam generators or reactor pressure vessels in public areas.

In Germany, such shipments were carried out in 2007 and 2008 as follows:

- steam generators from the Nuclear Power Plant (NPP) Stade to Studsvik/Sweden by road and sea
- reactor pressure vessel from NPP Rheinsberg to the interim storage facility near Greif-swald by railway and
- steam generators from the NPP Obrigheim to Greifswald as well by road and inland wa-terway.

The paper describes the experiences with these shipments including radiation dose assess-ments to transport workers as well as the main aspects of the applied regulatory procedure by special arrangement, for which the Federal Office for Radiation Protection (BfS) is the compe-tent authority in Germany. A high level of safety could be achieved for all involved modes of transport (road, rail, sea and inland waterways).

Based on these experiences some regulatory aspects will be discussed finally which include classification issues of large components within the current IAEA Transport Regulations, the safety concept and the use of special arrangements for such shipments and options for the fur-ther development of the IAEA Transport Regulations to achieve more specific and internationally harmonised conditions or requirements for shipment of large components.

ABSTRACT 43

Transport of Large Nuclear Power Plant Components - Experiences in Mechanical Design Assessment

STEFFEN KOMANN*, BERNHARD DROSTE, FRANK WILLE

In the course of decommissioning of nuclear power plants in Germany large nuclear components must be transported over public traffic routes to interim storage facilities.

In dependence of classification of the package it is necessary to

Abstracts – Tuesday 05 October 2010 : continued

subject the package to different mechanical tests according to the transport regulations.

Since it concerns surface contaminated objects (SCO) or low specific activity materials (LSA), a safety evaluation considering the IAEA transport regulations mainly for industrial packages (Type IP-2) is necessary.

For Type IP-2 packages the mechanical assessment under normal conditions of transport is required - a free drop of the package onto an unyielding target and a stacking test has to be investigated.

Large components are unique packages, therefore it is not possible to choose experimental testing as assessment method. The application of a complex numerical analysis for mechanical proof is necessary.

The assessment of the loads takes place on the basis of local stress distributions, also with consideration of radiation-induced brittleness of the material and with consideration of current scientific investigation results.

The large nuclear components have typically been transported in an unpackaged manner, so that the external shell of the component provides the packaging wall.

According to the present IAEA regulations the drop position is to be examined, which causes the maximum damage to the package.

In case of a transport under special arrangement a drop only in an attitude representing the usual handling position is necessary.

The paper will represent the methods, which are used for the evaluation of the mechanical integrity of the package for transport approval and will present recent transports of a reactor pressure vessel and of steam generators in Germany.

ABSTRACT 86

Qualification of Steam Generators for Shipment with Respect to the Requirements of TS-R-1

WILLI SCHIFFER, FRANZ HILBERT (MICHAEL KUEBEL PRESENTING)

The paper presents the technical and organizational measures for the qualification of two steam generators for shipment under special arrangement from the nuclear power station KWO to the interim storage site of EWN GmbH in Greifswald, Germany by road and barge on inland waterways.

The first issue to be solved was the categorization of the steam generators with respect to the Regulations. Based on contamination and radioactivity SCO-II was the only possible classification with one uncertainty left. Due to the design of the inner surfaces which are not accessible it could not be proved that all inner parts of the steam generators comply with SCO-II.

Secondly, regarding their mass and geometry the steam generators could not be fitted into a packaging. For this reason they were qualified themselves to fulfill the requirements towards an industrial package of type IP-2 except of the position which leads to maximum damage during the drop test.

The third problem to be solved was to comply with the dose rate limits for conveyances. One of the steam generators has been equipped with an additional shielding to meet the admissible value in a distance of two meter from the road vehicle.

The last issue was the development of a tie-down system for the large parts to the heavy cargo trailers used for the road transport and for the barge. NCS devised a solution which provided adequate safety, reduced radiation exposure of the personnel and was economically favorable.

As the radiation protection for such big components is of particular importance a special radiation protection programm was implemented for the shipment. All employees had been equipped with film badges to get an overview about the personal doses after shipment.

To comply with the requirements for quality assurance written instructions for handling and controlling during preparation, loading and unloading of the steam generators were set up.

ABSTRACT 223

Transportation of Solid Irradiated and Contaminated Non-fuel Radioactive Material in Large Transportation Package

MARLIN STOLTZ SR*, JAYANT BONDRE

Currently in the United States non-fuel bearing solid irradiated material generated during the operation of nuclear power plants is transported offsite in smaller transportation packages depending on the availability of the disposal sites. In the meantime, these materials are stored at the reactor sites generally in used fuel pools posing challenges for used fuel pool space and capacity management at some of the utilities. Examples of irradiated and contaminated hardware include: Control Rod Blades, Local Power Range Monitors, Fuel Channels and Poison Curtains for BWRs and Burnable Poison Rods Assemblies for PWRs. Decommissioned plants also have a need to dispose of segmented reactor vessel and internals. There is a need for disposal of these materials using a suitable larger transportation package.

Transnuclear Inc. belonging to Logistics Business Unit of AREVA has designed a Radioactive Waste Container (RWC) that can be used to package these materials on site. The RWC can then be transported offsite in the MP197HB Transportation Package as a payload to a disposal facility. Some used fuel pools are limited in available space for storage of RWC. In such cases, RWC is designed to be transferred from the used fuel pool to a temporary onsite storage facility. The unique designs of RWC and MP197HB Transportation Packages allow onsite storage of a partial or a fully loaded RWC at the plant site either in a used fuel pool or in a temporary onsite over pack before transportation. The design features of the RWC allow for repeated intermittent loadings of these materials for better packaging efficiency, higher packaging density for radiation dose, cost and space savings to the user.

This paper examines some of the unique design features of RWC and MP197HB Transportation Package that allow user to realize the benefits of using larger package for transportation and disposal.

ABSTRACT 360

The Transport Of Large Front End Facility Components From Decommissioning Operations

JURGEN WERLE*

The transport of large components from decommissioning operations raises a number of important issues in terms of regulatory compliance, safety and cost effectiveness. Transports under special arrangements are commonly used and are a viable option.

Abstracts – Tuesday 05 October 2010 : continued

However, the general perception of special arrangements needs to be improved. New methods and controls are proposed to facilitate characterisation of the contents and their classification. These new developments could eventually lead to regulatory change. WNTI has contributed largely to the new fissile exceptions being proposed during the 2009 revision cycle of TS-R-1. Many of these proposals have been made from WNTI member companies in order to improve efficiency while maintaining the same high level of safety. The transport of diffusers at the AREVA Georges Besse plant serves as an example to illustrate the practical challenges.

T15 - FRONT END AND RECYCLED MATERIAL

11:00AM – 12:40PM – TECHNICAL SESSION – CONFERENCE ROOM 1
CHAIR: MARC-ANDRE CHARETTE,
CO-CHAIR: AL STRATEMEYER

ABSTRACT 256

Front-end Transports: Challenges to 2020

PERRINE RUSSIAS*

During this decade, the front-end industry will face many challenges linked to the nuclear renaissance. In each sector of the nuclear cycle, the industry will invest, innovate, creating new nuclear sites and increasing the whole front end business, to answer the growing demand.

Period of growth are often difficult to manage, with uncertainties in the time frame, the investments amounts and their returns. Some of the investments planed as of today will become reality, some will be delayed, some will never be completed, and other could be launched.

The transport business will therefore have to follow this trend, and best anticipate it: new routes, new sites, new packages, while providing the best safety.

This paper will describe some of the challenges that front end transports will have to face before 2020 in this moving picture:

- Increasing volumes of the Uranium primary production: growth in Africa and Kazakhstan and associated logistics, maritime routes, packages …
- Feeding the new US Enrichment facilities: new balance with conversion capacity leading to road transport in North America and less transatlantic flows.
- Define adapted packages to deliver the fuel of the GEN3+ reactors like EPRTM
- New logistics scheme for tails material, according to tails management defined by the enrichment Companies,

In addition, evolution in the regulations and in the media means will also be part of the challenges.

The aim of the transport companies, and especially for the AREVA Logistics BU, is keeping the routes open and the industrial flows as fluent as possible, so that the nuclear industry could stay competitive and flexible.

The solutions are not yet defined, the transport companies and the industry have to work together to assure a smooth transition and take advantage of this period of growth to innovate: taking the best from our past 45 years experience and building the logistics for the next generation.

In that respect, the AREVA Logistics Business Unit has placed innovation as a core strategy.

Also, initiatives taken through WNTI and WNA for instance to address those challenges are to be supported for the benefit of all the industry.

ABSTRACT 249

Standard for Uranium Ore Concentrate Transport Drum

FABIEN PERRIN*, MARC DE SAILLY, PASCAL DE BASTIANI, WILLIAM MARTIN

A large number of mines spread all over the world provide few converters with uranium ore concentrate (UOC). The worldwide trade of UOC follows the usual commercial lines to optimize transportation cost and is mostly multimodal.

To comply with both mining and conversion facilities, UOC is often packaged in 210 L steel drums stowed in dry sea ISO containers. Drum's technical features and stowage system vary widely from origin, so does transport efficiency in terms of:

- Transportation cost: drum stacking – a very efficient way to reduce cost – isn't widely used.
- Internal contamination: transportation conditions impose rough constraints on drums, and containment of UOC powder is challenged.
- Environmental impact and dismantling: some producers use 800 kg of wood per container to block drums; this material needs to be dismantled at the arrival site.

Since 2007, AREVA has been reviewing its way of transporting UOC on its most constraining route from mines in Niger to conversion site in France.

The purpose of this article is to share AREVA experience on this field. After a presentation of the transportation constraints and industry current practices, this article will focus on the technical solution defined for drums and stowage inside the 20 foot ISO container. It will also include the description of the validation process which involved both laboratory and on-field tests.

This proven solution could be considered as a reference for optimizing UOC transportation and eventually become an industry standard.

ABSTRACT 103

Transportation of Reprocessed Enriched Uranium

FRANZ HILBERT*

In the last few years the volume of reprocessed enriched Uranium (RepU) transportation increased considerably and became an important issue. While the transport of enriched commercial grade Uranium is standard practice since decades the transport of RepU presents new challenges.

Reprocessed Uranium is specified in ASTM C 996 "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 % 235U". The limits given there for fission products and actinides are sufficiently low, however the limit specified for 232U of 0.05 µg/gU leads to considerably consequences.

Abstracts – Tuesday 05 October 2010 : continued

The A2-value of 232U is rather small and a concentration of roughly 2 % of the limit specified in ASTM C 996 will exceed the radioactivity allowed in type A packages which are used for the transport of commercial grade Uranium. A possible solution to this problem could be the declaration of RepU as LSA-II and the use of IP-2 packages containing fissile material. However, in some countries it is not allowed to approve IP-2 packages for fissile material. The consequence is hence to use type B(U)F packages.

A further problem is the decay of the aforementioned 232U. In the decay chain there are nuclides with a high gamma yield at high energies, especially the nuclide 208Tl. The gamma source intensity of fresh RepU increases in one year by a factor of roughly 200, in 2 years by a factor of 350, and in 10 years by a factor of 600!

This phenomenon is treated quite differently by different competent authorities. One certificate of package approval contains a rather restrictive limit for 232U which ensures that the limits for the dose rates are kept at all times. Other certificates specify no limit for 232U or restrict the time between reprocessing/enrichment and transport. In this case, even if the limit given in ASTM C 996 is not exceeded, the dose rates might exceed the limits specified in the Regulations by a factor of up to 3.

The presentation will illustrate the challenges mentioned and discuss the consequences for transportation of RepU.

ABSTRACT 78

Future Perspective for MOX Transport Based on Experience in JAEA

TAKAFUMI KITAMURA*, NOBORU TADOKORO, KAN SHIBATA, YUICHIRO OUCHI

Over 35 years, the Japan Atomic Energy Agency (JAEA) to promote nuclear basic research and nuclear fuel cycle done by its predecessors; JAERI and JNC (PNC) has been accumulating experiences to transport wide range of nuclear materials: MOX fuels and powder for fast breeder reactors, MOX spent fuel for post irradiation examination and uranium fuel elements for test / research reactors, etc..

This paper introduces our experiences in transporting MOX fuels and powder, and additionally technology and system in terms of transport method, emergency preparedness and quality assurance developed reflecting trends in Japan and world, then shows one of future perspectives for MOX transport.

Domestically, land transport of MOX fuels to the Experimental Fast Reactor (FR) "Joyo" and the Advanced Thermal Reactor (ATR) "Fugen" began in late 1970s, which leads to over 100 times shipments by 2008 of 7.9 ton Pu for Joyo and 45 shipments completed in 2007 of about 134 tons in total for ATR Fugen, respectively. Based on those experiences of safe and secure transport, land transport of MOX fuels for the Demonstration Fast Breeder Reactor (FBR) "Monju" started in 1992, which leads to 15 shipments by 2009. Internationally, 1.7 tons of returnable PuO2 under bilateral treaty was shipped from the reprocessing plant in France to Japan by a purpose built ship in late 1992.

In order to attain the safe and secure MOX fuel transport, not only safety measures complying with domestic regulations which incorporates the IAEA safe transport regulations; TS-R-1 but also physical protection measures corresponding to the Category I requirements by INFCIRC/225/rev4 as well as additional demands are considered and secured to a transport system as a whole. Those include emergency preparedness in accidents, training and exercise for transport workers, information control, etc. Development of packages which relate to transport meth-ods also should be addressed.

Currently, JAEA is planning to transport a large amount of MOX powder recovered from spent fuel of LWRs of electric companies from the Rokkasho Reprocessing Plant (RRP) in northern Japan to the Tokai Plutonium Fuel Production Facility (PFPF). In conclusion, future perspective for MOX transport will be discussed.

ABSTRACT 170

Packaging and Transboundary Transport of PuO and MOX Material

FRANCESCO D'ALBERTI*, ROBERTO DONATI, STEPHANIE LUTIQUE, ROBERTO VESPA

The Joint Research Centre (JRC) is a Directorate of the European Commission, distributed over sites in Belgium, Germany, Italy, the Netherlands and Spain. Its mission is to provide customer-driven scientific and technical support for the conception, development, implementation and monitoring of EU policies.

Peripheral to its fundamental mission are the Decommissioning and radioactive Waste Management (D&WM) activities associated with the JRC's nuclear installations constructed and operated on its research sites during the early years of the Centre. Since the 1980s, the JRC's evolving mission has progressively reduced the need for nuclear installations, that must now be decommissioned. The removal of nuclear material from an installation is a prerequisite to start the decommissioning.

This paper reports the successful experience at the JRC Italian site (Ispra) on the conditioning, packaging and transfer of Plutonium (PuO) and Mixed Oxides (MOX) reference materials used for over twenty years, from the Ispra PERformance LAboratory (PERLA) to the original owner abroad. In particular the design, licensing and construction of new glove box lines and auxiliary tools as well as the revision of safety provisions for material containers handling and transport off-site are described.

A significant return of experience has come from the preparation of the material for loading into the transport casks with particular reference to unexpected occurrence in the original packaging and preparation for recovery procedures in case of major unknown events.

T13 - CRITICALITY ANALYSIS

11:00AM – 12:40PM – TECHNICAL SESSION
– CONFERENCE ROOM 2
CHAIR: CECIL PARKS, CO-CHAIR: TBC

ABSTRACT 272

Imparting Realism to the Criticality Evaluation of a BWR Fuel Assembly Package

PETER VESCOVI*, TANYA SLOMA

The criticality evaluation is a demonstration of the most reactive configuration of the individual package in isolation, arrays of undamaged packages, and arrays of damaged packages. The most reactive configuration for the fuel assembly contents in a BWR package must take into consideration a number of parameters that include partial length fuel rods, neutron absorbing burnable absorber rods in the fuel bundle, rearrangement of the fuel bundle during accident transport conditions in the form of lattice expansion, and partial loadings of fuel rods. Packaging material composition and arrangement of packaging materials are also important to consider in the demonstration of maximum reactivity. Values must be assigned for these parameters that may not be known with a high degree of certainty, such as burnable absorber rod distribution, lattice expansion, and packaging material composition during a fire. Imparting realism to the criticality evaluation requires a thorough understanding of the effect that impact, fire and water immersion may have on the package configuration and material properties. Evaluating the sensitivity of neutron multiplication to intrinsic material property uncertainties can be accomplished by applying perturbation methods.

However, evaluation of sensitivity to other package configuration uncertainties is a more heuristic process that requires a detailed understanding of the fuel assembly design and package performance during accident transport conditions. There is no guarantee for a particular sequence of impacts or complete progression of a fire during a transport accident, yet intermediate conditions that result in the maximum neutron multiplication are often overlooked. A criticality evaluation of a BWR package has been done to demonstrate a realistic maximum neutron multiplication using values for parameters that takes into consideration credible intermediate transport conditions. Values for parameters used in the criticality evaluation are assigned in a manner consistent with constraints imposed by the fuel assembly design and performance of the contents and packaging materials during the sequence of mechanical, thermal and water immersion tests.

ABSTRACT 189

Perturbation Analysis for Demonstration of Reactivity in Criticality Safety Analyses

TANYA SLOMA*, PETER VESCOVI

A fissile package assessment per TS-R-1 guidance must be performed assuming that a contents specification provides the maximum neutron multiplication (keff) consistent with the fuel bundle design and transport conditions. In ensuring demonstration of most reactive and realistic contents specification for licensing criticality safety analyses, a variation of parameters is applied to evaluate the effects on reactivity. Perturbation theory is useful in studying the relative worth of a desired parameter, as it allows determination of the sensitivity of the eigenvalues with respect to changes in the system. This sensitivity analysis results in a simplified contents specification that minimizes any potential, unnecessary restrictions that transport package requirements would impose on the fuel bundle design.

The process of utilizing perturbation theory to determine the most reactive configuration defined by a set of realistic criteria is applicable to criticality safety contents and package evaluations. Through an optimization process, the parameter of effect is chosen, its function and impact investigated, and the relative worth of the parameter is evaluated by application of perturbation theory. Selection of the parameter value is determined by a set of realistic criteria specified by the application. These criteria add realism to the analysis, through basis on actual designs. By selecting the least worth or most reactive parameter value, the package contents can be specified in a manner that ensures maximum keff consistent with the transport condition.

This perturbation technique is applied to a BWR shipping package criticality safety assessment, which assumes the presence of integral burnable neutron absorber fuel (BA) rods. The effectiveness of the BA rods as a neutron absorber varies with the location of the BA rod within the fuel bundle lattice, and the sensitivity was quantified by the largest present absorber. Package criticality evaluations were performed using the TSUNAMI-3D module in SCALE6 code package, which automates the process of sensitivity and uncertainty analysis. Results justify selection of least worth reactivity locations for BA rods in the BWR lattice, while ensuring a demonstration of most reactive and realistic contents configuration for package evaluations.

ABSTRACT 351

Criticality Assessments Using Polyurethane Foam

JAMES LAM*

Rolls-Royce has designed a package to transport and store fresh fuel assemblies and anticipates approval from the regulators for the new package design in the near future. The space between the inner and outer steel shells is filled with shaped blocks of rigid polyurethane foam, of two different densities.

The criticality safety case for the fresh fuel package had to consider single packages and arrays of packages under routine, normal and accident conditions. IAEA regulatory requirements state that the criticality assessment must include investigations on the effect on the neutron multiplication factor (keff) due to impacts, flooding and fire. Sensitivity studies must also be carried out to determine the effects on the keff due to any uncertainties in the composition of the fuel and container materials. An important part of the criticality safety case is the treatment of the foam. The approach adopted to model the polyurethane foam is the subject of this paper.

The following were investigated:

- The effect on the keff of varying the elemental composition of the foam, including the removal of hydrogen.
- The experimental analysis of burnt foam.
- The effect of addition of water to the foam to simulate water absorption.
- A simple representation of crushed foam to simulate knock-back in the package.
- Extreme representations of burnt foam, such as replacing foam with solid carbon or as randomly distributed spheres of carbon to represent soot.

Abstracts – Tuesday 05 October 2010 : continued

These investigations were most informative and should be considered in any criticality assessments of transport packages containing large amounts of foam in the future.

ABSTRACT 275

Transport Criticality Assessment Methodologies the RWMD Spent Fuel Disposal Canister Transport Container

WILLIAM DARBY*

Over the past six decades, the UK has operated a number of reactors, including Magnox, AGR and PWR. Magnox and AGR spent fuel (SF) has been, and continues to be, sent to the Sellafield Site for reprocessing. However, other options, such as the transfer of SF into a central geological disposal facility are now being investigated.

The Nuclear Decommissioning Authority (NDA), Radioactive Waste Management Division (RWMD) is considering a number of options for the transport and disposal of SF. These options require the SF to be packed into dry and sealed disposal canisters, each provided with internal load bearing and supporting structures to locate the fuel elements. The canisters would be transported to a repository in a transport container. At the repository, the canisters would be unloaded and disposed of, with the transport container being re-used. An outline design of the transport container is currently being developed and is referred to as the "Disposal Canister Transport Container" (DCTC).

UK law requires the transport package (ie DCTC plus contents) to be designed to meet the requirements of the International Atomic Energy Agency (IAEA) Regulations for the safe transport of radioactive materials, TS-R-1. Under the IAEA Regulations there are several ways to demonstrate the nuclear criticality safety of a package.

The purpose of this paper is to report initial findings on the criticality safety issues that will arise in the assessment of the DCTC concept and which may significantly affect the final design of the transport package. The options that have been examined are:

- Restricting the number of fuel elements carried.
- Amending the package/disposal canister design to include neutron absorbing materials.
- Including multiple water barriers in the DCTC.
- Taking credit for fuel irradiation ("burn-up credit") in the criticality assessment.
- Hybrid approaches.

The paper will describe each of these in terms of nuclear criticality safety, and the arguments for and against each option, together with ancillary issues, are presented.

ABSTRACT 418

Fissile Exceptions – a General Scheme for Packages Based on CSI Control

NICHOLAS BARTON*, SAM DARBY, DENNIS MENNERDAHL, MICHELE NUTTALL

The IAEA regulations for the safe transport of radioactive materials allow exceptions from the regulations governing fissile materials for certain well-defined fissile materials.

Broadly speaking, exceptions can be made either for packages containing small quantities of fissile material or for specific materials which it is judged would be adequately criticality-safe under both the normal and credible accident conditions of transport. For example, TS-R-1 "excepts" packages containing 15g fissile material per package, with an overall consignment limit of 400 g.

In 2009, the IAEA accepted the need to revise the criteria for "fissile exception", firstly because of safety concerns about some of the criteria and secondly because it was becoming clear that the needs of the nuclear industry were changing (mainly in respect of wastes) and would be poorly served by the current arrangements for exception.

The task of defining a set of criteria for excepting fissile materials from the regulations is more difficult than might first appear. Any scheme must provide adequate (but not excessive) levels of criticality safety, be internally consistent and also consistent with the safety of licensed packages, whilst not unduly upsetting current shipping arrangements, unless truly necessary. The wider background and general approach to revising fissile exceptions will be described elsewhere in this conference (see paper by Cecil Parks).

The purpose of this paper is to outline a general scheme for excepting packages containing small quantities of fissile material. The scheme is based on the control of individual packages and consignment by CSI, and takes into account fissile material type (plutonium and uranium at selected enrichments), and other materials package type. The scheme was first discussed at a meeting of transport and criticality specialists in Chester UK, 2008, hosted by the UK DfT, and has been developed since. Criticality calculations have been undertaken to develop sub-critical package limits for a number of fissile and moderating materials and the paper will describe the modelling approach in detail.

T17 - IMPACT LIMITER MATERIALS/STRUCTURAL MATERIALS

11:00AM – 12:40PM – TECHNICAL SESSION – CONFERENCE ROOM 3

CHAIR: PETER PURCELL, CO-CHAIR: ROBERT VAUGHAN

ABSTRACT 48

A Comparison Between Mono-wall Body and Multi-wall Body Structures for a Large Scale Metal Cask

RYOJI ASANO*, YOSHIAKI MIYAJI, SHINTARO MIYAZAKI, AKIO NARA, HIROBUMI NUNOME

Regarding such as a large scale transportation cask for transporting spent fuels of a light water reactor, two kinds of body design are existing, one is a mono-wall structure using a large steel forging, and the other one is a multi-wall structure using the combination of steel plates, or the combination of steel plates and lead.

Historically, the multi-wall structure was used at first for the transportation casks body, and the mono-wall structure followed it.

Currently, it can be said that there is a tendency that the mono-wall structure is mainly used for European cask de-signs, and the multi-wall structure is used for the United States cask designs. Moreover, both types of cask design are used in Japan.

Assumed that the material of the mono-wall body is a low alloy large forged steel, and the materials of multi-wall body are a combination of stainless steel plates and lead, the methods of how to select and determine the cask body structure and materials is studied.

The merits and demerits of each factor of each body design are clarified from not only technical requirements view-points listed below but also from the fabrication and economical viewpoints.

(1) Structural strength design
(2) Heat removal design
(3) Shielding design

In addition, the study is considered the influence of the scale of demand for transportable storage casks to the deci-sion, which is several steps larger than that of transportation casks and is expected to rise more in near future. It is also considered the influences of both Nuclear Renaissance and the increase of a high pressure vessels demand in the chemical industries, which have been risen a few years ago.

The study is concluded that specifications of the contents (spent fuels) are one of key factors to determine the cask body structure.

ABSTRACT 101

Dynamic Fracture Toughness Tests of Dynamic Loaded Ductile Cast Iron

HANS-PETER WINKLER*, ROLAND HUGGENBERG, ANNETTE LUDWIG, GERHARD PUSCH, PETER TRUBITZ

For the use of cast iron with spherical graphite for the body of CASTOR® casks for the transportation and storage of radioactive materials a com-plete failure assessment including fracture mechanical analysis is neces-sary. Especially for accident conditions, where as a result of rupture load conditions rapid changes of stresses and deformations occur, characteris-tic dynamic fracture material values are necessary to calculate the stress conditions and to determine the margins against brittle fracture.

In the last years GNS realised a large test-program with more than 2500 specimens for the definition of dynamic fracture-toughness design values. Following the IAEA regulations these program includes -40°c tests, too.

The paper gives an overview about the test-program and presents the main results also under the aspect of the special behaviour of ductile cast iron under dynamic load conditions and the consequences for the test-method.

The main test-procedure to create fracture-toughness values by using small-scale specimens (10x10x55mm) from drilled bars from the cask wall is the determination of dynamic Crack-Resistance-Curves (R-curve) by using the so called Low Blow Method.

The experiences show, that for the experimental determination of fracture toughness values the existing rules deliver only first approaches, they are orientated on the rule for static tests. The investigation leads in the de-scription of a test method, that consider the special influence of the struc-ture of the behaviour of crack-propagation of ductile cast iron at low tem-peratures regarding the experimental determination of dynamic R-curves and the definition of physical and technical crack initiation values.

The test-program also includes the dynamic tests of large scale specimen (140x280x1350 mm), that had been tested form Competent German Au-thority BAM before. After this tests small-scale specimens had been ma-chined from large scale specimens and fracture toughness values were determined by using Low Blow Method. The paper will discuss the com-parison of the reached results from the both test methods.

Final the paper gives a prospect to further investigations.

ABSTRACT 93

Investigation of Availability of Rigid Polyurethane Foam as Shock Absorbing Material for Heavy Cask

JUN OKADA*, SATOSHI ASHIDA, AKIO OIWA, HIROAKI ARAI, MASAYUKI TANIGAWA

For the impact limiter of a transportation cask, wood has been mainly used as a shock absorbing material because it had enough capability with the limited volume. However, it has become difficult to procure wood in large quantities constantly with satisfying the characteristics specified in the design.

In this study rigid polyurethane foams (R-PUF) was chosen as an alternative shock absorbing material and avail-ability of the impact limiter using R-PUF was investigated by experiments.

Firstly, the shock absorbing performance of R-PUF with three different density types was investigated under differ-ent temperature conditions. Weight drop test using column shaped test specimens were performed to obtain the shock absorbing property. Weight drop test uses the heavy weight (about 300kg) and it freely drops on the speci-men from 3m in height.

Secondary, drop tests of a 1/3 scale cask model with R-PUF impact limiters were carried out. Availability of R-PUF as shock absorbing materials was evaluated by the comparison with the result of 1/3 scale drop test with impact lim-iter using wood which was reported at PATRAM2004.

Thirdly, the fire resistant performance of R-PUF was studied. In the test, cubic test specimens of R-PUF covered by Stainless Steel sheet were kept 800°C in the furnace for 30 minutes.

Results obtained by drop tests and fire resistant tests suggested that R-PUF worked well as a shock absorbing ma-terials, and would be useful for impact limiters of large-scale transportation casks.

ABSTRACT 152

Modeling of Polyurethane Foam Thermal Degradation within an Annular Region Subjected to Fire Conditions

MILES GREINER, JIE LI, SHIU-WING TAM, YUNG LIU, ALLEN SMITH*

Nuclear materials are placed in shielded, stainless-steel packagings for storage or transport. These drum-type packages often employ a layer of foam, honeycomb, wood, or cement that is sandwiched between thin metal shells to provide impact and thermal protection during hypothetical accidents, such as those prescribed in 10CFR 71.73. This work considers the modeling of thermal degradation of a polyurethane foam within an annular region during an 800oC fire. Measurement and analysis by Hobbs and Lemmon [Polymer Degradation and Stability, Vol. 84, pp. 183–197, 2004] indicates that at elevated temperatures, polyurethane foam exhibited

Abstracts – Tuesday 05 October 2010 : continued

a two-stage, endothermic degradation. The first stage produces a degraded solid and a combustible gas; the second stage reaction consumes the degraded solid and produces another combustible gas. As a result, during a prolonged fire, a gas-filled void develops adjacent to the outer metal shell and grows inward toward the inner shell and the containment vessel.

Because of radial symmetry in the drum geometry, a one-dimensional finite-difference model is constructed for the annular foam region. Heat flux is applied to the inner surface to model the decay heat of the contents in the containment vessel. Thermal radiation and convection boundary conditions with a specified environment temperature are applied to the outer surface. The material and reaction rate properties determined by Hobbs and Lemmon are applied to the foam. The annular region temperature and composition are determined as functions of radius and time, after the environment is changed from room temperature conditions to that of an 800°C fire. The rate of combustible gas generation is also determined. The effects of heat of reaction, the exterior void space, and the flow of product gases though the degraded porous foam on thermal protection are evaluated against the available data from burn tests of Model 9977 packagings.

ABSTRACT 87

Novel Reliable Hydrogen Risk Mitigation System for Transportation of Radioactive Materials

VALENTIN ROHR*, M.PARADIS , E.BILLOU,
J-M. MERIENNE , D. PINET

Major issue in the area of radiolysis of radioactive materials during transportation is reliability and safety of transportation casks, especially due to hydrogen build up in the containment vessel. In the past decades, several designing options were identified to ensure that the hydrogen concentration stays below the flammability limit during the whole duration of the transport. The most common option consists in over-dimensioning the volume of the containment vessel so that it can contain all hydrogen that may be released by radiolysis of the transported material, without reaching flammable concentrations. Another option consists in using catalytic hydrogen recombiners in the casks cavity. These recombine hydrogen with the oxygen contained in the gaseous mixture. However, the working duration of these catalysts is limited by the amount of oxygen in the containment vessel.

In order to overcome this problem, AREVA TN International together with SNPE and AREVA-ELTA have developed a hydrogen risk mitigation system consisting of hydrogen recombiners, oxygen generators and oxygen release control systems that can be used for long durations to ensure a hydrogen concentration below the flammability limit.

The present paper presents this mitigation system, in particular the oxygen generator and the oxygen release control systems. Details are given on the reliability and safety assessment of this system and its conformance with the IAEA-TSR-1 regulation.

P2 - REGULATIONS - A FUTURE PARADIGM

2:00PM – 3:40PM – PANEL SESSION – MAIN HALL
CHAIR: FRANK NITSCHE, CO-CHAIR: SYLVAIN FAILLE

ABSTRACT 311

Compliance Assessment for the Safe Transport of Radioactive Material. Russia Practice and Perspectives

VLADIMIR ERSHOV*, GENNADY NOVIKOV

IAEA TS-R-1 Regulations define requirements for RAM shipments, that are used in states without substantial deviations. Requirements are detailed and its interpretation do not cause contradictions and specific difficulties between various participants of RAM transport.

Regulations do not almost define bodies and conditions of controlling and confirmation of requirements fulfillments. Both Regulations and auxiliary document TS-G-1.5 mainly manages one body - Competent Authority. Whereas at practice various functions on control are carried out by number of bodies, in particular in Russia - at least 8 bodies. Control of fulfillment of some requirements are carried out by producers as well. Therefore if there are not clear distribution of functions, responsibilities and order of control procedures it is very difficult to avoid duplication and other difficulties. It is necessary to take into account that besides requirements of IAEA Regulations there are numerous requirements on security, route, emergency response, custom.

Federal law "About technical regulating" has installed in Russia conception of "compliance assessment" (CA) in technical areas and defined it like "direct or indirect determination of observance of requirements, claimed to object". Law does not restrict possible forms of CA, and installs requirements to some forms of CA such as mandatory and voluntary certification and others.

Purpose of law was to put in good order to various requirements for CA in various fields of technical activities, to exclude duplications of CA forms for object, to exclude excessive forms.

For realization of law analysis of current CA system for RM transport in Russia in comparing as well with recommendations of IAEA and experience of states was made. Results of analysis have to be implemented for modernization of CA system. Paper should be useful for specialists of states, connected with RM shipments to Russia and for international organizations developing requirements in RM transport field.

ABSTRACT 347

Lost and Found - Explanatory, Advisory and Fissile Materials

DENNIS MENNERDAHL*

This paper covers some of the history, current proposals and expected future of criticality safety in transport. The bases for the current requirements must be easily available to avoid future misunderstandings. The full paper will contain references that are necessary for understanding. Since the first 1961 edition of the IAEA Transport Regulations, there have been misunderstandings, often due

Abstracts – Tuesday 05 October 2010 : continued

to the combination of radiological and criticality safety. A constructive cooperation between criticality safety and other specialists between January 2008 and January 2010 appears likely to result in improvements concerning safety and usefulness of the Regulations. Other papers are expected to present the improvements in more detail.

Any radioactive material with trace quantities of fissile nuclides is now also a fissile material. This will change. Natural and depleted uranium that is not in the same package as other fissile material will remain excluded. Specific fissile materials in packages (not the materials themselves!) will be excepted from specific criticality safety control during transport. Each specific material must comply with a material design subject to multilateral approval or being included in the Regulations. Some people have a problem with the concept of material design even though it is used in current Regulations and keeps millions of other people busy.

The confinement system is another misunderstanding. A small problem was 'solved' in 1994 by a definition that is completely different to the intent. The evidence is now clear; the confinement system varies from nothing to five page descriptions. It is a safety problem since the confinement system is all that is needed to preserve criticality safety. The specialists agree that the confinement system has totally lost its original intent. It is becoming used as specified by the definition: to list items, features and controls that are relied on for criticality safety. The original intent was related to subcriticality of the containment system if it could be removed from the main packaging. The only IAEA member state that appears to have retained the original intent is also the only state that did not adopt the confinement system concept.

ABSTRACT 405

How Specific Should Be The Regulations?

PIERRE MALESYS*

The transport of radioactive materials has to comply with the requirements of the modal regulations for the transport of dangerous goods. These are based on the "Regulations for the Safe Transport of Radioactive Material" set forth by the International Atomic Energy Agency (IAEA).

When analysing these Regulations, it appears that the requirements can be roughly classified in three categories:

- (very) general requirements,
- "normal" requirements,
- (very) detailed requirements.

The second category does not induce any difficulty, by the definition of this level of requirements.

The first category leads to difficulties when implementing the Regulations. The industry may not know exactly what is expected by the Regulations and the regulators. Also, as these requirements are (too) general, they may be interpreted differently in various countries. Eventually, it can also be considered that they are not really requirements, but clauses establishing the background of the "true" requirements.

The third category may lead to safety issues. They can generally be considered as giving examples of what is requested in a "normal" requirement. But this may also be understood as clarifying a "normal" requirement: the detailed requirement is no longer understood as an example but as the true (and limiting) requirement.

The paper will provide examples of these three levels of requirements and the issues which are linked to these examples.

There is a will within the IAEA to improve the wording of the Regulations, to make the Regulations more easily understandable. This initiative should be strongly supported and the issues described above should be carefully taken into account when rephrasing the text.

In parallel, there is also a trend within the IAEA to define overarching requirements. The paper will explain our views on what could be overarching requirements, how they should interact with the other requirements (i.e. non overarching requirements), and what should be the limits with companion documents including explanatory and / or guidance material.

ABSTRACT 322

Risks and Regulations in the Transport of Nuclear Material by Sea

PHILIP ROCHE*

The focus of regulation of the transport of nuclear fuels has to date largely been on the technical and licensing requirements for shippers and carriers of such material. Since such regulations have come into force, it can be argued that the risk to nuclear shipments on the high seas has increased but the framework of legislation has not been updated to deal with this change.

The international requirements established under SOLAS and the IMDG Code lay down minimum criteria for safety and have been supplemented, since 2001, with the International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships (INF Code). A patchwork of national laws deals with terrorism and other risks.

Despite the design and technical requirements demanded by SOLAS and the INF Code, there remain many risks to ships engaged in the international transport of nuclear material by sea. The resurgence of the scourge of piracy on the high seas, the willingness of individuals and groups to resort to cross-border terrorism, the navigational risks presented by peaceful protestors and the absence of suitable security arrangements in the waters and ports of littoral States and particularly on the high seas are difficult issues for most ship owners and operators, but become acute when presented to those charged with the safety of carriage of nuclear material.

Outside the orbit of safety and security there are other issues in the domain of international law which must also be considered if the carriage and trade in nuclear material by sea is to be safely increased: the right of nuclear ships to innocent passage, the right of a safe haven or port of refuge in times of distress and how and when salvors are able to respond in the event of a maritime nuclear casualty.

In light of the foregoing, this paper asks whether the existing body of international law is sufficient to address the concerns faced by entities involved in the carriage of nuclear material or whether a new international convention addressing the transport by sea of nuclear material is required.

Abstracts – Tuesday 05 October 2010 : continued

ABSTRACT 126

Regulation of the Transport of Radioactive Materials

GEORGE SALLIT*

There are many ways to regulate the safe transport of radioactive materials. This paper reviews the range of approaches that can be used from a coercive regime to embedding regulators with operators. The pros and cons of the differing approaches are discussed. It is argued that one of the best arrangements is collaborative regulation which requires a considerable amount of trust and confidence from both operators and regulators.

ABSTRACT 28

Stability of Regulations Versus Confusion

FERNANDO ZAMORA*

Introduction - The stability of Regulations is essential to avoid confusion in their users: competent authorities and transport operators. In consequence, changes of Regulations should be sufficiently justified.

Modifications of Regulations are positive if they are really needed. Periodical review/revision cycles are necessary, since they give opportunities to introduce changes improving safety as soon as possible or adapt adequately the requirements to the reality.

Analysis - Changes in the IAEA TS-R-1 imply successive modifications in the UN Recommendations (Orange book), international Regulations like ADR (included their different language versions) and national Regulations. Finally, users of Regulations will have to modify procedures and carry out training to implement the changes. Then, a major o minor change in TS-R-1 and/or the Orange Book starts an â€˜in chain processâ€™ of modifications that spends many resources until the change is finally implemented; therefore, it is very important, essential, that the balance benefits/costs of any modification is as positive as possible.

However, sometimes revision cycles are used for introducing modifications not very well justified, where changes frequently minor only involve movements of paragraphs or simple variations of wording to reach a supposed improvement in the clarity of the text or the theoretical harmonization among regulations.

As an example, the 15th edition of the Orange Book introduced many changes affecting specifically to the radioactive material. In consequence, the ADR 2009 edition, which is binding in the European Union, introduced about 120 changes, where about 100 were exclusively changes of paragraphs or structural. In summary, approximately an 85 % of changes were not new requirements or substantial modifications. In addition, a lot of them involved revision of cross references.

Independently of this kind of processes is a source of mistakes in Regulations, it entails important disadvantages for their users, but confusion may be the main problem and it may lead the users to have low confidence in Regulations, insecurity in their application or/and a lack of compliance.

Conclusion - Instability of Regulations due to frequent unjustified changes may lead to confusion and to a lack of compliance and consequently to a problem of safety in the transport of radioactive material.

P1 - DENIAL AND DELAY OF SHIPMENTS

2:00PM – 3:40PM – PANEL SESSION – CONFERENCE ROOM 1

CHAIR: TBC, CO-CHAIR: CHRISTEL FASTEN

ABSTRACTS 9/10

Denials and Delays on Class 7

ANA SOBREIRA, NATHALIA ALBA BRAGA*

Although radioactive material is increasingly spread and necessary due to improvements on nuclear medicine, oil prospecting, power plants; issues on effective means of transportation still remain.

Concerning air delivery occurrences such denials of carrying class seven material by the aircraft pilot, delays on flights, air companies strikes, leaving packages behind on the plane’s basement, partially distributing volumes, deny the documentation… happen almost every week in Brazil. On 2008 there were 50 delays for many different reasons, and 3 denials.

Many of those processes were raised, focusing their enormous costs and implications.

The significant loss that could be avoided is increasing due to lack of training, few air companies that transport class seven and mixed up priorities.

ABSTRACT 204

Economic and Social Consequences of Denial and Delay of Shipments of Radioactive Material

MARIO MALLAUPOMA*, NATANAEL BRUNO, ANA SOBREIRA

The refusal of carriers in accepting radioactive shipments for transport produces are detrimental and causes series of consequences to the end users. One of these consequences refers to economic issues in the extent that the costs are severely increased. Additionally it produces lost of time and may impose a negative perception to the image of companies and individuals involved on this activity.

As part of the activities carried out by the Latin-American network on denials of shipment – the Montevideo Network - a methodology was developed to evaluate the economical impact of denials and delays on consignors, shippers and end users of radioisotopes. Four important relevant components are used to evaluate the economical impact: (i), labor cost; (ii) equipments, instrumentation and capital cost; (iii) material and services; and (iv) contingencies.

By encompassing relevant steps like loading, preparation of shipping papers, dispatch, carriage, stowage and in-transit storage and return of empty packages the methodology may be useful to consignors, shippers, consignees as well as suppliers of radioisotopes in evaluating the final cost for a single of multiple shipments and its consequences.

The proposed paper presents and describes the methodology and provides the outputs from a numbers of cases evaluated in Latin-America.

Abstracts – Tuesday 05 October 2010 : continued

ABSTRACT 160

Denial of Shipments

GEOFF LEACH*

The international transport of radioactive material by air is an essential element of healthcare across the globe. Bone scans, cancer treatments and sterilisation of medical equipment all depend on the expeditious transport by air of radioactive isotopes. The issue of the denial of shipments is a world-wide problem and comes in different forms e.g. from the individual pilot who is unhappy at seeing a package bearing a trefoil in the hold of his aircraft, and so offloads it, to the major airline which has decided, because of the costs associated with compliance, that it is simply not commercially viable to carry this type of goods. Both scenarios could fall under the heading of "Denial of shipment" but whilst the former may be relatively easily addressed by improved training, the latter is a far more difficult subject; whereas "Denial" is often thought of as a transport issue the costs of compliance with the transport requirements are only one part of the story because there are many other expenses relating to storage, worker safety, security etc which would still impact on commercial viability even if transport compliance costs were zero. Consequently, until the issue of denial is addressed holistically, involving all agencies with a vested interest, it is suggested problems will continue to be experienced for the foreseeable future.

Another common complaint is the failure of a consignment to pass an airline's acceptance check (which all consignments of dangerous goods undergo). Some such failures may be entirely justified, due to significant omissions or faults with a package. But there are also occasions when consignments are rejected for very minor reasons and whilst this is often blamed on the airline or their handling agent, the prime cause of the rejection is often because of the fear of punishment from an over zealous regulator.

I propose a presentation and panels to discuss the importance of: 1.engagement with all agencies either directly or indirectly involved in the air transport of radioactive material; and 2. the reasonable application by regulatory agencies of regulations.

ABSTRACT 383

The International Database of Problems Shipping Radioactive Material

JIM STEWART*

The IAEA and IMO have been recording reports of problems shipping radioactive material in a database. This paper outlines the results of an analysis of the records.

ABSTRACT 385

Regional Networks and their Effectiveness at Combating Problems Shipping Radioactive Material

JIM STEWART*

The IAEA has established National Focal Points in Regional Networks as an essential part of the efforts to combat problems shipping radioactive material. This paper sets out the way these networks work, and gives examples of how they have been and can be used.

ABSTRACT 404

Denial - Where are We Now

JIM STEWART*

There continue to be problems shipping radioactive material worldwide. This paper presents a global overview of the ability to ship radioactive material as well as providing an analysis of the global position today.

ABSTRACT 37

Difficulties in Transporting RAM - What it really means to Some People

STEVE WHITTINGHAM*

The effects of DoS and the inability to transport RAM by air due to the regulatory requirements.

If the subjects are combined then my proposal is split the panel session into two halves;

The session would be introduced by a short short film (5 minutes) featuring a village or town in an African country where the need for water sterilisation is not being met, where RAM sources cannot be repatriated and perhaps where medical procedures cannot be carried out because the material cannot be exported by sea due to the DoS issue which prevents a carrier taking the material due to restrictions imposed by other ports en-route. The film would be 'home movie' standard and not professional.

A discussion would then take place.

The purpose of the film would be to show how these issues are affecting real people and provide an insight into the effects that our deliberations in Vienna have on people and nations that often have no voice in our meetings.

For this we will need the support and help (by filming) from one or more NFP or RFP from Africa and for them to attend PATRAM to participate.

Statistics and probabilities are often used by some to defend the status quo and deter from changes to the Regulations. Perhaps statistics on mortality rates for children and adults due to contaminated water would make for some interesting comparisons as would the risks posed by not being able to remove sources from some MS.

P4 - QUANTIFICATION OF SAFETY IN TRANSPORT

2:00PM – 3:40PM – PANEL SESSION – CONFERENCE ROOM 2

CHAIR: MARC LEBRUN, CO-CHAIR: RUTH WEINER

ABSTRACT 325

Risk Based Model for Compliance Assurance Inspections for the Non-nuclear Sector

IAIN DAVIDSON*

In 2002 The UK was appraised by an IAEA team (TranSAS-3). Three recommendations were made. Recommendation 3 was '... the DfT

Abstracts – Tuesday 05 October 2010 : continued

should evaluate the adequacy of its inspection programme ... minor consignors and consignors of mobile sources should be more fully integrated into this programme. Priorities should continue to be risk based...'

Within the UK it's a legal requirement for organisations holding RAM to be licensed. The database of holders is approximately 3000. Potentially, all these organisations transport RAM.

DfT has 3 inspectors ensuring compliance in this sector. The team will audit approximately 120 organisations annually. Nominally, organisations are inspected every 3 years. We would inspect 360 organisations or 12% of our duty holders. It is essential that prioritisation is established.

In 2008 DfT sent a questionnaire to its duty holders. Answers were used to create a risk model that used the following parameters: number of RAM items held; amount of activity (relative to package type); number of movements made annually; if the organisations QA system is registered; if the QA system covers RAM transport; and if the organisation's aware of security requirements.

For each parameter a 'normal amount' was established and a 'normal' risk total proposed. A value above that total indicated an organisation worth visiting. It was recognised that some organisations below the cut-off should also be inspected to confirm the thresholds appropriateness.

A spreadsheet recorded information from the questionnaire and risks were calculated. Nil returns were given a default value that exceeded the 'safe' threshold. Inspectors then targeted the higher risk organisations

The inspection's output was a list of non-compliances. To validate the risk model we assigned an actual 'risk' to each inspected organisation. A numerical value was agreed at a team meeting and assigned to each category of non-compliance. We now had a hypothetical and objective risk value for each organisation. The risk model was then amended accordingly and continues to be used to guide the inspection programme and also to inform where education and dissemination of information is required.

ABSTRACT 113

Guide for Risk Assessment Studies Required for Transport Infrastructures

FRANCK KALOUSTIAN*, LAURENCE GOZALO, MARIE-THERESE LIZOT, GILLES SERT, CHRISTOPHE GETREY

The IAEA safety requirements are implemented in France for the transport of radioactive material transport. For use and storage of radioactive material, the applicable rules depend on the installation category: basic nuclear installations (INB), classified installations for protection of environment (ICPE), hospitals, etc. Transport infrastructures like harbours, marshalling yards, and truck parking areas are submitted to IAEA requirements but had no specific regulation relative to accumulation of dangerous goods and all the more of radioactive material. Recently, the national regulatory infrastructure has been completed with a requirement to provide for each installation a risk assessment dealing with health impact on populations in case of accident (French law of 30 July 2003 completed by the decree of the 3 May 2007 concerning the transport infrastructures).

This law, relative to the prevention of technological and natural risks and also to damage reparation, requires that the transport infrastructure operator develops a risk assessment of accident scenarios with estimation of probabilities, seriousness, kinetics and health consequences. Accident severity may exceed the regulatory accident conditions of transport. "Domino" effects are to be considered. The result will be appreciated in terms of seriousness and probabilities using a criticality matrix with acceptance criteria that will be fixed by authorities. Means to reduce the risk in compliance with these criteria are operational measures or procedures able to reduce either probabilities or consequences.

Transport infrastructure operators have to perform their risk assessments by May 2010. A guide will make easier and harmonized the expected studies. The Nuclear Safety Authority (ASN) and his technical support (IRSN) have been charged to produce the parts related to radioactive material.

First, it was decided to consider a consequence level from which risk should be characterized, valued at 50 mSv, considering the Q-system reference individual effective dose and the intervention level of 50 mSv recommended for public evacuation in the French national transport emergency plans

ASN and IRSN are considering 10 groups of packages; for each of them, severe but realistic scenarios will be provided with values of consequential doses.

ABSTRACT 330

A Multi-facet Approach for Evaluating Criticality Risks during Transportation of Commercial Spent Nuclear Fuel

ALBERT MACHIELS*, JOHN KESSLER

The U.S. industry's limited efforts at licensing transportation packages characterized as "high-capacity," or containing "high-burnup" (>45 GWd/MTU) CSNF, or both, have not been successful from the perspective of the fraction of existing spent fuel inventories that can actually be transported. A holistic framework is proposed for developing generic acceptance criteria for transporting CSNF, which considers transportation risks, spent fuel and cask-design features, and defense-in-depth in the context of present regulations as well as in the context of potentially future revisions of regulations that would reflect a risk-informed, technically state-of-the-art approach.

First, a probabilistic risk assessment (PRA) quantifies the frequency of criticality accidents during railroad transport of spent nuclear fuel in the U.S. The PRA shows that existing fuel burnup records and formal procedures for loading a 32-PWR transportation package make the likelihood of shipping a misloaded package on the order of 2.6x10-6 per shipment. When combined with historical evidence regarding train accidents and an estimate of the likelihood that an accident could breach and submerge a package, the calculated frequency of accident with potential for criticality is below 2x10-16.

The risk assessment assumes that a cask having any misloaded fuel would be subject to a potential criticality event. Work sponsored by EPRI and the U.S. NRC show that misloading of several fresh (unburned) fuel assemblies would be required for criticality to occur under any realistic fuel loading configurations. Given the distinct difference in the appearance of fresh and once-burned fuel assemblies, the estimate of the human error rate for transferring incorrect assemblies into the cask could be lower by orders of magnitude.

In addition, when fuel reconfiguration is assumed, "worst-case"

Abstracts – Tuesday 05 October 2010 : continued

scenarios show that the maximum reasonable reactivity increase is less than the administrative margin of 0.05 for scenarios involving physical changes to fuel assembly rod arrays. Fuel reconfiguration is actually more likely to result in substantial reactivity decreases.

Misallocation of regulatory requirements may lead to greater risks by overly restricting payloads. Additional CNSF shipments are required, and this results in larger non-radiological risks, the latter dominating risk assessments.

ABSTRACT 302

Development of a Web-based Routing Tool for Road Transport of Hazardous Materials

THOMAS MCSWEENEY, JAMES SIMMONS, AUTHUR GREENBERG, WILLIAM QUADE*

In 2007, the Federal Motor Carrier Safety Administration of the US Department of Transportation funded a project to develop a Web Based Hazmat Routing Tool to identify and select alternative routes for the transport of hazardous materials by truck. The current U. S. regulations provide criteria for selecting preferred alternative routes based on safety considerations and give State and Indian Tribes the authority to designate such prescribed routes. The goal of the project was to show the feasibility of developing a WebBased Routing Tool that routing authorities or other organizations, including shippers, could use to compare routes and select a preferred route based on safety and security considerations.

The tool was placed on a GIS platform that contains the routes, the population density along the routes and for the security assessment, the location of any iconic structures (national monuments or sports venues) and critical infrastructure (bridges and tunnels). The tool first evaluated the routes for safety by determining the length, accident rate and population density along the route. Security was evaluated by considering the distance from the routes to the iconic structures in comparison a weighted distance from police stations. Greater weights were given to structures with greater attractiveness. The paper will describe the methodology in greater detail and show how it was successfully used to compare and select prescribed hazmat routes based on both safety and security considerations. The tool was used to provide data used in a report to Congress on hazardous material routing.

ABSTRACT 419

Findings from Non-nuclear Small User Inspections in 2009 / 2010

DAVID ROWE*

Towards the end of 2009, the Dangerous Goods Division of the DfT began a programme of inspections at the premises of smaller organisations whose business involves the transport of radioactive material. These organisations included industrial radiographers, hospitals, road construction services and couriers.

The inspections were based on the requirements of the Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2009. The majority of the requirements of these regulations are referenced from the European ADR 2009 Agreement, which, in turn, is based on the requirements of the IAEA TS-R-1 with regard to radioactive material.

This paper presents a summary of the findings from approximately 100 inspections carried out over the past twelve months. Across the inspected organisations, there were a number of common non-compliances against the above regulations.

These non-compliances included:

- Emergency Arrangements not prepared or tested,
- Inadequate Instructions In Writing for drivers,
- Transport Documents with incorrect / incomplete requirements,
- Lack of training and awareness in transport security,
- Radiation Protection Programmes with inadequate structure and review,
- Package marking and labelling incomplete / incorrect,
- Package and Special Form Certificates out-of-date / missing,
- Insufficient Fire Extinguishers in the vehicle,
- References to out-of-date regulations,
- Driver training and certification incomplete,
- Instrument calibration out-of-date,
- Miscellaneous Equipment in the vehicle incomplete,
- No Dangerous Goods Safety Advisor appointed.

It is concluded that small organisations often do not have the capacity to implement complex regulations, and, where this is the case, there needs to be greater emphasis on appointing specialists to ensure that all transport-related activities are conducted in accordance with the applicable requirements.

P3 - CRUSH TESTING OF LIGHTWEIGHT PACKAGING

2:00PM – 3:40PM – PANEL SESSION – CONFERENCE ROOM 3
CHAIR: MATTHEW FELDMAN,
CO-CHAIR: MAKOTO HIROSE

ABSTRACT 332

Crush Testing at Oak Ridge National Laboratory

MATTHEW FELDMAN*

The dynamic crush test is required in the certification testing of some small Type B transportation packages. The IAEA regulations state that the test article must be "subjected to a dynamic crush test by positioning the specimen on the target so as to suffer maximum damage." Oak Ridge National Laboratory (ORNL) Transportation Technologies Group performs testing of Type B transportation packages, including the crush test, at the National Transportation Research Center in Knoxville, Tennessee, United States. This paper will document ORNL experiences performing crush tests on several different Type B packages. The paper will also discuss the interpretations of applicable testing requirements during these testing sequences and the perceived need for additional guidance from a tester's perspective.

This presentation at PATRAM 2010 will serve as the introduction to the subject of the dynamic crush test and will set the stage for subsequent presentations on the history of the development of the regulations requiring the crush test, current efforts to provide additional guidance on crush testing, and testers' experience performing the crush test. The presentation will be followed by a panel session on the crush test featuring the authors of the presented papers.

Abstracts – Tuesday 05 October 2010 : continued

ABSTRACT 336

What Constitutes a Valid Crush Test?

GORDON BJORKMAN*

In September 2008 a Consultant Services Meeting was held at IAEA Headquarters in Vienna. One objective of the meeting was to address proposed changes to the IAEA Regulations for Crush Testing and provide additional clarification as necessary. Paragraph 727.17 of TS-G-1.1 Rev 1 states: "For the crush test (drop III) the packaging should rest on the target in such a way that it is stable in the orientation selected to give the maximum damage. In order to achieve this it may be necessary to provide support, in which case the presence of the support should not influence the damage to the package." There was extensive discussion surrounding the intent and interpretation of this paragraph and what represents a valid crush test. To stimulate discussion, the author asked the question, "When is a crush test not a crush?" and introduced a sketch showing several flask orientations relative to the target and plate. The consultants considered this simple case of a cylindrical package supported in different orientations between horizontal and vertical. For each orientation and each impact position, they discussed whether it is a valid crush test as was originally intended and whether it would be a regulatory test or an extra-regulatory test. There was some agreement and some disagreement among the consultants, and the results of those discussions are summarized in a Figure presented in the paper.

It is the author's opinion that any description of what constitutes a valid crush test must be stated simply and unambiguously. To this end the author proposes the following.

A valid crush test is one such that,

1) At the instant of the plate's first contact with the package a direct load path (crush path) is established between the plate at the point of contact and the unyielding surface. (This load path must be capable of statically supporting 500 kg.), and
2) The drop location shall not produce a condition where the first action of the plate is to tear the package.

ABSTRACT 363

Historical View and Experiences with the Crush Test for Lightweight Packages

MARKO NEHRIG*, FRANK WILLE, THOMAS QUERCETTI, JORG-PETER MASSLOWSKI, BERNHARD DROSTE

The crush test for lightweight and low density type B packages was introduced for the first time into the 1985 edition of the IAEA-regulations.

In the early 1970s the need for an additional mechanical test besides or instead the well known 9 m drop test was deliberated. Various authors and test facilities, including BAM, were able to prove that the level of safety provided by IAEA drop and puncture tests in the regulations did not protect against dynamic crush forces to smaller packages. Already during the 3rd PATRAM symposium held in 1971 (Richland/USA), Robert F. Barker asked for "…a more strenuous crushing test for protecting small, lightweight packages…". BAM developed from research activities a proposal as to which types of packages should be subject to crush tests and how the crush tests should be performed, which was presented on the 5th PATRAM symposium held in 1978 (Las Vegas/USA).

At the IAEA, the possible need for a crush test was first mentioned in 1977. The subject for a discussion, besides the principal need for this test, was also the development of suitable set of crush test boundary conditions. It took more than 4 years of discussion until a dynamic crush test similar to today's test was recommended. Finally, after a rigorous evaluation process in which also the boundary conditions were determined, the crush test was proposed to be incorporated into the IAEA regulations.

BAM participated in the crush test development and implementation process right from the beginning in the early 1970s until its implementation in the IAEA regulations in 1985.

Today, BAM performs crush test procedures according to para. 727 (c) of TS-R-1, which have not been changed since their first implementation. Crush tests performed in 2002 at BAM will be discussed. These approval design tests where performed on birdcage pellet transport containers under normal and accident conditions according to the IAEA- Regulations.

ABSTRACT 46

Historical Background - Early Deliberations on and Assessments of the Need for a Dynamic Crush Test

RONALD POPE*, FRANK WILLE

Beginning in the late 1970s, arguments on the need for an additional test for some Type B packages were brought forth by US, German and other delegates participating in International Atomic Energy Agency (IAEA) Package Test Standards technical meetings. These included (a) early analyses by J.D. McClure of Sandia National Laboratories/Albuquerque (SNLA), and (b) testing of various packages to different crushing environments by various groups including (i) BAM/West Germany, (ii) Amersham International, United Kingdom, (iii) the US Atomic Energy Commission and the US Department of the Army , USA, and (iv) SNLA, USA. Consideration at the international level of these early deliberations and tests ultimately led to the inclusion in the IAEA Regulations for the Safe Transport of Radioactive Material of the dynamic crush test – the dropping of a 500 kg mass from 9 m onto a specimen positioned on an unyielding target so as to suffer maximum damage (para 727(c) of TS-R-1). The test is currently required to be performed on package designs having a low mass (less than 500 kg) and a low density (less than 1,000 kg/m3) other than special form contents in excess of 1,000 A2 para. 657(b) of TS-R-1). Recently discussions have been occurring as to what constitutes "positioning on an unyielding target", with a view to potentially changing the test requirement. Some of these changes could make the test more demanding than originally envisioned.

This oral paper will provide an overview of some of the very early thinking behind the crush test, including a review of the results of various tests performed in the US, UK and Germany from the mid-1960s through the early 1980s. This oral presentation will serve as part of an envisioned session at PATRAM 2010, leading to a panel discussion on this issue, and will serve as a complement two second paper whose lead author is to be F. Wille, BAM, Germany. Both papers will include the viewing of related videos. The concept of an oral session followed by a panel discussion at PATRAM 2010 is being organized by M. Feldman, Oak Ridge National Laboratory, USA.

Abstracts – Tuesday 05 October 2010 : continued

ABSTRACT 21

Crush Testing of 9977 General Purpose Fissile Packagings

ALLEN SMITH*

The 9977 General Purpose Fissile Package (GPFP) was designed in response to the adoption of the crush test requirement in the US regulations for packages for radioactive materials (10 CFR 71). This presentation on crush testing of the 9977 GPFP Reviews origins of Crush Test Requirements and implementation of crush test requirements in 10 CFR 71. SANDIA testing performed to support the rule making is reviewed. The differences in practice on the part of the US Department of Energy from those required by the NRC for commercial purposes is explained. The design features incorporated into the 9977 GPFP to enable it to withstand the crush test are described. Examples of crush tests performed on the 9977 are described and video included. Lessons learned from crush testing of GPFP packagings are given.

T20 - LONG TERM STORAGE STRATEGIES

4:00PM – 6:00PM – TECHNICAL SESSION – MAIN HALL
CHAIR: YVES CHANZY, CO-CHAIR: TOSHIARI SAEGUSA

ABSTRACT 61

Ageing Management for Long Term Interim Storage Casks

ANTON ERHARD*, HOLGER VOELZKE

The ageing management system for mechanical components, of nuclear power plants (NPP), must be established and used by the licensee in such a way, which is in accordance with this system the quality of safety relevant components is guaranteed for the whole designed lifetime of the NPP. This demands an extensive plant life management with special emphases at the knowledge of the degradation in material properties. The Basis Safety Concept (BSC) in Germany observed this circumstance. Lifetime extension of the German nuclear power plants is an aim of the current valid coalition agreement of the German government. Operational extension of interim storage facilities requires, in comparison to the ageing management system for NPP, an ageing management system adapted on the special circumstances of storage casks.

Extension of interim storage periods for spent fuel casks above the designed life time requires, in comparison to the components of NPP, an increasing knowledge of material degradation as well as the knowledge of integrity of the casks e.g. leak-tightness. Interim storage in Germany has been approved for 40 years. After that time, so the present strategy, a final repository should be available. But until now, such a final facility still not exists and the German exploration and licensing process is heavily delayed. Currently, discussions are continuing regarding further exploration of the Gorleben salt-mine. The willing to overcome this situation is clearly described in the available coalition agreement of the federal government. Anyway, a repository for heat generating radioactive waste in Germany will not be available until 2030. Therefore, what has to be done with the existing storage casks in the interim facilities? May these casks be fit for purpose, with an extension of the storage period? One option is, to have an ageing management system, which creates enough information about the technical condition of safety relevant cask properties as a basis for safety evaluation for extended storage periods.

In the present contribution the relevant ageing mechanisms for HLW storage casks will be discussed as well as the influence of the time dependent changes of the component properties.

ABSTRACT 260

The BSK3 Concept for Direct Disposal of Spent Fuel in Salt Using Borehole Emplacement Technology

STEFAN FOPP*, REINHOLD GRAF, WOLFGANG FILBERT, ROLAND HUEGGENBERG

The direct disposal of spent fuel - as a part of the current German reference concept, the so-called POLLUX® concept - was developed as an alternative to spent fuel reprocessing and vitrified HLW disposal. As an additional concept besides the POLLUX® concept, the BSK 3 concept was developed for the direct disposal of spent fuel in rock salt using borehole emplacement technology. It is based on the conditioning of fuel assemblies and inserting fuel rods into a steel canister which can be placed in vertical boreholes. The BSK 3-canister is designed to contain the spent fuel rods of three PWR-fuel assemblies with a maximum heat load of 6 kW. The BSK 3 concept simplifies the operation of the repository because the handling procedures and techniques can also be used for the disposal of reprocessing residues.

The emplacement system developed for the handling and disposal of BSK 3-canisters comprises an emplacement device, a borehole lock, a transport cart and a transfer cask which will shuttle between the aboveground conditioning facility and the underground repository. The transfer cask and the borehole lock provide appropriate shielding during the emplacement process and were designed by GNS.

Supported by the EU and the German Federal Ministry of Economics and Technology (BMWi), DBE TECHNOLOGY built an aboveground full-scale test facility to simulate all relevant handling procedures for the BSK 3 disposal concept. GNS provided the main components and its know-how concerning cask design and manufacturing. The test program was concluded after more than 1.000 emplacement operations had been performed successfully.

The BSK 3 disposal concept is expected to simplify disposal processes and to reduce operational risk without any compromise in long-term radiological safety aspects.

ABSTRACT 232

Impact on the Transportation Package Design for Transport First and Then Interim Storage Versus Interim Storage First and Transport

PETER SHIH*, PRAKASH NARAYANAN

In the US because of the unavailability of the repository, all the fuel from the commercial nuclear power plants are stored on sites using dry storage systems. Majority of these systems are dual purpose systems licensed for both storage and transportation. However, in some countries, fuel assemblies are stored in the interim storage

Abstracts – Tuesday 05 October 2010 : continued

facilities. In this case, the assemblies need to be transported from the nuclear plants to the interim storage sites first and then put into storage.

This paper examines the important aspects of the package design affected by the option selected. Some of the design considerations are the heat rejection and the radiation shielding capabilities of the package. In the Storage first option, the system is typically designed for the maximum possible heat load, therefore the system design is generally dominated by the heat rejection capability (limited by the acceptable maximum fuel cladding temperature). However, for the Transport first option, the heat load is limited for shielding considerations and thermal design is usually controlled by the maximum seal temperatures.

In addition, the crane capacities of the power plants also have a major effect on the system designs depending on the use of these cranes to handle transportation overpack with canister or onsite transfer cask with canister. Normally, the onsite transfer cask is lighter than the transportation overpack, therefore for the same crane capacity; the Storage first system can accommodate heavier weight (higher capacities) than the Transport first system.

This paper examines the important design aspects of the two options which include thermal, structural, shielding, and criticality. In addition, design loadings for each system and design interface requirements are also addressed.

ABSTRACT 163

Confirmation of Maintenance of Function for Transport After Long-term Storage Using Dry metal Dual Purpose Casks

TADAYOSHI TAKAHASHI*, MITSUO MATSUMOTO, TAKESHI FUJIMOTO

In Japan, currently preparations are underway for the interim storage of spent fuel away from reactor site using a dry metal dual purpose cask (transport and storage).

It is planned to confirm cask safety for post storage transport without opening the lid of cask and confirming the condition of cask contents (fuel).

In terms of safety for post storage transport, NSC(Nuclear Safety Commission of Japan) requests operators to accumulate data of spent fuel integrity in dry storage. Also, NSC requests the competent authority to establish reasonable rules of the inspections for post storage transport considering accumulated data of spent fuel and characteristics of the interim spent fuel storage facility.

As for operators' approach, Japanese operators are working to getting technical findings by the investigation of spent fuel integrity in dry storage at the nuclear power plants (NPP)[1]). Meanwhile, reasonable rules of the inspections are studied by NISA(Nuclear and Industrial Safety Agency).

In this paper, as the activities of NPP operators to ensure safety in the planed operations for the interim storage of spent fuel, the method of confirming (inspection ,etc) that dry metal dual purpose cask has the basic safety function and the structure strength is shown at each stage of transport from the NPP, the period of storage and transport from storage facility.

In the activities mentioned, It is important to accumulate data before storage (record of inspection at NPP) and data during storage (record of pressure between the lid ,surface temperature, appearance, etc.).

1) T. Fujimoto, M. Yamamoto, M. Matsumoto, K. Shigemune, H. Matsuo, "Investigation of Spent Fuel Integrity in Dry Storage at Japanese Nuclear Power Plants", PATRAM2010, Oct. 3-8 2010, London, UK.

ABSTRACT 226

Considerations for Transportation Licensing of Used Fuel Already in Interim Dry Storage

JAYANT BONDRE*, ROBERT GRUBB

Currently in the United States of America, used fuel assemblies from commercial nuclear power plants are in interim dry storage at various sites. The typical interim dry storage systems used for these assemblies are either storage casks or canisters stored in storage overpacks. Some of these systems were licensed and have had fuel assemblies in dry storage for a period of more than 20 years. The majority of the systems used for storage are also designed to comply with transportation regulations. However, some of the earlier vintage interim dry storage systems were not designed to be compliant with the requirements of transportation. It is desirable from a safety and economics point of view that these storage systems also be qualified for transportation. This qualification for transportation would eliminate the need to remove and repackage used fuel from these storage containers prior to eventual transportation.

The transportation regulations have several requirements that are different than those for storage. Additionally, transportation regulations have evolved over the time period that the used fuel has been in storage. Design analysis methods and computer codes have undergone significant changes over time. The current accepted practices and regulatory expectations have also evolved and are different than they were when these systems were designed and licensed. Therefore, if a user of these interim dry storage systems desires to have them qualified to meet current transportation regulations, evaluations are required to demonstrate that these systems are compliant with current transportation requirements and regulations.

This paper examines some of the challenges that a user might encounter during these evaluations. The differences in design and analysis methods including computer codes are discussed. Fabrication, testing and inspections requirements during fabrication, loading, operation and maintenance are examined to evaluate the suitability of these interim dry storage systems for transportation. The impacts on the already designed, fabricated and storage licensed containers due to changes in the current practices and regulatory expectations are presented.

ABSTRACT 357

Advanced Solution for Used Fuel Management

FREDERIC PATALAGOITY*, CAMILLE OTTON

When choosing the interim storage solution two ways of dealing with used fuel elements are available:

- wet interim storage inside pools
- dry interim storage inside casks

These solutions have a lifetime of 50 to 100 years. After these 50 or 100 years the used fuel elements will certainly have to be transported to a final storage site, to a recycling plant or to another interim storage site. To perform these transports means dealing with new laws, new regulations, new safety authorities' requirements and a new public acceptance environment, but also having access to new technologies.

Thus, the solutions needed today by the utilities to manage their used fuel have to be innovative and flexible.

This article will give elements of appreciation of the various solutions of management of used fuels proposed to the utilities towards the future problem of the movement of these materials.

T18 - WASTE MANAGEMENT

4:00PM – 6:00PM – TECHNICAL SESSION - CONFERENCE ROOM 1
CHAIR: DANNY VINCE, CO-CHAIR: MICHAEL CONROY

ABSTRACT 422

Soft Sided Packaging for Low Level and Hazardous Wastes

MIKE SANCHEZ*, STUART BOWE, PAUL MISKIMIN

This paper will describe the design and use of soft-sided packaging systems. Soft-sided packaging has been in use for storing, transporting, and disposing of radioactive wastes and hazardous wastes for more than 20 years. Soft-sided waste packages are widely used throughout the United States, with more than 12,000 units deployed. The safety record for these products is exceptional. The main soft-sided packages which will be described in this presentation comply with IAEA international packaging standards IP-1 and IP-2 and have been independently tested to confirm their compliance.

Soft-sided waste packages are now available in the UK and are attracting a lot of interest across the waste management community. This appears to be in recognition of a need for lower cost, more efficient, adaptable and environmentally friendly packaging, considering the volumes of wastes to be generated throughout the UK, primarily due to the Nuclear Decommissioning Authority's decommissioning activities. The volumes of waste to be generated under the NDA's programme alone are far beyond those experienced to date in the UK.

Standard practice in the UK has been to use half-height ISO containers for the storage, transport and disposal of LLW. While ISO containers work, they are very expensive by comparison to the equivalent soft-sided packaging, require grouting of their contents and sometimes the addition of "furniture" to manage the positioning of their contents within the container. ISO containers rarely achieve even a 60% loading factor, meaning that 40% of what is being disposed of is clean material.

Soft-sided packaging offers a useful alternative to ISO containers. It is thought that the initial uses of soft-sided packaging will be for the storage, transport and disposal of Very Low Level Waste (VLLW), and some LLW. Waste generators in the UK are already purchasing such products. The Low Level Waste Repository team, for example, is evaluating soft-sided packaging as part of their NDA mandate to look at innovative, efficient and lower cost packaging systems and products.

This paper will describe soft-sided packaging as a useful alternative to ISO containers, and compare the two packaging systems. It is likely that the two systems will be used side by side in the future, with the selection decision being made on the basis of which product best serves the clients and the waste streams.

ABSTRACT 224

Radioactive Waste Inventory Forecasting and Characterization Implications for Packaging and Transport

MARC FLYNN*

A large variety of process wastes arise in the nuclear fuel cycle industry, from mining, conversion, enrichment and fuel fabrication, reactor operations, reprocessing and more recently from decommissioning of a wide variety of nuclear facilities. These wastes vary greatly in their chemical, physical and radioactive properties and the degree of homogeneity is sometimes difficult to assess.

Traditionally, waste management has been mainly focused on the need to ensure safe storage of waste, either interim or long term, in the raw or conditioned state. The assessment of waste against the best practical environmental options for disposal; namely, saving valuable space in national repositories, is also important. However, it is important to note that all these waste streams will have to be transported eventually in some form or another and the IAEA Regulations for the Safe Transport of Radioactive Material, TS-R-1 [1], must be able to cater for these materials without imposing unjustified constraints which could result in significant operational difficulties and economic penalties. The World Nuclear Transport Institute (WNTI) has, therefore, formed an industry working group to share experiences amongst its members in the interest of focusing on the various issues affecting the future packaging and transport of radioactive wastes. This paper is concerned with one important issue - the forecasting of the inventory and the characterisation of low and intermediate level radioactive wastes which are essential precursors for packaging and transport operations for these materials.

The current transport regulatory position is discussed for characterising and classifying low and intermediate activity radioactive wastes for transport and the potential challenges the current regulations imply. Radioactive assay methods are also covered for characterising low activity and some intermediate activity radioactive wastes for transport and disposal. In addition, the implications of waste inventory forecasting and its importance on transport are considered.

ABSTRACT 57

Moving a Mountain by Rail!

ASHOK KAPOOR*, STEPHEN O'CONNOR, J. RITCHEY, W. RYAN, DONALD METZLER

This paper describes identification and resolution of regulatory considerations, construction challenges, development and implementation of the transportation plan and lessons learned from the movement of a mini mountain of uranium mill tailings in the United States that lies on the bank of the Colorado River in Moab, Utah.

In 1952, the discovery of significant uranium ore deposits on the Colorado Plateau and the subsequent nuclear arms race of the Cold

Abstracts – Tuesday 05 October 2010 : continued

War resulted in approximately 800 mines producing high-grade ore, largely used for building the nuclear arsenal for the United States of America. During the uranium mining heyday, the federal government and private industry built a number of ore processing sites on the Colorado Plateau. The waste from uranium mills consists of sandy material and contains the radioactive element radium. The mill tailings pose a potential hazard to public health and safety.

By the 1980's the uranium excitement was over and the mills were shut down. As a result, more than a dozen mill tailings piles remained at the processing sites. These piles range in volume from approximately 292,000 to 3.1 million cubic meters of mill tailings. The massive Moab pile is up to 28.5 meters high and contains 9.2 million cubic meters of material covering 162 hectares of land located only 230 meters from the Colorado River.

The Department of Energy has safely remediated all other piles and in 2005 decided on the final cleanup remedy for the Moab site. It was decided to relocate mill tailings and associated wastes to the Crescent Junction off-site waste disposal site, a distance of 48 kilometers using primarily rail transportation. The first trainload of uranium mill tailings was shipped from the Moab site to Crescent Junction site for disposal on April 20, 2009. The project is scheduled to be completed by 2025.

ABSTRACT 55

Identifying Opportunities for Process Improvements in Addressing Transportation Safety and Compliance Issues

JULIA DONKIN*, DANA WILLAFORD

In 1989, the U.S. Department of Energy (DOE) established the Office of Environmental Management to mitigate the risks and hazards posed by the legacy of nuclear weapons production and research. Many issues associated with these activities are unique, including the transportation of unprecedented amounts of contaminated waste, soil, and a vast number of contaminated structural debris. As the transportation of radioactive material increases, so does the potential of a transportation incident involving radioactive material. The challenge for DOE is ensure that thousands of its shipments annually are made in a safe and compliant manner.

The DOE recognizes the importance of radiological characterization for ensuring radioactive materials are shipped in compliance with national regulations promulgated by the U.S. Department of Transportation (DOT) in the Hazardous Materials Regulations. In December 2009, DOE became aware that certain radioactive materials were not being fully characterized for shipment. This resulted in a potential for shipments being made in noncompliance with DOT regulations. In response, DOE formed a working group to identify the source of the problem and develop solutions. This working group consists of waste characterization and transportation personnel involved with DOE radioactive material shipments. This group of subject matter experts is examining the multitude of requirements related to shipments, in particular: (1) waste acceptance criteria for the receiving site of the materials, and (2) transportation regulatory requirements. The group is reviewing existing documents, plans, procedures and processes used in preparing DOE-owned radioactive material for shipment, and are identifying areas for improvement. The group members seek to identify best practices, opportunities for improvement, and obstacles encountered by waste characterization and transportation personnel.

The working group findings and recommendations are being used by DOE to develop the necessary steps to ensure all radioactive materials are properly characterized and shipped in compliance with applicable requirements and regulations.

ABSTRACT 54

Radioactive Waste and Fissile Exceptions

BRUNO DESNOYERS*

The paper will present the use of exception for fissile material in the field of radioactive waste shipments. It will give an historical view of the changes introduced in the regulations for fissile exception, and will give an overview of the safety and operational issues generated by the current exceptions.

Examples from the industry will be developed to illustrate:

- the need for shipment of radioactive wastes containing fissile material;
- the quantities of wastes concerned by fissile material;
- the importance the criteria for definition of fissile-excepted material, and their durability in time, may have on the design and global cost of a complete process for radioactive waste conditioning, shipment and storage.

The work done, with the full implication of WNTI, to review completely the exception criteria and requirements for fissile-excepted material to increase together safety and fitness for the needs of industry, will be described.

The new criteria and requirements for fissile exceptions as proposed to be introduced in forthcoming new edition IAEA regulation for the transport of radioactive material will be presented, and illustrated with examples for wastes shipments.

Gain in safety margins as well as in transport capacities produced by these new exceptions for fissile material will be illustrated by some examples from the industry.

Difficulties which can come with the implementation of some of these new criteria will be listed and discussed.

ABSTRACT 266

Optimization of Alpha Contaminated Waste Transportation in France

JULIETTE VUONG*

Drums of waste contaminated with plutonium and uranium oxides generated by MELOX plant are transported to AREVA NC La Hague plant by TN International in TN™ GEMINI and RD26 packages. Up to 60 118-liter drums can be loaded in the TN GEMINI packaging, a 20-foot ISO container like packaging. One 118-liter drum can be loaded in the RD26 packaging. Up to 12 RD26 packages are shipped together inside a 20-foot ISO container. The TN GEMINI packaging was initially designed for low contaminated drums and the RD26 packaging for higher contaminated drums. This paper describes the main characteristics of these wastes and of these two packages.

So as to optimize the waste transportation while MELOX increases its MOX fuel production, TN International has implemented a strategy of sustainable development which is detailed in this paper. This is based on the one hand on the modular design of the TN GEMINI packaging:

Abstracts – Tuesday 05 October 2010 : continued

- the allowable amount of fissile material by drum and by package has been increased (criticality issue), hence a reduced number of drums and packages,
- the allowable heat power of waste by drum and by package has been increased (radiolysis issue), hence a reduced number of drums and packages,
- the RD26 payload has been included in the TN GEMINI package, hence an increased number of drums by shipment from 12 to 20, while warranting the safety of the package.

On the other hand, the limitation of fissile materials by shipment due to security issue has also been discussed with the French authorities, in case of waste contaminated with plutonium and uranium oxides.

Consequently, the decrease of the number of transports and of the number of packaging needed leads to:

- economic development with cost optimization for MELOX as consignor, AREVA NC La Hague as consignee and TN International as carrier,
- an environmentally friendly logistics plan, with a reduced greenhouse gas generation and less waste due to maintenance operations,
- social development with reduced radiation exposure for workers.

T37 - IP DRUM PACKAGES

4:00PM – 6:00PM – TECHNICAL SESSION
– CONFERENCE ROOM 2
CHAIR: MIKE WANGLER, CO-CHAIR: NATANAEL BRUNO

ABSTRACT 45

Status of US. Department of Energy Replacements for the DOT Specification 6m Shipping Containers

JEFFREY G. ARBITAL*, DREW WINDER, KENNETH E. SANDERS

US Department of Transportation (DOT) Specification 6M containers had been the workhorses for the U. S. Department of Energy (DOE) for over 20 years (specification source: U. S. Code of Federal Regulations, 49 CFR 178.354; 2003). The two most popular sizes of this specification container were the 55-gallon and 110-gallon models. As of September 30, 2008, all 6M specification containers were phased out by DOT because they did not conform to the latest transportation safety requirements in the U. S. Code of Federal Regulations, 10 CFR 71. The anticipation of this action prompted DOE to develop the ES-3100 and ES-4100 shipping containers as replacements. The ES-3100 development is complete and has been fully implemented to replace the 55-gallon 6M model. The ES-4100 development project began in late 2006 and is expected to be operational in FY 2011 as a replacement for the 110-gallon 6M model.

The ES-3100 was first licensed in April 2006 by the U.S. Nuclear Regulatory Commission (NRC). Since then, the license has been revised 9 times to add new material authorizations. The ES-3100 was operationally ready for use at several sites by September 2007, and is now used on a regular basis for materials that had previously been shipped in the DOT 6M 55-gallon model. In addition, the ES-3100 has been certified for air transport in support of foreign research reactor fuel supply and international nonproliferation efforts. This container recently obtained a license from the DOE Shipping Container Certifying Official (DOE Environmental Management Program), as well as a Competent Authority Certificate from the U. S. DOT. The ES-3100 license allows many forms of fissile material to be shipped internationally, and continues to be amended to authorize additional material types for a variety of users. The ES-4100 project has successfully completed all regulatory testing and the license application is planned to be submitted in the fall of 2010.

This paper discusses the detailed status of both of these Type B shipping containers. It gives the latest operational information for current users, and for those organizations who wish to become users in both the U. S. and worldwide.

ABSTRACT 27

The Transport of Uranium Swarf Immersed in Oil

DAVID WINDLEY*

Uranium swarf is considered to be a pyrophoric material liable to spontaneous combustion and is therefore immersed in oil when machined, stored or transported. It is typically packaged in a combination package comprising a plastic drum within a steel outer drum. The oil used is generally a water based emulsion and the material is stored in a non-corroding plastic drum. To avoid further handling of the material the same container is also used for transport.

As a radioactive material, transport is subject to the IAEA Regulations for the Safe Transport of Material. For a package carrying natural or depleted uranium for which the A2 value is unlimited the IAEA Regulations require transport within an IP-1 Package which is not subject to testing. Consideration must however be given to any other dangerous properties of the material which requires compliance with the packaging requirements of the relevant dangerous goods transport regulations which would normally require the use of a metal package for pyrophoric substances.

This paper will present a justification for the transport of uranium swarf in a plastic drum when the pyrophoricity is suppressed by immersion in oil. It will review the status of the payload by reference to the UN Recommendations on the Transport of Dangerous Goods (UNRTDG) to determine the classification of the material. The paper will review the requirements of the UNRTDG and the modal transport regulations and demonstrate that the proposed package complies with these requirements taking account of the other dangerous properties of the payload.

A further problem is that hydrogen may be produced as a result of radiolysis and corrosion. Since retention of hydrogen would raise further issues due to pressurisation of the package which would then be carrying a flammable gas, the hydrogen is vented. At the same time the oil must be retained to prevent spontaneous combustion of the uranium swarf. The paper will discuss these issues and the features adopted demonstrating compliance with the various regulations.

The paper will conclude by proposing a suitable package design that satisfies the various requirements and summarising the issues raised.

Abstracts – Tuesday 05 October 2010 : continued

ABSTRACT 112

Packing for Radioactive Waste Transport

ALBERTO ORSINI*, RENATO SANTINELLI,
NADIA CHERUBINI, SANDRO RIZZO

ENEA has established at the Casaccia Research Centre, with the help of NUCLECO, a technical unit which takes care of all radioactive waste produced by hospitals, research centres and industry, and, waiting for the national disposal facility, remains owner for them in the temporary storage.

NUCLECO covers all the operative phases, including packaging, transport, volume reduction treatment, conditioning for nuclear power plant waste too.

At the end of the conditioning procedures there are, for the scope of this paper, two types of waste packages: industrial ones for LSA-II and LSA-III and packages for high activity sources.

Packages containing LSAII and III are produced by solid volume reduction, performed through 220 l drums supercompaction, stacking of the pellets inside a metallic 380 l overpack and then pouring of a concrete compound.

Same packages are generated by treatment of liquid wastes, performed by a biological and chemical-physical process, mixing the sludge with cement inside a 200 l drum and get rid of the clear water.

The paper gives details on packages qualification referring to "the waste form", with compression test, thermal cycling, radiation resistance test, biodegradation resistance test, immersion test, water permeability, free liquids test, and referring to the "transport" with long term corrosion test, free drop, stacking and penetration tests.

Packages containing high activity sources must be mainly of Type B(U) and ENEA is studying several types of shielded containment systems for gamma and neutron sources, to ensure their characteristics over a long period. The design of containment system, as regards the choice of materials for the construction and the thicknesses used, is aimed to comply with transport regulations, taking into account the characteristics of mechanical and thermal protection guaranteed by external part of the package, a model of which was approved by the Competent Authority as Type B (U) for different contents.

The paper shows how we can guarantee packages complying with the waste characteristics as approved by the competent authorities for the storage facility and with the transport regulations, at present and in the next 30 years.

ABSTRACT 74

UK Low Level Waste Repository - Transport Package Designs Adapting to the Waste Management Hierarchy

MARC FLYNN*

The Low Level Waste Repository has been the UK's primary low level waste disposal facility since the 1959 and has undergone many strategic changes over 50 years of operation. Recent UK waste forecasting reveals that the Repository is being filled at an unsustainable rate. Waste management assessments are concluding that through successful segregation and diverting to landfill or recycling facilities the volume of waste sent to the Repository could reduce, by approximately 30%, extending the life of the Repository for another 50 years.

Since the 1980's very/low activity waste has been transported to the UK Repository in specially designed single use disposable IP-2 ISO Freight Containers, for final disposal in the concrete vaults. Adapting to the waste management hierarchy, LLW Repository Ltd is in the mature stages of developing new re-usable package designs for the transport of low activity wastes to treatment facilities for recycling and for transporting very low activity wastes to authorised land fill sites.

This paper briefly describes and discusses the two new package designs and focuses on the safety and operational considerations of designing re-usable packages to load, restrain, transport and unload raw or pre-containerised low activity wastes.

Design Number IP-2/LLWR/TC02 is a re-usable half height ISO Container designed to transport metal for recycling. The design comprises of a stainless steel containment barrier affixed to the inside of the mild steel outer structure. Internal content restraint is achieved by the integral interlocking stillage system, which excludes the need for man access to the package cavity.

Design Number IP-1/LLWR/TC11 is a re-usable steel framed outer with four disposable soft sided inner liners to transport very low activity wastes to landfill. The outer frame design is based on the half height ISO Container footprint with a panel -less type body. The liners comprise of specially manufactured and tested fabric with mechanical closure system forming the containment barrier.

ABSTRACT 16

Development of a Specific Activity Distribution Estimation Method for Large Low-Level Radioactive Waste Using Shape Measurement Technique

MICHIYA SASAKI*, HARUYUKI OGINO,
TAKATOSHI HATTORI

When a large low-level radioactive waste is to be transported, it is desirable to pack it into a large container instead of a drum to reduce the cost of cutting it into small pieces, and also to protect workers from unnecessary radiation exposure. According to the IAEA Safety Standards of No. TS-R-1, this kind of waste is regarded as the Low Specific Activity (LSA) -II material, and it is possible to transport as the Industrial Package (IP) when there is no extreme non-uniformity in the distribution of activity. The simple criterion of uniform distribution is suggested in the IAEA Safety Guide of TS-G-1.1: as â€œdifferences between the specific activities of portions, which are defined as one-fifth or one-tenth of the volume, should be within a factor of less than 10â€ .

In the previous study, it has been clarified that a usual non-destructive radiation measurement method will be inap-plicable for the judgement of the transport requirement of a completed large waste container because the uncertainty is too large when the filling rate is above approximately 10 %. Thus, the authors have developed a new specific activity distribution estimation method for large low-level radioactive waste.

In this method, a specific activity estimation of a segment, which means a certain amount of radioactive waste placed in a large container, is repeated until a waste package is completed by the use of gamma-ray measurement, mass measurement, shape measurement by photogrammetry and MCNP Monte Carlo calculation techniques. The specific activities of the portions can be estimated as a summation of the specific activities of the segments. The degree of the standard uncertainty in the specific activity estimation of the segment was

Abstracts – Tuesday 05 October 2010 : continued

evaluated experimentally by using standard radioactive sources and mock-metal waste samples with varying the filling rate, distance between the detector and the size of the segment. The combined standard uncertainty of the specific activity of the portion was also estimated by a Monte Carlo simulation.

ABSTRACT 415

An Overview of the Development of IP-2 packages ISO Freight Containers in the UK

ROBERT VAUGHAN*, R P HOWS

Croft was the prime mover in establishing within the UK the use of IP-2 packages for Low Level Waste (LLW) based on the design features and established ISO standards for ISO Freight Containers. The first designs were required for the shipment of LLW to BNFL, Sellafield and Drigg and for the emplacement of waste in the Drigg LLW vaults. These designs were established by Croft and adopted by an industry wide group called "The Club" which co-ordinated the technical and commercial requirements of the principal producers of LLW in the UK. Croft managed The Club, including managing the manufacture, storage and distribution of the first 2,000 IP-2 ISO Freight Containers built in the UK. Croft carried out all the design work, developed design approaches (especially leak testable seals and lid closure systems), and established standards which have since been adopted in a DfT design guide. Croft was also involved in the development of the special provisions for freight Containers that have been included in the IAEA TS-R-1 Regulations for the Safe Transport of Radioactive Materials. Since the early designs, there have been a number of different sizes of IP-2 ISO Freight Containers needed and a number of special designs for large heavy items requiring special internal tie-down arrangements. Croft has designed most of the IP-2 ISO Freight Containers used and currently in use in the UK for the transport and storage of LLW.

The paper will detail the key requirements for the design of IP-2 ISO Freight Containers, summarise the designs established since initial use 20 years ago, summarise designs currently in use in the UK and give outline details of shielded IP-2 ISO Freight Containers proposed for the UK radioactive waste repository.

T19 - STRUCTURAL BENCHMARKING

4:00PM – 6:00PM – TECHNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: CHI-FUNG TSO, CO-CHAIR: KOJI SHIRAI

ABSTRACTS 26/115

From Experiment to an Appropriate Finite Element Model - Safety Assessment for Ductile Cast Iron Casks Demonstrated by Means of IAEA Puncture Drop Test

MIKE WEBER*, FRANK WILLE, VIKTOR BALLHEIMER, ANDRE MUSOLFF

In the approval procedure of transport packages for radioactive materials, the competent authority mechanical and thermal safety assessment is carried out in Germany by BAM Federal Institute for Materials Research and Testing.

The combination of experimental investigations and numerical calculations in conjunction with materials and components testing is the basis of the safety assessment concept of the BAM.

Among other mechanical test scenarios a 1-m drop test onto a steel bar has to be considered for hypothetical accident conditions of Type B packages according to IAEA regulations.

In the framework of approval procedure for the new design of German HLW cask a puncture drop test was performed with a half-scale model of the cask at -40°C.

For independent assessment and to control the safety analysis presented by applicant, BAM developed a complex finite element model for a dynamical ABAQUS/Explicit analysis. This paper describes in detail the use of the finite element method for modeling the puncture drop test within an actual assessment strategy.

At first investigations of the behaviour of the steel bar are carried out. Different friction coefficients and the material law of the bar are analysed by using a "rigid-body" approximation for the cask body.

In the next step a more detailed FE model with a more realistic material definition for the cask body is developed. Strain verification is possible by results of the strain gauges located at the relevant points of the cask model. The influence of the finite element meshing is described.

Finally, the verified FE half-scale model is expanded to full-scale dimension. Scaling effects are analysed. The model is used for safety assessment of the package to be approved.

ABSTRACT 231

Verification of Computational Models by Comparison of Finite-Element Calculations and Experiments for the Model Cask CASTOR®HAW/TB2

WALTER VOELZER*, STEPHAN GLUTSCH, RONNY PEREZ-KRETSCHMER, PAVEL VRASTIL

The mechanical analyses in the context of the transport licensing of the type B(U)F package CASTOR®HAW28M Cask for Transport and Storage of Radioactive Material is based upon drop tests which are accompanied by calculations using Finite Element (FE) analyses. The drop tests have been performed with the model cask CASTOR®HAW/TB2 which is a 1:2 scale model of the original cask CASTOR®HAW28M.

The purpose of the drop tests was to verify the computational models by comparing numerical and experimental results and to comprehensively study the loading situation of the package under accident conditions with respect to stresses, strains, deformations as well as the kinematical behavior.

Seventeen drop tests with the same specimen were performed from 2005 to 2006 at the drop test facility of the Bundesanstalt für Materialprüfung und -forschung (BAM) located in Horstwalde near Berlin. The drop tests consisted of single drops as well as sequences which were selected according to the IAEA regulations. Combinations of 0.3m drop, 9m drops and 1m pin drops were considered at ambient temperature and –40 °C. Following each drop test, the integrity of the specimen was studied and the leakage rate was measured.

The drop tests were preceded by extensive material studies, where flow curves were measured for both tension and compression at various temperatures and strain rates. The materials studied include

Abstracts – Tuesday 05 October 2010 : continued

ductile cask iron (cask body), steel (bolts, trunnions), aluminum (side impact limiters), and wood (top and bottom impact limiters). The results were used to determine the parameters for the FE material models. A number of preliminary calculations were used to define the locations of strain gauges and accelerometers on the specimen.

The FE simulation of the drop tests was done taking into account the temperature of the specimen as well as the exact boundary conditions (drop height, orientation). Using virtually no adjustable parameters, perfect agreement was observed between computational and experimental results with respect to deformations, signals of strain gauges and accelerometers, and deformations. The high quality of these numerical results was the basis for the subsequent mechanical analyses for the serial cask type CASTOR®HAW28M.

ABSTRACT 355

Validation of Numerical Simulation Method Using a 1/3-Scale Model Drop Test of KN-18 SNF Transport Cask

KAP-SUN KIM*, JONG-SOO KIM, KYU-SUP CHOI, IN-SU JEONG

The KN-18 SNF (spent nuclear fuel) transport cask is a newly developed cask intended for the dry or wet transportation of up to 18 PWR spent nuclear fuel assemblies in South Korea.

The structural performance of the KN-18 transport cask in normal and hypothetical accident conditions was demonstrated in the SAR by the analysis using state-of-the-art finite element methods, and that is the subject of a separate paper.

A series of actual drop tests using 1/3-scale model were carried out to verify the numerical simulation method used in the analyses and to confirm the impact characteristics of the cask. Total of five 9 m drop tests and two 1 m puncture tests were performed using one 1/3-scale cask model with four sets of impact limiters. Basically 1/3-scale model has same design with real cask except some inevitable differences in geometry between the scale model and the real cask for the reason of fabrication, operation, and etc.

In order to provide a robust basis for verification, finite element analyses of the scale model cask in all the drop test conditions were carried out. The same numerical tool and analysis methodology used in the real cask analysis were used in the FE analyses of the scale model. To allow a robust comparison between test and analysis results, the analysis model represents the complete cask, and all of the components are explicitly modelled in three dimensions.

This paper presents the dynamic impact characteristics of the cask from test and analysis results and the validation of numerical simulation method by showing the correlation between test and analysis results.

In addition, several sensitivity analyses were carried out to investigate the effects of the various modeling and design parameters that can affect the impact characteristics of a cask and the accuracy of the numerical results.

ABSTRACT 222

Benchmarking of Analytical Methods and Analysis Software Used for Transportation Package Drop Analysis

RAHEEL HAROON*, PETER SHIH

10 CFR 71 regulations require the containment boundary of a transport package to be able to withstand a 9 m (30 ft) free drop onto a flat, essentially unyielding surface. Impact limiters (shock absorbers) are designed to limit the deceleration experienced by the cask and its contents during the impact. The design and testing of the impact limiters can be long and tedious. Often, the final design of the impact limiters can only be validated with a full or reduced scale drop test. However, performing the drop test is extremely costly and time consuming for larger transportation packages. This paper presents a free drop simulation using the LS-DYNA computer code to perform benchmark analyses as an alternative to performing a drop test.

A series of drop tests have been performed on a one-third scale mockup of the MP197 Transport Package equipped with impact limiters. The cask was dropped in three different orientations, including a 90° End Drop, 0° Side Drop and a 20° Slap Down. The 90° End Drop was performed with the impact limiters chilled to -20°F in order to analyze the effect of the low temperature on the impact limiter performance. The impact limiters consist of balsa and redwood blocks for energy absorption, enclosed by stainless steel plates to position and confine the wood blocks.

This paper presents detailed descriptions of how the LS-DYNA code was used to perform the benchmark analysis. This includes description of the finite element models of the cask, impact limiters, and unyielding surface, boundary and initial conditions.

The results of the LS-DYNA analysis in terms of peak filtered accelerations, maximum crush depths, and impact duration for the three drop cases are presented and compared to the drop test results. It was found that the LS-DYNA analysis results correspond well with the measured impact limiter drop test results. Therefore, the LS-DYNA code simulation can be used to accurately evaluate the free drop of a transportation package with impact limiters.

ABSTRACT 106

Verification of LS-DYNA Finite Element Impact Analysis by Comparison to Test Data and Classic First Principle Calculations

ANDREW LANGSTON*, VICTOR SMITH

The Global Nuclear Fuels (GNF) RAJ-II BWR fuel package is a second generation Type B(F) all stainless steel package designed to replace the older wooden/steel package designs. The RAJ-II is a collaborative effort between GNF, Westinghouse, and Areva.

During recent licensing activities in the EU, questions regarding the performance of the package during regulatory impact events were addressed. In order to evaluate the effectiveness of the RAJ-II BWR fuel container during a variety of impact conditions and orientations, a comprehensive LS-DYNA model was developed. LS-DYNA is a commercially available general-purpose multiphysics simulation software package used to solve highly nonlinear transient dynamic finite element analyses (FEA) using explicit time integration. To benchmark the model, analytical results were compared to drop test

data from the RAJ-II development program and similar first generation designs. Verification of the LS-DYNA analysis was performed using first principle calculations. Because the impact absorbing materials that protect the RAJ-II are comprised of simple shapes and materials that are well characterized, the classic calculations successfully bounded the LS-DYNA and test results. This study has shown that combining finite element analysis, drop test data, and classic first principle calculations is an effective approach for evaluating and optimizing package design.

ABSTRACT 67

Numerical Simulation and Experimental Testing of Brit Lead Cask (BLC)

DHIREN SAHOO*, JOTIRAM MANE, VINAY BHAVE, PIYUSH SRIVASTAV, ANIL KOHLI

BRIT lead cask (BLC) has been designed as 'Type B (U)' package to transport 100kCi of Co-60 sealed sources. The main design feature is its inclined fins on the outer surface. The fins are designed to protect the cask from damaging effects of 9m drop on rigid surface in accident conditions and to enhance heat dissipation during normal conditions of transport. The performance of cask after impact is numerically simulated to assess the structural integrity using explicit Finite Element code PAM – CRASH specialized for nonlinear dynamic simulations. The most damaging orientation was found out from the various possible drop situations by numerical analysis. Experimental drop of the prototype cask on unyielding target was conducted followed by 8000C fire test.

Radiometry test of the damaged cask was carried out to ascertain the shielding integrity of the cask. The paper presents comparison of numerical simulation and the experimental drop test results. The results are found to be in good agreement. The results of Radiometry before and after the accident conditions are also discussed.

Keywords: Inclined fins, Structural integrity, Fire testing.

Abstracts – Wednesday 06 October 2010

T23 - COMPETENT AUTHORITY ACTIVITIES

9:00AM – 10:40AM – TECHNICAL SESSION – MAIN HALL
CHAIR: FRANZ HILBERT, CO:CHAIR: FERNANDO ZAMORA

ABSTRACT 7

The Association of European Competent Authorities for the Safe Transport of Radioactive Material

STEVE WHITTINGHAM*, LORIS ROSSI

In the coming years there will be a significant increase in the number of shipments of packages containing radioactive material. The main contributors to this increase will be:

- the medical sector;
- the transport of low level non-nuclear wastes to disposal sites;
- the nuclear industry from decommissioning activities and the revival of the nuclear sector.

Clearly many of the challenges faced by a Competent Authority (CA) to demonstrate that the levels of safety provided by the transport regulations are being achieved in practice are common in many countries and it would be the most efficient and effective solution to share CA resources.

It is to meet this challenge that the creation of an â€˜Association of European Competent Authoritiesâ€™ has been made in February 2008. At the moment the EU countries involved represent more than 75% of the overall EU territory and population.

Broadly speaking the mission of the Association is to develop co-operation between Competent Authorities (CA). All the subject matters that are the responsibility of national Competent Authorities can be addressed in principal in the framework of the Association.

We propose to present the mission and objectives of the Association, the results achieved so far and the future developments envisaged.

ABSTRACT 171

Development and Implementation of the “Joint Canada-United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages”

MICHELE SAMPSON*, KARINE GLENN, MICHAEL CONROY

In both the United States and Canada, for Type B and fissile material transportation package designs approved by another country, a technical review is typically performed by the Competent Authority prior to issuing approval to allow use in their respective country. In 2005, the United States Nuclear Regulatory Commission (NRC), Department of Transportation (DOT) and Canadian Nuclear Safety Commission (CNSC) embarked on an effort to develop a framework to facilitate the United States/Canadian validation of each other’s certificates with limited additional technical review. After nearly 5 years of work, the framework was finalized and the first application under this streamlined process was received in late 2009.

There are two components of this framework, a guidance document and an Administrative Arrangement between NRC, DOT, and CNSC, for the sharing of information related to the guidance document. The guidance document, “Joint Canada-United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages” was issued in the United States as NUREG-1886, and in Canada as RD-364. A draft of the document was issued for public comment in both the United States and Canada in May 2008, and, after comment resolution, the final guidance document was published in March 2009. The Administrative Arrangement will formalize the manner in which the guidance is to be managed and integrated into each organization’s regulatory programs.

This paper provides an overview of the process for development of the guidance document and Administrative Arrangement. The paper discusses how differences between the United States and Canadian regulations are addressed in the guidance document. In addition, the paper describes the process for implementation of the framework, the review of initial applications, and plans for periodic bi-lateral review of the effectiveness of the framework. The successes realized in the development of this bi-lateral approval process framework may provide a path forward for additional agreement, further streamlining the process for approval of package designs by multiple Competent Authorities.

ABSTRACT 91

How the UK Competent Authority Has Developed a Risk Based Strategy for Carrying Out Non-Nuclear Small User Inspections

MICHAEL TURNER*

The IAEA carried out a Transport Safety Appraisal Service Mission in June 2002 of the implementation of the Transport Regulations in all relevant transport activities in the UK. One of the recommendations stated that “It is recommended that the Department for Transport (DfT) should evaluate the adequacy of its audit and inspection programme and that the necessary resources should be provided for audits and inspections. Specifically, minor consignors and consignors of mobile sources should be more fully integrated into this programme. Priorities should continue to be risk based to maximize the effectiveness of the limited resources”.

Since 2002, the Department for Transport has developed and evolved a ‘risk based’ strategy to carry out non-nuclear small user inspections as part of its overall responsibility to ensure compliance with GB legislation. These include but are not limited to: industrial radiographers, hospitals, road construction services and couriers. There are over 2500 organisations in GB registered as holders of radioactive material.

The inspections are based on the requirements of the Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2009 and previous regulations. The majority of the requirements of these regulations are referenced from the European ADR 2009 Agreement, which, in turn, is based on the requirements of the IAEA TS-R-1 with regard to radioactive material.

This paper presents a summary on how the system has evolved from 2002 to the present day and plans for the future:-

- Enforcement strategy.
- Risk based inspections.
- How greater emphasis has been placed on questionnaire sets to assess risks, and how these have evolved with time and experience.

Abstracts – Wednesday 06 October 2010 : continued

- More open dialog with other Agencies and enforcement bodies who have a role in either licensing premises for holding radioactive material or working with radioactive material.
- Inspector training.
- Additional security requirements.
- How information technology can help.
- Identifying industry shortfalls and implementing industry awareness training.

ABSTRACT 51

Effective Analysis Submissions – A Regulator Perspective

JOSEPH OYINLOYE*

Transport regulations require that the safety of packages containing radioactive materials be demonstrated deterministically by subjecting the package to a number of physical loadings which have been selected to simulate those which the packages may experience during routine, normal and accident conditions of transport. Such demonstration can be by analysis, physical tests or by reasoned argument.

This paper discusses analysis submissions (i.e. those in which demonstration of safety is by means of analysis) from a regulator's perspective, with the aim of explaining what regulators expect to see in such submissions. This perspective is based on the author's current regulatory experience of "good" and "bad" submissions and on over thirty years previous experience as an analyst.

It is expected that the more complex analyses will not be undertaken by the applicants themselves, but via a specialist, third-party contractor, with the applicants playing the role of an "intelligent customer". The paper therefore also discusses the effective organisation of the analyses for this particular situation, and the required interactions between the regulator, the applicant and the analysis contractor.

ABSTRACT 17

Verification of Activity Release Compliance with Regulatory Limits within Spent Fuel Transport Casks Assessment

ANNETTE ROLLE*, BERNHARD DROSTE, SVEN SCHUBERT, FRANK WILLE

BAM is the German competent authority for mechanical, thermal and containment assessment of packages liable for approval. Admissible limits for activity release specified in the IAEA regulations (10-6 A2 per hour for normal and A2 per week for accidental conditions of transport) have to be kept by appropriate function of the cask body and its sealing system.

In general direct measurements of activity release are not feasible. Therefore the most common proof method is relating permissible activity release of the radioactive contents within a containment system to equivalent gas leakage rates under specified test conditions. Applicable procedure and calculation methods are summarized in the International Standard ISO 12807.

Differing from the methodology described in this standard the permissible standardized leakage rate is not related unconditionally to the test leakage rates measured before transport or after a mechanical or a thermal test. Design data for leakage rates have to consider the sealing behaviour including modifications of groove geometry under normal and accidental conditions of transport in the most conservative way and have to be deduced from results of different test situations with real casks combined with results of test series with containment components, like elastomeric or metal seals compressed between flanges. BAM has to check if design data for standard leakage rates specified are conservative especially in cases were test results from one design are transferred to another similar design.

The specification of releasable activity from cavity to the environment is a further important point. The potential influence of relevant fuel characteristics or higher burn-ups to the rate of failed spent fuel rods, the airborne fraction of solid or the fission gas fraction has to be considered and assessed by BAM.

This paper gives an overview about methodology of activity release calculation and correlated boundary conditions for assessment.

T24 - DRY STORAGE ISSUES

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 1

CHAIR: HOLGER VOELZKE, CO-CHAIR: JAYANT BONDRE

ABSTRACT 215

Meeting the Challenges of International Projects

GARCIA JUSTO*

Used fuel storage is a common issue in all countries with nuclear reactors. Notwithstanding considerable efforts to increase the efficient use of nuclear fuel and to optimize the storage capacity, delays in realizing geological repositories in most countries or in implementing recycling in some countries entail in increased used fuel storage capacity needs in combination with longer storage durations. This trend combined with more sophisticated fuel design and higher enrichment requires providing innovative solutions tailored to each customers.

Facing these challenges, AREVA Logistics Business Unit, through its subsidiaries TN International in France, Transnuclear Inc. in the USA and Transnuclear Ltd. in Japan, has decided to launch an extensive innovation process to create the new generation of transport and storage systems and also to renew the management of the projects. Thanks to its international subsidiaries, AREVA Logistics Business Unit could form multicultural project teams. Each team consists of several experts from three continents in order to propose to customer the more efficient teams in term of knowledge and know-how. These multicultural teams could understand more efficiently the customer's needs as well as the different regulatory requirements of each country.

The purpose of this paper is to present these experiences, and furthermore to underline our know-how and ability to provide high efficient solutions to our customers.

Abstracts – Wednesday 06 October 2010 : continued

ABSTRACT 15

Kozloduy Dry Spent Fuel Storage

ANDY ANDREWS*

The project comprises the design and construction of a dry spent fuel store together with the provision of 34 spent fuel storage casks and the associated handling equipment for Kozloduy Nuclear Power Plant Plc. Funding is provided by the EU and is administered by the European Bank for Reconstruction and Development via the Kozloduy International Decommissioning Fund.

GNS have provided the expertise to prepare the detailed design, licensing application and manufacture of the storage casks. In addition GNS will provide the entire specialist handling equipment required to transfer the spent fuel assemblies from the storage ponds in to the casks and subsequently relocate the casks to the dry store.

Each cask is designed to hold up to 84 VVER 440 fuel assemblies in an inert atmosphere for a nominal storage period of 50 years. The casks are sealed using a combination of bolted and welded lids to ensure long term security.

At the time of writing, initial deliveries of the casks have commenced and the project is entering the next major stage which requires the installation of the cask handling equipment in the existing Wet Storage Facility. The facility will then be tested and commissioned with a target date of August 2009 for the loading and transfer of the first cask.

The paper presented will provide an overview of the issues and challenges surrounding the design, licensing and construction of the casks and their associated equipment. It will discuss the critical success factors that should be considered when working with a multinational decommissioning project team on a large international project

ABSTRACT 338

Influence of ISFSI Design Parameters on the Seismic Response of Dry Storage Casks

GORDON BJORKMAN*

Many nuclear utilities have considered using upright cask systems for the dry storage of spent nuclear fuel. These casks are in most cases free standing and rest on a reinforced concrete pad in a variety of arrays. Stability requirements to prevent incipient tipping and sliding of the casks are often based on the cask not exceeding specific limits on either the ZPA of the site ground spectrum or the acceleration at the cask/pad interface (top of pad). Implicit in the use of either the ZPA or the acceleration at the top of the pad, is the assumption that the acceleration at the top of the pad is the same as the acceleration at the center of gravity of the cask, and, therefore, no amplification occurs between the top of the pad and the cask's center of gravity. In contrast to this assumption, the author's experience in the evaluation of ISFSI sites has shown that the cask/pad/soil system can significantly amplify the acceleration response at the cask center of gravity to levels well above the acceleration at the top of the pad.

This paper presents the results of an investigation to determine the influence of three parameters on cask response: pad flexibility (i.e., pad thickness), soil properties and cask layout. A total of 18 SSI analyses were performed with various combinations of these parameters using the SASSI program. The results show that the most important parameter affecting cask response is the out-of-plane flexibility and rigid body rotation of the pad, and that this parameter can significantly amplify cask acceleration response at the cask center of gravity. Graphs and tables showing the influence of each parameter on response are presented. These results should be helpful to engineers making preliminary or confirmatory seismic response evaluations of ISFSI sites and design parameters. However, it is important to point out that these results only apply to the prediction of the onset of sliding or tipping. Once tipping or sliding has occurred these results no longer apply, and a non-linear analysis, much like that of NUREG/CR-6865, must be performed.

ABSTRACT 408

Thermal Evaluation of Loading and Drying Operations of a High Capacity Spent Fuel Storage Canister

MIKE YAKSH*, CHRISTINE WANG

Higher capacity designs for storage of spent fuel provide an advantage of ALARA during loading operations and optimal use of space on the storage location. In this paper the thermal analyses for a transfer system to store 87 BWR fuel assemblies are presented. The assemblies are stored in a welded canister whose fuel assembly's positions are maintained by a non-welded basket configuration. The operations to complete the drying operations and to move the canister with in the plant employs a shielded transfer cask which also uses a cooling system to remove heat from the system. Heat rejection internal to the canister is primarily accomplished by natural convection, and heat rejection from the canister surface uses circulating water. Two separate thermal analysis models are employed for this evaluation. During the drying phase, a detailed three-dimensional model of the basket and canister determines the transient thermal response of the fuel. This defines the duration of the vacuum drying cycle. Segments of the final operations permit the helium backfill in the canister to reject heat by convection from the fuel assembly to the canister surface. This is simulated using a two-dimensional CFD model which incorporates porous media modeling to represent the hydraulic resistance of the fuel rods and fuel assembly grids. Certain transient conditions require the results of the three-dimensional conduction model to define the initial conditions of the two-dimensional CFD transient model. The simulation methodology makes efficient use of both technologies to determine the thermal response of the system for all operational conditions. The results confirm that the maximum component temperatures remain within their allowable temperatures.

ABSTRACT 217

Applying Optimization Methods and Stochastic Analysis in Evaluating a Storage Accident

WALTER VOELZER*, ROBERT GARTZ, MATTHIAS HECK, THOMAS SEIDER, MARCO GROSSE

During CASTOR®-cask handling in interim storage facilities, a hypothetical crane failure has to be considered. In this case, the cask falls onto the transport vehicle and, subsequently, into the reception area. The impact onto the ground will be cushioned by a cavity filled with porous concrete. Should the cask tilt, however, it cannot be excluded that it hits the ground with its top-end outside the area of porous concrete. For this type of accident, the load on the cask needs to be evaluated.

To evaluate realistic impact scenarios, the entire complex drop sequence needs to be simulated. Furthermore, the drop sequence is influenced by numerous parameters, so that the worst situation in terms of loads cannot be determined by theoretical consideration.

For this matter, the drop sequence was simulated on the one hand as a simplified and physically plausible FE-model which represented the cask as a rigid body. On the other hand, a sensitivity analysis with the program OptiSLang identified those parameters which contribute significantly to a high rigid-body deceleration and therefore cause adverse load situations. OptiSLang offers efficient methods for sampling and stochastic analysis. It eases through the high given number of parameters with the lowest possible number of computation runs to arrive at a resilient statistical evidence and further contributes to the automatization of the computation process.

A stress analysis for some selected impact scenarios can be conducted by means of an adequately detailed and sufficiently discretized FE-model. Here, it is possible to position the cask immediately prior to the impact and to initialize the kinematic, which has been determined on the basis of the simplified model, at this point in time.

The complete FE-model is composed of individual partial models which need to be adequately realistic. This is especially important for the partial model of the porous concrete. In order to simulate the energy dissipation of this compressible material, a complex material model had to be used. The material characteristics required for this were calculated with optiSLang. This calculation here means an optimization task in order to minimize the deviations of a test and the test re-calculation.

T22 - SOURCES AND RADIOPHARMACEUTICALS

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 2
CHAIR: PAUL GRAY, CO-CHAIR: TBC

ABSTRACT 233

Transport of Radiopharmaceuticals, Cradle to the Patient

CHARLIE CARRINGTON, EUGENIE ROELOFSEN
(PRESENTED BY ROB DEKKERS)

This presentation looks at the constraints and challenges of moving Radiopharmaceuticals, from the raw material to the end customer, the patient.

Most Radiopharmaceuticals tend to be short half life nuclides used to diagnose or treat disease.

This paper will concentrate on the nuclide molybdenum-99 with the daughter product technetium-99m and Iodine-123.

Technetium-99m is mixed with inactive (cold) material and injected into the patient, depending on the cold material, the radioactive material will concentrate in the organ (s) to be evaluated. After scanning the patient, the doctor can then make a diagnosis.

Iodine-123 is diagnostic agent used for the detection of tumours and other diseases.

Many millions of scans are undertaken each year across the globe.

Through the supply and manufacturing chain for molybdenum 99, from reactor to manufacturer, chemical process to produce sterile material and then from the manufacturer to the end user has to be well choreographed. Any delays in any of the transport routes causes a lose due to decay of Molybdenum 99. Molybdenum has a half life of 66 hours and delay of one day means ~30% lose of material and therefore subsequent doses for use with the patient. In contrast Iodine-123 has a half life of 13.2hours, the process of making the raw material using a cyclotron, processing and transporting must happen on the same day. Any delay invariable means that the product will not be able to be used. To add to the problems of the product being radioactive, it also has to meet the strict requirements of injectable drugs.

Delays have an impact on the Healthcare system, not just monetary for the manufacturer but also the cost to Healthcare system for re-scheduling, lose of scanner time and of the course the disruption to the patient.

ABSTRACT 64

Existing Practices for Safe Transport of Radioactive Sources in Bangladesh

ABDUS SATTAR MOLLAH*

Radioactive materials are being transported throughout the world in ever increasing quantities. All of these materials are transported on land, sea or in the air in compliance with regulations which are now universally based on the IAEA Regulations for the Transport of Radioactive Materials. These regulations are designed to ensure adequate radiation protection of transport workers, the public and the environment against the hazards of external radiation, the spread of radioactive contamination and security measures. Bangladesh Atomic Energy Commission (BAEC) issued its Nuclear Safety and Radiation Control (NSRC) Rules-97 which also incorporates IAEA Transports Regulations.

Among the most important rules to be considered in monitoring compliance for the usual shipments of radioactive are the requirements for information in shipping papers and on package labels, affixing warning placards to every side of the transport vehicle, and limiting the potential radiation exposure in terms of label categories and the transportation index (TI). Keeping in view the IAEA Transport Regulations, a number of radioactive sources of large activity imported abroad were transported inland over large distances. These sources have been imported for research in agriculture, industry and medicine. Since these sources are to be considered to be a "strong radioactive sources" any unprecedented accident may result serious radiation hazard to those directly working with it and to the general population of the area.

The safety measures are, to the extent possible, adopted in packaging the materials for transport. The transport workers who, in general, will have n specialized training must observe some relatively simple rules concerning the stacking of the packages and their segregation from persons. According to the BAEC Rules no person or cargo ship or aircraft carrying radioactive material on board is allowed by the port authorities to enter Bangladeshi ports unless permission had been granted in advance by the BAEC in this regard. This paper briefly describes such transport aspects as packing and package design requirements, categories, labeling and marking of containers

Abstracts – Wednesday 06 October 2010 : continued

during storage and transport, and also discusses the IAEA Transport Regulations vis-à-vis Bangladesh and data is presented to illustrate transportation experience in Bangladesh.

ABSTRACT 395

Controlling Sources and the Transport Implications

JIM STEWART

The control of sources, the transboundary movement of sources and the problems experienced when finding sources in scrap are investigated in this paper.

ABSTRACT 229

Sustaining Reliable Maritime Shipments of Radioactive and Nuclear Materials

PETER LAMBOURNE*

The reliable transport of radioactive and nuclear materials affects all of us, from the supply of electricity to our homes and hospitals, the many consumer products we buy every day that contain small amounts of radioactive materials and the use in medical applications. For example, 45% of the world's single use medical supplies are sterilized using gamma radiation from Co-60 sources. Commercial scale radiation processing saves lives and benefits society in many other surprising ways:

Much food packaging is sterilized before use to minimise adulteration of food products with surface bacteria. Radiation processing is also playing its part in environmental protection. Some high-risk hospital waste is irradiated to sterilize it before disposal. Maintaining reliable shipping routes for a total consignment of only a few thousand containers of class 7 cargo per year is a challenge for suppliers of radioactive and nuclear materials.

In the face of increasing legislation, many ports have become unwilling to manage the intermittent transit class 7 cargos. Not surprisingly this has a big impact on the ability of carriers to accept such class 7 cargo. For 40 years commercial quantities of radioactive materials have been transported to users around the world without radiological incident. The Flasks for the transportation of commercial quantities of radioactive materials must be approved under the IAEA Regulations for the Safe Transport of Radioactive Materials (TS-R-1, 2009).

This paper will examine and some of the problems shippers face and the consequences to mankind caused by ports and shipping companies that do not accept radioactive materials. It will explain the actions industry has undertaken to alleviate the problems and how industry, the International Maritime Organisation and International Atomic Energy Agency work to overcome these difficulties.

There are clearly strong humanitarian reasons for port authorities and shipping companies to review the management and implementation of shipping legislation for the shipment, transit and trans-shipment of these materials.

ABSTRACT 47

Packaged Material in Foreign Countries

CRISTY ABEYTA*, JAMES MATZKE

The Global Threat Reduction Initiative's (GTRI) Off-Site Source Recovery Project (OSRP), which is administered by the Los Alamos National Laboratory (LANL), removes excess, unwanted, abandoned, or orphan radioactive sealed sources that pose a potential risk to health, safety, and national security. In total, GTRI/OSRP has been able to recover more than 19,000 excess and unwanted sealed sources from over 750 sites. In addition to transuranic sources, the GTRI/OSRP mission now includes recovery of beta/gamma emitting sources, which are of concern to both the U.S. government and the International Atomic Energy Agency (IAEA). This paper provides a synopsis of cooperative efforts in foreign countries to remove sealed sources by discussing three topical areas:
1) The Regional Partnership with the International Atomic Energy Agency; 2) Challenges in repatriating sealed sources; and 3) Options for repatriating sealed sources.

T21 - SHIELDING CALCULATIONS

9:00AM – 10:40AM – TECHNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: TBC, CO-CHAIR: CATHERINE WEBER-GUEVARA

ABSTRACT 23

Comparison of Monte Carlo Codes MCNP and MONACO for Applying to Shielding Calculation of Transport/Storage Cask

HIROAKI TANIUCHI*

To design practical and efficient transport/storage casks for loading higher neutron and gamma source such as spent fuels, shielding calculation is the most important and Monte Carlo codes may be the most powerful tool for the calculation. But so far, there are still difficulties for many users to obtain reasonable and reliable results without many trials. In worst case, he fails to get reliable results. Then, it is important to know what the good usage method of Monte Carlo codes for many users is.

In this study, two major Monte Carlo codes such as MCNP and MONACO are compared with the view point of the user of shielding calculation of casks. MCNP is the most popular Monte Carlo code and is used in the world for plenty of applications. MONACO is not so popular now, but it is based on MORSE code and may have advantage in using for shielding calculation of casks because the MONACO is belong to SCALE 6 code system which has new variance reduction option called MAVRIC.

The important points for using Monte Carlo codes for shielding calculation are 1) easy to make model configuration and easy to find out geometric errors 2)easy to introduce effective variance reduction technique 3)easy to judge re-liability of the results, and 4)reasonable total calculation time including input preparation. Several calculations using such as simple cask geometry and the complicated cask geometry with detailed shielding structure are performed to check the characteristics of both codes concerning the points mentioned above.

With respect to neutron shielding calculation of cask model, reliability and total calculation time of both codes are good enough to apply for safety calculations of casks. On the other hand, considerable techniques are necessary to introduce them into gamma shielding calculation because the attenuation of gamma dose rate inside of the cask is several orders higher than that of neutrons. Regarding gamma shielding calculation, variance reduction method in MAVRIC seems more powerful than Weight Window Generator in MCNP. The detailed discussion is shown in the paper.

ABSTRACT 76

A Comparison of the TRITON and ORIGEN2 Source Generation Programs

RICK MIGLIORE, PHILIP NOSS*

The Battelle Energy Alliance Research Reactor (BRR) Package has been designed to ship high-enriched aluminum-plate fuels from several research reactors. The reactor sites have a long history of using the ORIGEN2 computer program for generating neutron and gamma source terms. Consistent with this experience, source terms for the various research reactors were developed using ORIGEN2. ORIGEN2 is an older computer program that is no longer supported by the code developers. For this reason, the United States Nuclear

Regulatory Commission (NRC) would not accept the shielding analysis without significant justification of the ORIGEN2 reactor libraries used, which were typically developed for low-enriched pressurized water reactors (PWR). The NRC also expressed their dissatisfaction with using such an old program when newer programs were available. Because justifying the data libraries used in ORIGEN2 would be difficult, the source term was regenerated using the TRITON sequence of the SCALE6 code package. TRITON allows a two-dimensional representation of the fuel elements, and generates the data libraries in a problem-specific manner. When the source terms were regenerated using TRITON, the gamma source was similar to the gamma source developed by ORIGEN2.

However, the neutron source in some cases increased by several orders of magnitude. This paper compares the neutron and gamma source terms for both ORIGEN2 and TRITON, and suggests possible reasons for the differences.

ABSTRACT 324

Study of Cross Section Libraries for Shielding Design of Spent Fuel Cask and Cask Storage Facility

TAKUYA TAKAHASHI*

When defining the shielding design of a spent fuel storage cask or the cask storage facility, nuclear cross section library is one of the most vital factors of calculations, such as source term estimations of the spent fuel, dose evaluations of bulk shielding or it around the cask, streaming or skyshine around cask storage facility.

Sn Transport Codes or Monte Carlo Codes are used with various cross section libraries to these calculations. JENDL3.3 and ENDF/B- , updated nuclear data libraries, are available to use for previously mentioned calculations, however, relatively old libraries are employed for it.

The purpose of this study is to examine the suitable cross section libraries for shielding design of these objects.

1. Libraries for the dose calculations with Sn Transport Code DLC23/CASK library, based on relatively old nuclear data, is generally used in the licensing shielding definition of the cask. Other libraries, based on newer nuclear data, are employed to design the cask storage facility however these libraries are also available for the shielding design of the cask only. MATXSLIB-J33 based on JENDL3.3 data,VITAMIN-B6 based on ENDF/B- , and BUGLE can be used for shielding design of these objects. Doses were calculated by using Sn Code with the above three libraries mentioned.

Calculation results were then compared with shielding database as benchmarking.

2. Libraries for a source term calculation with ORIGEN2 Code ORIGEN2 Code and BWRU library included into it is generally used in combination for source term estimation of a BWR spent fuel in Japan. However, BWRU library is based on old nuclear data. The specification of this library is also different from the current core conditions. Therefore, applicability of new libraries or a new code for shielding design is studied. Source term calculations have been conducted by using ORIGEN2&BWRU, ORIGEN2&ORLIB-J33 based on JENDL3.3, and ORIGEN-ARP Code included in SCALE system. Calculation results are compared and suitable libraries for source term estimations are discussed.

ABSTRACT 228

Impact of Higher Burn-ups on the Transportation Package Design: Radiation Shielding Perspective

PRAKASH NARAYANAN*

The improvements in the fuel designs and fuel performance in the operating nuclear power plants have led to an increase in the discharge burnup of the used fuel assemblies. This increase in burnup also has resulted in the improvements in the design of used fuel storage and transportation systems. The impact of higher burnups on the design of transportation packages offers unique challenges from a radiation shielding perspective.

The transportation cask radiological design is characterized by the applicable dose rate limits on and around the package surface under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC). For most high burnup, large capacity transport casks, the 10 mrem/hour at 2m from the vehicle edge during NCT controls the radiation shielding design.

Used fuel assemblies with acceptable combinations of burnup, enrichment and cooling time after discharge (BECT) are considered eligible for loading in a transportation cask. The increase in fuel assembly burnup results in an increase in the decay heat of the fuel assembly and also results in an increase in the neutron source term of the fuel assembly. Therefore, cask designs to accommodate fuel assemblies with higher burnups need to be modified to enhance neutron shielding. Zone loading of fuel assemblies to effect self-shielding within the basket is also necessary for this purpose.

This paper examines the important considerations for qualification of high burnup fuel assemblies from a radiation protection perspective. This paper also evaluates the impact of higher burnups on the source terms and discusses the enhancements to the dose rate calculation methods. The qualification of fuel with higher burnups is

Abstracts – Wednesday 06 October 2010 : continued

truly an optimization problem that balances shielding material design and fuel assembly parameters (burnup, enrichment and cooling time after discharge) selection with proper application of shielding analysis methodologies.

T27 - DENIALS OF SHIPMENT

11:00AM – 12:40PM – TECHNICAL SESSION – MAIN HALL
CHAIR: STEVE WHITTINGHAM,
CO-CHAIR: KASTURI VARLEY

ABSTRACT 361

Maritime Shipments of Radioactive Materials

STEFAN HOEFT*

The main routes for shipments of class 7 material by sea are;

- Europe – North America (westbound/eastbound)
- Europe – Russia (westbound/eastbound)
- Europe – Far East
- Europe – South America
- Australia – Europe – North America
- Australia – Far East
- Africa – Europe
- Africa – North America
- North America – Far East
- South America – North America

With the main class 7 products shipped being;

- Enriched UF6 (max. 5 % Uranium-235)
- Natural UF6
- Uranium Ore concentrates (U3O8)
- Uranium Oxide (pellets, powder)
- Various sources (e.g. Co60)
- Equipment (cylinders - empty, heeled state; SCO-material)

However there are limited carriers available worldwide who are willing to accepting radioactive material with the majority of carriers only accepting non-fissile material on board of vessels.

This paper will examine and explain the consequences of the different types of maritime services available to the transporters of radioactive materials including liner, fixed routes with fixed ports of call and schedules; charter, flexible routes and overall service; tramp, flexible routes and schedule depending on cargo volume and agreements with charterers. It also draws on examples from the radioactive transport industry on how WNTI helps to alleviate some of these issues.

It will also consider the different modes of transport, comparing how radioactive materials are shipped in the containerized 20, 40 container, flat racks and platforms; in break-bulk (lash barges, vessel holds) and in Ro/Ro (Mafi-trailer)

There are many things that have to be considered when transporting radioactive materials from the basic regulatory framework: - IMDG code (International Maritime code for Dangerous Goods) and supplement codes in combination with the laws and national requirements at ports of calls, licenses, handling permits, package approvals/validations, insurance needs and the list goes on. This paper provides examples of lack of harmonisation and suggests ways of a more common approach to the transport of radioactive materials by sea.

ABSTRACT 190

Denial of Shipment of Radioactive Material

PAUL GRAY*, GRANT MALKOSKE

The International Source Suppliers and Producers Association (ISSPA) is comprised of most of the world's major manufacturers of sealed sources. In 2009, ISSPA was the Chair of the IAEA International Steering Committee on the Denial of Shipment of Radioactive Material.

We are all impacted in some way by the peaceful uses of radioactive materials. Currently, more than 35 million nuclear medicine procedures are performed annually around the world using short lived radioisotopes. Cobalt-60 sealed sources are used for external beam radiation cancer treatment with more than 45,000 treatments per day provided in some 50 countries around the world. In addition, Cobalt-60 is used to sterilize some 45% of all single use medical disposable products, and in the food industry for preservation of food or sterilization of food packaging materials. Further, radioactive sources are used in industrial applications for the checking of weld and structural integrity; in industrial facilities for process control, and in numerous other industrial, agricultural and home applications.

Use of these products is dependent on safe, secure, timely and cost efficient transportation both within and between countries. Delay or denial of shipment of these products has a direct and potentially serious and life threatening negative impact on industry, on health care and on individuals around the world. Reported denials now number in the hundreds – where and why? What are some of the key issues causing denial of shipment? What are international agencies, regulators and industry doing about it? How has this problem been resolved in some jurisdictions? What are some of the key activities that the IAEA International Steering Committee is engaged in? What accomplishments were achieved and what are some areas that still require international effort amongst key stakeholders? This presentation will answer these questions and more, and will provide industry perspectives on how the issues causing denial of shipment can be addressed

ABSTRACT 38

The Role of National Authorities in Minimizing Denials of Shipments

NAT BRUNO*, ARANGURAN NANDAKUMAR, MICHAEL WANGLER

Denials and delays of radioactive shipment have occurred and probably will always do. Reducing instances of denials to acceptable levels may be a tangible goal that consignors and consignees should envisage and pursue. It should be recognized that the basis for solving denials and delays includes the participation of not only the national competent authorities but other authorities too. Shippers, consignors, carriers and consignees should include in their consultations the whole range of those authorities which APPARENTLY have no role to play. These "other authorities" who represent various State authorities may play an important role on this issue.

Successful cases where port authorities, air transport agencies, airlines and other carriers have agreed to cooperate to minimize the problems are documented. These cases should be broadcast and addressed taking into account regional/national characteristics and reality.

Some successful cases documented in Brazil, a country where the national competent authority for transport of radioactive material declares itself outside the problem, may be useful for this purpose.

This paper describes and discusses some national experience and how IAEA Member States may use this experience to reduce the instances of denials and delays of radioactive shipment by developing similar models appropriate to their conditions.

ABSTRACT 135

Experience on the Management of the Regional Network in the Mediterranean Basin for the Denials of Shipments of Radioactive Material

SANDRO TRIVELLONI*, MARTIN FERNANDO ZAMORA, BERNARD MONOT

The Regional Network in the Mediterranean Basin (Rome Group) was established in the framework of the Regional Networks created by the IAEA in response to a resolution by the 51st regular session of the IAEA General Conference as one of the actions to try to solve the problem of denials of shipments of transport of radioactive material. It was created during the workshop held in Rome (14 to 16 May 2008). The main objective of the Rome workshop was to develop a Regional Action plan on the basis of the general action plan established by the International Steering Committee tacking into account the different characteristics and needs of the countries of the network to remove the causes of real or potential denials of shipments . The responsibility to coordinate the actions of the Regional Plan was assigned to the Regional Coordinators (France, Italy and Spain).

One of the first action of the Regional Coordinators was to try to quantify the phenomena of denials in the Mediterranean Basin, particularly for sea transport, both by the data recorded into the IMO database contained into the Global Integrate Shipping Information System (GISIS) and on the basis of a questionnaire that was distributed to the countries of the network. The experience of these two year of life of the Mediterranean network put in evidence the difficulties to act only at general level by lobbing, training, etc., to solve the phenomena of denials and encouraged to act also on case by case when the denial of shipment is reported by the operators. The paper will illustrate the actions and the results of two years of functioning of the Mediterranean Network.

ABSTRACT 253

Nuclear Renaissance, Nuclear Transports : the Communication Challenge: Front end Experience

BERNARD MONOT*

This presentation will elaborate on the nuclear transport challenge facing a growing flow of nuclear transports to fuel the Renaissance and the associated communication issues.

In a first part we will describe the condition to set up the Renaissance of nuclear energy in the world. As the first Renaissance in the XVIth century started with the figures of Leonardo da Vinci and Laurent of Medicis, the nuclear one's has to be fuelled by Innovation and Money. The question is: what are the prerequisites in term of communication to gain the global acceptance for these developments?

Nuclear transport is the most visible part of the nuclear fuel cycle. As for others means of energy transports, nuclear ones are not very popular: they don't bring any advantages stable nuclear plants can offer: jobs, money, local equipments with the associated consequences: quite good image, large neighbouring relationship, and easier opponents containments.

The different stakeholders involved in transports, have not the same agenda: some are supportive, some are neutral others are opponents and their positions could move easily under external pressures, rumours and misunderstandings.

So far the question is: why speak? We would better live unnoticed. We are part of the communication flow addressing the possible accidents, fighting rumours and misinformation.

The front end is characterized by :

- a natural dispersion of the uranium mines in few countries in the world unfortunately far from the enrichment facilities which means long and numerous transports of heavy tonnages.
- A concentration of enrichment facilities in few countries for financial and non proliferation reasons which leads once again to a flow of UF 6 transports
- Due to the enlargement of the world nuclear fleet, a huge stream of fuels is to be loaded in the power plants.
- And more over the large amount of material has to use commercial shippers and/or airlines and could face Denials and Delays.

The presentation will address all the communication challenges including evolution of the messages to be delivered and how to address the denials and delays challenges.

ABSTRACT 425

Documentation Requirements for Class 7 Transports

ROB VAN UFFELEN*

This paper will examine in detail shipping documents that are required to be completed for an international maritime shipment of radioactive material. The paper will provide a summary of the required fields within the documentation, and it will further explain why these are required by the competent authority; shipping company; freight forwarder; port authority; customs, and others. This summary will provide an understanding to those not familiar with shipping documents. The study also will highlight where duplication of information appears in the documentation, and any lack of harmonisation in presentation where the same information is required between different organisations, authorities and countries. It will highlight the difference in time for preparation of the documents for a Class 7 radioactive cargo compared with other cargoes.

The study will compare the typical documentation required for a successful unhindered international maritime transport of Class 7 radioactive materials with those required for other dangerous goods, and non-dangerous goods.

Importantly, the study will suggest where harmonisation between the shipping documents could be made, and recommend reductions of the documentation burden.

Abstracts – Wednesday 06 October 2010 : continued

T26 - NON SPENT FUEL PACKAGE DESIGN

11:00AM – 1:00PM – TECHNICAL SESSION
– CONFERENCE ROOM 1
CHAIR: KEN SORENSON, CO-CHAIR: TBC

ABSTRACT 366

Cage Design, Impact Analysis and Experimental Testing of Teletherapy Source Transportation Flask

J. V. MANE*, S. SHARMA, V. M. CHAVAN, D. C. KAR, R. G. AGRAWAL

A lead shielded Teletherapy Source Transporation Flask has been designed and used to transport cobalt-60 radioactive source from the source manufacturer's site to the location of the teletherapy machine. Thermal shield made up of ceramic wool is provided around this flask to avoid lead melting under 8000C fire test. As this is a delicate thermal shield, it is required to design a structural cage to protect it and intern cask itself under accident condition mechanical tests of IAEA regulation. This paper explains cage design and impact analysis of cask with cage using F.E. Software PAM-CARSH. Also experimental impact and fire test are performed on full scale model cask. The result of numerical simulation and experimental testing are presented and these are found in good correlation which demonstrates compliance with regulatory tests.

ABSTRACT 73

The DN30 Overpack - a New Solution for the Transport of Commercial Grade and Reprocessed Enriched UF6

FRANZ HILBERT, WOLFGANG BERGMANN*,
FREDERIC NOYON

The transport of enriched uranium hexafluoride is one of the most essential parts of the nuclear fuel cycle. For these transports 30B cylinders are used worldwide as primary containment of the material. For protection against mechanical and thermal impacts from accident conditions of transport the cylinders are enclosed in Protective Structural Packages (PSPs).

Recently, the certificate of package approval USA/4909 expired once and for all reducing the number of usable PSP designs considerably. To ensure reliable transportation and continuous availability of PSPs NCS decided to design, license and manufacture the new DN30 PSP. This PSP is going to be licensed for commercial grade and reprocessed enriched Uranium up to an enrichment of 5 wt.%.

The DN30 PSP consists of a top and a bottom part which are connected during transport by a robust and easy to handle closure system. The bottom part is equipped with feet for tie-down compatible with existing designs and allowing the transport of four loaded PSPs on a 20' flatrack as well as loading of the PSPs by removing the top part only.

The design pays special attention to the mechanical and thermal properties of the PSP during accident conditions of transport. Extensive FEM calculations and design optimizations by experts from DAHER, the mother company of NCS, have been carried out to ensure that (Regulations para. 677 (b)) "there is no physical contact between the valve and any other component of the packaging other than at its original point of attachment …". In order to achieve this the DN30 is equipped with an integrated valve protector (pat. pend.). On the plug side the design guarantees as well the "no contact" condition during accident conditions, hence a CSI of 0 applies.

The presentation will provide an overview of the DN30 design. Important aspects of the safety analysis approach and the FEM calculations will be discussed. Finally, the drop test program will be presented

ABSTRACT 147

Recent Approval of the UX-30 as a Type B Package

MARK WHITTAKER*

The UX-30 packaging has recently been authorized by USNRC as a Type B package for transport of greater than Type A quantities of recycled UF6 in 30B or 30C cylinders. The contents are limited by: the quantity of uranium hexafluoride, the fission product gamma activity and the transuranic alpha activity, the hydrogen to uranium atomic ratio, and the total activity. The use of the packaging for transport of Type B quantities of uranium hexafluoride requires that the 30B or 30C cylinder containing the uranium hexafluoride meet the leak test criterion of "leak tight". A maintenance requirement has been added to specify annual testing of 30B or 30C cylinders used for Type B quantities of reprocessed uranium hexafluoride. The USDOT has endorsed this approval and approvals have been requested in various other countries. The paper discusses the specific requirements of the newly authorized contents along with the additional leak test conditions that apply to this expanded content. The current status of endorsement by other countries is also presented.

ABSTRACT 138

FCC NG – Transatlantic Design

FRANCOIS MARVAUD*, PASCALE FAYE, MICHEL DOUCET

AREVA, as a worldwide PWR fuel provider, has to have a fleet of fresh UO2 shipping casks being agreed within a lot of countries including USA, France, Germany, Belgium, Sweden, China, and South Africa … and to accommodate foreseen EPRTM Nuclear Power Plants fuel buildings. To reach this target the AREVA NP Fuel Sector together with TN International decided to develop an up to date shipping cask gathering experience feedback of the today fleet and an improved safety.

As this cask is intended to a worldwide use, it must be a global container which implies that it should be licensed as a minimum in France, Belgium and United States.

This package must therefore comply with both IAEA TS-R-1 regulations and with the United States Code of Federal Regulation, Title 10, part 71 (10 CFR 71).

The requirements of those regulations are not fully identical; therefore the package design, manufacturing and licensing options must comply with the two regulations and their associated code and standards.

The safety options for the package assessment should also fit to each regulation and national requirements:

Abstracts – Wednesday 06 October 2010 : continued

- package type will be different in Europe and in the United States,
- mechanical options synthesises both IAEA and 10 CFR 71,
- thermal as well as criticality assessment are different in Europe and in the United States.

After a brief presentation of the cask, a compared analysis of IAEA and 10 CFR 71 regulations regarding fresh UO2 fuel shipment will be performed and the design, manufacturing and safety options will be discussed.

ABSTRACT 201

Outline of Fresh MOX Fuel Transportation in Japan and Development Status of Transportation Cask for LWR

NORIHIKO TAMAKI*, AKIRA OE, KAZUNARI ONISHI

In Japan, the operation of JNFL MOX fuel fabrication plant for LWR (J-MOX plant) will be started in June 2015, and the preparatory work by JNFL is proceeding with steady steps. The introduction of the MOX fuel into the LWRs in Japan is aimed from the viewpoint of the policy on nuclear energy, and it is in progress certainly.

MOX fuels are transported from J-MOX plant in Rokkasho to the LWR plants in Japan. The transportation cask is type B(M) package, and Physical Protection categorization is category 1. The transportation cask for JNFL MOX (J-MOX cask) is designed to increase the volume of contents as much as possible within the handling restriction of the cask in J-MOX plant.

J-MOX casks consists of two types designed for BWR and PWR fuels, which are of about 6m in length × about 2.4m in diameter including the shock absorber, and of about 25 tons in weight including the contents. The capacity of the J-MOX cask for BWR is 12 MOX fuel assemblies; 8×8 type or 9×9 type, and J-MOX cask for PWR is 4 MOX fuel assemblies; 17×17 type.

BWR fuel is loaded in the fuel holder with fuel tightening function before loaded into the cask. On the other hand, the fuel holder is not necessary for PWR because the cask tightens the fuel directly.

As characteristics of J-MOX cask, multistage cylindrically shaped shock absorbers are used to absorb impact from all direction. Propylene Glycol water solution is used as a neutron shielding material in the lateral of the cask. He-lium gas is filled inside the cask to improve conductivity of decay heat during transportation.

Safety analysis is performed by a proven analysis code for B(M) package permits.

It is scheduled to apply for design approval by the Japanese Authority in the near future, and the transportation test using the dummy fuel will be executed before actual transportation

ABSTRACT 32

CASTOR® HAW28M – Development and Licensing of a Cask for Transport and Storage of Vitrified High Active Waste Containers

ANDRE VOSSNACKE*, RAINER NOERING

From 1997 to 2006 GNS has returned vitrified high active waste (HAW) from reprocessing in France to Germany by using the GNS cask CASTOR HAW 20/28 CG. The cask has a capacity of 28 canisters with a maximum total thermal power of 45 kW. 74 casks of this type were loaded at the reprocessing plant in La Hague, France and have been shipped to the storage facility in Gorleben (TBL-G). The remaining HAW canisters at La Hague site exceed the technical limits of the CASTOR HAW 20/28 CG cask concerning heat capacity and radioactive inventory.. Therefore GNS developed a new cask generation, named CASTOR HAW28M, which ensure the further return of the HAW to Germany. Hence it is possible to load 28 HAW canisters with a maximum total thermal power of 56 kW.

For the CASTOR HAW28M new materials and new design methodologies were developed and applied., Furthermore, increased heat capacity and sophisticated shielding measures were considered as a consequence of the reprocessing of fuel with increased enrichment and burn up. The cask design had been subject to a comprehensive drop test programme performed by the Federal Institute for Materials Research and Testing (BAM) following a complex validation of numerical models to realise a transparent proof of the package performance. For normal and hypothetical accident conditions it was demonstrated that a safe transport and storage of the waste will be ensured also with the new CASTOR cask design.

The CASTOR HAW28M complies with the IAEA (International Atomic Energy Agency) regulations for Type B package designs containing fissile material and fulfils the acceptance criteria of the TBL-G in terms of radiation protection, heat dissipation and safe confinement under both normal and hypothetical accident conditions.

Already in the spring of 2003 the application for approval of the Type B package design containing fissile material was submitted to the competent authority. In September 2009 the design approval certificate was issued by the Federal Office for Radiation Protection (BfS).

The paper gives an overview on the design characteristics, the cask materials developed and the safety analyses performed.

T28 - IMPACT TESTING

11:00AM – 1:00PM – TECHNICAL SESSION – CONFERENCE ROOM 2
CHAIR: KARSTEN MUELLER, CO-CHAIR: TOM DANNER

ABSTRACT 359

Package Testing To Demonstrate Safety with Added Features

PIERRE MALESYS*

Industry has noted that there are different views in the approach about how the transport and/or handling of frames, where they exist, should be taken into account in the safety analysis. This subject is crucial when having to comply with the requirements of paragraph 611 in the 2009 Edition of the IAEA Transport Regulations for the Safe Transport of Radioactive Material (TS-R-1): "Any features added to the package at the time of transport which are not part of the package shall not reduce its safety".

This topic was considered in IAEA during the fifteenth meeting of the TRANsport Safety Standards Committee (TRANSSC 15) in October 2007. It was discussed during a Technical Meeting at the beginning of the year 2008 resulting in convening a Consultants Meeting in September 2008. In support of the discussions of this Consultants Meeting, members of the World Nuclear Transport Institute (WNTI) met in April 2008 to expand their common position. In the Technical

Abstracts – Wednesday 06 October 2010 : continued

Meeting held in January 2010 the output from the above meetings were further discussed and reflected in the draft of the 20xx Edition of TS-R-1, which is now subjected to the Member States and international organisations 120-day review.

The paper considers the following aspects, with an industry perspective:

- What are the technical and administrative difficulties when dealing with the requirements of paragraph 611 in TS-R-1
- What are the boundaries of the package. What is a feature added to the package at the time of transport
- How to assure safety of transports when considering the added features
- What is the outcome of the 2007 and 2009 review/revision cycle of TS-R-1
- How to go further

ABSTRACT 149

Certification Testing of the TRUPACT–III Package

RICHARD J. SMITH*, PHILIP W. NOSS

The Transuranic Package Transporter Model III (TRUPACT-III) is the newest transportation package to be added to the Waste isolation Pilot Project (WIPP) fleet of packagings. The TRUPACT-III package is a 2.50 meter (98 inches) x 2.65 m (104 inches) x 4.29 m (169 inches) rectangular package that has a gross weight of 25,000 kg (55,116 lbs). The package is designed to transport a large rectangular standard large box (SLB2) that contains only contacted-handled transuranic (CH-TRU) waste. The package will transport the CTU-TRU waste from the various DOE sites to the WIPP site for disposal. This paper describes the various certification tests that were performed in support of obtaining the U.S. Nuclear Regulatory Commission (NRC) Part 71 Certificate of Compliance, which is required for all packages utilized for the WIPP site. Included in the testing was one half-scale engineering test unit (ETU), and two full-scale certification test units (CTUs). Due to its size and weight, this paper will include discussion of the unique conditions that were required to be addressed during the testing program.

ABSTRACT 81

Prototype Test of a New MOX Powder Transport Packaging

YASUHIRO KAWAHARA*, TOKUO TAKE, TAKAFUMI KITAMURA, KAN SHIBATA, YUICHIRO OUCHI

The Japan Atomic Energy Agency [JAEA] is planning to transport uranium and plutonium mixed oxide [MOX] powder for the prototype fast breeder reactor ''MONJU'' and experimental fast reactor ''JOYO'' from the commercial Rokkasyo Reprocessing Plant [RRP] operated by Japan Nuclear Fuels Limited [JNFL] in Rokkasyo-mura at the northernmost prefecture on the Japanese mainland of Honshu.

The design and analyses for the new packaging for transporting the MOX from the RRP to the JAEA Plutonium Fuel Fabrication Facility [PFPF] in Tokai-mura, Ibaraki Prefecture, started in 2002.

The packaging configuration is approximately 1.4m in diameter and 2.2m in height; package weight is approximately 4 tons.

The contents are storage canisters containing three MOX powder cans. The design concept is to have the simple and compact design satisfying the required technical standard as much as possible in consideration of operation limitation of its weight and size in the facilities.

The demonstration performance tests according to the requirements for Type BU-F package in IAEA TS-R-1, 2005 Edition were conducted using a 1/1 full scale prototype packaging from 2007 to 2009. It was confirmed to meet the technical standard.

The research for the test facilities in advance has been carried out in consideration of test items. The test data to be measured and these positions have been investigated carefully too. The logic of the safety analysis method was confirmed by these test results.

When designing and manufacturing the packaging, explanation to the basic policy of quality control and manufacturing detail and the examination relates to the quality control are done in the system of law in Japan.

In the fabrication of the prototype packaging, the quality control was done in the fabrication of special material such as neutron shield and welding. And it was confirmed that the manufacturing difficulty would not be foreseen. In the paper, the test results and the logic of the safety analysis method are presented.

ABSTRACT 172

Drop Test Program with the Half-Scale Model CASTOR HAW/TB2

ANDRE MUSOLFF*, THOMAS QUERCETTI, KARSTEN MUELLER, BERNHARD DROSTE, STEFFEN KOMANN

BAM is the competent authority for mechanical and thermal safety assessment of transport packages for spent fuel and high active waste (HAW). In context with the design licensing procedure of the new German HLW cask CASTOR HAW28M several drop test series were performed by BAM with a half-scale model CASTOR HAW/TB2. The cask is manufactured by GNS (Gesellschaft fuer Nuklear Service mbH) and was tested under accident transport conditions on the 200 tons BAM drop test facility at BAM Test Site Technical Safety.

The sequences of the tests comprise cumulations of nine-meter free drop onto an unyielding target in various most damaging drop orientations to cause maximum package damage, and a drop from one meter onto a steel punch onto the most sensitive part of the container. The subsequent release of radioactive substances must not exceed a value specified in the dangerous goods transport regulations, and radiation shielding and nuclear safety must be guaranteed.

For this purpose the prototype CASTOR HAW/TB2 was instrumented on 21 measurement planes with altogether 23 piezo-resistive accelerometers, five temperature sensors and 131 tri-axial strain gauges in the container interior and exterior, respectively. The strains of four representative lid bolts were recorded by four uniaxial strain gauges per each bolt. Helium leakage rate measurements were performed before and after each test sequence.

The paper presents experimental results of the half-scale CASTOR HAW/TB2 prototype (15,000 kg) and provides insight into the complexity of test procedure, extensive test methods and expensive measurement data logging.

Finally, the paper reports about the performance of the lid closure system before and after drop testing and give an assessment about the optimisation of measuring technology and method of testing.

Abstracts – Wednesday 06 October 2010 : continued

ABSTRACT 220

Mechanical Safety Analyses of Cast Iron Containers for the KONRAD Repository

UWE ZENCKER*, MIKE WEBER, LINAN QIAO, BERNHARD DROSTE

At BAM within the last years numerous drop tests with prototype casks made of ductile cast iron were carried out onto targets according to the requirements for final disposal of non-heat generating waste in the German KONRAD repository. The results have shown that the target specifications in the acceptance criteria have to be defined more accurately to get reproducible test results with high precision. Hence a suitable test stand foundation was developed with much effort. The integrity of the upper concrete layer of this target must be preserved during a test.

Recently the geometrical properties of a tested cubic cast iron container led to a concentration of the impact forces beneath the container walls. The target was damaged strongly with the consequence of inadmissible reduction of cask stresses. For that reason the target construction was modified. However, the basic design was not changed. A prefabricated concrete slab was still joined by a mortar layer to the IAEA target of the BAM drop test facility. In the course of the optimization of the test stand foundation the concrete slab dimensions and the reinforcement were enlarged. During the drop test repetition the target kept intact. Additionally, the mechanical behaviour of the cast iron container and the target was analysed by finite element calculations.

This improved target construction is suggested as a reference target for drop tests according to the KONRAD repository acceptance criteria with casks whose mass and base area are covered by the container types VI or VII respectively.

The measurements during the drop tests with the cast iron casks have provided the strains at the cask surface at selected positions. This allows the verification of finite element simulations of the drop tests which show the stress distribution also inside the component. In September 2008 a drop test was carried out with a cylindrical cast iron cask containing an artificial material defect which was designed under consideration of critical stress states in the cask body. This drop test could demonstrate the safety against failure by fracture of a cask made of a special cast iron with reduced fracture toughness.

ABSTRACT 373

Demonstration of the Impact Performance of the Windscale Pile Fuel and Isotope Waste Package

CHI-FUNG TSO*, JOHN CLIFFORD

In accordance with UK Government policy, the approach promulgated by the UK nuclear industry regulators is for radioactive waste to be converted into ‘passively safe’ forms as soon as is reasonably practicable taking full account of the long-term disposability requirements. For intermediate level waste this generally involves conditioning the waste into a solid matrix within a standardised waste container.

The waste management strategy for the fuel and isotope material from the Windscale Piles – two of Britain’s earliest nuclear reactors which were constructed shortly after the Second World War - is to package them into 500 litre drums which comply with the waste package standard defined by the Radioactive Waste Management Directorate of the Nuclear Decommissioning Authority (NDA-RWMD).

It is envisaged that the waste will be characterised, segregated and encapsulated in an organic polymer in one of a variety of liners depending on the waste’s characteristic. The liners will then be placed centrally in a 500 litre drum and encapsulated in a 3:1 PFA:OPC grout. The 500 litre drum will then be closed with a bolted lid.

Among the performance requirements specified by NDA-RWMD on 500 litre drum packages, is performance in impact accident scenarios. The waste packages are required to retain its radioactive contents and remain suitable for safe handing after normal handling impacts. After the transport accident and repository accident impacts, the release of radioactive contents from the waste package is required not to result in the relevant regulatory radiation dose limits to workers or to members of the public being exceeded.

The waste package for the fuel and isotope material from Windscale Piles 1 and 2 has been designed adopting the best practice to achieve the best integrity in such impact accident conditions. Its performance has been demonstrated by finite element analyses, benchmarked against drop tests of prototype packages, in combination with breakup tests of the wasteform materials, and with material behaviour from material tests.

This paper presents a summary of the work carried out and discusses best practice in the impact performance design of waste packages.

T25 - SHIELDING MATERIALS, BASKET MATERIALS

11:00AM – 12:40PM – TECHNICAL SESSION – CONFERENCE ROOM 3

CHAIR: BERNHARD DROSTE, CO-CHAIR: HERVE ISSARD

ABSTRACT 90

A New Use for the Vyal-B Neutron Absorbing Resin

GUILLAUME FOUSSARD*

TN International proposes to its customers a wide range of solutions for the transport of nuclear material and takes part of all the phases of the Nuclear Fuel Cycle. Thus TN International has developed neutron shielding materials for various duties adapted to the characteristics of the transported nuclear materials. The shielding material is fitted on the casks as an external layer of the containment shell in the radius gap with another external shell. For example the TN™ Resin VYAL-B (made by mixing a thermoset resin and two minerals fillers) is applied on casks that required maximum temperature. This resin can afford 160°C.

At the moment, the state of the art of the manufacturing process is to pour the mixture directly in the cask external shell under temperature control. This process need to be surveyed and checked very precisely for two reasons:

On one hand the pouring operation is not reversible. After pouring the resin there is no way to remove it (except destroying the external shell) or to control it directly (not possibility to cut a sample). On the other hand some resin (for instance TN™ Resin VYAL-B) need to stay in a temperature range during the polymerisation. As the casks have a great thermal inertia and are located in boil maker workshops (not temperature

Abstracts – Wednesday 06 October 2010 : continued

regulated) the pouring becomes a complex operation that needs to be performed by qualified experts when the weather is not so cold or warm.

TNI manufacturing and research and development departments are commonly developing alternative solutions to the pouring in-situ. One of those solutions is to manufacture the resin layer as blocks that are poured in separated moulds and installed after on the cask. This new process enables to control the resin after pouring and afford to discard any non-conformed blocks before their mounting in the cask. This process impact the design of the casks and leads to a new approach for the safety analysis. It is also used for the EPRTM reactor neutron protection provided by TN International on Olkiluoto 3 NPP.

ABSTRACT 173

Radiation Induced Structural Changes of (U)HMW Polyethylene with Regard to its Application for Radiation Shielding

KERSTIN VON DER EHE*, MATTHIAS JAUNICH, DIETMAR WOLFF, MARTIN BOEHNING, HARALD GOERING

The special properties of high molecular weight polyethylene (HMW-PE) and ultra high molecular weight polyethylene (UHMW-PE) result basically from their extreme chain lengths and their high degree of crystallinity. As high-performance polymers, they are used for a variety of applications. UHMW-PE in particular is often utilized for endoprothesis (due to its excellent slip and wear properties) and due to its high hydrogen content as a neutron moderator in casks for storage and transport of radioactive materials.

To prepare the material for instance for its use as a total joint replacement, it is exposed to radiation for several reasons, such as sterilization and crosslinking, leading to partial improvement of the mechanical properties (e.g. fracture toughness, crack propagation resistance, wear resistance) and better chemical stability.

To be applicable for long term radiation shielding purposes for instance over a period of 40 years, PE has to withstand any type of degradation affecting safety relevant aspects.

The scope of our investigation comprises an estimation of the radiation impact on the molecular and supra molecular structure of the two types of PE and to what extent these changes are detectable by thermo-analytical (TA) methods, such as Differential Scanning Calorimetry (DSC), Thermo Mechanical Analysis (TMA), Dynamic Mechanical Analysis (DMA) and Thermo Gravimetric Analysis (TGA). Additionally density and gas sorption measurements were carried out.

Due to the poor solubility of HMW-PE and UHMW-PE, some classical analytical techniques are not applicable. But TA-methods represent a feasible approach to detect structural and morphological features of these materials as well as changes caused by external influences, such as thermal treatment and/or irradiation. With the combination of the applied TA-techniques it is possible to distinguish between crosslinking and degradation.

ABSTRACT 136

Thermal Ageing of Vinylester Neutron Shielding Used in Transport/Storage Casks

FIDELE NIZEYIMANA*, V. BELLENGER, PASCALE ABADIE, HERVE ISSARD

A mineral filled vinylester composite used in radioactive materials transport/storage casks as a neutron shielding has been studied. The matrix is a highly hydrogenated novolac-type vinylester resin and two mineral fillers are zinc borate and aluminium tri hydrate. During the use of casks, neutrons are slowed down by hydrogen atoms contained in both resin and fillers, and then absorbed mainly by boron atoms contained in zinc borate. Aluminium tri hydrate is used for its fire resistance and self-extinction properties. Due to fillers water content and residual heat from radioactive materials, three types of ageing can be taken into account: thermal degradation, irradiation and hydrolysis. In this paper, only thermal ageing results are discussed.

The composite is cast under room temperature, and then cured at 160°C for 4 hours before ageing experiments. Gravimetric analysis data were used to evaluate weight loss of the composite after exposure to different 105 MPa of oxygen pressure. temperatures ranging from 140°C to 170°C, under 2 During the first 48 hours of ageing, a fast weight loss was observed due to low molecular mass molecules volatility. Then, over ageing time, the rate of weight loss decreased. Fourier transform infrared (FTIR) spectroscopy measurements were carried out in transmission mode to investigate C-H aliphatic bonds consumption and carbonyl functions build up. In addition, glass temperature (Tg) was monitored using differential scanning calorimetry (DSC) and dynamic mechanical thermal analysis (DMTA). To investigate the effect of thermal ageing on the composite mechanical properties, flexural modulus and ultimate stress were determined prior and after exposure.

ABSTRACT 70

Experimental Study of Heat Removal Ability and Lead Slump of Lead-Type Multi-Wall Cask

SATOSHI ASHIDA*, JUN OKADA, SHINTARO MIYAZAKI, KOJI KITAMURA, DONG HUI MA

Numbers of the lead-type multi-wall casks have been used so far. The lead-type cask might be economically advantageous, because it is not so influenced by the price fluctuation as the forged type cask.

At present, conservative design is adopted to the lead-type cask because the experimental data about heat removal ability and lead slump is insufficient.

The purpose of this study is to establish the practical design of the lead-type cask by experimental study.

To examine the heat removal ability of lead type multi-wall cask, a cylindrical model composed with three layers (carbon steel – lead – carbon steel) was used.

The inner surface of the cylinder model was heated with an electric heater and the temperatures of the surface and each layer of the model were measured.

To examine the lead slump, the same model used for heat removal ability test above was also used for the free drop test, and the displacement of the lead layer surface was measured.

The heat transfer characteristics and the amount of displacement

of lead layer were confirmed from the experimen-tal data above, the analytical study for these measurements will be executed.

These results will be reflected for more practical design of the lead-type multi-wall cask.

ABSTRACT 262

The Role of Metamic®-HT – Industry's First Nano-Particle Based Material - in Fuel Basket Design

K.P. SINGH, I. RAMPALL, T. G. HAYNES (PRESENTED BY WILLIAM WOODWARD)

A modern storage system or transport package for spent nuclear fuel requires a thermally efficient and structurally robust fuel basket. Development of thermally capable (high heat rejection capability) and a structurally competent fuel basket design that can meet the 10CFR71.73 free drop or non-mechanistic tipover requirements has been stymied for lack of a suitable basket material. Austenitic stainless steel, the staple material for fuel baskets, suffers from relatively poor thermal conductivity. Carbon steel, another material of choice for cost conscious users, is corrosion prone. Borated Aluminum, more common in Europe, has poor structural strength at typical operating temperatures in casks.

Metamic-HT (trademarked), a nano-particle-strengthened aluminum containing Boron Carbide powder, developed by Holtec International, removes the above limitations that have hampered the progress in cask design. This paper provides a summary of the three-year test program carried out to quantify the minimum guaranteed properties of Metamic-HT. The tests were carried out on coupons of Metamic-HT in three physical states:

i. Raw (as-extruded)
ii. Thermally conditioned to simulate 40 years of cask thermal environment
iii. Thermally conditioned and irradiated to simulate 40 years of fluence

The key thermo-mechanical properties of Metamic-HT (listed below) have been characterized by a testing program similar to the qualification regimen for an ASTM material.

i. Creep Strength
ii. Yield and Ultimate strengths
iii. Youngs Modulus
iv. Elongation
v. Area reduction
vi. Charpy impact strength and lateral expansion
vii. Fracture toughness
viii. Thermal conductivity, expansion coefficient, and heat capacity
ix. Emissivity

A large number of coupons (well over 1,000) were tested to perform statistical correlations on the data obtained on each property. The results of the test program were shared with the USNRC in Docket No. 71-9325, which provided the basis for the certification of the fuel basket in HI-STAR 180.

In this paper, the essential characteristics of Metamic-HT are presented along with its design embodiment for a transport cask fuel basket. A summary of typical structural response of the Metamic-HT basket under a 10CFR 71.72 accident event shows that the safety margins are greater than those in typical stainless steel baskets.

P5 - LIABILITY AND INSURANCE

2:00PM – 3:40PM – PANEL SESSION – MAIN HALL
CHAIR: DONNA GOERTZEN, CO-CHAIR: SERGE GORLIN

ABSTRACT 285

Nuclear and Third Party Liability Insurance for Nuclear Transport

MIKE PEACH

The aim of the session will be to discuss the special nature of nuclear liability in terms of insurance and in particular understanding the responsibilities of various stakeholders including despatchers, transporters and receivers of consignments and the various jurisdictions and states that may be part of the itinerary.

In general nuclear liability is channelled back to the operator so that all other insurance policies are subject to a general exclusion from nuclear risks.

The concept is set down in international conventions (Vienna, Paris and CSC) and then implemented in national law for signatory countries. The implementation in different states differs in detail, examples are direct or commercial channelling of liability and details of what may constitute nuclear materials

In terms of the development of the market, the prime vehicles for providing nuclear liability cover have been the national pools, but other players include mutuals and captive. The terms of reference of these will also vary from country to country.

The transport of nuclear materials therefore presents particular challenges and needs. The implications of mixed jurisdictions that may be encountered en route, the need for internationally valid certification of financial security and characterisation of the material in transit and dealing with countries that are not signatory to international agreements in this respect.

In dealing with possible incidents it will be important to distinguish between responsibility of third party liability and property damage to the consignment or transport itself, which may constitute a significant loss for its owner.

MARK RICHARDS

The presentation will focus on the work of the World Nuclear Association Task Force on Liability which has examined the implications of the current international liability regime for industry, including international transport operations. Among its findings, the Task Force has concluded that the lack of coherence between liability treaties and certain national policies, together with general lack of industry understanding, is creating an obstacle to investment in the field. The presentation will look at some of the measures required to redress this, including the need for education, as well as redoubled governmental efforts to provide a unified liability regime, and greater treaty adherence.

REGIS MAHIEU

The purpose of the presentation is to provide a carrier's perspective on the concerns and issues faced during the transport of nuclear and radioactive materials in relation to Nuclear Civil Liability and insurance. The presentation will examine how these issues are handled in practice, using real examples.

Abstracts – Wednesday 06 October 2010 : continued

ANTHONY WETHERALL & NATHALIE HORBACH

Through a scenario, the presentation highlights the common benefits and advantages for potential victims, industry and states arising from treaty relations between States on third party liability for nuclear, particularly in the context of a global regime on third party nuclear liability.

P8 - PUBLIC ACCEPTANCE

2:00PM – 3:40PM – PANEL SESSION – CONFERENCE ROOM 1
CHAIR: BERNARD MONOT, CO-CHAIR: LORNE GREEN

ABSTRACT 240

Communicating the International Transport by Sea of Nuclear Material

RUPERT WILCOX-BAKER*

With a strategy of proactive stakeholder communications and issues management, INS aims to match its world leading transport capability with a complementary approach to the delivery of its communications.

Undertaking the global sea transport of some of the most controversial and emotive hazardous materials requires a clear strategy and the application of tried and tested tactics across global political, media, government and industry stakeholders to contribute to the successful fulfilment of INS's customers' requirements.

INS's communications approach for transport operations is as follows:

- Openness – be open and transparent whenever possible, recognising commercial and security constraints
- Authority – timely communications with key stakeholders to establish relationships and authority
- Service – recognise the needs of customers, regulators, media, government officials and politicians
- Clarity – straightforward messages on safety, security, legitimacy of the business
- Context – explain security framework and restrictions on releasing information

As part of the INS operations project team from the start, the communications team draws on internal and external support to deliver programmes of overseas public acceptance missions to en route states; welcomes key stakeholders to its terminal in Barrow-in-Furness to visit its vessels and facilities; works in partnership with its customers to understand their requirements – all well in advance of any transport operations.

Integral to the transport operations, INS delivers comprehensive communications plans that encompass local, national and international stakeholders to contribute to the successful completion of the shipment.

ABSTRACT 241

Public Acceptability for International Sea Shipments of High Level Waste and MOX Fuel

GAVIN CARTER (PRESENTED BY ALISTAIR BROWN)

Over the past two decades, nuclear fuel cycle companies have successfully delivered on their business commitments despite heightened public scrutiny of recycling, waste management and the associated international transportation activities and campaigns by opponent groups.

One critical factor in this success has been the way in which fuel cycle companies have addressed public acceptance. The companies have taken responsibility for actively promoting public understanding of the benefits of the fuel cycle and the safety and physical protection regimes that are in place.

By putting the business in its proper context, and by being prepared to disseminate information in littoral countries, the companies have been able to provide reassurance and perspective. They have also maintained a flexible approach by constantly developing communications tools and messages that have demystified the industry and cast it as, in many ways, others that attract little or no public concern.

Through their public acceptance activities, the companies have reached out to different audiences in more than a dozen nations, including government officials, regulatory bodies, academics, associated businesses and journalists.

The companies recognize that this commitment to providing information must be on-going to ensure that the most positive perceptions of fuel cycle operations are maintained.

ABSTRACT 271

Transparency of Operations – Working with Stakeholder Groups

PAUL HARDING*, HENRY-JACQUES NEAU

Transport is very much in the public domain and there are frequent calls for transparency. Indeed some countries laws put requirements on operators to provide information.

Created by the French Law on Nuclear Safety & Transparency in June 2006, the High Committee for Transparency & Information on Nuclear Safety (HCTISN) has provided the public with an organisation in charge of guaranteeing openness on the part of the nuclear industry operators in relation to their operations.

It has equally provided operators with an additional vehicle for communication with the public and with a forum in which essential issues in terms of communication can be addressed and debated.

In June 2008 the HCTISN examined a transport of plutonium oxide between Sellafield and La Hague, performed earlier that year by International Nuclear Services and TN International.

The High Committee was concerned with ensuring information relating to the safety of the shipment, particularly to the safety of the ships used, could be made available to the public.

Whilst much information is already made available by operators in documentation and on web sites in their drive to be as open as possible, some data such as that relating to assessment of ship safety may be held by the national authorities of the flag state in charge of those assessments and is not systematically communicated to every nation in which a vessel may call. Because of the very nature of

Abstracts – Wednesday 06 October 2010 : continued

maritime transport, international agreements provide for mutual recognition of ship certifications between flag states and for Port State Inspections by the authorities of a port in which a vessel makes call.

Further constraints arise from the need to protect certain information, the release of which may be considered detrimental to the security of the shipment.

This paper describes how International Nuclear Services and TN International worked with the HCTISN to ensure information could be made available to provide adequate public reassurance on the safety of shipments, without undermining existing international maritime conventions or indeed security.

ABSTRACT 384

A Communication Tool-Kit to Combat Problems Shipping Radioactive Material

JIM STEWART

As part of the tools to combat problems shipping radioactive materials a tool-kit for communication has been developed. This paper sets out the contents of the tool-kit and how it is applied.

ABSTRACT 382

The Benefits of Simple Short and Clear Training for Targeted Audiences

KASTURI VARLEY*

Material of use in training of those involved in the transport of radioactive material have been developed. This material provides the basis for short training of specific audiences. In addition efforts have been made to provide presentation material which can be easily adapted to any language. The benefits include the ability to incorporate training in the transport of radioactive material into existing training courses and thus provide a basic level of knowledge to a widespread audience. This paper sets out the target audiences, gives details of the material and the presentation material developed as well as highlighting some successes.

P9 - LONG TERM STORAGE AND TRANSPORT - TECHNICAL ISSUES

2:00PM – 3:40PM – PANEL SESSION – CONFERENCE ROOM 2

CHAIR: STEVE BELLAMY, CO-CHAIR: TARA NEIDER

ABSTRACT 34

Code Cases of Basket Material for Spent Fuel Transport/Storage Packagings in the Japan Society of Mechanical Engineers

MAKOTO HIROSE*, TOSHIARI SAEGUSA, KATSUHIKO SHIGEMUNE

The Japan Society of Mechanical Engineers (JSME) established the Rules on Transport/Storage Packagings for Spent Fuel as a part of Codes for Construction of Spent Nuclear Fuel Storage Facilities in 2001, and revised in 2007. The revised Rules provides a material and design code for baskets made of aluminum alloy or borated aluminum alloy. Further, the Rule includes guidelines for the application of aluminum alloy and borated aluminum alloy as a new basket material for the spent fuel transport/storage packagings.

In accordance with the guidelines proposals of code cases for new basket materials including 4 types of aluminum alloys and one type of borated stainless steel were submitted to JSME in late 2007. In January 2008 the Subgroup on Spent Fuel Storage Facilitues witin the Subcommittee on Nuclear Power in the Power Generation Code Committee of the JSME establised the Working Group on Packaging Material Evaluation with experts from universities, research organizations, material manufactureres and utilities to entrust the assessment of these applications. The Working Group met 8 times in a half year to investigate and discuss intensely the applications, and reported to the Subgroup that applied materials were comply with the guidelines in July. For aluminum alloys each set of allowable stress has been set forth with consideration to time and temperature effects such as creep or overageing. The code cases were approved by Subgroup in the end of July, by the Subcommitte in August, and approved for public comments by the Code Committee in December 2008. With no public comment the code cases were finally approved by the Code Committee in March 2009, and published as followes.

Code Nos.	Materials	Descriptions
JSME S FA-CC-001	Borated Aluminum Alloy	1 % borated Type A-6061-T6 and –T651 aluminum
JSME S FA-CC-002	Aluminum Alloy	Type A-6061-T6 and –T651 aluminum
JSME S FA-CC-003	Aluminum Alloy	Type A-5083FH-O aluminum
JSME S FA-CC-004	Borated Stainles Steel Sheet	1 % borated Type 304 stainless steel
JSME S FA-CC-005	Blrated Aluminum Alloy	Up to 9 % B4C added Type A6N1 aluminum (ASME Code case N-673)

These materials will be used in basket designs for transport/storage packagings for the Mutsu Recycle Fuel Storage Facility, the first away-from-reactor interim spent fuel storage facility scheduled to be put into operation in 2012.

In the presentation outline of the materials and major discussions on them within the JSME Committees will be introduced.

ABSTRACT 348

How to Transport a Cask Which Has Been Loaded Then Stored for Several Decades?

PIERRE MALESYS*

One option for the storage of spent fuel which is not reprocessed is the use of dual purpose casks suitable for the transport and the storage. As well, waste or residues can be stored in dual purpose casks before an appropriate repository has been commissioned.

Some authorities license the storage facility with the conditions that the package design is approved according

Abstracts – Wednesday 06 October 2010 : continued

to the IAEA Transport Regulations (TS-R-1).

Whilst the storage facility needs to be licensed for a middle term, package design (transport) approvals are issued with a limited period of validity (typically 3 to 5 years).

One issue is then the "maintenance" of the design (transport) approval. What happens if after a certain period of time the approval cannot be renewed / prolonged (or re-issued if there was no need for a transport approval during the storage period), either because of a revision in the Transport Regulations or because of a new safety review? Is it possible to synchronize the expiration dates of the transport approval (short term) and of the storage facility license (middle term)?

Several options have been considered by the World Nuclear Transport Institute (WNTI) Waste Transport Industry Working Group:

- To issue package design (transport) approval with a validity of several decades for dual purpose casks, and to synchronize the expiration of the transport approval with the validity of the storage facility license;
- To stabilize the Transport Regulations for the dual purpose packages;
- To revise the Transport Regulations in order to allow dual purpose casks prepared for transport not later than a given date under a certain edition of the Regulations to continue in transport, whatever is the latest edition of the Regulations;
- To separate the storage license and the package design (transport) approval, and at least to avoid requiring maintenance of the transport approval along the life of the storage facility, and then to allow the final transport through a new package design approval or through the special arrangement procedure.

The paper will discuss these options in detail.

ABSTRACT 354

A Preliminary Look at Used Nuclear Fuel Transportation Options to a Repository Site in Canada

ULF STAHMER*

The used nuclear fuel produced by Canadian power reactors is currently stored in interim wet and dry storage facilities at the reactor sites where it is produced. Adaptive Phased Management (APM), Canada's chosen plan for the long term management of used nuclear fuel is built around containment and isolation of the used nuclear fuel in a centralized deep geological repository (DGR). Implementation of APM necessitates the transport of the used nuclear fuel from the seven interim storage locations to the repository.

This report re-examines the transportation options previously presented (reference here) and puts context to hypothetical DGR locations located in the 4 provinces that have benefitted from the nuclear energy cycle. These provinces are Saskatchewan, Ontario, Québec and New Brunswick. Since the eventual site of the DGR is still unknown, the 4 nuclear provinces have been arbitrarily divided up into 9 geographic zones. Transport from existing sites to each zone is analyzed. The investigation examines the following:

- Transport viability of the fuel from the interim storage sites by road, rail and water, or combinations thereof;
- Route considerations, on a high level, from each of the current interim reactor storage sites to the centralized used nuclear fuel deep geological repository;
- Existing transport infrastructure at the interim sites, along potential routes and in the 9 geographical regions; and
- Potential additions or improvements to transport infrastructure required to enable transport of used nuclear fuel to the DGR.

The paper concludes that the viability of transport depends very much on the location of the nuclear licensed site in question and that of the disposal facility.

ABSTRACT 157

Accelerated Corrosion Testing of Aluminum/Boron Carbide Metal Matrix Composite in Simulated PWR Spent Fuel Pool Solution

DAISUKE NAGASAWA*, HIDEKI ISHII, KAZUTO SANADA, VALENTIN ROHR, HERVE ISSARD

MAXUS Al/B4C MMC plate, produced using a powder metallurgy process and having a sandwich structure with Al/B4C MMC core and thin Aluminum skins, was developed by Nippon Light Metal as a neutron absorber for basket plates in dry storage casks. The material's relatively low weight, high thermal conductivity, high mechanical strength, and other properties make it especially attractive for this application. However, because the casks must be im-mersed in the pool during loading of the spent fuel, it is necessary to confirm the corrosion resistance of MAXUS in the pool environment.

The same accelerated corrosion testing as neutron absorbers for spent fuel storage racks was performed to inves-tigate the corrosion resistance of MAXUS plate in the pool environment. Samples for corrosion testing were cut from plates having a 15 wt%-B4C core. Anodized and non-anodized samples were tested in a simulated PWR spent fuel pool solution at 363 K over a 6-month period in accordance with ASTM G31-72.

The Aluminum surface cladding and its anodic film were slightly hydrated after accelerated corrosion testing; how-ever, the surface of both anodized and non-anodized samples had no blisters or pitting corrosion. For the anodized samples, the thickness of the anodic films on both skin and core showed almost no change with testing. For the non-anodized samples, the side of specimens, at which Al/B4C MMC core was exposed to the simulated pool solu-tion, had no pitting corrosion. These results demonstrate that MAXUS has good corrosion resistance in the simu-lated PWR spent fuel pool solution.

P11 - MANAGEMENT CONTROLS

2:00PM – 3:40PM – PANEL SESSION
– CONFERENCE ROOM 3
CHAIR: MICHEL HARTENSTEIN, CO-CHAIR: TBC

ABSTRACT 234

Emergency Response without Borders

ALAN BACON*

The IAEA Safety Guide TS-G-1.2 (ST3) provides guidance to the many organisations, including consigners and transporters, on the subject of planning and preparing for Emergency Response to transport accidents involving radioactive material.

The guide details responsibilities for both the consignor and the carrier for preparedness, assistance, provision of information and the alerting of appropriate authorities of an event.

International Nuclear Services sees its Emergency Response function as part of a fully integrated emergency management system. The six building blocks of this approach are Mitigation, Prevention, Preparedness, Response, Recovery and Review.

The transport activities of International Nuclear Services are carried out across the globe and this in itself introduces challenges to the Emergency Response function which are not usually encountered by the emergency agencies that operate only within the borders of their host nation.

Some of these challenges include time differences, language barriers, country specific special laws, coastal state concerns and expectations, training and exercising, contracting services, response times and the demonstration of this planning and preparedness to both regulators and stakeholders.

The object of this paper will be to review and discuss these challenges and detail how the solutions meet the responsibilities laid out in the guide (ST3).

One of the most crucial parts of the International Nuclear Services' Emergency Response capability is to have personnel and equipment ready to deploy within and outside of the UK 24hrs a day, 365 days per year to deal with the unlikely occurrence of an event during its transport activities. During August 2009 International Nuclear Services undertook a live exercise to deploy two teams and associated equipment overseas utilising the full range of resources and services that were both internal and external to the organisation. This paper will demonstrate the success of the INS approach.

ABSTRACT 3

Transport of Radiopharmaceuticals and Labelled Compounds in Cuba

ZAYDA HAYDEE, AMADOR BALBONA*, SAUL PEREZ PIJUAN, MIRTA BARBARA, TORRES BERDEGUEZ, FERNANDO ENRIQUE, AYRA PARDO

The Centre of Isotopes (CENTIS) is the main consignor and carrier of radioactive material in Cuba. The purpose of this paper is to describe the Radiation Protection Program (RPP) implemented inside a Quality Management System, to achieve and maintain an optimized standard of protection in the accomplishment of these functions. All those areas involving radiation exposures are considered (e.g. design of type A packages, packing, loading, handling, in-transit storage, road transport and inspection and maintenance of packaging). The quality assurance requirements for packaging components were established using a grading process. A material to absorb twice the volume of the liquid contents is tested and its water absorptivity, grammage and capillary rise were estimated. Categories and transport indexes for 56 packages of radiopharmaceuticals incorporating radioiodines, 32P, 188Re and 90Y and technetium generators, are determined. Tests for demonstrating compliance with requirements for type A packages with liquid and solid radioactive content and for air transport are performed and documented. A numeric code for each package by consignee is registered and controlled in each step of the process and as a guarantee of its traceability. Safety and security of radioactive materials during storage in transit and transport are supervised. Training of workers trough periodic curses and emergency exercises is implementing. Individual Licensing of this staff is conducted by CENTIS and presented to the Cuban Regulatory Authority. The effective annual doses distributions are reported since 1996 to 2008. Occupational exposure is acceptably low and less than 6mSv, which is the dose constrain adopted. It has not been reported any incident in about two thousand road shipments carried out. CENTISâ€™ RPP has been under review, detailed appraisals and audits. The Certification of the management system by ISO 9001:2008 has been identified as a goal and a way for the continuous improvement.

ABSTRACT 161

The Role of the Dangerous Goods Safety Adviser and Improving Compliance with the Radioactive Material Road Transport Regulations Amongst Users in the GB Industrial Sector

SIMON JAKES*

Over the last few years the Department for Transport (DfT) has been enforcing the radioactive material road transport regulations with a new found vigour and has now conducted many inspections across the country. Unfortunately these inspections have served to highlight the often poor levels of compliance amongst small users in the UK and it appears one requirement, more than any other, is unlikely to be met: the requirement to appoint a DGSA. In the DfT's Transport of Radioactive Material Newsletter from November 2009 they state that 'most organisations don't have one'. In light of this and the other findings of the DfT's inspections, the aims of this paper are twofold:

The first is to discuss reasons why there are so few DGSA appointments by small users, before the author clarifies where the requirement comes from, what the role of the DGSA is and which organisations should appoint one.

The second is to highlight the fact that the level of understanding of the requirements of the radioactive material road transport regulations amongst small users is often poor. The author, as a DGSA working in industry, suggests reasons why this is the case, before moving on to provide his view on the current state of compliance amongst small users and suggesting how common compliance problems can be addressed by small users and how the DfT may help.

In summary, compliance currently varies considerably between small users in industry. Whilst some are fully aware of recent developments in ADR, such as security in transport and changes to documentation, vehicle and personal protective equipment requirements, some are still failing to ensure compliance with the basic requirements of TS-R-1. The author believes the overall level of compliance amongst small users will not improve significantly until the number of DGSA appointments increases significantly too.

ABSTRACT 273

10 CFR PART 71 Quality Assurance and Inspection Experience

EARL LOVE (PRESENTED BY ROBERT TEMPS)

The Nuclear Regulatory Commission's (NRC's) regulations impose quality assurance (QA) program requirements on entities that use or design Type B and fissile material packages for the

Abstracts – Wednesday 06 October 2010 : continued

transportation of radioactive materials. Prior to commencing activities subject to the QA requirements, a QA program description must be submitted to the NRC for review and approval. While it is incumbent on each Quality Assurance Program Approval and Certificate of Compliance (CoC) holder to ensure proper implementation of their NRC-approved QA program description, NRC conducts independent periodic safety and compliance inspections of CoC holders, and companies that have an NRC approval for their QA program for Title 10 of the Code of Federal Regulations (CFR) Part 71 activities. Specifically, it is the responsibility of SFST to inspect Part 71 designers and fabricators to assess the adequacy of program implementation. Inspections may be reactive; i.e., they may occur in response to a specific event, or, as is normally the case, they may be conducted at periodic intervals. NRC has developed a program and procedures for these reactive and planned inspections and is responsible for their implementation. This paper discusses SFST inspection experience of NRC-licensed activities involving the transport, design or fabrication of radioactive material packagings.

ABSTRACT 392

Applying the Good Practices Identified in IAEA TranSAS Missions

JIM STEWART*

Over several TranSAS missions there have been a number of good practices identified. This paper sets them out and examines how they might be used by competent authorities to improve safety and efficiency.

ABSTRACT 65

Canadian Emergency Response Requirements and Cameco's Experience

JOHN ZAIDAN*, MARC-ANDRE CHARETTE

Cameco Corporation based out of Saskatoon is a world leader in the mining of uranium and it's processing. Cameco transports and receives front end nuclear fuel cycle materials from many locations around the world. Within Canada there exists a regulatory requirement for Cameco to provide assistance in the event of a transport incident involving not only the products shipped by Cameco but also product destined for Cameco.

This requirement is part of the Federal government of Canada's Transportation of Dangerous Goods Act and Regulation and is referred to as an Emergency Response Assistance Plan (or ERAP). The assistance provided may range in nature from information given over the telephone to the immediate deployment of a trained team of emergency responders. This paper will outline the regulatory requirements that exist in Canada and the efforts that have been undertaken by Cameco to provide assistance to public response agencies. Examples of actual transport incidents and the lessons learned are presented in the paper.

Given the vast and remote geography of Canada, coupled with the possibility of extreme weather conditions, Cameco faces a variety of challenges in responding to transport incidents. As such Cameco depends on third party contractors to support efforts in reaching and assisting with the initial response and subsequent clean up efforts. The training and management of this network of for-hire emergency responders is essential to providing timely, safe and vital emergency response. Additionally, and since 2003, Cameco has conducted an outreach program to provide training to public first response agencies and others along the main transport corridors in an effort to familiarize them with Cameco's products and what support to expect from Cameco. These efforts have proven to be beneficial and critical in several incidents.

Abstracts – Thursday 07 October 2010

T31 - REGULATIONS AND GUIDANCE

9:00AM – 10:40AM – TECHNICAL SESSION – MAIN HALL
CHAIR: FRANK NITSCHE, CO-CHAIR: GEORGE SALLIT

ABSTRACT 397

Clear Regulations

JIM STEWART*

The need for clear and unambiguous regulations is essential in order to encourage their application. This paper discusses the reason for regulation and explores the difference between regulation and guidance, providing suggestions for future developments.

ABSTRACTS 118/380

The Environmental Conditions Experienced by Packages during Routine Transport

SARAH FOURGEAUD*, KARIM BEN OUAGHREM, GILLES SERT, IGOR LE BARS, JIM STEWART

Packages carrying radioactive material experience a range of environmental conditions during routine transport including temperatures, pressures and shocks. This paper examines the current requirements in the IAEA Regulations for the safe transport of radioactive material (TS-R-1) and the associated guidance (TS-G-1.1), and outlines a research project to confirm the validity of the current requirements and improve the existing guidance.

As for the mechanical loadings the packages are subjected to, the advisory material TS-G-1.1 indicates that, due to the differences in transport infrastructures and practices, the recommended acceleration factors, which represent the package inertial effects, could differ from one country to the other and that the package designer should confirm the acceptability of those factors.

In this context, IRSN performed a bibliographical study relative to accelerations measured on packages or conveyances during transport. This study shows some variations with the acceleration factors mentioned in TS-G-1.1. It also highlights areas where data are missing. In these areas, further measurement campaigns should be performed.

The international project under the auspices of IAEA will provide opportunities for collecting a large set of results and facilitating the needed international consultation.

ABSTRACT 129

Onsite Transport Regulations: How to Adapt International Regulations?

LAURENT HANSEL*, YVES CHANZY

Onsite transports of radioactive materials are usually performed using specific rules approved site by site by the competent authority. They are necessary to transfer radioactive materials from one building to another during the different phases of a transformation process.

The French competent authorities have asked the largest nuclear operators in France to set up a working group in charge of writing a draft regulation to be approved by the regulators before coming in force on the different French nuclear sites: RTIR (Regulations for onsite transportation of radioactive materials).

The transport being performed onsite, it is easier to characterize the different transport conditions according to the usual categories:

Routines conditions - Transports are no longer than a few kilometres. It is therefore easier for the driver to acquire a good knowledge of the path. This enables also to easily guaranty, except for potential very sudden storm or hail storm, the weather and traffic conditions during the transport. It is also possible to identify all possible hazards, including co-activity and traffic along the path.

As concerns radiation protection and contamination, the exposure for public (visitors) is limited in time and minimized by the fact that people are well aware of the risks when visiting a nuclear site.

Thus, the operator can precisely define the conditions of performance of the transport. This leads to introduce in the regulations the possibility to exempt the package designs from non relevant testing, or to limit the number of prescriptions for the transport.

Incidental and accidental conditions – On each nuclear site, health physics specialists are available with a good knowledge of the specific hazards inherent to the materials transported. Thus, in case of incident or accident, those specialists are available to rapidly make a diagnosis and take all relevant dispositions to limit the possible consequences.

Thus specific rules can be established for the design of the packaging as well as for operational conditions of transport. Those rules may be less stringent than for transport outside the sites, but providing the same safety level.

The paper will describe the content of the draft regulation prepared by the nuclear operators.

ABSTRACT 371

TCSC 1086: Good Practice Guide to Drop Testing of Type B Transport Packages

CHI-FUNG TSO*, BILL SIEVWRIGHT

The Transport Container Standardisation Committee (TCSC) is a UK nuclear industry club whose main function is to maintain and develop codes of practice relating to radioactive materials transport. It's role is 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.

Drop tests, calculations and reasoned arguments could be used on their own or in combination to demonstrate that a transport package meets the relevant impact requirements of the Transport Regulations.

Drop tests are often time consuming and are expensive exercises, and therefore it is important that not only are they carried out correctly, but that they are performed efficiently.

TCSC 1086 Good Practice Guide to Drop Testing of Type B Transport Packages has been developed by TCSC to provide practical guidance on good practice in the planning, executing and analysing of drop tests. It is a companion volume to TCSC 1087 Good Practice Guide - The Application of Finite Element Analysis to Demonstrate Impact Performance of Transport Package Designs which discusses good practice in using finite element calculation techniques to

Abstracts – Thursday 07 October 2010 : continued

demonstrate compliance with impact performance requirements. Its contents include the planning of a test programme, design of test specimen, design of a test matrix including choice of drop orientation and selection of test sequence, instrumentation, photography, metrology, leak testing, developing a test specification, and requirements for test reporting.

The Advisory Material for the IAEA Regulations for the Safe transport of Radioactive Material contains useful advisory and guidance material to satisfying the requirements of the Transport Regulations. TCSC 1086 aims to supplement the Advisory Material and covers specific areas in more detail.

Although the context of this guide is the application for approval from the UK Competent Authority, it is also relevant to applications for approvals from other Competent Authorities.

This paper presents a summary of TCSC 1086.

ABSTRACT 368

The Facilitation Of Criticality Safety Assessments For Fuel Assemblies

MICHEL DOUCET (PRESENTED BY SAM DARBY)

Packages for the international shipment of fissile nuclear fuel cycle materials require multilateral approval in respect of criticality safety. This involves not only certification and validation by the Competent Authority of the country of origin but also approval by each country through, or into which, the consignment is to be transported. Although the regulations in TS-R-1 which govern criticality safety are unambiguous, the interpretation of these regulations and the assumptions which form the basis of the assessment may differ significantly between the applicant and the various Competent Authorities. As a result, gaining a full set of approvals can sometimes be a lengthy and expensive business.

The World Nuclear Transport Institute, WNTI, believes that the approval process can be made more efficient. An improved approach to criticality safety case preparation would be to use consistent methodologies and more realistic assumptions based on reliable data. This could lead to significant reductions in costs and timescales for both applicants and regulators.

WNTI has established a working group of criticality experts from its member companies to explore ways to improve the preparation of criticality safety cases. To begin with the working group has concentrated on fuel cycle materials, with the aim of identifying ways to facilitate consistency, reduce the effort and shorten the time involved in obtaining approvals.

The main features of the WNTI study have been presented elsewhere [1, 2]. The initial focus has been on new and spent fuel assemblies because often criticality safety cases for these materials present technical challenges. Fuel lattice expansion and fuel pin cladding failure under impact accident conditions have been studied and the principal issues and conclusions reported in [1,2]. Recently other criticality methodologies for important topics relevant to the criticality safety case for new and spent fuel elements, namely enrichment mapping, water ingress, burn-up credit, the deformation of internal components of the package and safety margins have been reviewed. The findings are described below.

T30 - EMERGENCY RESPONSE (SESSION 1)

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 1
CHAIR: BETTY BONNARDEL-AZZARELLI,
CO-CHAIR: GILLES SERT

ABSTRACT 56

Transport Emergency Preparedness – Lessons Learned from WNTI

MARC FLYNN*

Regulations only have a beneficial impact on safety when they are fully and properly implemented at the operating level. During recent years, there have been increasing signs of improved consistency in interpretation of the requirements. However, a single interpretation has the potential to remove any flexibility and can be viewed by some as tightening the requirements and by others as relaxing the requirements. The paper analyses this phenomenon, using emergency response requirements as a case study.

The IAEA Regulations for the Safe Transport of Radioactive Material require National/International Organisations to develop emergency response capability. IAEA Safety Guide TS-G-1.2, “Planning and preparing for emergency response to transport accidents involving radioactive materials” provides guidance on emergency planning and preparedness for dealing effectively with transport accidents involving radioactive material. Interpretation and implementation of this requirement vary, where emergency response systems in some countries are state managed, and others require consignors to develop their own systems. Differing interpretations of just what is required may lead to varying capabilities and efficiencies amongst the national crisis management organisations. It also has a direct and potentially costly impact on organisations consigning radioactive material. It is important that industry shares its knowledge, and collaborates in the development of consolidated positions.

The World Nuclear Transport Institute (WNTI) formed an industry working group to share experiences amongst its members on transport emergency response preparedness. Over the last two years, the working group has prepared questionnaires and held workshops in France, UK & Russia.

This paper gives an insight into how the regulations can be interpreted differently and whether this level of flexibility may help or hinder the global efficiency of crisis management organisations. In particular, it addresses the consequences for the transport operational stakeholders, with respect to emergency preparedness.

Finally, it summarises the lessons learned so far from the WNTI Emergency Response Industry Working Group; discussing emergency incident communication processes and tools to encourage best practice.

Abstracts – Thursday 07 October 2010 : continued

ABSTRACT 314

RADSAFE: Meeting the Industry Needs for Transport Emergency Arrangements

TERENCE KELLY*, JONATHAN HARRISON, GARETH DAVIES, ANTHONY WETHERALL

Emergency arrangements are required for a number of reasons, which include:

- Legal
- Commercial
- Moral

This lead to a number of transport plans being developed within Great Britain such as

- IFTFEP, Irradiated Fuel Transport Flask Emergency Plan, England and Wales
- SNIFTFEP, Scottish Nuclear Irradiated Fuel Transport Flask Emergency Plan
- NIREP, Nuclear Industries Road/Rail Emergency Plan and the voluntary long stop emergency response
- NAIR

All of the plans have been based on the IAEA Transport regulations which have been enacted through British legislation.

RADSAFE became operational on 1st August 1999 as a single radioactive material transport emergency response scheme for the British nuclear industry, combining the best aspects of the previous emergency response plans. The current RADSAFE members are:

- British Energy Safeguard International
- Magnox North Westinghouse
- Magnox South Imperial College
- GE Healthcare Reactor Sites Restoration Ltd
- United Kingdom Atomic Energy Authority Urenco
- Low Level Waste Repository Dounreay Site Restoration Ltd
- GE Healthcare Sellafield Sites
- MOD

The implementation of a single RADSAFE scheme has lead to many benefits for its members which include:

- Cost savings
- Standardisation of response
- Reduced confusion about what plan to activate
- Raising of awareness throughout the industry and emergency services
- Standardised training for RADSAFE members and the emergency services
- A focussed working group responsible for a budget, driving development of the scheme

RADSAFE is now a standalone company which is self regulating and is responsible to its owners and members. The Department for Transport, Radioactive Materials Transport Division is supportive of the work that RADSAFE has undertaken to deliver a consistent approach transport emergency arrangements.

This paper describes the development of RADSAFE and the learning that has taken place over the years which can be used as a template anywhere.

ABSTRACT 281

DSTL RADSAFE Exercise

BRIAN CORBETT*, WILLIAM BLANCHARD

RADSAFE is a UK mutual support company which provides an emergency response in the event of a road/rail transport accident involving radioactive materials belonging to a company member. DSTL (Defence Science and Technology Laboratory) forms part of the emergency response cover on behalf of the Ministry of Defence membership.

As part of its obligations as a RADSAFE member, DSTL undertook an exercise on its Porton Down Range in Wiltshire in December 2009. The aim was to test its RADSAFE 'Level 2' response to a road traffic accident involving radioactive material. This paper describes the exercise, including how it was organised and implemented.

Good communications and decision-making are of vital importance in emergency situations. An essential part of the exercise was the interaction of DSTL's responders with the emergency services, particularly the Fire and Rescue Service. Information exchange with the owner of the package involved was also a key element. Real radioactive sources were used to inject some realism into the scenario.

The exercise was a successful demonstration of DSTL's ability to respond to a transport accident. Valuable lessons were learned, in terms of both exercise organisation and emergency response.

ABSTRACT 166

Technical Basis for Transport of Radioactive Materials Emergency Planning

SANDRO TRIVELLONI*, LUCIANO BOLOGNA, GIORGIO PALMIERI, ANTONIO SANTILLI, PAOLO ZEPPA

The transport of radioactive materials is an activity strictly linked with their use in nuclear installations, industry, medicine, agriculture and research.

Emergency preparedness in case of an accident is one of the main aspects related to the protection of workers, people and of environment from the risks arising from ionizing radiations during transport. The operational organization of emergency response in Italy relies on government's local representative, the prefect. On the basis of national legislation (Legislative Decree n. 230/1995 and Governmental Decree of the 10th February 2006) the prefect shall adopt a local plan for emergency response prepared by a local advisory committee. This advisory committee shall take into account the accident scenarios and the evaluation of the radiological consequences contained in a Technical Report, applicable to the whole national territory, issued by ISPRA (National Institute for Environmental Protection and Research), which in Italy, among other duties in the field of environmental protection, has the role of Regulatory Body for nuclear activities and Competent Authority for transport of radioactive materials.

The paper shows the contents of the Technical Report. The reference accident scenarios for emergency planning depends on different factors: mode of transport, nature of accidents, type of materials, shipment and packages, etc. The choice of the reference accident scenarios is based on conservative assumptions and on the statistical analysis of radioactive material transports data available from ISPRA database. The information derived from the data elaboration were used to define the source term associated to the

Abstracts – Thursday 07 October 2010 : continued

accident scenarios. The results of the evaluations in terms of release of radioactive material and radiological consequences allowed to define the kind of protective measures and actions to be envisaged in the emergency planning. The analysis were performed for all modes of transport and considering both radioactive and fissile materials.

In case of transport of irradiated fuel a specific technical report is prepared by the authorized carrier for each transport campaign and evaluated by ISPRA. The paper presents also the key elements of national experience in the preparation of these reports and associated emergency planning.

T29 - PACKAGE DESIGN AND STRATEGIES (SESSION 1)

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 2
CHAIR: MALCOLM MILLER, CO-CHAIR: HEINZ GEISER

ABSTRACT 105

Considerations in Developing a New Fissile Transport Package

TIM GLEED-OWEN*

This paper identifies and illustrates the typical activities and skills involved in the design and development of a new package. The activities described are modelled on those undertaken by Rolls-Royce in designing, developing and licensing the NMTSP package for carrying fresh fuel. It illustrates the wide range of skills required and the need for a flexible approach in deriving the final design. Topics considered are:

- Details of the payload to be carried, which includes assessments of containment boundary, fragility, thermal durability and means of criticality suppression;
- Space and weight constraints for the final package design;
- User requirements, eg stacking, ease of use, maintainability;
- Lifetime and whether used for transport only or storage and transport;
- Permeation and humidity control;
- Material choices, future-proofing, and the trade off between initial and through-life costs;
- Transport modes and the effects on design;
- Design ambient temperature range;
- Pressurisation;
- Testing for material characterisation;
- Structural testing on design features;
- Thermal testing of barrier materials and sections;
- Scoping calculations for impact;
- Lifting and tie-down features;
- Lid joint development, including bolting sizing;
- Lid bolt testing;
- Adverse material property combinations;
- Detailed impact analysis and predictions for drop testing;
- Criticality modelling and confinement boundary for normal and accident conditions;
- Modifications through manufacture;
- Test programme, including cumulative damage for normal and accident conditions;
- Development of drop target;
- Drop and stacking test results;
- Correlation and validation between test and impact analyses;
- Ancillary equipment, eg lifting and transport;
- Licensing.

This list is not exhaustive, and not every step will be required for all package designs, but the intent is to illustrate a typical process.

ABSTRACT 417

Multiple Barriers: Application to Package Design for Used Fuel Elements

STEPHANE BRUT *

Packages for the transport of radioactive material have to comply with national and / or international regulations. These regulations are widely based on the requirements set forth by the International Atomic Energy Agency (IAEA) in the "Regulations for the Safe Transport of Radioactive Material". In this framework, packages to transport used fuel assemblies have to meet the requirements for packages containing fissile material. The applicant for a package design approval shall demonstrate that the package remains sub-critical in all conditions of transport.

According to the IAEA Regulations, the sub-criticality of a package may be demonstrated assuming water exclusion from the containment system, if and only if the design is based on a multiple high standard water barriers. It is widely accepted by Competent Authorities that a double watertight barrier design is enough to comply with this requirement. Application of this requirement to used fuel package may result in different types of design.

In most cases, used fuel packagings are made of a thick forged vessel (a shell and a welded bottom) made of steel or cast iron, to lower the radiation level around the package. Thus, an application of the double watertight barriers is to design a double lid system. Another way of application is to design a double vessel cask.

The paper will describe our experience with the implementation of the multiple barriers requirement, and compare pros and cons of each type of design: either a double vessels design, or a double lids design. Examples of package designs with such features will be shown, as well as the approvals which were granted in various countries.

ABSTRACT 134

Description of Fuel Integrity Project Methodology Principles

MAURICE DALLONGEVILLE*, ARAVINDA ZEACHANDIRIN, PETER PURCELL, ANTHONY CORY

TN International and International Nuclear Services (INS) have started the Fuel Integrity Project (FIP) in early 2000s, which goal is the development of a methodology to evaluate as a safety requirement the nature and the extent of fuel assemblies (FA) damage during accident drops of a packaging.

From TN International previous knowledge acquired from fresh FA behaviour during drop tests, a mechanical tests programme including testing on fresh and used fuel rod samples has been planned by both companies and executed by INS. Tests results analysis has led to the elaboration of FIP methodology by TN International.

Considerable experience on fuel was collected from the tests

Abstracts – Thursday 07 October 2010 : continued

programme and the main mechanical phenomena arising from a drop have been clearly identified and quantified. As a result, the FIP methodology, structured in flow charts, gives guidelines to study the effects of a lateral or axial drop of a packaging loaded with fresh or used FA of PWR or BWR types.

The flow charts of methods have the same philosophy: several pessimistic mechanical evaluations based on direct calculations or dimensionless comparisons with appropriate reference tests allow determining the damage of FA, gradually increasing with their acceleration. First, elastic models distinguish the null or slight damage cases; then, plastic models allow ruling out cases with extreme FA damage that lead to unacceptable criticality hypotheses; finally, other plastic models quantify the extent of fuel rods deformations in moderate FA damage cases.

FIP methodology application to a given case leads to the following output, used as criticality hypotheses for the safety analysis: existence or not of fuel rods rupture, their number, their location, and the associated amount of released fuel material, extent of fuel rods array deformation and sliding.

All the knowledge arising from the FIP is synthesized in the Technical Guide, which presents extensively the methodology and builds up all background experimental data.

The application of the methodology is currently valid for fresh and used FA provided that brittle fracture risks are excluded, as FIP tests results on used fuel allow taking into account only partially dynamic loading and irradiation effects on fuel rods.

ABSTRACT 369

Transportation Implications of a Closed Fuel Cycle

RUTH WEINER*, KEN SORENSON, MATTHEW DENNIS, SAMUEL BAYS, MILES GREINER

Transportation for each step of a closed fuel cycle is analyzed in consideration of the availability of appropriate transportation infrastructure. The United States has both experience and certified casks for transportation that may be required by this cycle, except for the transport of fresh and used MOX fuel and fresh and used Advanced Burner Reactor (ABR) fuel. Packaging that had been used for other fuel with somewhat similar characteristics may be appropriate for these fuels, but would be inefficient. Therefore, the required neutron and gamma shielding, heat dissipation, and criticality were calculated for MOX and ABR fresh and spent fuel. Criticality would not be an issue, but the packaging design would need to balance neutron shielding and regulatory heat dissipation requirements.

ABSTRACT 162

Current Practise and Experience of Shipping Bulk Powders and How this is Relevant to the Transport of Uranium Ore Concentrates

MARC-ANDRE CHARETTE*, AL STRATEMEYER, GUY KARRER

Natural uranium ore concentrates (UOC) have been shipped safely and successfully in open head steel drums for over 50 years via a combination of road, rail and sea. These drums are appropriate and met the regulatory requirements for the packaging and transport of natural uranium ore concentrates. However, improved packaging options have not been seriously evaluated to replace this technology in recent times. The UOC transport industry believe it to be timely, relevant and aligned with the concept of product stewardship to bench mark this current practices against current world best practices for transporting bulk high density industrial based powders.

The World Nuclear Transport Institute (WNTI) has funded a paper analysing the various means of transporting bulk powders. The study included a review of the different types of packaging used for the transport of bulk powders, how these are handled, used, tooling and handling equipment requirements and experience with these packages.

This paper will present the result of the study and how this compares with the transport of Uranium Ore Concentrate.

T32 - SEAL BEHAVIOUR

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 3

CHAIR: BILL SIEVWRIGHT,

CO-CHAIR: HANS-PETER WINKLER

ABSTRACT 25

Influence of Mechanical Vibration in Transport on Leak-Tightness of Metal Gasket in Transport/Storage Cask for Spent Nuclear Fuel

TOSHIARI SAEGUSA*, KOJI SHIRAI, HIROFUMI TAKEDA, MASUMI WATARU, KOSUKE NAMBA

Transport casks of spent fuel will receive mechanical vibration in transport. It has been known that the leak-tightness of metal gasket is influenced by large external load or displacement. Quantitative influence of such vibration has not been known, but is crucial information particularly if the cask is stored as it is after the transport. It is noted here that more stringent leak-tightness is required for storage than transport.

Mechanical vibration in sea transport of spent fuel shipping cask has been measured and analyzed to result in possible cyclic displacement of Â±0.02mm to the metal gasket. In order to obtain a relationship between amount of lateral sliding of the lid and the leak rate, a 1/10-scale model of a lid structure of metal cask with a metal gasket of double O-ring type was manufactured. The gasket had a diameter of 10 mm and was coated with aluminum sheet.

Static one-directional loading experiments showed that no leakage was observed if the lateral displacement was less than 0.1mm. If the displacement increased up to 3mm, the leak rate increased up to 10E-6Paâ€¢m3/sec, but recovered to 10E-8Paâ€¢m3/sec in 72 hours.

Cyclic loading experiments showed the leak rate did not increase permanently if the lateral displacement was within Â±0.02mm. The leak rate increased permanently, if the displacement increased to more than Â±0.035mm.

Dynamic one-directional loading experiments showed that the leak rate after the maximum displacement coincided with that at the same displacement by the static one-directional loading experiments.

Abstracts – Thursday 07 October 2010 : continued

Namely, the leak rate did not depend on the loading rate nor displacement rate.

These results indicate that the mechanical vibration in transport would influence the leak-tightness of the metal gasket required for storage if the amplitude of the vibration exceeded a threshold value.

This study was conducted in a contract from NISA, METI of the Japanese government.

ABSTRACT 41

The Influence of Thermal Expansion on Package Tightness during Fire Test

FRANK KOCH*, JENS STERTHAUS, CLAUS BLETZER

Packages for the transport of radioactive materials are subjected to a fire test of thirty minutes at 800°C according to TS-R-1. As a result of the fire test, significant temperature gradients usually occur within the package. This is supported by thick walled designs of those packages, in particular. Temperature gradients lead to different thermal expansion, which results in displacements as well as stresses. This has an impact on the package wall but also on the leak tightness because of different influences of thermal expansion on the package wall and the lid.

The paper provides approaches based on analyses. Starting with well known analytical correlations for thin and thick walled pipes, a finite element model will be developed to analyse the problem on a numerical basis also. After the verification of the numerical model with respect to the results of the analytical approaches, the finite element model will be adjusted to be nearer to real package designs. In particular, the link to the lid system and the evaluation of leak tightness will be made. Finally, design aspects will be included in the finite element model. Results of the analyses are presented and conclusions are drawn.

As a result, shock absorber design and the design of the lid system are important aspects to improve the safety of the package with respect to leak tightness. The effects of thermal expansion during fire test can be analysed numerically. Analytical approaches are suitable for basic estimations and to support first design steps. Experimental verification is a difficult task because the event will occur during the fire test and the leak tightness could be rebuilt after the fire test is done and the measurements take place. Nevertheless, the opening of the package can last a significant time. Therefore, analyses are important to explore this effect and to demonstrate safety.

ABSTRACT 108

Evaluation of Sealing Performance of Metal Cask Subjected to Vertical and Horizontal im-pact Load due to Aircraft Engine Crash

KOJI SHIRAI*, TOSHIARI SAEGUSA, KOSUKE NAMBA

In Japan, the first Interim Storage Facility of spent nuclear fuel away from reactor site is being planned to start its commer-cial operation around 2012 in use of dual-purpose metal cask. It is important to develop the knowledge for the inherent security of metal casks under extreme mechanical-impact conditions, especially for increasing interest since the terrorist attacks from 11th September 2001.

To investigate the integrity of the lid structure of the metal cask during the extreme impact loads due to aircraft engine crash, two impact tests by aircraft engine missile onto the metal cask without impact limiters were carried out considering both, a ver-tical impact onto the lid structure and a horizontal impact hitting the cask. In the test, simplified deformable missiles were used considering the rigidity of the actual aircraft engine and accelerated to 60m/s.

In the horizontal test, the scale model with the single lid was used. The leak late from the lid measured during the impact test. Although the leak rate value from the lid increased by 5 orders of magnitude during the impact immediately, the leak rate shows the goodness of the leak-tightness at the lid, as the value is under 1.0x10-5 Pa•m3/s. In the vertical test, the full-scale lid structure with the primary and secondary lids was used. At the impact in the test, the leak rate, inner pressure between the lids and displacement of the lids were measured. The leak rate of the secondary lid exceeded to 1.0×10-3 Pa•m3/sec at the instant of the impact. However, as no residual lid opening displacement was occurerred after loading, the leak rate was recovered to less than 1.0×10-6 Pa•m3/sec after 3 hours from the impact test.

From these experimental results, it seems that the loss of the inner pressure of the metal cask cavity may be avoided in the im-pact due to aircraft engine crush.

ABSTRACT 8

Non Competent Authority Approved Packages – Methods for Leak Testing

GERRY HOLDEN*, MARC FLYNN

In preparation for TRANSCC 18 in 2009 several aspects of leak test methods and their associated pass/fail criteria for Industrial and Type A packagings (ie non Competent Authority Approved packages) were discussed and proposals submitted to change the criteria. It emerged that whilst there was existing guidance on this subject and that the regulations specified â€˜no lossâ€™ the practicalities of testing had not been addressed sufficiently, since â€˜no lossâ€™ was not a justifiable pass criteria. It was suggested that a briefing paper should be produced to present what the current leak testing practice was for packages that do not require Competent Authority approval (Excepted, IP-1,2 ,3 & Type A). The paper below forms the spine of that briefing paper and draws on the experience of Gravatom and LLWR for the testing of such package types.

The paper considers and describes;

- the existing regulatory status and the associated guidance;
- qualitative methods including liquid immersion/bubble detection, pressure and particulate stimulant;
- the applicability of methods for a range of designs including vials, drums, boxes and Freight Containers;
- the accuracy, advantages, practicality and limitations of the qualitative methods presented;
- the relationship of such methods to package types permitted to be used under the alternative arrangenments for IP-2/3 package types e.g.UN tested packagings;
- economics of the methods

The paper will then describe several case studies based on a range of packagings, e.g Type A liquid vial packages, IP-2 Freight Containers etc.

Abstracts – Thursday 07 October 2010 : continued

T39 - RADIATION PROTECTION ISSUES

11:00AM – 1:00PM – TECHNICAL SESSION – MAIN HALL
CHAIR: ASHOK KAPOOR, CO-CHAIR: FLORENTIN LANGE

ABSTRACT 396

The Interface between the IAEA Basic Safety Standards for Radiation Protection and the IAEA Transport Regulations

JIM STEWART

The IAEA Basic Safety Standards for Radiation Protection are expressed in the IAEA transport regulations. Both documents are currently being updated, and this paper examines the links between the two documents.

ABSTRACT 130

The Revision of the New International Basic Safety Standards and its Effect on the IAEA Regulations for the Safe Transport of Radioactive Material

CHRISTEL FASTEN, FRANK NITSCHE

The last version of the "International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources" (BSS), IAEA Safety Series No. 115, was incorporated into the IAEA Regulations for the Safe Transport of Radioactive Material, Edition 1996.

The main changes which were implemented into the transport regulations at that time refer to revised A1 and A2 values and new radionuclide specific exemption values.

Within the IAEA in cooperation with other international organisations WHO, NEA, UNEP, PAHO, EU and ILO a new draft of BSS as DS 379 was developed based on the recommendation of the International Commission on Radiation Protection (ICRP) No.103.

It is foreseen, that the work for the new BSS is finished in 2010. Afterwards the requirements of the new BSS shall be incorporated into national and international rules and orders in the radiation protection field as well as into the IAEA Regulations for the Safe Transport of Radioactive Material. The paper describes these changes coming from the latest draft of BSS which will have an effect on the IAEA Regulations for the Safe Transport of Radioactive Material TS-R-1. In particular the new exemptions values are being discussed with the focus on keeping them harmonized with the exemption values in TS-R-1.

ABSTRACT 168

Review of Methodologies and Development of Software to Calculate A1 and A2 and Exemption Values

TIBERIO CABIANCA*, KELLY JONES, MIKE HARVEY, TRACEY ANDERSON, IAIN BROWN

The purpose of this work is to carry out a review of the methodologies used to determine the A1 and A2 and exemption values. A1 and A2 values are given in the IAEA 2005 transport regulations and are the activity limits for Type A packages for special form and non-special form material respectively. The exemption values of activity and activity concentration, if either is not exceeded, define materials in practices which are exempted from the requirements of the IAEA Basic Safety Standards, and the IAEA transport regulations. The review includes:

- A review of the assumptions made in the calculations and identification of any improvements that can be made;
- A review of the treatment of decay products and a comparison with the approach taken to calculate exemption values.

The second part of the work is to produce a user-friendly software application that will be able to generate A1 and A2 and exemption values. Existing data, scenarios and methodologies have been used to calculate values that are currently listed in the IAEA transport regulations and Basic Safety Standards. However the application will be developed so that alternate parameters can easily be introduced. Following the publication of the revised IAEA Basic Safety Standards, and new ICRP decay data and dose coefficients, the software application can be used to calculate new exemption values as input into international discussions at IAEA. The application will also be capable of producing A1 and A2 values for radionuclides that are not listed in the 2005 Edition of the IAEA Transport Regulations, provided suitable data are available for those radionuclides.

A prototype of the application is available and the project will be completed by the spring of 2011.

This work is being funded by the UK's Department for Transport.

ABSTRACT 251

Review of Material Requirements of the IAEA Transport Regulations for LSA-II and LSA-III

WENZEL BRUCHER, UWE BUTTNER, FLORENTIN LANGE

In accordance with the safety concept of the IAEA Transport Regulations the package as combination of packaging and contained radioactive material has to provide required safety functions in all conditions of transport. Since Type IP-2 packages and Type IP-3 packages only have to protect against loss or dispersal of their contents under normal conditions of transport, IP packagings for LSA-II and LSA-III have only limited accident resistance. Consequently, the material properties required for LSA-II and LSA-III are mainly based on accident considerations and limitation of potential radiological consequences. Examples are:

- limiting the specific activity of LSA material to 10-4 A2/g for solid LSA-II and to 2x10 3 A2/g for LSA-III,
- homogeneity requirements regarding the distribution of radioactivity within the LSA material,
- for LSA-III only solid materials are allowed and powders are explicitly excluded,
- the requirement of a leaching test for LSA-III materials,
- the dose rate limit for the unshielded LSA-II or LSA-III material,
- conveyance limits for the transport of combustible LSA-II and LSA-III materials.

In the case of LSA-II material it is quite apparent by review of the historical development and current advisory material that the specific activity limit of 10-4 A2/g was introduced regarding transport and handling accidents connected with airborne release. In contrast, the corresponding reasoning for the justification of the 20-fold higher specific activity limit of 2x10-3 A2/g for LSA-III materials is not that

Abstracts – Thursday 07 October 2010 : continued

evident since the leaching test examines a material property which has hardly any connection with the majority of accident sequences and with airborne release.

The LSA-II and LSA-III material requirements were introduced into the Regulations in the early 70s of last century. In the meantime much progress has been achieved internationally regarding knowledge of material characteristics of LSA-II and LSA-III and release behaviour in accident conditions. It is shown that the factor of 20 in specific activity of LSA-III material compared to LSA-II has sufficient safety margins and results from other currently required material properties than those involved in the leaching test.

ABSTRACTS 390 / 391

The Results of a Coordinated Research Project into the Surface Contamination of Packages / The Practical Application of the Results of a Coordinated Research Project into the Surface Contamination of Packages

YONGHANG ZHAO

A detailed study was carried out to determine whether the current requirements for surface contamination of packages remain adequate. This paper presents the results of this study.

The results of the study into the surface contamination of packages suggests that there may be different ways to regulate the surface contamination of packages. This paper examines the options considered and presents the problems implementing them.

ABSTRACT 399

Safety of Transport of Naturally Occurring Radioactive Material

KASTURI VARLEY*, ULRIC SCHWELA, TIBERIO CABIANCA

In 2006 the International Atomic Energy Agency (IAEA) established a Coordinated Research Programme (CRP) on Safety of Transport of Radioactive Material in response to the recommendation made by the IAEA Transport Safety Standards Committee (TRANSSC) "to examine the adequacy of the current safety standards pertaining to the transport of naturally occurring radioactive material (NORM)". The CRP is considered research studies from nine countries related to the transport of a variety of NORM. The research areas provide full coverage of the subject, including exposures to both the public and workers. In addition, the CRP has discussed issues of a less technical nature relating to denial of shipment and perception of harm from the transport of NORM, a subject of great interest to modern commerce.

One important consideration to the work of the CRP is consideration of the basis of the current exemption levels for transport of NORM; namely, the basis for the 10× larger exemption level of 10 Bq/g for NORM not intended for the extraction of radionuclides, is unclear. Consideration is given to the appropriateness of the 30× factor for LSA-I in Para 226 of TS-R-1.

Details of the research, outcome and conclusions will be presented at this conference. The final research report, conclusions and recommendations from this Coordinated Research Project will be subject to review and decision on the way forward by the IAEA's Transport Safety Standards Committee during 2010.

T36 - STRUCTURAL ANALYSIS

11:00AM – 1:00PM – TECHNICAL SESSION
– CONFERENCE ROOM 1
CHAIR: GORDON BJORKMAN, CO-CHAIR: UWE ZENCKER

ABSTRACT 303

Numerical Analysis on Ship-Ship Collision Resistance in Design of 'KAIEI-MARU' and Classified as INF 3 Ship

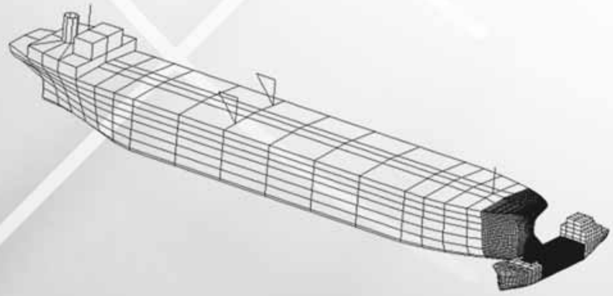
AKIHIRO YASUDA*

It is well known that the structural design for collision resistance is very important to assure the security of the package on a ship which transports spent nuclear fuels. As for the regulation of safe carriage of the spent nuclear fuel, INF code is commonly referred, which covers specifications of damage satiability, structural consideration and so on. However, in this code, nothing is mentioned about collision resistance hull girder. On the other hand, the official notice "Kaisa 520" issued by the Ministry of Transport includes the statement about the collision resistance hull girder.

In "Kaisa 520", Minorsky method is applied for design of collision resistance structure of this kind of ships. This simplified method is effective for the structural design on the initial stage because of comparatively few computational tasks. However, in Minorsky method T2 tanker is adopted as the striking ship, which is not in service very much now, and the outer shell is not taken into account as the structural member of the collision resistance. To achieve more accurate evaluation of collision resistance of hull girder, a numerical procedure introducing the nonlinear finite element analysis of the ship-ship collision was proposed by the regulation research panel of the Shipbuilding Research Association of Japan. In the procedure, VLCC is adopted as the striking ship. By using the procedure, some results of numerical analysis were already published.

Mitsui Engineering & Shipbuilding (MES) has accumulated experience to analyze and evaluate the collision resistance of ships. Based on these experiences, the analysis procedure was applied to actual design of the exclusive ships which transports spent nuclear fuels for the first time. This paper describes the numerical computation conducted in the design of Kaiei-Maru, which was constructed by MES in 2006. The analysis procedure and the method to evaluate collision resistance of hull structure are presented as an example, which has not been given before in the design of this kind of ships.

Figure 1 shows the numerical model of ship-ship collision analysis. Dynamic motion and material failure of structure for each ship are taken into account in this analysis.



Abstracts – Thursday 07 October 2010 : continued

ABSTRACT 372

Analyses to Demonstrate the Structural Performance of the KN18 in Hypothetical Drop Accident Scenarios

CHI-FUNG TSO*, KAP-SUN KIM, JONG-SOO KIM, KYU-SUP CHOI

The KN18 is a new cask design by KONES for KHNP for the dry or wet transportation of up to 18 PWR spent nuclear fuel assemblies in South Korea.

The containment vessel consists of a cylindrical thick-walled forged steel body, closed by a stainless steel lid with bolts. Spent fuel assemblies are located in a basket which consists of a tube disc system. Two pairs of trunnions are attached for lifting, manoeuvring and tie-down. A pair of impact limiters manufactured from wood and encased in steel cladding provide impact energy absorption during the hypothetical accident conditions.

The package complies with the requirements of 10 CFR Part 71 for Type B(U)F packages and is expect to enter service in 2011.

Structural performance of the package in the normal and accident conditions were demonstrated against the requirements of 10 CFR Part 71 by analysis including extensive calculations by state-of-the-art finite element methods, and confirmed by tests carried out on a 1/3 scale test model which were also used to verify the numerical tool and methods used in the analyses.

For the analyses of the hypothetical accident drop conditions, the models consisted of the complete package - including the impact limiters, the containment structure and the basket – which was modelled explicitly in detail and in three dimensions, to take into account the complex interaction between the components and the non-linearities in the geometry, the material behaviour and overall behaviour. The analyses were carried out using the explicit transient finite element method so that the transient behaviour could be robusted simulated.

This paper presents a selection of the analyses that were used to demonstrate the performance of the package in the hypothetical accident drop scenarios, discussing the analyses methodology, modelling technique and evaluation methodology, as well as analyses results and package response.

1/3 scale model drop testing and benchmarking of the model to the scale model tests are the subject of a separate paper.

ABSTRACT 409

Structural Evaluation of a Shielded Transfer Cask System for Intra Plant Spent Fuel Transfer

MIKE YAKSH*, MARC GRISWOLD

Intra Plant transfer movement of spent fuel between units can be accomplished using shielded transfer cask anchored to a high capacity trailer supported by pneumatic tires. The spent fuel is contained in a stainless steel canister with a bolted lid to allow for loading and unloading of the spent fuel. In this paper two separate evaluations are presented to confirm the safety of the system. The seismic event controls the design of the tie down system to prevent the transfer cask from tipping over. Since the wheeled vehicle has a broad band of possible properties, a series of time histories are perform to define the envelope of loading that could be imposed on the tie down system. The system is assumed to be subjected to a non-mechanistic tip over. A subsequent evaluation is performed using a separate model to ensure that the canister maintains confinement and that the internal structure (basket) maintains the position of the fuel during this event. This analysis employs a detailed model of the shielded transfer cask, bolted canister, basket, and fuel. To capture the nonlinear behavior of the system, each lead brick and the individual elements of the basket was modeled. Results for the transfer cask, canister and basket evaluations show that the geometric positioning of the fuel is maintained during design basis impact loading for both storage and transport configurations.

ABSTRACT 92

Mechanical Assessment Criteria of Spent Fuel Assemblies Basket Design

CHRISTIAN KUSCHKE*, VIKTOR BALLHEIMER, FRANK WILLE, STEFFEN KOMANN

Packages for the transport of radioactive material are generally equipped with specific structures (basket) to support the radioactive content in defined position. The safety function of the basket depends upon the kind of transported inventory. In case of transport cask for spent fuel, the basket design has to ensure the subcriticality of the fissile material in all conditions of transport in particular. Therefore the evaluation of structural integrity and neutron absorption capability of the basket is an important part of complete safety analysis. Sufficient heat transfer to maintain fuel assembly and cask temperature within allowable limits has to be verified as well. Corrosion resistance is an additional requirement on basket materials owing to contact with water during loading and unloading operations.

Computational and experimental methods or their combination along with additional material and component tests can be used to analyse the mechanical and thermal basket behaviour under transport conditions defined in IAEA regulations. By deciding between the analysis methods, the design features (including material selection concept) as well as specific safety function should be accounted.

In approval procedures of transport packages for radioactive materials, the competent authority mechanical and thermal safety assessment is carried out in Germany by BAM. Some questions of safety evaluation of basket designs are discussed in this paper based on the BAM experience within approval procedures. The paper focuses primarily on the mechanical behaviour of baskets with regard to the assumptions that have to be used in the criticality safety demonstration. The state of the art methodologies for computational basket stress and deformation analysis as well as for interpretation of drop tests results are presented.

ABSTRACT 205

Analysis Methodology and Assessment Criteria for Bolted Trunnion Systems of Type B Packages for Radioactive Materials

JENS STERTHAUS*, VIKTOR BALLHEIMER, FRANK WILLE

Packages for the transport of radioactive material are generally equipped with particular components for crane operations and

Abstracts – Thursday 07 October 2010 : continued

supporting the package during transport. The paper described the bolted trunnion systems of type B packages as an example of such devices.

The analysis of functional capability of trunnion systems is a constituent part of package safety design. The components of the trunnion systems (trunnion, fastening bolts) have to be analysed focusing on the assembly state, strength under maximum loads and, if necessary, fatigue in view of the overall load history. Safe handling of the package during crane operations (lifting, tilting) and secure package tie-down to the transport vehicle, if the trunnions are used as attachment point during conveyance, have to be ensured.

According to the BAM guideline draft for analysis and assessment of bolted lid and trunnion systems, the finite element method is to be used preferably in the analysis of such structures to obtain more accurate and detailed information about their loading. The finite element model of trunnion system should envelop the trunnion, the bolts and an appropriate part of the container wall with necessary contact conditions on all interfaces between these components. The application of solid finite elements, which is generally recommended in the BAM guideline draft, leads to local stress and strain fields as an result of the calculation. However the assessment concept and the corresponding safety factors in the technical standards, which have to be considered in case of the trunnion system, are usually based on nominal stresses.

This paper will discuss some aspects of finite element modelling of the trunnion system. The approaches to preparation and interpretation of calculation results in connection with local or nominal assessment criteria will be discussed with general reference to BAM experience in the approval procedure of type B packages.

ABSTRACT 133

Numerical Simulation of Dynamic Deformation of Air Transport Package in High-Speed Accidental Impact

ALEXANDER RYABOV*, VLADIMIR ROMANOV, SERGEY KUKANOV, VALENTIN SPIRIDONOV, DENIS DYANOV

In accordance with IAEA regulations, a package for air transportation of radioactive materials (a Type C package) must meet certain strict requirements. One of these requirements is that the package must be strong enough to withstand an impact on a hard surface at any angle and at a speed of at least 90 meters per second (m/s).

On the one hand experimental testing of resistance of air transportation packages is very expensive; therefore, experimental tests with real packages need to be curried out only after overall detailed computer simulations of dynamic behavior of the structure during impacts at different angles, which define the “weakest” elements of the structure and the most dangerous direction of impact. On the other hand it is necessary to involve any available experimental data for verification of results of the modeling because the problem is really complicated. Dynamic deformation of the structure under high-speed impact to a hard surface is an extremely non-linear process, which has several specific aspects as follows:

- large displacements (huge changes of initial structure shape);
- high levels of plastic strains;
- multiple contact interactions between structures elements and hard target.

Practical solution of this problem with acceptable accuracy could be obtained by using finite element code LEGAK-DK developed at RFNC-VNIIEF. The code is oriented to simulations of 2D and 3D high speed dynamic deformation problems.

The brief description of the code LEGAK-DK and its application for analyzing of large deformations dynamic response of structures is presented in the paper. The results of numerical investigations of a Type C package in accidental impact with hard surface at a speed of 90 m/s are also presented. Comparison of the calculated deformed package shape with experimental data shows that they are in a good agreement.

T7 - FISSILE EXCEPTIONS

11:00AM – 1:00PM – TECHNICAL SESSION
– CONFERENCE ROOM 2
CHAIR: DENNIS MENNERDAHL,
CO-CHAIR: HIROAKI TANIUCHI

ABSTRACTS 388/389

Influence on Transport of Fissile Material by Proposed Changes to TS-R-1

YONGHANG ZHAO*, JIM STEWART

Exceptions from the requirements applying to fissile material have existed since 1961. This paper sets out the development of the exceptions over the years.

As part of the developments of the next Edition of the IAEA Transport Requirements major changes have been proposed to the exceptions from the fissile material requirements. This paper explains the process undergone to develop the new regulatory text and outlines the proposed changes.

ABSTRACT 328

Overview of Proposed Modifications for Exceptions to the Requirements for Transport of Fissile Material

CECIL PARKS*, NICHOLAS BARTON, SAM DARBY, BRUNO DESNOYERS, MAKOTO HIROSE

The transport of fissile material is currently performed consistent with one of two sets of provisions within IAEA TS-R-1. The first set of provisions (termed Set I for this abstract) describes the testing and criticality safety assessment requirements for package designs to be certified for fissile material contents. Successful assessments provide a Criticality Safety Index (CSI) for the design to limit package accumulation and assure a safe subcritical value. These “(F)” package designs are labeled as FISSILE. The second set of provisions (Set III for this abstract) are those currently in para 417 where fissile material can be classified such that it is excepted from the provisions of Set I if specified characteristics and limits are met. Unfortunately, the Set III provisions were identified as posing potential safety issues, chiefly because several provisions rely on consignment mass limits (allowed up to ½ minimum safe critical mass) with no control on consignment accumulation.

Thus several Member State representatives have worked to address these safety concerns and have recently reached agreement on a set of provisions that can be forwarded to the IAEA Transport Safety Committee (TRANSSC) for review. The proposed Set III provisions allowing full exception from the Set I provisions are less flexible and more limiting than the current Set III provisions. In order to address the accumulation control concerns regarding the current Set III provisions, a new set of provisions (Set II) are proposed. These provisions require the use of a CSI control, where the formula for the CSI is based on a consensus safety assessment performed by the Member State representatives to identify a safe subcritical mass depending on fissile material specification (e.g., 235U) and the packaging (e.g., ability to meet normal conditions of transport) selected to contain the fissile material. Thus, the proposed Set II provisions classifies the fissile material as FISSILE, requires CSI control to limit accumulation to a defined safe subcritical mass, but does not require an "(F)" package. The full paper will review the requirements of the current and proposed provisions in TS-R-1. The advantages and perceived concerns with this approach will be discussed.

ABSTRACT 127

Changes in the Transport of Fissile Material Resulting From the Latest Proposed Revision of the IAEA Transport Regulations

INGO REICHE*, FRANK NITSCHE

Most provisions in the IAEA transport requirements TS R 1 regarding the classification and transport of fissile material have been in the regulations for a long time with only a small number of changes. However, the regulatory framework as well as the transport practice and needs have changed over the years. Therefore activities have been carried out by IAEA to reassess the complete system of classification and transport of fissile material. In March 2009 the IAEA initiated a review of TS-R-1 and associated guidance (TS-G-1.1) which will lead to a new revision of these standards. The draft of these new regulations contains significant changes to the provisions related to fissile materials. These changes, if getting approved, will considerably influence the practice of transporting fissile material.

In the paper at first the current alternatives for shipping fissile material are recalled. Then it is analysed how these different types and quantities of materials could be transported under the provisions of the latest draft of the new revision of TS-R-1. The schemes for transition from the old to the new system are intended to provide some guidance for continuing the transport of fissile materials under the proposed new provisions.

ABSTRACT 403

Competent Authority Approved Fissile Exceptions – one Regulator's View

NICHOLAS BARTON*

The IAEA transport regulations contain criteria for excepting packages containing enriched uranium and/or plutonium from the requirements pertaining to fissile material.

Many of these fissile exception criteria were designed to ensure a subcritical k∞ so that criticality safety was ensured regardless of packaging and without the need to control accumulations of packages or material.

Concerns were raised over the adequacy of some of these criteria, in particular over the continued adherence to a criterion under accident (or even normal) conditions of transport. Material might initially meet a criterion, however, following an impact or fire the disposition of the fissile isotopes could change leading to the criteria not being met and consequently it may no longer be possible to guarantee criticality safety.

With the introduction of TS-R-1 in 1996 a limit on the total mass of fissile material in a consignment was imposed on top of some of the fissile exception criteria in order to address these concerns.

The fissile exception criteria have been under review for some time and one aim has been how to exempt materials containing fissile isotopes mixed with much larger quantities of non-fissile material and which do not pose a realistic criticality hazard.

One provision in the proposed new fissile exceptions is for a material to be approved by the Competent Authority as fissile excepted and therefore posing no criticality safety concern. Such an approval would be given only following the successful assessment of a safety case justifying that material would remain subcritical under normal and accident conditions.

This paper gives a view of the type of material that might be approved under this scheme, the criteria against which an application for approval may be assessed and the type of arguments that an applicant would be expected to present.

ABSTRACT 365

Why Considering CH2 Moderation for Excepted Fissile Material?

IZASKUN ORTIZ DE ECHEVARRIA DIEZ*,
LUDYVINE JUTIER, STEPHANE EVO

Paragraph 417(a) of the Regulations for the Safe Transport of Radioactive Material (2009 Edition) imposes mass limits for packages and consignments where the disposition of the fissile material under NCT or ACT cannot be guaranteed. Therefore, it is proposed to revise this exception in paragraph 672 to strengthen these mass limits by requiring the use of CSI.

To be consistent with the safety level required for certified packages, the calculated CSI must ensure that two groups of packages, each having a total CSI of 50, contain a subcritical mass. Under NCT, five groups of these packages must be safe.

Thus, proposed subparagraph 672(a) requires that a group of packages with a total CSI of 50 contains less than 1/5th of a given subcritical mass but the release of fissile material from the package is not limited. Proposed subparagraphs 672(b) and (c) require that a group of packages with a total CSI of 50 contains half a given subcritical mass and that the packages don't release their content under NCT. The values of the subcritical masses are provided in Table M for two cases: with restrictions (based on H2O moderation) and without restrictions (based on CH2 moderation).

This paper discusses the criticality issues related to the use of CSI calculated in case of restrictions. Indeed, since packages do not necessarily survive the ACT or even NCT, ruined packages configurations cannot be excluded. Then, considering the fact that material used for transport may contain HDPE (retention trays, containers, bottles, pallets, etc.) or that packages could be transported alongside any other package carrying a CSI regardless of how the CSI was derived, a fissile material from packages whose CSI

Abstracts – Thursday 07 October 2010 : continued

has been calculated with restrictions can actually become moderated by a material whose hydrogen density exceeds the hydrogen density in water.

Different scenarios of consignments mixing packages approved using different cases of Table M have been studied. Results show that the mass of CH2 per consignment has to be lower than 500 g. Moreover if materials used for transport are taken into account, only the case without restrictions should be considered.

ABSTRACT 188

Bases for the General Licenses for Fissile Material and Exemptions from Classification as Fissile Material in 10 CFR Part 71

JEREMY SMITH*, ANDREW BARTO, CECIL PARKS

Included within Title 10 of the Code of Federal Regulations, Part 71 (10 CFR 71), are general licenses for fissile material shipments and criteria that allow exemptions from classification as fissile material for transportation package evaluation. These provisions in Part 71 were modified in a 2004 rulemaking to address potential safety concerns, as well as to provide a more straightforward set of criteria, consistent with other portions of 10 CFR 71. These rule modifications were based, in part, on the recommendations provided by NUREG/CR-5342, "Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses Within 10 CFR Part 71."

This paper will summarize the recommendations made by this NUREG/CR, as well as detail the bases for the version of the fissile material exemptions and general licenses that were subsequently adopted in the regulation. Additionally, this paper will provide examples that illustrate the intent and practical application of each fissile material exemption and general license.

T40 - CHARACTERISATION OF ENERGY ABSORBERS

11:00AM – 12:40PM – TECHNICAL SESSION – CONFERENCE ROOM 3

CHAIR: PETER SHIH, CO-CHAIR: WALTER VOELZER

ABSTRACT 153

Evaluation of Influence of Temperature Below 80ºC and Strain Rate on Compressive Property of Wood for Shock Absorber

KOJI SHIRAI,* KOSUKE NAMBA, YOSHIYUKI FUJITA

Generally, the mechanical properties of the shock absorbing materials at the normal temperature were used to evaluate the impact analysis at the accidental condition in the Safety Analysis Report of casks. However, there is a possibility that the impact acceleration increases because the shock absorbing material woods becomes soft with the increase of temperature and as a consequence these materials reaches to a bottoming-up point due to the re-markable deformation of impact limiters.

In this study, as a representative material of the shock absorbing materials used for the casks in Japan, three mate-rials (Oak wood, Fir-plywood, Balsa wood) were selected, and data concerning the temperature dependence (20, 50, 80oC) and deformation speed (0.1, 10, 1000mm/sec) on the compressive property was acquired by the static and impact compression tests. Test specimens were cured in the furnace at the designated temperature during 168 hours and its moisture content ratio was controlled less than 10%. According to the results, while the compressive strength reaches to 70-80% of the strength at 20oC with the increase of the temperature, the dynamic strength in-creases up to 1.2 times as high as the static strength with the increase of the strain rate over 0.1/sec.

Moreover, to investigate the affect of the temperature and strain rate effects of the shock absorber materials on the impact response of the ductile cast iron cask, the sensitivity horizontal drop analysis with LS-DYNA code were per-formed for the 9m horizontal drop test. It is concluded that in case of the use of the wood model considering the temperature and strain rate effect at the high temperature, the dynamic behaviour of the cask during the impact loading might not be significantly affected comparing the impact response using the wood model at room tempera-ture, in which the temperature and strain rate effects are not considered.

ABSTRACT 175

Contribution to Further Development of Simulation Methods for Impact Limiting Materials and Structures - a Report on the Situation from the German Quest-Project

EGBERT SCHOPPHOFF*, ROGER VALLENTIN, MANFRED STEEGMANNS, ROLAND HUEGGENBERG

The safe transport and the storage of radioactive materials from nuclear plants has to be guaranteed by the package-design. For example the loads as they appear during a drop onto different targets from predefined heights have to be examined.

The specification of the boundary conditions and the size of the loads are of special importance for the experimental and numerical analyses. In this context the energy dissipation of the involved components and materials are important. The behaviour of these materials within the relevant loading speed and the temperature boundary conditions have significant influence.

In the QUEST-project, conducted in close cooperation with the ENREA-project of BAM (German Federal Institute for Material Research and Testing), basic investigations in the area of describing the behaviour of material-specimen exposed to high-speed loading are carried out. Samples from wood, polyurethane-foam and damper concrete for interim storage facilities are subject to investigate. Thus, aim of all efforts is to achieve validated numeric methods and procedures for the improvement of simulation techniques for shock absorbers. The intention will therefore be a reduction of the physical loads during critical drop-scenarios.

By developing new material models and checking their validity with the experimental collected data it will be possible to obtain a more precise prediction by future numeric simulations. The material samples are subject to a variety of additionally appearing physical parameters, so that the mechanical behaviour can exactly be measured and analysed. The efforts in this project are also necessary to achieve modern analysis methods for developers, users and manufacturers of transport and storage casks.

The paper will give a brief overview on the progress within the

Abstracts – Thursday 07 October 2010 : continued

QUEST-project of WTI (Wissenschaftlich-Technische Ingenieurberatung) and GNS (Gesellschaft für Nuklear-Service). The tasks of supplying the project with suitable specimen and materials and the efforts to ensure quality-control during delivery are completely finished. The construction and the check of the test-facilities are also completed. First simulations of pre-tests were successful and have shown the precision of the test-facilities and the numerical analyses. The more sophisticated simulations of the various examined materials have been started and will be going on.

ABSTRACT 367

Waste Container Drop Tests onto a Concrete Target

THOMAS QUERCETTI*, ANDRE MUSOLFF, BERNHARD DROSTE, NAKAGAMI MOTONORI, KYOSUKE FUJISAWA

The paper shows technical details and experimental results of drop tests carried out with a 'Yoyushindo-Disposal ' waste container for intermediate depth disposal.

In context with Japanese disposal container safety assessment of L1-containers for intermediate depth disposal the German Federal Institute for Materials Research and Testing (BAM) performed drop tests contracted by Kobe Steel, Ltd. and a consortium of Japanese electric power plant companies. The tests were carried out in 2008 at the 200-tons drop test facility situated on the BAM Technical Test Site (TTS) nearby Berlin, Germany.

The drop tests were conducted with three specimens of the so-called 'Cubic Disposal Container Type L1' which is produced by Kobe Steel, Ltd.. The specimens are built in original scale. The outer dimensions are 1600 mm x 1600 mm x 1600 mm and the steel made walls have a thickness of 50 mm. The lid is welded to the container. The content of the containers was simulated by massive steel layers stacked together embedded to the side walls with mortar. Under the lid a free volume was created for the filling with zirconium oxide powder of approximately 20 kg as indicator substance for the particle release measurements, which were carried out after the drop tests. The total weights of the specimens were 20,000 kg and 28,000 kg because of different content masses.

The drop test program comprises three single 8-m drop tests in corner edge orientation onto a concrete slab. The specimens temperatures were equal or less 0°-Celsius. The drop tests were accompanied by extensive and various measurement techniques: strain and deceleration measurements to obtain the structural, cinematic and kinetic impact responses, high-speed video to visualize and analyze the impact scenario, temperature measurements to observe the cooling process, leakage testing of the container's lid system, optical 3D-metrology of the impacted corner edge and particle release measurements.

ABSTRACT 114

Effect of Dynamic Loading on Compressional Behaviour of Damping Concrete

EVA KASPAREK*, ROBERT SCHEIDEMANN, UWE ZENCKER, DIETMAR WOLFF, HOLGER VOELZKE

In drop scenarios related to assessing and licensing the storage procedure of spent fuel and high active waste, the casks under examination are generally not equipped with impact limiters. Hence, the extent of their potential stresses in case of an assumed handling accident is largely affected by the foundation properties of the reception hall floor in the specific storage facility.

Damping concrete which is formed by combining mineral aggregates with a polymer performs quite well in such applications as it features high stiffness as well as high energy absorption due to the filler pore volume. However, its softening effects are not sufficiently exploited in current finite element (FE) calculations due to missing advanced material models for simulating its impact response. An implementation of qualified concepts that account for plastic, strain rate dependent behaviour requires additional information that has to be provided by systematic test series.

BAM recently started a research project to generate such data, subsequently to develop and to improve numerical methods for the analysis of impact limiters and damping foundation material and thus to optimize safety assessment tools for the design of transport and storage casks.

A major part of this research concerns dynamic compression tests of variably shaped specimens conducted at a servo hydraulic 1000kN impact testing machine as well as at a BAM facility for guided drop tests. This presentation focuses 100mm damping concrete cubes deformed vertically at constant rates under different constraint conditions. For example, a special fitting holder was constructed to subject the specimens to multiaxial loading. Thereby a deformation of 70% could be applied.

Simulation was conducted by FE program Abaqus® based on the material model "Concrete damaged plasticity" allowing to define rate sensitive stress-strain relations in compression outside the elastic range following a yield function from Lubliner. It provides likewise tensile and compression damage variables that were used to reproduce crushing under uniaxial loading.

ABSTRACT 123

A Material Testing Program to Characterize the Concrete Behavior under Static and Dynamic Loads

JOE MAGALLANES, RUBENS MARTINEZ, ALOYSE NESER*, DIETMAR SCHREIBER, UWE ZENCKER, MIKE WEBER

In this paper a comprehensive material testing program is described to characterize a German final repository concrete material (BERB1) subjected to static and dynamic loads. The BERB1 material was developed und specified by the Bundesanstalt für Materialforschung und –prüfung for targets used for drop tests according to the KONRAD requirements. The testing consists of three concurrent laboratory experimental programs performed in the USA. An extensive preliminary testing program in Germany was performed to assure the required concrete specifications during fabrication and curing. The first is the static basic material test program consisting of structural tests to quantify macroscopic concrete properties under quasi-static loads. Cubic and cylindrical specimens are investigated under this effort. The second set of test data is from the static complete material test program, where cylindrical concrete specimens are subjected to a variety of quasi-static axial and radial stress and strain paths using a high-pressure hydraulic tri-axial chamber. The material is characterized for confining pressures up to 400 MPa. The third set of data is from the dynamic complete material test program, which uses a modified Split-Hopkinson Pressure Bar to induce dynamic compression and tension waves into cylindrical concrete

Abstracts – Thursday 07 October 2010 : continued

specimens that generate strain-rates up to 400 s-1. The ensemble of data generated in these experiments provides a complete set of data that effectively describes the behavior of this concrete and can be used to develop a constitutive calculation model. In addition, the results of the tests show a significant effect of moisture on the strength and rheology of this concrete for quasi-static pressures greater than 50 MPa. Numerical drop test simulations with the developed constitutive model in the Lagrangian finite element code, LS-DYNA, show good agreement with the experimental results from a BAM research project.

P7 - LONG TERM STORAGE AND TRANSPORT REGULATORY ISSUES

2:00PM – 3:40PM – PANEL SESSION – MAIN HALL
CHAIR: JIM STEWART,
CO-CHAIR: DOUGLAS AMMERMAN

ABSTRACT 139

Long Term Storage of Used Nuclear Fuel

KEN SORENSON, HOLGER VOELZKE, TOSHIARI SAEGUSA, MIKE WATERS

National policies around the world associated with the back end of the commercial nuclear fuel cycle are driving a need for very long term storage (e.g., > 20 years) of used nuclear fuel. In Germany, a referendum was passed in 2001 to phase out nuclear power in the 2020's. The policy for used fuel is to store at the reactor sites until a licensed repository is opened and operating. The currently licensed interim storage period in Germany is 40 years. In Japan, spent fuel shall be stored until reprocessing occurs. Interim storage is demanded, since the amount of spent fuel generated exceeds the current reprocessing capacity. The current safety guideline assumes the storage period of 40 to 60 years. The current U.S. policy is to store used fuel in-place (e.g., at the reactor sites) pending research and development to assess advanced fuel cycles and new disposal options. U.S. regulations license storage for 20 years with possible extensions up to 40 additional years.

Because there is no licensed repository operating in the world for disposal of used fuel, long term storage is the default alternative. Extended storage of used fuel suggests that material degradation and aging management are key topics for the safety evaluation of fuel elements, as well as storage casks. Other issues that need to be addressed with these key topics include material degradation in a marine environment and transportability of used fuel after long term storage.

Germany, Japan, and the U.S. have active programs addressing the technical and regulatory issues associated with the long term storage of used nuclear fuel. This paper will discuss the status of these national investigation programs and will provide a framework for international understanding in addressing these issues.

P6 - SECURITY ISSUES

2:00PM – 3:40PM – PANEL SESSION
– CONFERENCE ROOM 1
CHAIR: ANN-MARGRETH ERIKSSON EKLUND,
CO-CHAIR: YUNG LIU

ABSTRACT 299

Developing a Memorandum of Understanding Regarding Transportation Security in the United States

JOHN AHERNE*, RICK BOYLE, AL TARDIFF

In the United States, three federal agencies have statutory authority to ensure the security of radioactive materials in transport. These agencies are the U.S. Department of Homeland Security, the U.S. Department of Transportation and the U.S. Nuclear Regulatory Commission. Given that three agencies have authority, this could result in overlapping and conflicting requirements for ensuring the security of radioactive materials in transport. These agencies are working together to avoid this situation, which is consistent with the Obama administration's expectation for harmony across the U.S. government. This paper will describe the statutes that define each agency's role regarding secure transport of radioactive materials an how these agencies are working together to develop a memorandum of understanding (MOU) to identify each agency's responsibility and authority, to ensure consistent application of the statutes, and to avoid duplication of efforts. The major program elements, such as inspection, enforcement and development of requirements, are elaborated upon in the paper.

ABSTRACT 245

Securing the Transport of Nuclear or Radioactive Material

BRUNO AUTRUSSON*, OLIVIER LOISEAU, PIERRE FUNK

A nuclear or radioactive material transport could be a target of malicious acts which include:

- the theft or diversion with the aim to use it later, for instance, as an explosive device
- the sabotage in order to generate, directly or not, consequences to the human health and environment

The time aspect allows distinguishing these two items. In case of theft and diversion, the radioactive materials could be used a long time after the malicious act, whereas for sabotage the potential consequences could be immediate or almost immediate.

The aim of the paper is to present the protection of radioactive or nuclear material during transport against theft and sabotage. It applies to transports which occur on public or private domain (transfer within the fence of a facility). The role of actors is detailed. A graded approach to physical protection is defined and requires provisions depending on the level of potential consequences. Thus a graded approach to protect from sabotage needs to rely on a proper evaluation of the potential radiological consequences regarding certain sabotage scenarios. It differs from the graded approach to protect against theft and diversion for which the categories are simply based on the attractiveness of material for the potential construction of a nuclear explosive device.

Abstracts – Thursday 07 October 2010 : continued

Appropriate physical protection measures should then be designed and implemented for both in an integrated manner. These two sets of measures for protection against unauthorized removal of nuclear material and sabotage should be considered and implemented in a consistent and non-conflicting manner in order to achieve an adequate level of physical protection.

ABSTRACTS 297/298

Transportation Security Rulemaking Activities at the US Nuclear Regulatory Commission

RICHARD CORREIA, MARK SHAFFER, MICHAEL LAYTON, ADELAIDE GIANTELLI*

Since the formation of the U.S. Nuclear Regulatory Commission (NRC), the agency's mission has been to license civilian, peaceful uses of radioactive material in a manner that ensures adequate protection of the public health and safety, and to promote the common defense and security, and protection of the environment. The events of September 11, 2001, heightened our concerns about the use of special nuclear material, spent nuclear fuel (SNF), and other radioactive materials in a malevolent act. The theft or diversion of such materials, in particular SNF and IAEA Code of Conduct on the Safety and Security of Radioactive Source (Code of Conduct) Category 1 and 2 materials, during transport could lead to their use in a malicious act. Since 2001, the NRC evaluated its regulations, identified areas where security could be enhanced and, as an interim solution, issued several orders imposing additional security beyond the existing requirements found in Title 10 of the Code of Federal Regulations (CFR). With an interim solution in place, the NRC is moving forward with rulemaking to both enhance and put in place additional in-transit security requirements in the CFR.

This paper will focus on the NRC efforts to improve in-transit security of SNF and Code of Conduct Category 1 and 2 radioactive materials. In general, these enhanced in-transit security requirements will address areas such as preplanning and coordinating shipments, advance notification of shipments to the NRC and States through which the shipment will pass, control and monitoring of shipments that are underway, trustworthiness and reliability of personnel, and information security considerations.

ABSTRACT 213

Radioactive and Nuclear Material Transport Security

ANN-MARGRETH ERIKSSON EKLUND*, RICHARD RAWL

Transport of radioactive and nuclear material is highly regulated and the transport safety regulations have been in effect for decades. Transport security recommendations for radioactive material were published in 2008 by the International Atomic Energy Agency (IAEA) as an implementing guide, "Security in the Transport of Radioactive Material", and are just now being implemented in many countries. On the other hand, nuclear material transport security has been governed for many years on the basis of a binding international convention, the "Convention for the Physical Protection of Nuclear Material", and its supporting document "The Physical Protection of Nuclear Material and Nuclear Facilities" INFCIRC/225, Revision 4 (corrected).

Experience in implementing the radioactive material transport security recommendations has been gained by countries as they make decisions on specific security provisions to require, provide training to their regulatory staff and licensees, and begin reviewing and approving transport security plans. This experience has led to the development of practical approaches that minimize impacts as the recommendations are put into practice.

The nuclear material transport security requirements contained in INFCIRC/225 are being revised to update them and to incorporate requirements based on the recent amendments made to the Convention. This revision will include development of a new recommendations document within the Nuclear Security Series of documents.

The interface between the nuclear and radioactive material transport security documents is important in order to ensure that appropriate security measures, based on both the nuclear and radioactive properties of the material being transported, are defined and implemented.

This paper will provide up to date information on the development of the IAEA transport security documents and will present information on implementation of the radioactive material transport security recommendations. It will explain how the documents interface with each another and provide examples of how they should both be used in defining transport security requirements for shipments.

ABSTRACT 214

The IAEA Assistance and Training Programme for Transport Security

RICHARD RAWL, ANN-MARGRETH ERIKSSON EKLUND (PRESENTED BY MARK HAWK)

The IAEA Office of Nuclear Security is working cooperatively with the U.S. Department of Energy's Global Threat Reduction Initiative, European Union and Australia to provide transport security assistance to countries throughout the world

Assistance is available to countries in reviewing and upgrading their transport security programs at all levels

- National level (regulatory and other government agencies)
- Operator level (shippers and carriers)

Assistance is directed at implementing a consistent level of security throughout the life cycle of radioactive material (same level of security during transport as when in a fixed facility)

Upgrade assistance can include:

- Expert advisory missions to provide advice and guidance
- Training courses for regulatory, governmental and industry personnel
- Transport security awareness
- Detailed training on designing and implementing transport security programs
- Planning to identify and prioritize needs (developing security approaches and plans)
- Developing model security plans and procedures
- Equipment (vehicles, packages, command and control equipment, etc.)

Country visits are now being scheduled to initiate transport security cooperative activities

A training course has been developed to assist countries in developing and implementing transport security programs. The training course has been given as a national training course (three

Abstracts – Thursday 07 October 2010 : continued

times) and as a Regional training course (three times). The course addresses recommended security provisions for the transport of all radioactive material.

The course does not address additional physical protection provisions that may be needed for the transport of nuclear material subject to requirements that emanate from the Convention on the Physical Protection of Nuclear Material (CPPNM) and its amendment.

The Goal of the Training Course is to illustrate the need for adequate security during the transport of radioactive material, how to define levels of security with appropriate security measures, and how to effectively implement transport security programs.

ABSTRACT 193

Report on Radio Frequency Identification 2010 Category I Vault Testing Program

RICHARD KOENIG*, TERENCE WILLONER, HANCHUNG TSAI, YUNG LIU, DANIEL LEDUC

The U.S. Department of Energy (DOE) (Environmental Management [EM], Office of Packaging and Transportation [EM-45]) Packaging and Certification Program (DOE PCP) has developed a Radiofrequency Identification (RFID) tracking and monitoring system, called ARG-US, for the management of nuclear materials packages during transportation and storage. The performance of the ARG-US RFID equipment and system has been fully tested in two demonstration projects in April 2008 and August 2009. Planning has also been underway for field testing and applications of the ARG-US RFID systems at selected DOE sites and national laboratories, including the Savannah River Site (SRS), the Nevada Test Site (NTS), Argonne, Los Alamos, Oak Ridge, and Sandia National Laboratories. With the strong support of DOE-SR and DOE PCP, a field testing program has been authorized to be performed in Savannah River Site's K-Area Material Storage (KAMS) Facility, an active Category I Plutonium Storage Facility, in 2010.

The primary objective of the 2010 Category I Vault Testing Program is to demonstrate the operating capabilities and functionality of the ARG-US RFID equipment and system under realistic environment in the KAMS facility. Deploying the ARG-US RFID system lends to a reduced need for manned surveillance and increased inventory periods by providing real-time access to status and event history traceability, including environmental condition monitoring. The successful completion of the testing program will provide field data to support a future Facility Operators' campaign to finance and deploy the ARG-US RFID equipment and system for increased Operation efficiency and cost effectiveness for vault operation.

This paper will report progress in the field testing of the ARG-US RFID equipment in KAMS, the operability and reliability trend results associated with the applications of the system, and discuss the potential benefits in enhancing safety, security and materials accountability.

P10 - EMERGING REGULATORY ISSUES

2:00PM – 3:40PM – PANEL SESSION
– CONFERENCE ROOM 2
CHAIR: STEVE WHITTINGHAM, CO-CHAIR: EARL EASTON

ABSTRACT 326

A Discussion on the Secure Stowage of Packages

IAIN DAVIDSON*

The meaning of 'secure stowage of packages' within the UK (and elsewhere) is not consistently understood. There are obviously variations across industry with each organisations specialist preferring a particular method. There are also variations in package stowage (or tie-downs) within package types, for example Type B package are routinely held rigidly in place by trunnions or lashings and chocks or occasionally, more freely, in a well area that the package can not topple or bounce out of. There are greater variations across package types e.g. Type A or excepted packages in the medical sector are often freely placed in the boot of a car or a Type A nuclear density gauge may be loosely tied in the back of a van with rope. Further variations can be found in the techniques used to demonstrate 'secure stowage' some form of dynamic or static stress analysis either by hand calculation or numerical analysis usually takes place. Again, there are many variations as to what is a conservative analysis. More variations in understanding occur across the modes where each mode has its own guidelines which may not always concur with the IAEA guidelines and are also enforced by different organisations. Significant variations (or the risk of misunderstanding) can occur between the Design Safety Report, the Operating, Handling and Maintenance Instructions and the Local Work Instructions. Finally two conditions must be considered for secure stowage: 1. routine and 2. normal and accident conditions where the tie-down points, if they fail, must fail in such a way so as not to reduce the capacity of the package to meet the regulations.

The intention of the author is to explore these variables and produce guidance for industry on best practice and also to highlight areas that may still not be clear so that they may be resolved in the future.

ABSTRACT 39

Use of Vehicle Radiation Portal Monitors and Transport Regulations in Canada

SYLVAIN FAILLE*

In Canada, vehicle radiation portal monitors have been used at metal recycling facilities for many years to detect the presence of radioactive material in the material coming into the facility. Within the last few years, vehicle radiation portal monitors have also been installed at landfill sites and waste transfer stations.

In some cases, it is not practical or possible to identify or search through the material at the location where the material is detected due to lack of space, appropriately trained personnel or proper instrumentation. When dealing with waste, the task of locating and identifying the material is further complicated with the potential health hazards associated with garbage. Often, the facility simply denies entry of the material into the facility.

When there are limitations of space, it is more appropriate to move the vehicle to another location where the material can safely be surveyed. The Packaging and Transport of Nuclear Substances (PTNS) Regulations which make reference to the IAEA TS-R-1 Regulations, provides exemptions under which radioactive material can be exempted from the application of the regulations but without identifying the isotope and the activity contained within the load, it is not possible to determine if the radioactive material is regulated under the transport regulations or not.

This paper presents an overview of the issues related to the discovery of radioactive material in waste and scrap metal and the efforts undertaken by the Canadian Nuclear Safety Commission (CNSC) to address the issue related to the movement of unidentified radioactive material while preventing unnecessary exposure to workers by maintaining the protection of the health, safety and security of Canadians and the environment.

ABSTRACT 413

Designing Tie Down Systems for Heavy RAM Packages - Should Revised Design Criteria Apply?

PETER PURCELL*

The safe transport of Radioactive Materials (RAM) is the number one priority for all stakeholders and achieved through strict compliance with the IAEA Transport Regulations TS-R-1. These regulations cover all aspects of RAM transport and the full range of package types, irrespective of their radioactive contents.

Many of these package types are relatively light in weight but there are a significant number of larger RAM packages which weigh over 80 tonnes.

The object of this paper is to discuss issues that arise from the transport by road and rail, of these heavier packages, in particular the design and operation of the tie down systems acting between the RAM package and its conveyance.

The TS-R-1 regulations do not specify acceleration factors for RAM package tie downs but the supporting advisory material TS-R-1.1 gives indicative factors frequently applied by tie down designers for light packages. However, the tie down designer is permitted to use lower acceleration factors, subject to agreement with competent authorities and transport modal organisations. In the case of heavy RAM packages this is frequently essential to ensure a compliant tie down system can be designed.

This paper will demonstrate, in the case of heavy RAM packages, there can be issues arising from designing tie down systems to comply with modal organisations standards, which may lead to over design without real increase in transport safety. Indeed, in some cases achieving compliance with certain transport loading conditions could potentially increase risks to operating personnel. It will be demonstrated that where the RAM package weighs more than the conveyance vehicle, under specific loading conditions, it is the weight of the conveyance vehicle that determines the stresses in the tie down system and not the weight of the package itself.

Bearing in mind that tie down systems are designed to meet normal conditions of transport, in the specific case of heavy RAM packages, perhaps it is time to apply a pragmatic approach to tie down systems, taking account of realistic transport induced forces and the weight of the conveyance vehicle itself. This paper argues the case for this and demonstrates there will be no reduction in safety as a consequence.

ABSTRACT 200

Consideration on Safety Requirement for Large Component Transport with Q System

HIROSHI SUZUKI*, HIROMITSU MOCHIDUKI, MAKOTO HIROSHI, MANABU URAGAMI, MASANORI ARITOMI

A large component would be best categorized as SCOs. Para 523 of TS-R-I says "SCO-I may be transported unpackaged but SCO-II may not". About SCO-II level objects, competent authorities have reviewed the safety analysis reports on "surface contamination level over the internals" and "free drop", and they have approved them as "special arrangement" transports. On this paper, we considered "What conditions are the same level as requisite standards of safety established by TS-R-1?" with safety analysis reports for special arrangement. Under the normal conditions of transport, a component would be "fallen off the platform of a vehicle" as minor mishaps. In most cases packages would be relatively undamaged and would continue their journey after these minor mishaps. Under the accident conditions of transport, there are a fire and a collision as accidents. For Type A /IP packages, TS-R-1 requires little damage effect with contents limits.

If a large package would maintain the same level of safety as Type A or Type IP packages, an activity intake for a person in the vicinity of the accident should be approximately in the same level as that of that of Type A or Type IP packages, which is considered as a value of 10-6 A2. An activity intake for a person in an accident is given in the following equation:

QINT = QIV x FSCRAP x FRSUS x FREL x FINT

QINT: intake activity of radionuclides (Bq), QIV: actibity in a package, FSCRAP: fraction of surface area that is scraped in an accident, FRSUS: fraction of activity in a form of respirable aerosol, FREL: activity release fraction, and FINT: factor of activity intake for a person

Parameters FSCRAP, FRSUS and FREL are sensitive, and should be demonstrated to be appropriate through literatures, tests or reasoned arguments. Parameter FINT may be a value of 10-4, which is used in the Q system. Simple scenario such as "10% of internal activity will be re-leased from the component, and 1% of particles will be in the respirable size range may be adapted, when justified, then activity limit will be 10A2 for fixed and non-fixed surface contamination.

ABSTRACT 36

Transport of Abnormal Indivisible Radioactive Loads

DANNY VINCE*

During the decommissioning of nuclear installations many contaminated pieces of equipment are removed from the installation which, were they not to be further dismantled, would be far larger than the loads normally allowed to be transported on the transport infrastructure. Dismantling of such equipment can, however, lead to a risk of release of the activity they contain into the environment. There is often, therefore, a strong ALARP argument that these large components should be transported and disposed of without being further dismantled. Movement of such a load is often as much a transport logistics problem as it is a radiological one. In November 2007 the UK Department for Transport hosted a seminar

Abstracts – Thursday 07 October 2010 : continued

focussed on this issue as it affects the UK. Although that seminar was focussed on UK issues, many of the points raised during it will be of interest to the international community. This paper summarises the key points from the 2007 seminar and also provides an update on the proposed large load provisions for the IAEA Regulations for the Safe Transport of Radioactive Material.

T45 - FINITE ELEMENT MODELLING ASME

2:00PM – 3:40PM – TECNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: FRANK WILLE, CO-CHAIR: MIKE YAKSH

ABSTRACT 400

Flat Plate Puncture Test Convergence Study

DOUGLAS AMMERMAN*, MIKE YAKSH, CHI-FUNG TSO, DAVID MOLITORIS, SPENCER SNOW

The ASME Task Group on Computational Mechanics for Explicit Dynamics is investigating the types of finite element models needed to accurately solve various problems that occur frequently in cask design. One type of problem is the 1-meter impact onto a puncture spike. The work described in this paper considers this impact for a relatively thin-walled shell, represented as a flat plate. The effects of mesh refinement, friction coefficient, material models, and finite element code will be discussed. The actual punch, as defined in the transport regulations, is 15 cm in diameter with a corner radius of no more than 6 mm. The punch used in the initial part of this study has the same diameter, but has a corner radius of 25 mm. This more rounded punch was used to allow convergence of the solution with a coarser mesh. A future task will be to investigate the effect of having a punch with a smaller corner radius. The 25-cm thick type 304 stainless steel plate that represents the cask wall is 1 meter in diameter and has added mass on the edge to represent the remainder of the cask. The amount of added mass to use was calculated using Nelm's equation, an empirically derived relationship between weight, wall thickness, and ultimate strength that prevents punch through. The outer edge of the plate is restrained so that it can only move in the direction parallel to the axis of the punch. Results that are compared include the deflection at the edge of the plate, the deflection at the center of the plate, the plastic strains at radius r=50 cm and r=100 cm , and qualitatively, the distribution of plastic strains. The strains of interest are those on the surface of the plate, not the integration point strains. Because cask designers are using analyses of this type to determine if shell will puncture, a failure theory, including the effect of the tri-axial nature of the stress state, is also discussed. The results of this study will help to determine what constitutes an adequate finite element model for analyzing the puncture hypothetical accident.

ABSTRACT 377

Propped Cantilever Mesh Convergence Study Using Hexahedral Elements

CHI-FUNG TSO*, DAVID MOLITORIS, SPENCER SNOW, DOUG AMMERMAN

The Task Group on Computational Modelling for Explicit Analyses in the ASME Boiler and Pressure Vessel Code committee was founded in August 2008 to develop a quantitative finite element modelling guidance document for the explicit dynamic analysis of energy-limited events. This guidance document will be referenced in the ASME Boiler and Pressure Vessel Code Section III Division 3 and NRC Regulatory Guide 7.6 as a means by which the quality of a finite element model may be judged.

In energy limited events that the guidance document will address, ductile metallic materials will suffer significant plastic strains to take full advantage of their energy absorption capacity. Accuracy of the analyses in predicting large strains is therefore essential.

One of the issues that this guidance document will address, is the issue of the quality of a finite element mesh, and in particular, mesh refinement to obtain a convergent solution. That is, for a given structure under a given loading using a given type of element, what is the required mesh density to achieve sufficiently accurate results.

One portion of the guidance document will be devoted to a series of element convergence studies that can aid designers in establishing the mesh refinement requirements necessary to achieve accurate results for a variety of different elements types in regions of high plastic strain. These convergence studies will also aid reviewers in evaluating the quality of a finite element model and the apparent accuracy of its results.

The first convergence study consists of an elegantly simple problem of a cantilevering beam, simply supported at one end and built in at the other, loaded by a uniformly distributed load which is ramped up over a finite time to a constant value. Three different loads were defined, with the smallest load to cause stresses that are entirely elastic and the largest load to cause large plastic deformations. Material properties, loading rates and boundary conditions were also defined.

A number of the members of the Task Group analysed the problem. The results were collated and compared, and this paper presents the results of this study.

ABSTRACTS 335 / 340

Mesh Convergence Studies for Thin Shell Elements, Developed by the ASME Task Group on Computational Modeling

GORDON S. BJORKMAN*, DAVID P. MOLITORIS

The ASME Task Group on Computational Modeling for Explicit Dynamics was founded in August 2008 for the purpose of creating a quantitative guidance document for the development of finite element models used to analyze energy-limited events using explicit dynamics software. This document will be referenced in the ASME Code Section III, Division 3 and NRC Regulatory Guide 7.6 as a means by which the quality of a finite element model may be judged. One portion of the document will be devoted to a series of element convergence studies that can aid designers in establishing the mesh

refinement requirements necessary to achieve accurate results for a variety of different elements types in regions of high plastic strain. These convergence studies will also aid reviewers in evaluating the quality of a finite element model and the apparent accuracy of its results.

In this paper the authors present the results of a convergence study for an impulsively loaded propped cantilever beam constructed of LS-DYNA thin shell elements using both reduced and full integration. Three loading levels are considered; the first maintains strains within the elastic range, the second induces moderate plastic strains, and the third produces large deformations and large plastic strains.

ABSTRACT 353

Use of Computational Modeling Software for Evaluation of Structural Integrity

JASON PIOTTER*

The storage and transportation of spent nuclear fuel and other radioactive materials in dry storage casks and transport packages are regulated under 10 CFR Part 72 and 10 CFR Part 71, respectively. Due to the impracticalities and cost involved in physical testing of spent fuel storage casks and transport packages, licensees are increasingly using computer codes, defined here as Computational Modeling Software (CMS), to analyze and evaluate the structural integrity of these casks and packages to meet the regulatory requirements. This, in turn, has led to increasing challenges for both licensees and reviewers.

The current Standard Review Plans (SRPs -NUREG -1536, NUREG-1567, NUREG-1609, and NUREG-1617) do not provide sufficient detail on what information the staff should review in a Safety Analysis Report (SAR) and what supporting documentation is needed to describe the specifics of computer modeling to adequately capture cask or package performance. In order for the staff of the Division of Spent Fuel Storage and Transportion (DSFST) within the Nuclear Regulatory Commission (NRC) to efficiently review cask and package analyses, sufficient detail is necessary for the staff to effectively audit results or perform confirmatory analyses. Because cask and package analyses contain many parameters that can change the results of the analyses if treated inappropriately, situations exist where the staff will need to verify the validity of an applicant's analysis model, the methodology and assumptions used to create the model, or perform confirmatory analyses.

This Interim Staff Guidance (ISG) provides the technical position of the DSFST on what an acceptable structural analysis utilizing CMS should include to subsequently be reviewed by the staff in a request for licensing action. Areas considered in the ISG include Files, Model Development Process, General Features of the Models, Geometry Input, Material Models, Meshing, Boundary Conditions and Contact, Loads, Results, and Examples of Common Errors. The guidance presented in this ISG on the use of CMS for licensing spent fuel storage casks and transportation packages provides a coherent method for developing quality CMS models and a technical framework for licensing packages and casks in a timely manner.

ABSTRACT 219

Finite Element Mesh Design of a Cylindrical Cask under Puncture Drop Test Conditions

UWE ZENCKER*, MIKE WEBER, FRANK WILLE

Transport casks for radioactive materials have to withstand under mechanical accident conditions the 9m drop test, 1m puncture drop test and dynamic crush test according to the IAEA regulations. The safety assessment of the package can be carried out on the basis of experimental investigations with prototypes or models of appropriate scale, calculations, by reference to previous satisfactory safety demonstrations of a sufficiently similar nature or a combination of these methods. Computational methods are increasingly used for the assessment of mechanical test scenarios. However, it must be guaranteed that the calculation methods provide reliable results. Important quality assurance measures at BAM are given concerning the preparation, run and evaluation of a numerical analysis with reference to the appropriate guidelines.

Hence, a successful application of the finite element method requires a suitable mesh. An analysis of the 1m puncture drop test using successively refined finite element meshes was performed to find an acceptable mesh size and to study the mesh convergence using explicit dynamic finite element codes. The finite element model of the cask structure and the puncture bar is described. At the beginning a coarse mesh was created. Then this mesh was refined in two steps. In each step the size of the elements was bisected. The deformation of the mesh and the stresses were evaluated dependent on the mesh size. Finally the results were extrapolated to an infinite fine mesh or the continuous body, respectively. The uncertainty of the numerical solution due to the discretization of the continuous problem is given. A safety factor is discussed to account for the uncertainty. The calculation results are compared with experimental data from a puncture drop test with a half-scale model of a cylindrical cask.

This paper supports the convergence studies of the Task Group on Computational Modeling for Explicit Dynamics reporting to the ASME BPV Code Working Group on Design Methodology (SC III).

T35 - TRANSPORT SYSTEMS

4:00PM – 5:40PM – TECHNICAL SESSION – MAIN HALL
CHAIR: JUSTO GARCIA, CO-CHAIR: TBC

ABSTRACT 362

Land Transport Issues for the Industry

DONNA GOERTZEN*

Each year, several thousand packages of radioactive materials are transported by road or rail safely, securely and efficiently in countries around the world. Land transport is used to move a large variety of radioactive materials, for the medical sector, industrial uses and at various stages of the nuclear fuel cycle. It could even be said that every shipment starts and finishes its journey by being transported by land.

International transport by road and rail is by essence regional. In some instances, national requirements reflect the provisions developed in regional agreements, such as the European agreement concerning the International Carriage of Dangerous Goods by Road

Abstracts – Thursday 07 October 2010 : continued

(ADR), contracted by forty-five states; in South America, the Mercosul/Mercosur agreement regulates the transport of dangerous goods by rail and road.

The World Nuclear Transport Institute (WNTI) represents the collective interests of the radioactive materials transport sector, with member companies drawn from a wide range of industry sectors, including major utilities, fuel producers and fabricators, transport companies, and the supply of large sources. Through its semi-annual members meeting, WNTI members have been able to share their experiences of land transport; for example in Europe and North America.

The land transport of radioactive materials faces issues specific to this mode, but which are not necessarily country or region-specific. This paper will present some of these issues which impact on transport operations and which seem to have common features across the globe. With the North American experience as a case study, issues such as long distances through a varying landscape, driver shortages, the trade-off between road and rail, the emergency preparedness and response arrangements will be discussed and suggestions for solutions be given. Security and economic sustainability are also of important concern to those involved in the safe land transport of radioactive materials.

ABSTRACT 292

The Safe, Secure, and Efficient Transportation for Shipping Urania (“Yellow Cake”) From the Republic of Kazakhstan to Western Europe

AARON WIENER*

The safe, secure, and efficient transportation for shipping Urania (“Yellow Cake”) from the Republic of Kazakhstan to western Europe is affected by many factors within the transportation system including inadequate rail systems, inclement weather, unforgiving geography/topography, volatile governments, and strict European Union regulations governing the transit of radioactive materials through member countries.

That’s the type of Hazardous Material (Hazmat) shipping challenge Global Transportation Systems, Inc. (GTS Group) faces every day.

Our presentation will focus on the solutions GTS Group uses to ensure successful delivery in this challenging environment including: identification of alternate rail transfer points; multiple modes of transport; the use of European controlled connection points and hubs; and the use of real time tracking and tracing to ensure in-transit visibility.

In addition, we will discuss how project freight coordination best practices—proven through two decades as an industry leader—are adapted to support proper dangerous goods transit planning handling, and management. We will also describe how GTS Group leverages:

- Fully trained and certified (IMDG regulations, and IATA DGR and Radioactive Materials Handling) operations staff to ensure compliance with international transport regulations
- Strict companywide quality and reliability standards (including ISO 9001:2008 Certification) to meet and exceed customer expectations
- Secure web access for real time shipment monitoring for our customers

Speaker: Aaron M. Wiener, Operations Specialist GTS Group.

Mr. Wiener joined GTS Group in 2001. He manages and provides full service international import/export support and logistics for hazardous materials, air charters of various DGR, including “forbidden” and license controlled commodities. Mr. Wiener received a bachelor’s degree in business management from Radford University in Radford, Virginia. He has completed IATA DGR and Radioactive Transport Training and U.S./TSA Air Carrier Security Training.

ABSTRACT 306

Designing, Building and Delivering a Modern Approach to Consigning Radioactive Materials

MARTIN PORTER*, SONYA GRATTAN, ANGELA PARKER

Sellafield Limited (SL) consigns radioactive material (RAM) as part of its regular, daily business. Each year over 1,000 RAM packages are consigned from the Sellafield Site and around 10 of these represent significant overseas exports. In recent years there have been a number of challenges to these consignment activities, not least the restructuring of the UK nuclear industry brought about by the publication of 2003’s Energy White Paper. In early 2007 the Nuclear Decommissioning Authority (NDA) launched the competition to secure a Parent Body Organisation (PBO) for Sellafield, a site which represents about two thirds of the NDA’s nuclear liabilities. The ensuing competition, with the prize of entry into the UK nuclear clean-up market, was characterised by bids from numerous world-leading international companies. On 24 November 2008 NDA signed the new Parent Body Agreement for Sellafield Limited with Nuclear Management Partners (NMP) a joint venture between URS (Washington Division), Amec and Areva. Working together in partnership, NDA and NMP set about delivering significant improvements in operational efficiency, project management and cost control at Sellafield.

From day one, the new management team had a very clear view of the need to enhance consigning capability and this vision was swiftly realised. A new Operating Unit, Transportation Logistics, was formed containing a new department which was christened Consignment & Validation. Recruitment and re-organisation followed which embedded a suitable, sufficient and sustainable capability within Sellafield Limited as a measure to allow the Company to meet its duty as a major UK Consignor of RAM. The new organisation quickly designed, built and tested new approaches to the consignment of RAM (The 5P Consignment Process) and the validation of these activities (Consignment Command & Control Arrangements). These new, modern approaches have been piloted by Sellafield Limited, working with their stakeholders, and now form the basis of consignment activity across Sellafield. Deployment of the new system is governed by an overseeing body, the Office of the Consigning Authority.

This paper describes in detail the background to the required changes in approach, the new organisation/process and how this was delivered and implemented in collaboration with partners and regulators.

Abstracts – Thursday 07 October 2010 : continued

ABSTRACT 98

A New Information System for Transportation

PHILIPPE BAGONNEAU*

In a multi-business and also multi-entities context, in order to handle mono or multimodal, national or international transports, a strong and integrated information system has to federate these various businesses and entities around common concepts, to share information and to make processes more fluid, such are the main objectives of the project aiming to modernize the "Transport" information system of the AREVA group.

From transport order recording, up to real-time follow-up of this transport during its execution, through freight forwarding, fleet management (casks and associated equipments, transportation means as trucks, wagons), risk analysis of transportation schemas, preparation of the useful documentation in case of crisis and organization of the truck drivers missions, such is the functional scope of the project.

Among the business stakes linked to this project, we can quote :

- consolidation of the control of radioactive materials transport for the whole AREVA group (and more generally transports of dangerous goods), in particular by generalizing the transports oversight,
- industrialization of crisis management processes,
- support of AREVA Logistics Business Unit development while contributing to a high level of competitiveness,
- "distribution" of best practices in the field of radioactive materials transports (and more generally transports of dangerous goods) to the whole group.

Considering the international dimension of the project, ergonomics of the software solutions must be particularly looked after: the tool must remain easy to use while being flexible enough to adapt itself to the various scenarios. Furthermore, in case of international transports, actors of various entities can be brought to share centralized, reliable and secured data.

Finally, regarding its transversal nature, this project is also a "business project" by whom actors of various profiles will have to share processes and "business objects" (transportation schemas, transport files, suppliers, casks...).

ABSTRACT 116

Approach for Safe Transport of the Sample Including Nuclear Material

KEIICHI MORITA*, TADAHIKO YAMASHITA, DAISUKE TOGURI

The most of the sample included nuclear material, i.e. U, Pu and Th, is transported as Excepted package or Type A package with non fissile material.

In this transport, the packages must be conformed to technical requirements of the transport regulation, and can be transported. However, since it is not needed to apply any licensing or approval to competent authority for this trans-port, the shipper must check its conformity to the technical requirements of the transport regulation. After that, the package can be shipped out and transported. The checking of conformity to the technical requirements of the transport regulation before shipment is very important.

Therefore, the packaging, which is used for this transport, is checked for the conformity to the technical require-ments of the transport regulation, before the sample is packed in the packaging.

After the sample is packed in the packaging, the package is checked for the conformity to the technical require-ments of the transport regulation by radiation level survey, checking the label, marking, etc.

In this paper, through our experience of the transport of sample, the procedure and way of these checking will be introduced, and the improvement of our approach for the safe transport of the sample will be introduced.

T34 - EMERGENCY RESPONSE (SESSION 2)

4:00PM – 5:40PM – TECHNICAL SESSION – CONFERENCE ROOM 1
CHAIR: NICHOLAS BARTON,
CO-CHAIR: VERONIQUE BAYLAC

ABSTRACT 264

The Macarthur Maze and Newhall Pass Fires and their Implications for Spent Fuel Transport

EARL EASTON*, CHRIS BAJWA

In 2007, two severe transportation accidents, involving primarily long-haul tractor trailers, occurred in the State of California. In the first, which occurred in Oakland in the "MacArthur Maze" section of Interstate 580, a tractor trailer carrying gasoline impacted an overpass support column and burst into flames. The subsequent fire caused the collapse of the overpass onto the remains of the tractor trailer, due to the loss of strength in the steel exposed to the fire, in less than 20 minutes. The second incident was a chain-reaction accident involving several tractor trailers in the I-5 "Newhall Pass" truck bypass tunnel in Santa Clarita. This accident also involved an intense fire that damaged the concrete walls of the tunnel and required the tunnel to be shutdown for repairs. The US Nuclear Regulatory Commission (NRC) has studied both of these accidents to examine any potential regulatory implications related to the safe transport of spent nuclear fuel in the United States. This paper will summarize the work completed to date by the NRC on these accidents.

ABSTRACT 119

Lessons from Transport Events Involving Radioactive Materials Occurred in France between 1999 and 2009

LAURE CARENINI*, GILLES SERT, MARIE-THERESE LIZOT, CLAIRE SAURON

This paper presents a synthesis of the transport events involving radioactive materials occurred in France from 1999 to 2009, which have been notified to the competent authority. For each of them, about 70 parameters have been collected from the analysis of the notifications and reports of the events (type of event, type of package, level on the International Nuclear Event Scale...). The annual evolution of the number of transport events according to their nature

Abstracts – Thursday 07 October 2010 : continued

and their seriousness is presented as well as the summary of incidents with radiological implication. Two significant events are described more in detail:

- the one that involved in 2001 overexposure of the Paris-CDG airport handling personnel,
- the event, occurred in 2007, that involved a type B package in a fire.

The results from the analysis of these events have been used by the French Nuclear safety authority (ASN) and the French Institute for Radioprotection and Nuclear Safety (IRSN), to propose measures aiming at reducing the risks related to these transports. Indeed, areas of improvement have been identified relating to package designs and transport operations, as well as regulatory modifications and priority topics have been retained for inspections led by ASN.

In many events, human error has been cited as a cause or contributing factor. "Human error" mechanisms are part of the ordinary spectrum of human behavior. Such mechanisms are usually assessed by methods with fault tree analysis. It is important to think about what can be stated in the regulation to limit the associated risk.

ABSTRACT 376

Reviewing the Impact of the Revised INES Manual on Transport Activities

GARRY OWEN*

The International Nuclear Event Scale was developed in 1990 by international experts convened jointly by the IAEA and the OECD/NEA with the aim of communicating the safety significance of events at nuclear installations.

The INES has included 'transport' in its scope since 1992, however more recent revisions of INES have begun to 'increasingly' focus on transport activities and incidents involving sources.

The latest manual adopts a hazard rating system (D values) originally intended for incidents where facilities which may be redundant or otherwise have lost control of radioactive sources capable of significant harm.

The basis for the D values system is well founded and reflects several 'real life events'. Typically these cases are a result of radioactive sources being 'innocently' found and collected by an individual(s), subsequently taken to their home or workplace, sometimes causing radiation damage to themselves their families and others, through close proximity to the source.

For several reasons the reality of a transport event is likely to be quite different, primarily due to the inherent multiple barriers which are required to be established for transport.

The author considers the suitability and correlation between 'D values' and 'A values' is worthy of further study and investigation.

The proposed paper investigates the impact of using 'D values' over the 'A values' hazard rating system normally associated with the transport of radioactive materials in the public domain.

The paper also discusses the 'appropriateness' and longer term impact of the D value system in assessing and categorizing transport events.

ABSTRACT 398

A Transport Risk Assessment Package

KASTURI VARLEY*, ARUNGUNRAM NAGARAJAN NANDAKUMAR, CLIFFORD JÃRNRY

A Coordinated Research Programme (CRP) on Probabilistic Safety Techniques Related to The Safe Transport of Radioactive Materials was established by the International Atomic Energy Agency (IAEA) in the early 1990s. The CRP resulted in the development of a risk assessment package, INTERTRAN2. The INTERTRAN2 Package includes (a) INTERTRAN2, a handling program to assemble and manage input databases, construct input files for INTERTRAN2-RT4 , and execute INTERTRAN2-RT4 cases (b) INTERTRAN2-RT4, based on RADTRAN4.019IOSI (date January 20, 2000), an SI-unit version of RADTRAN4.019 (c) TRANSAT, An Atmospheric dispersion model and (d) a pre-processor which compiles the input file. This is a PC-based Code.

The Code calculates the radiation dose received by transport workers and the members of the public as a result of transport of radioactive material. The Code calculates the dose under incident-free conditions of transport and accident conditions of transport. The dose under accident conditions of transport is calculated from the assessed release of the radioactive content from the package. Such release is a function of the ability of the package to withstand an assault and the severity of an accident. In the assessment of the dose the code takes into account the probability of occurrence of an accident of a given severity and the probability of failure of a package of a given design in the accident.

The user of the Code has to obtain the accident data from reliable sources for the relevant mode of transport and classify the accidents into different severities on the basis of the mechanical impact and the thermal impact of the accident. The user should further determine the probability of failure of the package (which is considered in the transport risk assessment) for each accident severity. The reliability of the results obtained from using the Code would largely depend on the accuracy of the data used in the input. It is, therefore, necessary for the potential user to be familiarized with the use of the Code and the development of reliable data.

ABSTRACT 53

Hazardous Materials Commodity Flow Survey

WILLIAM SPURGEON*

The US Department of Energy, Office of Packaging and Transportation, has conducted a number of Commodity Flow Surveys along the major shipping corridors it uses for shipment of hazardous (radioactive and chemical) materials, in response to requests from its transportation stakeholders.

The objective of a Commodity Flow Survey is to collect data which can be analyzed to provide clear images, over time, of the types and amounts of hazardous materials shipments moving past a point along a transportation corridor. The information produced by such a survey can be an indispensible tool in helping emergency planners to understand and identify the planning, training, and resource requirements needed to effectively respond to a transportation incident involving hazardous materials.

This paper discusses requisites for conducting a useful Commodity Flow Survey and presents lessons learned on how results are developed and how insights on hazardous materials movements are gained through use of a survey. A key issue in the process is the care that must be exercised in planning and executing the survey, such that high data quality is achieved, with the result that the survey findings are clear and useful to transportation emergency planners. Among the activities described and discussed are: establishing survey location and objectives cooperatively with local emergency planners; establishing and training an effective survey team; survey data interpretation and analysis; and presentation of results as clear and persuasive images.

T33 - PACKAGE DESIGN AND STRATEGIES (SESSION 2)

4:00PM – 5:40PM – TECHNICAL SESSION – CONFERENCE ROOM 2
CHAIR: SANDRO TRIVELLONI,
CO-CHAIR: VLADIMIR ERSHOV

ABSTRACT 104

Innovation: Ahead of the Pack(aging)

MICHEL HARTENSTEIN*, CELINE FONTANET, HERVE ISSARD

Customers want the packaging to be more capacious, to accommodate hotter materials, to reduce dose rates, to have quicker approvals, not to mention to come cheaper. On the other side, Safety Authorities want augmented safety, thorough justifications, and essentially no new concepts. To make ends meet, the key is innovation on all fronts. AREVA's business unit Logistics has introduced a new concept, internally known as ID school. It is a unique combination of innovation tools and methods. It blends an all-round technological watch, a system for collecting internal ideas, an extensive R&D program, state-of-the-art research and problem-solving software, a network of experts, an array of innovation methods and facilities that promote an innovative spirit. These tools apply to engineering, freight forwarding, organization... and favour both individual initiative and teamwork.

This paper describes each of these aspects, stressing the successes and expected benefits to come.

This paper discusses some of the future developments in the ID school concept, and is also a plea for Safety Authorities to look neutrally or better favourably upon innovative features and designs, which often offer safety and radiation exposure benefits.

ABSTRACT 274

Developments of New Radioactive Transport Packages of Type B within the Current EMBAL Plan in CEA

EMMANUEL RIGAUT*, ALAIN JOUDON, SEBASTIEN CLAVERIE-FORGUE, THOMAS CUVILLIER

The fundamental principle applied to the transport of radioactive materials is that the protection comes from the design of the package regardless of the radioactive contents to be transported. At CEA, the development of nuclear research programs and the objectives fixed on decommissioning operations of nuclear facilities lead to a renewal of the ancient packages. In the long process of development, radioactive contents and packaging have to comply with the recent requirements and changes in the Regulations.

The designs of the packages have to take into account both the constraints of the nuclear facilities and the various natures of radioactive contents to be carried through areas outside such facilities. According to such needs for the nuclear activities, CEA has initiated in 2001 a program of renewal of its specific containers. This program called EMBAL plan has now 10 years of safety development process with a large feedback on conceptual design studies by referring to Type B packages. 28 projects have been analyzed with the high safety approach in the radioactive material transportation activities. In 2010, 13 packages have been manufactured with the EMBAL plan for the CEA's programs and have the approval certificate of the Competent authority.

The purpose of the paper is to present, for three designs, how the new CEA packages meet the applicable requirements of the recent Regulation : IR800 package for the transportation of the fresh nuclear fuels ; LR144 tank with the material performances for the prevention of high corrosion effects due to the chemical properties of radioactive liquid wastes ; TIRADE design in development for the transport of radioactive solid wastes with the possible hydrogen explosion in the cask due to the radiolysis risks. The main characteristics will be described and the reasons associated to the technical options will be explained according to the available systems, the test materials and the safety demonstration requirements.

Finally, the paper will show the complexity arisen during a safety assessment by an example: a qualified method may not be shared as a common reference between the applicant and the experience feedback of the safety expertise on the topic.

ABSTRACT 254

Transport: a Most Sensitive Link for the Nuclear Industry

MARC LEBRUN*, PASCAL CHOLLET

AREVA, world leader in the nuclear field, is involved in all sectors of the industry and has strategically organized its assets in order to provide power reactors with integrated services. As such, AREVA soon recognized that the transport activity is a fully integrated part of the delivered product. Consequently, stringent quality requirements applying to products and management equally apply to transport activities. Transport and logistics is considered to be a vital link between the various sectors of the industry, from mining, conversion, enrichment, fuel fabrication, fuel management and recycling. Yet, transport being an international activity, media sensitive, relying upon multi-disciplinary operations and expertise, and being performed within the public domain, transport is also considered to be one of the industry's weakest link.

This paper will review how AREVA has applied risk management practices to its vital transport activities and how it is being used on a day to day basis to conduct operational and strategic transport related decisions.

Four pillars, supported by a specifically designed information system, compose the methodology which has now been implemented within the company during the last four years:

1. Control of Sub-contractors' Chain of Transports
2. Risk Analysis and Risk Management
3. Studies and Support

Abstracts – Thursday 07 October 2010 : continued

4. Emergency preparedness and response

By reviewing the latest four years experience in this field, the paper will demonstrate that applying risk management practices to transport activities benefits the company, as well as the industry, as it contributes to improving security, image, and overall performance.

ABSTRACT 370

Transportation Scenarios for Risk Analysis

RUTH WEINER*

Transportation risk, like any risk, is defined by the risk triplet: what can happen (the scenario), how likely it is (the probability), and the resulting consequences. This paper evaluates the development of transportation scenarios, the associated probabilities, and the consequences. The most likely radioactive materials transportation scenario is routine, incident-free transportation, which has a probability indistinguishable from unity. Accident scenarios in radioactive materials transportation are of three different types: accidents in which there is no impact on the radioactive cargo, accidents in which some gamma shielding may be lost but there is no release of radioactive material, and accident in which radioactive material may potentially be released. Accident frequencies, obtainable from recorded data validated by the U.S. Department of Transportation, are considered equivalent to accident probabilities in this study. Probabilities of different types of accidents are conditional probabilities, conditional on an accident occurring, and are developed from event trees. Development of all of these probabilities and the associated highway and rail accident event trees are discussed in this paper.

ABSTRACT 178

Risk Management in the Design, Licensing and Fabrication

CHARLES TEMUS, RICHARD J. SMITH*

The adoption of the 1996 IAEA Regulations by the US Nuclear Regulatory Commission (NRC) required the decommissioning of many casks that had been in service in the United States. One such cask, the BMI 1, served a number of research reactors associated with university and US Department of Energy programs. A new package, the Battelle Energy Alliance (BEA) Research Reactor Cask (USA/9341/B(U)F 96), was designed, licensed and fabricated in an expedited manner to replace the BMI 1 cask.

The risks associated in this program came from a lack of well defined source terms, handling systems that were different for various facilities, and the need to be able to ship several different families of spent research reactor fuels. To meet the desired schedule, many of the activities were done in parallel. Established methodologies and materials were used to manage the risks associated with such a project. A previously licensed package was used as a basis of the design. The use of large design margins, conservative evaluation methods, and scale testing allowed the design and testing to move forward even though the contents were not fully defined.

By addressing the areas of risk both in design and in licensing, appropriate allowances were made to minimize the impact on the project. The project required the delivery of one complete package with interchangeable baskets for the different families of fuel, and a complete set of lifting, handling, tie down, vacuum drying and leak testing equipment.

This paper discusses the management of the project risks and the lessons learned during the design, licensing and fabrication of the new package and auxiliary equipment.

T38 - SPENT FUEL TRANSPORT

4:00PM – 5:40PM – TECHNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: ANTONY CORY, CO-CHAIR: TAKAFUMI KITAMURA

ABSTRACT 293

Italian-French Experience in the Transportation of INF for Reprocessing

FERNANDA DI GASBARRO, JEAN PASCHAL, DAMIEN SICARD, ROBERT VESPA, ROBERTO DONATI, GIANRICO LOMBARDI*, MARTINE VALLETTE-FONTAINE

SOGIN the state Company founded in 1999 to manage the closure of the nuclear fuel cycle and the decommissioning of the Italian NPPs and nuclear research centers, signed, in April 2007, a contract with AREVA for transport and reprocessing of approximately 235 tons of INF stored at Caorso, Trino and Deposito Avogadro sites.

One hundred ninety tonnes was stored in the pool of Caorso site located in northern of Italy.

The international transport activities, coordination and cask provision were entrusted to TN International (AREVA group). The multimodal transport in the Italian territory, the preparation activities, the coordination and execution of transports were subcontracted by TN International to the Italian authorized carrier MITNucleare;

The execution of the transport required significant preparation efforts both for the consignor site and the carriers with an interesting feedback to learn from.

In December 2007, just seven months after the signature of the contract, two TN17/2 casks containing 17 BWR fuel-element each left Caorso.

The loaded casks moving by road from Caorso up to the new road-rail transfer site, were transshipped on special 8 axes wagons, transported up to Valognes (F) rail terminal and then transferred by road to the AREVA La Hague reprocessing plant.

During preparation activities, adaptation of the plant for receiving and loading TN17/2 casks and transport infrastructure were performed. In addition an area of 4,000 m2 disused area, close to the Caorso rail station, was purchased by Sogin to realize a private rail terminal with a 120 tons capacity frame crane.

The emergency plan for transport was developed with the local Governmental Authorities while the Physical Protection System was designed, implemented and authorized by Authorities involved and the Ministry of Home Affairs.

Some of the activities, such as frame crane and road vehicles, were made significantly in advance, while other activities, such as procurement of cask loading equipment and the transfer site design and construction, were accomplished on a very tight time schedule. Such an ambitious goal was reached only thanks to the joint efforts of all the actors involved.

Abstracts – Thursday 07 October 2010 : continued

ABSTRACT 121

Return of the Fuel from the German Compact Sodium-Cooled Nuclear Facility KNK II with the CASTOR® KNK

ROGER VALLENTIN*, IRIS GRAFFUNDER, OLIVER PATZOLD, DIETMAR BRAUER

The Compact Sodium-Cooled Nuclear Reactor Facility KNK II, located at the Karlsruhe Institut for Technology (KIT), former Research Centre Karlsruhe has been operated from 1977 to 1991 as a prototype facility for the Fast Breeder Reactor SNR 300. The fuel of the KNK II consisted of fuel assemblies (FA) with highly enriched Uranium-/Plutonium-MOX fuel (up to 93 % 235U enrichment and up to 35 % plutonium in the heavy metal).

The FA were transported to C.E.A. (France) in 1993 for reprocessing. However, due to the low solubility of the MOX fuel 2413 fuel rods from 27 FA could not be reprocessed. They were encapsulated and stored in a pool of the French research centre Cadarache operated by the Commisariat à l'énergie atomique (C.E.A.). In a German-French fuel return project organized by Wiederaufbereitungsanlage Karlsruhe GmbH (WAK), these fuel rods will be returned to Germany to be stored for a maximum of 40 years in the interim storage facility ZLN operated by Energiewerke Nord GmbH (EWN). For the return and the interim storage of the fuel, 4 transport and storage casks of the type CASTOR® KNK were designed and fabricated by GNS Gesellschaft für Nuklear-Service mbH especially for this project. The transport is investigated by Nuclear Cargo + Service GmbH (NCS) and will be performed in one batch by road and rail. The project started in September 2001, the planned project deadline is at the end of 2010.

The paper will give an overview of the already finished project milestones (submission of the German cask certificate and storage approval for the CASTOR® KNK, validation of the cask certificate in France, fabrication of the 4 casks and the equipment, cask cold handling in France and Germany and the additional encapsulation of the fuel in Cadarache) on the one hand and the open milestones on the other hand (transport from France to Germany, storage of the casks in the ZLN).

ABSTRACT 350

Transport of Irradiated Fuel Pins

XAVIER BAIRIOT*, FABIEN LABERGRI

The transport of irradiated fuel pins is important to be able to investigate the behavior of the fuel after irradiation; in the Western Europe, these transports are using the BG 18 since 2001. The R 72 packaging has been taken in operation in 2009.

The presentation will focus on different aspects :

- Experience gained with the BG 18 packaging. This aspect will give more information about the lessons learned during the last 3 years of transports, mainly related to the organization of international transports and the associated difficulties, but also the loading and unloading on-site activities.
- The use of the new R 72, approved in 2009, from cold testing to transport operations. More information will be given on the broad possibilities offered by this new packaging for loading and unloading.
- The difficulty of leaking fuel pins. When utilities encounter situations of leaking fuel, it is important to investigate the causes leading to the leak, and therefore transport the leaking fuel pins to hotlabs. The difficulty to obtain an approval for leaking fuel pins will be explained, leading today to a situation where these transports are not possible with the R 72 packaging.
- The specific case of the first use of the R 72 packaging will be outlined; due to the old infrastructures, the loading of the R 72 in Avogadro (Italy) was in a first step considered as impossible. After all, the loading has been done in a very unusual way, by means of a upside-down shielding, taking the fuel pins with a pneumatic gripping device trough an opening on the shielding. This shielding as been docked to the R 72 packaging in vertical position, allowing the fuel pins to be lowered in the R 72 with minimal gaps. The story of one year preparation for one week on-site work.

ABSTRACT 296

Air Shipment of Spent Nuclear Fuel from Romania to Russia

IGOR BOLSHINSKY, KEN ALLEN*, LUCIAN BIRO, ALEXANDER BUCHELNIKOV

Romania successfully completed the world's first air shipment of spent nuclear fuel transported in Type B(U) casks under existing international laws and without shipment license special exceptions when the last Romanian highly enriched uranium (HEU) spent nuclear fuel was transported to the Russian Federation in June 2009. This air shipment required the design, fabrication, and licensing of special 20-foot freight containers and cask tiedown supports to transport the eighteen TUK-19 shipping casks on a Russian commercial cargo aircraft. The new equipment was certified for transport by road, rail, water, and air to provide multimodal transport capabilities for shipping research reactor spent fuel. The equipment design, safety analyses, and fabrication were performed in the Russian Federation and transport licenses were issued by both the Russian and Romanian regulatory authorities. The spent fuel was transported by truck from the VVR-S research reactor to the Bucharest airport, flown by commercial cargo aircraft to the airport at Yekaterinburg, Russia, and then transported by truck to the final destination in a secure nuclear facility at Chelyabinsk, Russia. This shipment of 23.7 kg of HEU was coordinated by the Russian Research Reactor Fuel Return Program (RRRFR), as part of the U.S. Department of Energy Global Threat Reduction Initiative (GTRI), in close cooperation with the Rosatom State Atomic Energy Corporation and the International Atomic Energy Agency, and was managed in Romania by the National Commission for Nuclear Activities Control (CNCAN). This paper describes the planning, shipment preparations, equipment design, and license approvals that resulted in the safe and secure air shipment of this spent nuclear fuel.

Abstracts – Friday 08 October 2010

T42 - MARINE TRANSPORT

9:00AM – 10:20AM – TECHNICAL SESSION – MAIN HALL
CHAIR: NAOTERU ODANO, CO-CHAIR: ALASTAIR BROWN

ABSTRACT 381

Responding to a Maritime Emergency Involving Radioactive Material

KASTURI VARLEY*

In the event of an emergency involving a ship carrying radioactive material there may be the need to involve coastal states in emergency response. While the emergency may not involve the radioactive material it is possible that its presence could lead to complications. This paper outlines a concept document developed to provide guidance to coastal states to help avoid such complications.

ABSTRACT 145

Planning, Licensing, Modifying, and Using a Russian Vessel for Shipping Spent Nuclear Fuel by Sea in Support of the DOE RRRFR Program

MICHAEL TYACKE*, IGOR BOLSHINKSY, SERGEY NALETOV, WLODZIMIER TOMCZAK

The Russian Research Reactor Fuel Return (RRRFR) Program, which is part of the U.S. Department of Energy's Global Threat Reduction Initiative, began returning Russian-supplied high-enriched uranium (HEU) spent nuclear fuel (SNF), stored at Russian-designed research reactors throughout the world, to Russia in January 2006. During the first years of making HEU SNF shipments, it became clear that the modes of transportation needed to be expanded from highway and railroad to include sea and air. This expansion was needed to meet the extremely aggressive commitment of completing the first series of shipments by the end of 2010. The first shipment using sea transport was made in October 2008 and used a non-Russian flagged vessel. The Russian government reluctantly allowed a one-time use of the foreign-owned vessel into their highly secured seaport, with the understanding that any future shipments would be made using a vessel owned and operated by a Russian company. ASPOL-Baltic of St. Petersburg, Russia, owns and operates a small fleet of vessels and has a history of shipping nuclear materials. ASPOL-Baltic's vessels were licensed for shipping nuclear materials; however, they were not licensed to transport SNF materials. After a thorough review of ASPOL-Baltic's capabilities and detailed negotiations, it was agreed that a contract would be let with ASPOL-Baltic to license and refit their "MCL Travel" vessel for hauling SNF in support of the RRRFR Program. This effort was funded through a contract between the RRRFR Program, Idaho National Laboratory, and Radioactive Waste Management Plant of wierk, Poland. RWMP, in turn, signed a contract with ASPOL-Baltic to prepare the vessel to transport SNF. This paper discusses planning, Russian and international maritime regulations and requirements, Russian authorities' reviews and approvals, licensing, design, and modifications made to the vessel in preparation for SNF shipments. A brief summary of actual shipments made using this vessel and experiences and lessons learned also are described.

ABSTRACT 261

The Logistic of the MOX Transport from France to Japan

PATRICE FORTIER*, YLAN TOUBOL

The plutonium issued from treatment is returned to Japan under the MOX fuel form for a future use in peaceful life in the commercial reactors of the Japanese Utilities. MOX transports towards Japan are most likely known through a media communication and because of the nature of the product which requires a high level of security protection in accordance with the International regulations.

Alongside the specific security measured which are applied by the governments including Japan, UK , USA and France, for the transport of Plutonium and Mixed Oxide fuel, the paper is proposing to provide a little bit more of industrial information on the logistics involved for the transport and repacking effort for the delivery of the MOX fuel.

After its transport from Melox plant, the fuel is repacked in AREVA NC La Hague recycling plant where the sea transport casks are prepared for their delivery to Japan. Despite the material transported is nuclear and highly protected for security reasons the industrial scheme applied is close to other industry domain. The transport means, the handling devices and human skill do not require complicated structures.

For the transport from La Hague to Japan, means used, even if they are dedicated, are commercial means such as the port, gantry crane and ships. No Military or State infrastructures are required to cope with the logistics and repacking activity.

Specific care is taken during the transport considering the fuel Integrity. The monitoring of fuel integrity is performed with on-shelves self-power devices which record data according to predetermined parameters. Such devices are also used in other industry fields which want to monitor their valuable goods during a transport over the world.

In conclusion the success of such a transport is based on the effort provided by all the parties involved from the management to the working level.

ABSTRACT 349

The R 74 Packaging : the Lightweight Packaging for HLW

XAVIER BAIRIOT*, FABIEN LABERGRI

Most of the packagings designed to transport HLW generated by reprocessing of spent fuel are heavy packagings, capable of taking high quantities of material, and designed to be transported by rail. The aim is to reduce the quantity of transport and increase as much as possible the quantity of HLW in one transport.

Due to the fact that we had to transport HLW issued from reprocessing activities on Belgian fuel from UK to Belgium, the transport has in any case to cross the Channel, making the transport by train or road of a heavy packaging not the most effective way of transporting in regard of the costs and the safety. it needed for example to overload in total 4 times for rail transport : near Dounreay (UK) as Dounreay is not connected to the grid, in the UK port, in the Belgian port, and finally near Mol (B) as Mol is neither connected to the grid.

A first feasibility study concluded on the advantages of a transport by truck, within weight and dimensions limits allowed for normal road

transport, and crossing the Channel on Ro-Ro vessels.

The presentation will focus on the difficulties encountered when designing the packaging, as well as the difficulties encountered when organizing the transport (mainly the difficulty of Ro-Ro sea transport)

1. The design of the R 74 was a challenge : due the size of the waste and the necessity to transport as much as possible waste, a very large opening was necessary. The associated closure devices created different difficulties to fulfill all requirements for accidental conditions.
2. When organizing the transport, all regular lines between UK and Belgium, Netherlands and north of France were contacted. The presentation will give more information about all actions taken, and the difficulties raised by the IMO regulations.

T43 - TRANSPORT EXPERIENCE

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 1

CHAIR: ERIK WELLEMANN, CO-CHAIR: DAMIEN SICARD

ABSTRACT 66

Loss of Control Incidents in Transportation: Challenges and Opportunities

SHAMSIDEEN ELEGBA*, NASIRUDEEN BELLO

Nigeria has a very robust petroleum industry, which is largest importer and user of radioactive substances in the country. The sources are used mainly for industrial radiography and nuclear well logging. The itinerant nature of these practices coupled with the location of oilwells and pielines on land, in the creeks and in the ocean increase the vulnerability of the sources during transportation. Since all the radioactive sources are imported, the vunerability of the sources is further aggravated during transportation in and out of the country. Safety and security of radiation sources and nuclear materials are well enshrined in the Nigerian Nuclear Safety and Radiation Protection Act 19 of 1995. Authorization, inspection and enforcement are three of the components of the regulatory control programme for radiation sources derived from the Act.

This Act provides for the establishment of the Nigerian Nuclear Regulatory Authority (NNRA) which was established in 2001. This Regulatory Control Programme has been tested and failed once in May 2003, when sources were exported as scrap metal. The second incident was in December 2004, when a duly authorized export of a spent radioactive source was re-packaged and mis-declared as "mould". These challenges have resulted in great opportunities for collaboration between NNRA and several national organizations as well as between Nigeria and the international community. Furthermore, two National Regulations on Transportation of Radioactive Sources and on the Safety and Security of Radioactive Sources have been gazetted since 2006. Similarly, annual training courses are conducted for front line officers and Radiation Portal Monitors are being installed at the major airports and seaports. Finally, transportation of radioactive sources is now to be tracked via satellite from the moment they enter Nigeria, through use and final export out of the country.

ABSTRACT 146

Safe and Secure Life Cycle Management of Radioactive Sources

GRANT MALKOSKE, PAUL GRAY

The International Source Suppliers and Producers Association (ISSPA) is comprised of most of the world's major manufacturers of sealed sources. The mission of ISSPA is to ensure that the beneficial use of radioactive sources continues to be regarded by the public, the media, legislators, and regulators as a safe, secure, viable technology for medical, industrial, and research applications.

Radioactive sources are used globally for a wide range of beneficial applications in health care, in industrial exploration and development, as well as in basic scientific research and discovery. In the health care industry, applications include sterilization of medical products, radiation therapy for cancer treatment, nuclear medicine and food irradiation. Other applications, such as oil exploration and industrial radiography make extensive use of radioactive sources. This presentation will elaborate on the extent to which sources are used in some of these applications.

The safe and secure use of radioactive sources in these sectors requires collaboration amongst all key stakeholders that affect the various operational aspects of the supply chain. The list of key stakeholders includes suppliers and producers, users, carriers, and regulators. They are involved during the useful operational life of radioactive sealed sources as well as the appropriate disposition of them at the end of their useful life.

How can safety and security be enhanced during the source life cycle? What measures are suppliers and producers taking to enhance security during source transportation? What are some of the operational impacts that need to be considered when implementing safety and security measures? What is needed to ensure that sources are and can be managed effectively at the end of their useful life? What is the role of key stakeholders in this process? In particular, how important is availability of appropriate infrastructure in providing options to users and suppliers for safe and secure source management? These are topics that will also be covered in the presentation.

ABSTRACT 235

Anomolies and Challenges of the IAEA Regulations that Effect the Transportation of Radiopharmaceuticals

CHARLIE CARRINGTON*, EUGENIE ROELOFSEN

The IAEA regulations have since their inception ensured that the transport of radioactive material is safe. This is testified by the number of packages that are transported annually with no problem with no safety issues to the public.

As the IAEA regulations have been updated to take into account changing technology and policies e.g. Quality Assurance, some of the older redundant requirements have not been removed.

One case is exclusive use shipment limits for normal radioactive material. For radioactive material the radiation protection programme (RPP) covers the requirements for dose control to workers and to the general public. When the RPP was instigated within the regulations the exclusive use requirements stayed as they are They were originally

Abstracts – Friday 08 October 2010 : continued

in place as a simple protection programme to the workers and the general public. Now that the RPP has been in place for a number of years, the exclusive use limits to vehicles carrying normal RAM should be removed for conveyances. In some countries the exclusive use limit is used as a limit per vehicle.

The need to limit the Transport Index per road vehicle is not required as the vehicle dose limits must be measured and be within the regulatory requirements. Revoking this requirement would allow more flexibility in moving RAM and utilise vehicles better and be greener.

The other change that would help the industry, help it to be greener with packaging and have no detrimental effect to the safety in transport is a review of the 20% increase in radiation level of any external surface for Type A packages after testing, paragraph 646 (b) TS-R-1.

This in reality is easy at high dose rates, for instance if the package has a starting surface dose rate (SDR) of 1500 Sv/h 20% is an increase of 300 Sv/h. At low levels SDR say 2 Sv/h the increase is 0.4 Sv/h, extremely difficult to measure. Our suggestion is that the SDR should be limited to an increase of up to the category level for Cat I and Cat II packages and 20% for Cat III packages.

ABSTRACT 406

Radiation Level Changes at RAM Package Surfaces

ERICH OPPERMAN*, MARK HAWK, RONALD NATALI, ASHOK KAPOOR

This paper will explore the regulatory requirements that limit variations in the radiation levels at external surfaces of radioactive material packages. The radiation level requirements at package surfaces (e.g. TS-R-1 paragraphs 531 and 646) invoke not only maximum radiation levels (e.g. 2 mSv/h), but also strict limits on the allowable increase in the radiation level (e.g. 20%). This paper will explore the regulatory requirements by quantifying the amount of near surface movement and/or payload shifting that results in a 20% increase in the radiation level at the package surface. Typical IP-2, IP-3, Type A and Type B packaging and source geometries will be illustrated. Variations in surface radiation levels typically are the result of 1) changes in the geometry of the surface due to an impact, puncture or crush event, or 2) shifting and settling of radioactive contents.

ABSTRACT 386

The Results of a Coordinated Research Project into the Severity of Air Accidents

JIM STEWART*

An extensive study was carried out to determine whether the Type C hypothetical accident conditions were representative of real life. This study involved the analysis of all significant air accidents and comparing them to the test requirements. This paper presents a summary of this study.

T44 - MANUFACTURING

9:00AM – 10:40AM – TECHNICAL SESSION – CONFERENCE ROOM 2

CHAIR: RYOJI ASANO, CO-CHAIR: STEPHANE COMPERE

ABSTRACT 14

Ignalina NPP Defuelling and Dry Fuel Storage Systems

GARETH WATKINS*

In order to provide safe on site storage for spent fuel elements from its 2 x RBMK1500 reactors, the largest of their type in the world, Ignalina Nuclear Power Plant (INPP) contracted with GNS mbH for the initial supply of CASTOR and later CONSTOR design dry spent fuel storage casks

The initial contract was for the supply of Casks for use in INPPâ€™s existing dry fuel store and this allowed continued operation of their nuclear power plant

A precondition to Lithuaniaâ€™s accession into the EU was to commit to a programme of closing both reactors. Unit 1 was closed at the end of 2004 and Unit 2 was closed on December 31st 2009. As part of the package of support to the country the Ignalina International Decommissioning Support Fund, managed by the European Bank for Reconstruction and Development, provided funding for a new, state of the art dry spent fuel store

The contract to build a new store and supply new casks in support of the decommissioning of INPP, was awarded to a consortium of GNS mbH and Nukem GmbH following a 2 stage competitive tendering process

GNS undertook to supply a redesigned cask with a higher storage capacity than previous designs in order to optimise the store footprint and overall project costs. The new design of cask required a full structural, thermal and criticality analysis and a new license application to be processed by VATESI, the Lithuanian Nuclear Regulator.

Due to the increased size and weight of the new RBMK1500 M2 cask all of the existing cask handling systems in the reactor pools are also being redesigned and either upgraded or replaced

The paper presented will provide an overview of the issues and challenges surrounding the design, licensing and construction of the casks and their associated equipment. It will discuss the critical success factors that should be considered when working with a multinational decommissioning project team on a large international project

ABSTRACT 97

Manufacturing of a New Transport Cask for MOX and UO2 Fuel.

RIPERT HERVE

TN International has designed a new cask TN112 for transport of used MOX or UO2 assemblies for PWR power plants. The capacity of this unique cask is 12 fuels. A mix of the two types of fuels in any proportion can be loaded. Its heat load can reach 48 KW with a total maximum weight of only 114 t. A B(U) type approval according to AIEA 2005 has been granted. It is intended for the transport between 900 MW reactor plants and La Hague AREVA NC reprocessing plant. The first cask is already in operation and the second unit of the fleet is in construction. The high level of performance has been achieved thanks to very specific materials such as: high strength forged austenitic stainless steel, reinforced gamma shielding materials, Vyal B resin for neutron shielding, soft aluminium for shock absorption and aluminium metal matrix materials as neutron absorber. The most significant step of the manufacturing are addressed in this paper such as preparing the technical specification, procurement of the main materials, destructive and non destructive testing, special process qualification, welding and assembly, final tests, quality surveillance.

ABSTRACT 242

CASTOR HAW28M - Fabrication and Cold Trials of a Cask for Transport and Storage of Vitrified High Active Waste Containers

ANDRE VOSSNACKE*, THOMAS HORN

From 1997 to 2006 GNS has returned vitrified high active waste (HAW) from reprocessing in France to Germany by using the GNS cask CASTOR HAW 20/28 CG. The cask has a capacity of 28 canisters with a maximum total thermal power of 45 kW. 74 casks of this type were loaded at the reprocessing plant in La Hague, France and have been shipped to the storage facility in Gorleben (TBL-G). The remaining HAW canisters at La Hague site exceed the technical limits of the CASTOR HAW 20/28 CG cask concerning heat capacity and radioactive inventory. Therefore GNS developed a new cask generation, named CASTOR HAW28M, which ensure the further return of the HAW to Germany. Hence it is possible to load 28 HAW canisters with a maximum total thermal power of 56 kW.

Already in the spring of 2003 the application for approval of the Type B package design containing fissile material was submitted to the competent authority. In September 2009 the design approval certificate was issued by the Federal Office for Radiation Protection (BfS). Accompanying the licensing procedure the manufacturing/assembly of prototypic components was investigated and the fabrication of 21 packages had been initiated.

In 2007 one brand-new, empty CASTOR HAW28M cask was shipped from the GNS cask assembly facility in Muelheim to the TBL-G for cold trials. With this cask, GNS had demonstrated the transhipment of casks at the Dannenberg transfer station from rail to road, transport to and reception at the TBL-G as well as incoming dose rate and contamination measurements and preparation for storage. Another cask was shipped to La Hague in 2008 to demonstrate all handling operations during transhipment and loading at that site too. During the cold trials all important handling steps which have to be carried out at the cask loading plant and at the reception area of an interim storage facility in Gorleben were performed, witnessed by the licensing authorities and their independent experts.

The paper gives an overview on the manufacturing/assembly of prototypic components investigated, the status of fabrication and the cold trials performed

ABSTRACT 268

Performance and Restrictions of Non-destructive Testing (NDT) Within the Quality Surveillance During Manufacturing of Type B- Packages

UVE GUENTHER*, MANFRED DR. BADEN, STEFFEN KOMANN, THILO NITZ

The Competent Authority BAM (Federal Institute for Materials Research and Testing) is responsible for the supervision of quality assurance and quality control in Germany.

BAM is supported by the independent experts of TÜV Rheinland. The main work scope of these experts is the surveillance of the manufacturing process of components or whole packages for the transport of radioactive materials.

Today the necessary material quality and in particular the acceptance limits of indications is determined in many cases by interpretation according to fracture toughness standards or new analysis concepts. The paper describes several important points to get agreed concepts between the package design analysis and the quality assurance. As an example it will explained how the establishment of a new approach of fracture mechanical analysis was accommodated with the extensive 3-D ultrasonic inspection.

The NDT has to prove the acceptance criteria surely. It is an important part of the quality control and design safety assessment of Type B-Packages.

Sensitivity and reliability must guarantee the preclusion of inadmissible flaws.

The paper explains following examples of NDT practice:

- Ultrasonic examination of thick-walled cask body made of ductile cast iron (Type CASTOR)
- Ultrasonic examination of the bottom to shell weld from forged components (Type TNI Cask) and
- Surface crack testing of trunnions from martensitic corrosion resisting steel

The possibilities and limitations of NDT as well as a view of the development of appropriate method to catch the state-of-the-art technology will be presented.

ABSTRACT 412

Radioactive Packaging Spares Management

DAVID MCWILLIAM*, GEOFF ROBINSON

Spares used on all radioactive packages for International Nuclear Services Ltd transports must be controlled and managed to ensure the original design basis of the packaging is maintained.

They must comply with the original design intent, that is, the package will fundamentally be exactly the same as it was originally, no matter how many components are replaced. This is assured by meeting quality requirements with regard to appearance, dimensions, materials, testing, storage, and fitting.

To be assured that these criteria are maintained, INS manages all aspects of spares from procurement and storage through to their issue and use.

The procurement process involves purchase via reputable suppliers. This is assured by the use of controlled listings of suppliers who have been vetted and verified by the Quality Assurance and the Environment, Health and Safety Departments within INS. The vetting

Abstracts – Friday 08 October 2010 : continued

process ensures the supplier is capable of doing the work required by INS, to a suitable standard. This may be by desktop assessment e.g. for small, low value items to regular on-site audits at the supplier e.g. for more complex items. This process includes proprietary 'off the shelf' items and a supplier can not be used unless they have been vetted. However, this does not prevent competition during the procurement process of new suppliers being added at any time.

During the procurement process the spare is graded in accordance with IAEA Safety Guides which will determine the manufacturing process and levels of inspection. For bespoke components the supplier receives detailed specifications and drawings. Depending on the quality grading level independent qualification of the manufacture may be required by,required inspections on behalf of INS both before and during manufacture, as well as prior to delivery.

INS also maintain trace-ability of spares, so that should it be found that a spare is faulty for whatever reason it can be determined where the component was manufactured, if there are likely to be any others from that source and whether they may be in-service on other packages.

During the lifetime of a package its design may change. Therefore, INS manages the removal of 'out of service spares and replace with current specification components.

T41 - SEALS, SNF ENCAPSULATION

9:00AM – 10:40AM – TECHNICAL SESSION
– CONFERENCE ROOM 3
CHAIR: TBC, CO-CHAIR: SARAH FOURGEAUD

ABSTRACT 151

Alternative Frequency for Periodic Leak Rate Testing

SHIU-WING TAM*, HANCHUNG TSAI,
YUNG LIU, JIM SHULER, YUNG LIU

According to the American National Standards Institute (ANSI) N14.5 Standard – Radioactive Materials, Leakage Tests on Packages for Shipment, Section 7.5, the purpose of periodic leakage rate testing is to confirm that the containment capabilities of packagings built to an approved design have not deteriorated during a period of use. Periodic leakage rate testing must be performed for all containment boundary seals, closures, valves, rupture disks, and other components. At present, the accepted frequency of periodic leakage rate testing is 12 months before each shipment.

The basis of the current frequency of periodic leakage rate testing can be traced to a paper by Lake in 1978 that deals specifically with closure designs using elastomer O-ring seals. Lake addressed both component and system reliability. Component reliability may be defined as the probability of that component performing as required under specified environmental conditions over a specified period. System reliability may be defined similarly, but the definition is expanded to encompass an interacting system of components. An example of an interacting system of components is a closure system with redundant seals and a test port/plug, such as the Models 9975, 9977, and 9978 Type-B transportation packagings for radioactive materials.

This paper describes methodologies that may be used to extend the interval applied to the periodic leakage rate testing of Models 9975, 9977, and 9978 packagings from 12 months to 5 years. The extended intervals are based on acceptable test results of O-ring performance and data from the continuous monitoring of environmental conditions of the packagings provided by the ARG-US radiofrequency identification (RFID) monitoring system, which was developed by Argonne for the Department of Energy (DOE) Packaging Certification Program (PCP), Office of Packaging and Transportation. Extending the interval between periodic leakage rate testing of the packagings, without compromising safety, reduces radiation exposure to workers and cuts annual costs by $2,500–3,000 per package.

ABSTRACT 169

Understanding the Low Temperature Properties of Rubber Seals

MATTHIAS JAUNICH*, KERSTIN VON DER EHE, DIETMAR WOLFF, HOLGER VOELZKE, WOLFGANG STARK

Rubbers are widely used as main sealing materials for containers for low and intermediate level radioactive wastes and as additional component to metal seals in spent fuel and high active waste containers. The save encapsulation of the radioactive container contents has to be guaranteed according to legislation and appropriate guidelines for long storage periods as well as down to temperatures of -40 °C during transport.

Therefore the understanding of failure mechanisms that lead to leakage at low temperatures is of high importance.

It is known that the material properties of rubbers are strongly temperature dependent. At low temperatures this is caused by the rubber-glass transition (abbr. glass transition). During continuous cooling the material changes from rubber-like entropy-elastic to stiff energy-elastic behaviour, that allows nearly no strain or retraction, due to the glass transition. Hence rubbers are normally used above their glass transition but the minimum working temperature limit is not defined precisely, what can cause problems during application. Therefore the lower operation temperature limit of rubber seals should be determined in dependence of the material properties.

The results of Differential Scanning Calorimetry (DSC) and Dynamic Mechanical Analysis (DMA) are combined with the results of standardized measurements as the compression set according to ISO 815. To reduce the test time of the standard tests a faster technique was developed.

Additionally, the breakdown temperature of the sealing function of complete O-ring seals is measured in a component test setup to compare it with the results of the other tests. The experimental setup is capable of measuring the leakage rate at low temperatures by the pressure rise method.

The materials that were selected for this investigation were ethylene propylene diene (EPDM) and fluorocarbon rubbers (FKM) as they are often used for radioactive waste containers. Some materials (seals and test sheets) were purchased from a commercial seal producer and some materials were compounded and cured at BAM in form of rubber sheets.

Abstracts – Friday 08 October 2010 : continued

ABSTRACT 334

Long-Term Mechanical Behaviour of Rubber O-Rings for MOX Fresh Fuel Transport Packages by Experiment and FEM Numerical Simulation

AKIHIRO MATSUDA*

Japanese transport regulation requires leak-tightness test as Periodic Inspection Check (PIC) on the trans-port packages package for MOX fresh fuels within a year. However the interval of the leak-tightness test should be settled reasonably considering the performance of the packages.

The durability of containment system of the transport package for MOX fuels depends on stress relaxation behaviour and plastic deformation of the rubber O-rings.

Establishment of evaluation method for long-time behaviour of the rubber O-rings would be useful, when the PIC periods would be required to be extended.

The purpose of this study was to evaluate the long-time mechanical behaviour of selected rubber O-rings for transport package by experiment and FEM numerical simulation.

We applied the accelerated thermal aging tests to the experimental method. The concept of the thermal accelerating test is that rather higher temperature than real one is applied to the specimen though the duration time to shorten the test period. This method is based on that the plastic deformation and stress relaxation of rubber follow the Arrhenius equation. From the result of accelerated thermal aging tests of the rubber O-ring, it can be concluded that the rubber O-ring which is used in real packages, will keep its integrity, at least, for five years at temperature of 70 degree Celsius.

Additionally, the finite element analysis was applied to the evaluation of the effect of lid groove on the mechanical deformation of rubber O-rings. The material model of rubber O-rings was obtained from the tensile tests of the O-rings and the hyperelastic model was applied to finite deformation analysis.

ABSTRACT 71

Encapsulation of Fuel Rods for Transport

SIMON STANKE*, FRANZ HILBERT

Due to a lack in benchmarks, currently used computer codes for fuel depletion and decay are not validated for very high burn-up values. Furthermore, there is no sufficient information about the mechanical properties of cladding and fuel available to assess the behavior of fuel rods with very high burn-up under normal and accident conditions of transport. Therefore, for modern designs of packages encapsulation of fuel rods with a very high burn-up as well as for damaged fuel rods is required by the competent authorities. Additionally, utilities, fuel manufactures and plant operators have a high demand to transport fuel rods with these very high burn-up values.

The package NCS 45 is licensed for the transport of fuel rods with a burn-up of up to 120 GWd/MgU. Fuel rods with a burn-up of more than 62 GWd/MgU must be encapsulated, with even more stringent requirements in USA. The cans must not contain water and must be sealed by welding or an adequate method.

The production of welded cans in Hot Cells is standard practice. However, the encapsulation of fuel rods in a fuel pool was not possible in the past. For this problem NCS developed a solution (pat. pend.) comprising an encapsulation device, cans and proper procedures to produce under water dry and leak-tight cans containing fuel rods. On the contrary to cans produced in Hot Cells, NCS uses a brazing method to close the cans.

The presentation will provide an overview of the encapsulation device and respective procedures. It will describe the approval procedure in Germany as well as the validation in other countries. Finally, experience gained during approval tests and commissioning will be presented.

PATRAM 2010 Posters

MANAGEMENT CONTROLS

ABSTRACT 6

Survey and Evaluation of the Safety Measures Applicable to the Radioactive Dangerous Goods Transportation in Slovenia

THOMAS BREZNIK

Due to industrialization and simultaneous increasing heavy traffic is the risk assessment of hazardous materials transport more and more inportant. Every year approximately around 150 radioactive material packages (consumer goods excluded) are transported within, into and out of Slovenia by all modes of transport. The transport involves many types of nuclear and non-nuclear radioactive materials and radiation sources for applications in research, medicine, industry, hydrology, geology, education and nuclear power production in quantities ranging from miniscule amounts to very large quantities of radioactivity.

This article is an account of work undertaken with the purpose to establish a national radioactive material transport event review and analysis system and presents the principal findings and conclusions of a comprehensive analysis and evaluation of abnormal occurrences and irregularities involving radioactive material shipments in the review period from about 1997 - 2007 in Slovenia. The 10-year transport event review and analysis results will provide evidence if implementation and application of the existing national and international regulatory controls and safety requirements to ensure a satisfactory level of protection and safety in transport of radioactive material in Slovenia. Safety measures of hazardous materiela, i.e. radioactive material have recently emerged as critical need and several models andapproaches have appeared. Survey and evaluation of the safety measures applicable to the radioactive dangerous goods transportation in Slovenia may be achieved by addressing in considering of the following basic objectives:

- Survey and evaluation Â»screeningÂ« of the existing relevant systems and their application,
- Identify requirements and consider international (EU) links,
- Determine appropriate analyses,
- Develop our own criteria and formats,
- Produce a full report.

ABSTRACT 20

Data Base for Reports of Cases of Denial and Delay of Shipments in Latin America and Caribbean countries

MARIO MALLAUPOMA, WALTER CASTILLO, NATANAEL BRUNO

Modal transport have specific regulations which are based on the UN Model Regulations and, for the Class 7 â€" radioactive material - are also based on the Regulations for the Safe Transport of Radioactive Material, published by the International Atomic Energy Agency. Transport of life saving radioactive material is particularly affected by delays and denials of shipments. The refusal by airlines and shipping companies in accepting radioactive material for transport results in hardship to patients undergoing diagnosis and treatment with radioisotopes. Those relying on ionizing radiations to sterilize material are facing similar problems.

At the first meeting of the International Steering Committee on denials of shipments held in Vienna late in 2006 emphasis was given to the need of keeping records of instances of denials and also to the success cases in avoiding refusals by carriers and forwarders. A database developed by the International Maritime Organization and joint administered by IAEA and ICAO served as starting point. After the first workshop on denials held in Uruguay Latin-America and the Caribbean countries have considered submit as a report of all those cases produced in the region. In order to implement the recommendations given by the International Steering Committee, a form was developed to enable to record and store data and to allow related analysis. The proposed paper describes a database developed in Latin-America and show how this database can help in identifying, avoiding and preventing cases of denials and delays of shipment of radioactive material.

ABSTRACT 83

Current Status and Future of Cask Maintenance at NFT

HIROSHI KANAZAWA, TSUTOMU MATSUMOTO

1.Introduction

NFT currently owns about 70 casks of wet type and have been using them for transpor-tation of the spent fuel from Nuclear Power Plants to Rokkasho Reprocessing Facility in Japan since 1998 . During this time, NFT have performed periodical inspections (PI) to maintain and manage their performance as well as routine voluntary inspection for packagings.

We introduce the plan of repair and maintenance and maintenance and management of performance that took into consideration a long-term maintenance plan based on future high aging deterioration.

2. Results of maintenance

NFT's routine maintenance is performed on the packages at every year, every five years and ten years, in addition to the PI performed one or more times per year. We evaluated the performance of the package which has been used for 10 years in order to evaluate the effect of aging deterioration, by measuring the temperature and dose rate of actually fuel loaded package.

As the result, we confirmed that all the packages measured did not suffer any aging de-terioration by comparing the actual measurement with the analytical values based on a specification for actual fuel.

3. A long-term maintenance plan

We are developing a maintenance plan taken into consideration of aging deterioration for the purpose of maintenance of a long term safety performance of the packages.

PATRAM 2010 Posters : continued

ABSTRACT 132

Integrating Business Continuity with Emergency Response

PETER BENTLEY

The Nuclear Industry has had emergency response procedures in for many years, however the need to ensure Business Continuity has recently become more apparent. With a higher reliance on technology and on members of the organisations supply chain this has increased the potential for interruptions.

Business Continuity is a relatively new and evolving discipline. With new technology lie new business risks. Our dependencies upon 'hidden technologies' only become clear when things go wrong. An organisation must be prepared to respond effectively and efficiently in the event of a crisis, maintaining customer and stakeholder confidence is critical within the nuclear industry.

Business Continuity Management has evolved as an organisational response to such situations. However get it wrong or fail to prepare by adequately training and exercising and the result will be catastrophic. This can create an inability to identify some of the latent, emerging and unknown risks facing the organisation.

Although there are many theories as to why organisations might fail in times of crisis this paper will avoid a debate about the merits of Natural Accident Theory versus High Reliability Theory. This paper will draw upon those theories to show how people, communications and organisational culture can be regarded as the biggest variables in business continuity planning. Critical to organisational failure are,

- Key business functions were unconnected to a recovery plan.
- Early signals that things were going wrong or were about to go wrong were not interpreted correctly.
- The interdependency of key business functions was not fully appreciated.

This paper will outline the importance of effective business continuity planning for INS. This will ensure that the organisation, culture and people are not merely part of the response, they form the very backbone to dealing effectively with the crisis and making the seemingly impossible, possible.

PUBLIC ACCEPTABILITY

ABSTRACT 265

Potential Effects of Historic Rail Accidents on the Integrity of

EARL EASTON, CHRIS BAJWA

The US Nuclear Regulatory Commission (NRC) completed an analysis of historical rail accidents (from 1975 to 2005) involving hazardous materials and long duration fires in the United States. The analysis was initiated to determine what types of accidents had occurred and what impact those types of accidents could have on the rail transport of spent nuclear fuel. The NRC found that almost 21 billion miles of freight rail shipments over a 30 year period has resulted in a small number of accidents involving the release of hazardous materials, eight of which involved long duration fires. All eight of the accidents analyzed resulted in fires that were less severe than the "fully engulfing fire" described in the NRC regulations for radioactive material transport found in Title 10 of the Code of Federal Regulations, Part 71, Section 73, as hypothetical accident conditions. None of the eight accidents involved a release of radioactive material. This paper describes the eight accidents in detail and examines the potential effects on spent nuclear fuel transportation casks exposed to the fires that resulted from these accidents.

REGIONAL EMERGING ISSUES

ABSTRACT 19

Denial and Delay of Shipments of Radioactive Material - Experience in Latin America and the Caribbean Countries

NATANAEL BRUNO, MARIO MALLAUPOMA

The increasing difficulties in having nuclear and other radioactive material accepted for transport, the so called phenomenon â€œdenials and delays of shipment are generating both social and economic problems. This problem needs to be addressed in both global and regional perspective. Short, medium and long terms strategy are being implemented but they required commitment of stakeholder.

The regional work should join forces to develop alternatives of solutions in common problems of denial and delays Once regional progress in alleviating denials are noted, it would be then time to promote a greater integration with different regions. Participation of government organization and private sector in finding alternatives to keep transport a sustainable activity will be crucial.

Social and economic development encompasses the use of radioisotopes and denials and delays of shipment are imposing restrictions for such development. The proposed paper will describe the experience of Latin-American countries (the Montevideo network) in implementing the recommendations given by the International Steering Committee on Denials and /delays of shipment of radioactive material.

PACKAGE MANUFACTURING AND TESTING

ABSTRACT 4

Mathematical Modelling Of Immobilization Of Radionuclides 137cs And 60co

ILIJA PLECAS

Transport phenomena involved in the leaching of a radioactive material from a cement composite matrix are investigated using an empirical method employing a polynomial equation. To assess the safety for disposal of radioactive waste-cement composition, the leaching of 137Cs and 60Co, from a waste composite into a surrounding fluid has been studied. Leaching tests were carried out in accordance with a method recommended by IAEA . Determination of retardation factors, KF and coefficients of distribution, kd, using a simplified mathematical model for analyzing the migration of radionuclides, has been developed. Transport phenomena involved in the leaching of a radioactive material from a cement composite matrix are investigated using an empirical method employing a polynomial equation. In our experiment we have analized mechanism of 137Cs

PATRAM 2010 Posters : continued

and 60Co leaching values during a period of 60 days. Results presented in this paper are examples of results obtained in a 25 year mortar and concrete testing project, which will influence the design of the engineered trenches system for a future central Serbian radioactive waste storage center.

ABSTRACT 164

Tests and Measurements for Radioactive Material Transport and Storage Packages

GEORGETA IANCSO

This paper describes tests and measurements available at "Horia Hulubei" National Institute for Physics and Nuclear Engineering (I.F.I.H. – HH), Romania, for radioactive waste transport and storage packages

Testing and Nuclear Expertise Laboratory of IFIN-HH is accredited by national competent authorities to perform tests for radioactive material transport and storage packages.

Transport and storage packages which are tested in Testing and Nuclear Expertise Laboratory are of two types: containers and type A packages - which could be for solid radioactive waste (which was mainly tested in the last years) or for liquid radioactive waste.

For the above mentioned packages, Testing and Nuclear Expertise Laboratory performs certification tests and batch tests or as called, Final Quality Control tests.

Tests and also methods, stands and equipments used for tests, comply with National Commission for Nuclear Activities Control and IAEA Regulations.

The activities in this field are checked and supervised by Quality Assurance Department, working under the incidence of the International Quality Standards.

Testing and Nuclear Expertise Laboratory elaborates Test Reports for the packages which have to be certified and Quality Certificates for the packages which are certified and have to be sent to the National Radioactive Waste Deposit or which must have an intermediate storage.

This year, it will be started the decommissioned activities for the Nuclear Reactor for research and radioisotopes production, whose operates it has been stopped since 1997. Considering this and the Testing and Nuclear Expertise Laboratory experience in testing radioactive material transport and storage packages, is required an entire strategy which is expected to find solutions concerning the specific tests for new kinds of radioactive material transport and storage packages.

ABSTRACT 167

Validation of Design and Design Changes for a Type B(U) Transport Package After Drop Testing Using Computational Analysis

SEAN DUVALL, PHIL ROBBINS, DAVID ROGERS

In order to determine the mode of failure of the containment system in the R7021 transport container, during IAEA drop testing, finite element analysis was carried out, using PATRAN for pre/post-processing and LS-DYNA as the solver. The structure was interrogated for strains in excess of the failure criteria, indicating a weld failure in the drain tube assembly. The paper includes benchmarking of the finite element model against the drop test results. It then explains how a series of design modifications were tested to determine their behaviour and the level to which they improved the design. Finally it shows how simplified and detailed finite element analysis has been used to prove the design for regulatory drop tests in support of a license application.

ABSTRACT 195

Evaluation of Inaccessible Surface Contaminations Inside of Large Components (2) Contamination Estimation from the Exterior Measured Dose Rate Distribution

YOSHIHIRO HIRAO, AKIKO KONNAI, HIROMITSU MOCHIZUKI, MASAMI ISOBE, NAOTERU ODANO

Japan's utilities who operate light water reactors have conducted a feasibility study of transporting large components (LC) by sea without being segmented for disposal or recycle purposes. Current transportation needs are mainly due to a component degradation requiring replacement, in which a used PWR steam generators (SG) is included. SG is so large and massive that it is not readily amenable to transport under the current Regulations. While it is apparent that SG is not essentially activated and contains only surface contamination, it is not certain that the SCO limits for inaccessible surfaces can be met due to non-uniform contamination; nor can the surfaces be readily surveyed without on-site dismantlement.

Some countries, however, have experiences of shipping SGs without segmentation in an unpackaged manner over the course of a decade. They have been shipped under special arrangements approved by competent authorities only when they are technically justified that contamination levels are grouped into SCO-II and the outermost shells are considered as the packaging wall equivalent to Type IP-II packages. Japanese study was also based on the former examples. Therefore, it is important to verify whether the contamination level fits within SCO-II under given conditions in Japan.

In this study, we have evaluated the inaccessible surface contamination levels from gamma dose rate distribution at the exterior surfaces. A dose rate distribution was measured in the axial and circumferential direction at a SG storehouse in domestic power plant. A gamma dose calculation is carried out using the point kernel integration code QAD-CGGP2R. First, a contamination of the tube bundle area is estimated by a dose rate of the exterior center. For the exterior of channel head or U-bend tube part, source contribution from each interior area is examined. An overall axial distribution of the exterior is calculated using a reference knowledge of the relative distribution among interior areas. Then, by comparing the above results with the measurement, adjusting the area distribution and repeating a calculation, a range of a contamination level is approximately determined for each area. Finally, compliance matters are discussed regarding this SG case.

PATRAM 2010 Posters : continued

ABSTRACT 198

Evaluation of Inaccessible Surface Contaminations Inside of Large Components (1) Measurements of Dose Rate Distribution around a Steam Generator

OSAMU SATO, HIROSHI SUZUKI

The heavy radioactive components, such as feed water heaters or steam generators, have been transported as unpackaged IP-1 for SCO-I and as "Special arrangement" for SCO-II in US and Europe. We have no experience of transportations of such components from Japanese NPPs. We have started investigation on the feasibility of the transport of the components to provide the flexible planning of the D&D or the maintenance. As a part of the study we have developed an evaluation method for the surface contamination deposited inside of the large components from the measured radiation dose rate distribution on the surface of the components.

The method was demonstrated by applying to the surface contamination evaluation of a replaced steam generator. The schematic flow of the method is shown in Fig.1. The dose rate distribution on the surface was measured by us-ing glass dosimeters in "Glass Badge" as shown in Fig.2. Three belts with the eight couples of glass dosimeters were bandaged around the body of the steam generator to measure the dose rate distribution around the shell of the steam generator as shown in Fig.3. One of a couple of glass dosimeters was covered with a lead shield and another was bare to subtract background components. The examination results show the applicability of the method to evaluate the surface contamination inside large components by measuring the dose rates from outside.

ABSTRACT 218

Brand New Fire Test Facilities at "BAM Test Site Technical Safety"

BERNHARD DROSTE, ARMIN ULRICH, JOERG BORCH

Fire Testing is an essential part of the hypothetical, cumulative mechanical and thermal accident test conditions that shall guarantee package safety in severe accidents. Following the guideline "Safety in technology and chemistry" BAM as a scientific and technical German federal institute operates a 12 km^2 large open air test facility for experimental testing of dangerous goods and their containments.

On an area besides the well-known 200 ton-drop test facility we recently put into operation new fire test facilities. The facility provides two fire test stands that are utilized with liquid Propane as fuel from a central earth-covered 60 m^3 LPG storage tank. From that storage the Propane is pumped via pipelines to the test stands where the gas is released from nozzles, and ignited by ignition burners. The fire exposed test facility areas are 12 m x 8 m. Fire test facility B (with gas release nozzles below water-covered ground) is designed for fire testing of containers that may burst during the test. Fire test facility A (with ring burner systems) is designed for heavy test objects up to 200 tons, e. g. for full-scale spent fuel casks.

The paper will present a detailed description of the facility, insight into first test performances, and results of calorimeter fire tests, using containers of various sizes, to verify the absorbed heat fluxes, demonstrating that regulatory fire test conditions are met.

ABSTRACT 345

Testing Packages for Transport and Storage of Radioactive Material (RAM), in Romania

GHEORGHE VIERU

Regulating nuclear and radiation safety is a national responsibility and, as a consequence, each Member States (MS) have to adopt the IAEA Recommendation on safety standards to be used in their national regulations and nuclear laws..

To assure a high level of safety and security in the use of Radioactive Material (RAM) and radioactive sources, the IAEA safety standards provide a consistent and reliable means of ensuring the effective fulfillment of obligation under the international conventions in nuclear field.

The transport and storage of radioactive material is an activity of maximum importance. One of the important safety measure is to prove the quality of the packages to be used for transport and storage of RAM.

Romania built a new testing facility for testing of packages type A, B and C.

The paper, to be presented at PATRAM 2010, will underline the main qualifications testing performed over the packages in order to assure the conformity with design and to achieve the goals of a safe practical activity in the field of transport and storage of RAM.

An overview over the new Romanian testing facility will be also presented.

The paper will present also the main characteristics of the radiation protection programme applied in the field of packaging, transport and storage of RAM in Romania as an important issue to protect people and the environment and for continuing utilization of nuclear facilities for peaceful scopes, for the benefit of mankind in a safe manner in which the IAEA Safety Standards playing a base role.

ABSTRACT 378

Construction of an Unyielding Target for Large Horizontal Impacts

DOUGLAS AMMERMAN, NEIL DAVIE, ROBERT KALAN

Sandia National Laboratories has constructed an unyielding target at the end of its 2000-foot rocket sled track. This target is made up of approximately 5 million pounds of concrete, an embedded steel load spreading structure, and a steel armor plate face that varies from 10 inches thick at the center to 4 inches thick at the left and right edges. The target/track combination will allow horizontal impacts at regulatory speeds of very large objects, such as a full-scale rail cask, or high-speed impacts of smaller packages. The load-spreading mechanism in the target is based upon the proven design that has been in use for over 20 years at Sandia's aerial cable facility. That target, with a weight of 2 million pounds, has successfully withstood impact forces of up to 25 million pounds. It is expected that the new target will be capable of withstanding impact forces of more than 70 million pounds. During construction various instrumentation was placed in the target so that the response of the target during severe impacts can be monitored. This paper will discuss the construction of the target and provide insights on the testing capabilities at the sled track with this new target.

PATRAM 2010 Posters : continued

ABSTRACT 414

A Systematic Analytical Approach to the Marine Transport of Heat-Generating Nuclear Packages

BENJAMIN ACKER

Sea transportation of radioactive materials (RAM) presents unique challenges, made greater when the material is of a heat generating nature. The structural integrity required of an INF ship for its buoyancy and its stability restricts natural air circulation within the holds. Consequently, a cooling system may have to be operated when the cargo comprises a package, or packages with a significant heat load.

In comparison with the regulations applicable to land transport of RAM, the regulations dedicated to sea transportation are less prescriptive. An accurate study of the heat effect and the efficiency of the cooling is therefore important as generally, transport packages are considered to be directly exposed to ambient conditions of temperature and insolation prescribed by the Transport Regulations. Enclosure within the hold of the ship distances the ambient conditions and imposes local environmental conditions which may result in higher package temperatures. With reference to the IMDG requirement, the package containment of any sea transportation must be maintained without controlling the hold environment. However, package temperatures which are not sufficiently high to affect containment could lead to other safety issues which have to be considered, such as the degradation of shielding material, or thermolytic hydrogen generation.

The justification of forced ventilation or refrigeration to cool the package during shipment at sea requires a comprehensive study combining the characteristics of the RAM package or packages, the 'containment' of the ship's hold, and the specification of any cooling system used. This is considerably more demanding than the analysis requirements for land transport of the equivalent package, and may require the application of advanced CFD techniques.

With more than 2000 packages shipped, our organisation has always placed the safety of the ship crew, the environment and the population which may be at close proximity to the ships as its first priority.

This paper illustrates how INS Ltd considers the thermal issue during a shipment of radioactive material, explaining the background with a brief summary of the regulations associated with this topic. The principle of defence in depth is demonstrated, focusing on the thermal issue and finally, the paper will be supported with some typical examples.

ABSTRACT 416

Stress-state Modified Strain Based Failure Criterion for Evaluating the Structural Integrity of an Inner Eutectic Barrier

DAVID HARDING, LILI AKIN, RICHARD YOSHIMURA, DAVID MILLER

A slight modification of a package to transport solid metal contents requires inclusion of a thin titanium liner to protect against possible eutectic formation in 10 CFR 71.74 regulatory fire accident conditions. Under severe transport regulatory impact conditions, the package contents could impart high localized loading of the liner, momentarily pinching it between the contents and the thick containment vessel, and inducing some plasticity near the contact point. Actuator and drop table testing of simulated contents' impacts against liner/containment vessel structures nearly bounded the potential plastic strain and stress triaxiality conditions, without any ductile tearing of the eutectic barrier. Additional bounding was necessary in some cases beyond the capability of the actuator and drop table tests, and in these cases a stress-modified evolution integral over the plastic strain history was successfully used as a failure criterion to demonstrate that structural integrity was maintained.

Empirical evidence shows that ductility in metals decreases with hydrostatic pressure. Brozzo, et al. in 1975 developed a failure index which matched test-to failure data at varying pressure stress states. Sandia National Laboratories modified this empirically-based ductile failure criterion with an exponent of 4 to better fit highly notched tensile specimens and still match previous test data. This criterion is compared to a critical "tearing parameter" value easily determined from a tensile test to failure:

$$TP = \int_0^{\varepsilon_f} \left\langle \frac{2\sigma_T}{3(\sigma_T - \sigma_m)} \right\rangle^4 d\varepsilon_p$$

The Heaviside brackets only allow the evolution integral to accumulate value when m is the average of the threesthe maximum principal stress is positive (principal stresses), since failure is never observed under pure hydrostatic pressure, where the maximum principal stress is negative.

Detailed finite element analyses of myriad possible impact orientations and locations between package contents and the thin eutectic barrier under regulatory impact conditions have shown that not even the initiation of a ductile tear occurs. Although localized plasticity does occur in the eutectic barrier, it is not the primary containment boundary and is thus not subject to ASME stress allowables from NRC Regulatory Guide 7.6. These analyses were used to successfully demonstrate that structural integrity of the eutectic barrier was maintained in all 10 CFR 71.73 and 71.74 regulatory accident conditions. The NRC is currently reviewing the Safety Analysis Report.

SUSTAINING SHIPMENTS

ABSTRACT 44

TRADAWEB - A Web System to Collect and Process Data on Transport of Radioactive Material

SANDRO TRIVELLONI, PATRIZIA CAPORALI, GIORGIO PALMIERI

Italian licensed carriers are requested, by the national legislation, to provide data on their transport of radioactive material on regular basis. The data to be provided are established by a fixed format and contain the type of package, activity of the material, TI (if applicable), package dimensions, etc. To collect the information on systematic mode the carriers have different ways :

1.use a software called TRIME2004 provided by ISPRA on free basis;

2.use their own software according with the fixed format (this is the case for carriers that transport thousands of package/year);

3.use a data sheet

PATRAM 2010 Posters : continued

All the data collected by the carriers shall be transmitted to ISPRA every three months. To facilitate the transmission of the data and to improve the "quality" of the data a web system called TRADAWEB (TRAnsport DAta WEB) has been developed and implemented at ISPRA web site.

The carrier can access to a dedicated area at ISPRA web site by a user name and password and carries out different operations by a user-friendly system. During the data transmission TRADAWEB performs some checks of the data and at the end of the transmission send back to the carrier a report containing the results of the checks. If some mistakes are discovered by the system (i.e. if the activity declared in a Type A package is grater than A1 or A2) a warning will appear on the report indicating which is the record containing the mistake. In this case the carrier can evaluate the warning, make the correction and send again the data. TRADAWEB was established on 1st January 2008 and up to 31st of December 2008 more than 60% of the licensed carriers have used this system to transmit the data. The system guarantees a high quality and the immediate availability of the data that can be elaborated and stored into ISPRA database.

The poster will show how the transmission of data has a legal acknowledgment and some elaborations of the data related to radioprotection aspects in transport.

ABSTRACT 263

New INF3 Ships

PAULINE WOODS

International Nuclear Services manages a fleet of INF 3 class marine vessels through a subsidiary Pacific Nuclear Transport Ltd (PNTL), of which INS is the majority shareholder. PNTL is the World's most experienced transporter of nuclear materials, having travelled more than 5 million nautical miles without a single release of radioactive material.

The last PNTL INF3 vessel was built in 1987 and therefore investment was committed to introduce a new fleet of 3 ships.

The brief was to design a new fleet of vessels that would utilise modern technologies with innovative design and minimise environmental impact, whilst maintaining the existing high safety standards of the current fleet. The paper will discuss the operational and environmental improvements that have been introduced in the new ships.

Although all new builds are required to comply with the regulatory requirements of the UK and International regulators where appropriate, our design, where possible incorporates any known potential future regulatory changes, whilst also incorporating modern technology and minimising environmental impacts effected from the operation voyages

The first of class Pacific Heron was handed over to PNTL at Tamano Shipyard, Japan on the 10th of April 2008 and is indeed a truly innovative and modern vessel - the pride of the PNTL Fleet! The second ship, the Pacific Egret, was launched in January 2010 and is due to be completed and delivered in June 2010, with the third ship due to be delivered in October 2010.

INTERIM STORAGE

ABSTRACT 77

Strategy for use of Dual-Purpose Metal Casks in the Dry Storage Installation at Hamaoka Nuclear Power Station

TAKEFUMI MIKATSURA, KEISUKE ASAI, YUUICHI FUJIMORI, TETSUHIRO ITOU, TAKESHI NARAMA

At-Reactor Dry Storage (ARDS) is a simple and reasonable solution for capacity shortage of fuel pools in nuclear power stations. Two ARDSs, which use metal casks, have been installed at nuclear power stations in Japan. Because the metal casks used in the existing ARDSs are dedicated to use for storage not for transport outside the stations, the spent fuels inside them will be reloaded into transportation casks just before carrying-out to reprocessing plants.

The first off-site Interim Storage Facility (ISF) has been designed and under review of safety analysis. It does not have a hot cell to handle radioactive materials, and will employ a dry storage method with Dual-Purpose Metal Cask (DPMC), which is designed and licensed for both transportation and storage of spent fuels. The loaded spent fuels will be transported to reprocessing plants without breaking the seals of DPMCs and reloading them into any other transportation casks.

We, Chubu Electric Power Co., Inc., have been designing an ARDS for the Hamaoka Nuclear Power Sztation. It is the first project to introduce DPMC into a nuclear power station to streamline ARDS in Japan. Since we do not need to reload the spent fuels into transportation casks before carrying-out outside the power plants, exposure dose on workers and possibility of a radioactivity-releasing accident can be kept down. The design considerations of the ARDS with several types of DPMCs have been underway for a few years to satisfy the requirements on manufacturing, operation, monitoring and maintenance policy equivalent to those of the ISF. It will contain 61 DPMCs, which corresponds approximately to 700 tU of BWR spent fuels, and start to operate by FY 2016 (current plan).

Here we present a conceptual design of the ARDS and several types of DPMCs for Hamaoka Nuclear Power Station.

ABSTRACT 216

An Innovative Solution for the Transportation and the Storage of Used Fuel

JUSTO GARCIA, HELENE SONNENMOSER

TN International and Transnuclear Inc. (AREVA group) have proposed for more than 2 decades the leading storage systems of UOX used fuel in use today:

- the TN™24 cask family which features by metallic casks used both for the transport and storage of used fuel.
- The NUHOMS® system which is the most popular storage overpack in the United States. The NUHOMS® system allows naturally the transportation of the used fuel.

These systems have mainly been supplied in Europe: Belgium, Switzerland, Germany, Italy, but also in the US and in Japan. The PWR, BWR or VVER fuel characteristics may have various enrichment value up to 5%, various cooling time down to 2 years and various burn-ups up to 60 000 MWd/tU.

PATRAM 2010 Posters : continued

Facing the current international trend towards expanding Used Fuel Interim Dry Storage capabilities with higher performances, TN International and Transnuclear Inc. decided to launch an extensive innovation process to create the new generation of transport and storage casks.

The TN NOVA solution is the result of an extensive process to develop innovative and cost effective storage overpack which could be transported to the storage facilities as well as other sites like reprocessing facilities or geological repositories.

The first application of the TN NOVA system has been sold in January 2009 to Axpo AG (previously NOK) for the storage of used fuel produced by the NPP of Leibstadt in Switzerland.

The purpose of this paper is to present our experience, and furthermore to underline the main advantages of the TN NOVA overpack.

ABSTRACT 236

Thermal and Shielding Benchmark Analysis of Dry Storage Cask in Tokai no.2 Power Station

MASAHIRO YAMAMOTO, TAKESHI FUJIMOTO,
NAOSSHI AOTA, KAZUO IWASA

Tokai No.2 Power station (BWR 1,100MWe) commenced spent fuel dry storage in 2001 and 915 spent fuel assem-blies are stored in 15 metallic dry casks (61 assemblies a cask) at the end of 2009.

The objective of this study is to investigate the margin of current dry cask design by comparison of measured data of dry storage casks in Tokai No. 2 power station with calculation and to contribute a more practical cask design . The targets of this benchmark are thermal and shielding analysis. For this purpose, temperature and dose equiva-lent rate of the dry casks were measured in detail.

The temperature data were measured at several points of the surface of the target cask, the adjacent casks, the ambient air, etc. in the dry storage facility. The dose equivalent rate data were measured at several points of the surface and 1 m from the surface of side, top and bottom of the target cask.

The thermal and shielding analysis based on the current design method were carried out for the comparison with the measurement data. The thermal analysis was performed using ABAQUS code and the shielding analysis was performed using DOT3.5 code.

It was confirmed that the current methods of the thermal and shielding analysis were conservative enough as a re-sult of the comparison.

Furthermore, additional analysis was performed to adopt more precise data of not only the dimensions and the ma-terial density of the shielding materials by using the inspection records of the storage cask, but also the decay heat and the radiation source strength by using the axial burn-up distribution calculation data of the spent fuel assem-blies. The comparison between measured data and calculation is shown in table 1. As a result of this analysis, the design margin based on the conservativeness of input data for analysis was confirmed.

An advanced methods of the thermal and shielding analysis of the dry storage cask are proposed based on these evaluation results.

Table 1 Comparison between measured data and calculation

Axial position	Top	Center	Bottom
Temperature (Cask surface) [°C]			
Measured data	36.5	46.0	35.9
Calculation	53.1	70.7	58.2
Dose equivalent rate (Cask surface)[μSv/h]			
Measured data	20.0	34.0	6.6
Calculation	49.0	57.2	20.6

ABSTRACT 307

Development of Spent Fuel Transportable Storage Cask (1) Design of the Cask

TSUTOMU MATSUMOTO, KAZUO KAWAKAM,
YOUJI SAKATA, DAISUKE SHIMIZU, HARUAKI KIKUCHI

1. Introduction

We report progress of the design of spent fuel transportable storage cask which we reported at PATRAM2001. Major features of this cask are as follows.

- It has a simple body structure with forged carbon steel for the gamma ray shielding and propylene glycol water solution for the side neutron shielding.
- The fuel basket is built of machined aluminum alloy plates, and there is no welding construction part. Borated aluminum alloy plates are used for neutron-absorbing material.
- The lid closure system is consists with double lids and metalic gaskets on each lid. The seal function is confirmed by monitoring the pressure between the double lids during the storage period.
- The third lid with elastomer O ring that make a sealing boundary is installed while transportation.

A detailed design of the cask was performed according to the progress of the intermediate storage plan. Some tests were performed.

2. Outline of cask

Externals of the cask are the cylindrical geometries, and main bodies without the shock absorber are about 2.5m in diameter, and about 5.4m in length. The weight as the transportation cask is about 110 tons.

The cask is held on the transport frame in horizontal position during transportation as shown in Figure 1.The third lid and shock absorbers are installed.

The main body is a double cylinder consists of the main body and the outer shell. Neutron shieldings are the propylene glycol water solution filled with the annulus between the body and the outer shell and the resin set up in the lid plate and the bottom plate. Shock absorbers were designed to reduce the impact acceleration smaller than the existing casks by using comparatively soft woods as shock-absorbing material.

The cask is stored on the storage skid in vertical position as shown in Figure 2. The third lid is removed, and the protection cover is installed during the storage period.

3. Test item

Following tests were performed, and the performance was confirmed.

- Drop test by scale model
- Heat transfer test of propylene glycol water solution
- Corrosion test of propylene glycol water solution body material

Fig. 1 Cask in Transportation

Fig. 2 Cask in storage

ABSTRACT 308

Development of Spent Fuel Transportable Storage Cask (2) 9m Drop Tests by Scale Model

YUICHI MOTEGI, KAZUO KAWAKAMI

1. Introduction

In the design of a spent fuel transportable storage cask, it is required that the cask keeps structural soundness against the 9m drop test. In order to minimize impact acceleration, larger shock absorbers were designed and soft wood that has low crushing load property was used for shock-absorbing material.

To confirm the design validity, the drop test was performed using 1/4 scaled model. In addition to the drop tests, the dynamic analysis by three dimensional dynamic analytical code LS-DYNA was conducted. The validity of a dynamic analysis technique was confirmed by comparing analysis results to the drop test results.

2. Outline of drop test

The shock absorbers and the main body of the test specimen were almost exactly downed the scale by 1/4. The contents were simply simulated the weight.

The height of the drop was 9m, and the direction of the drop was two cases (the top vertical drop and the horizontal drop). The impact acceleration of the cask was measured with the acceleration sensor. And, the deformation of shock absorbers and the strain of lid bolts were measured.

3. Test results and comparison with analysis results

Figure 1 and Figure 2 show the cask after the drop. The drop posture was excellent, and there was very little inclination. It was confirmed that there was no damage on the cask body, and there was no problem about structural soundness of the cask.

Measurement results of the cask body acceleration and the analysis results by LS-DYNA are shown in Table 1. Analysis results indicate a little conservative values about the maximum acceleration. And, shock absorber deformation results and the analysis results are shown in Table 2. Analysis results are corresponding to test results well.

The design validity and the validity of the analysis by LS-DYNA were confirmed.

Tab. 1 Maximum acceleration (unit : m/s2)

Drop direction	Test results	Analysis results
Top vertical	1.7×103	2.4×103
Horizontal	1.8×103	2.2×103

Tab. 2 Maximum deformation (unit : mm)

Drop direction	Test results	Analysis results
Top vertical	55	57
Horizontal	66	67

Fig. 1 The cask after the top vertical drop

PATRAM 2010 Posters : continued

Fig. 2 The cask after the horizontal drop

ABSTRACT 309

Development of Spent Fuel Transportable Storage Cask (3) Heat Transfer Behavior Study on Convection of Propylene Glycol Water Solution

HARUAKI KIKUCHI, KAZUO KAWAKAMI, TSUTOMU MATSUMOTO

1. Introduction

Our designed spent fuel transportable storage cask uses propylene glycol water solution (PG water) as the side neutron shielding material. There is an expansion chamber at upper area of side neutron shielding. Expansion chambers are surrounded with PG water. The convection of PG water is the heat transfer mode from the cask body to the outer shell..

In order to confirm this convective heat transfer performance, the heat transfer tests in vertical and horizontal positions were performed respectively. In addition, the effectiveness of the heat flow analysis technique was confirmed by comparing the analysis results by the heat flow analysis code FLUENT with the test results.

2. Outline of test

The test body for the vertical position test was 1/1 scaled model of 1/4 (90degree) portion of the actual side neutron shielding structure. In addition, the model had an expansion chamber and boss of trunnion. (Refer to Figure 1.)

The test body for the horizontal position test was 1m length of 1/2 scaled slice model, simulating the middle part of actual side neutron shielding structure.

Major test conditions are as follows.

- PG water was used for the fluid.
- The heater with uniform heat generation density was put in inner cylinder.

3. Test results and comparison with analysis results

As the test results for the vertical position, the axial temperature gradient was produced in the test body due to the convection of PG water. Because the difference of temperature between cylinders is constant along the height, there was no decrease in the heat transfer coefficient around the expansion chamber, and it was confirmed to have an highly heat transfer performance. (Refer to Figure 2.) And, analysis results of axial temperature distribution showed a good agreement with the test results, and the temperatures of analysis results were little higher than the test results.

As the test results for the horizontal position, the circumferential temperature gradient in test body was also produced, and the temperatures of analysis results were higher than the test results.

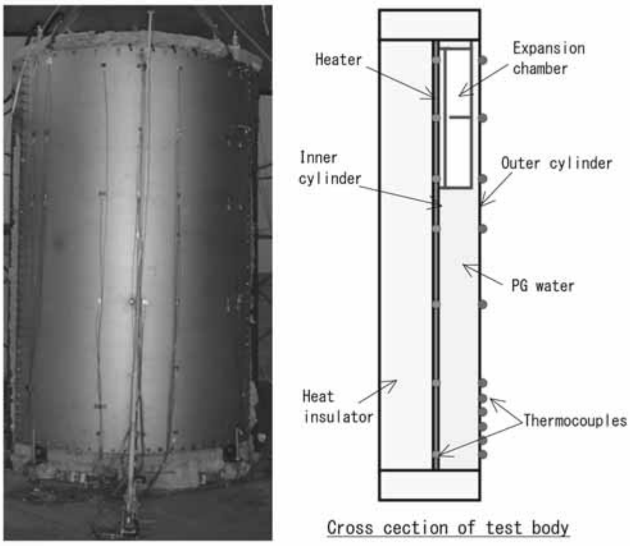


Fig.1 Test body for vertical position

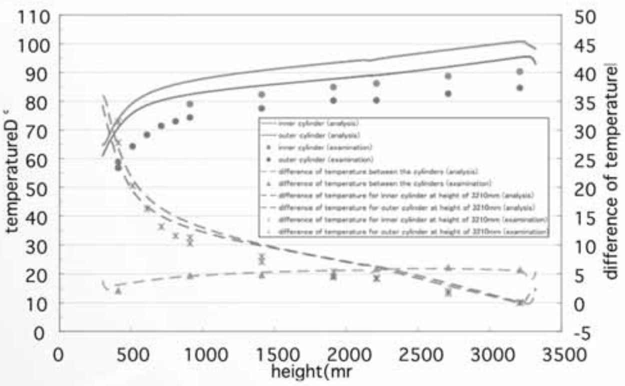


Fig. 2 Comparison between analysis results and test results

ABSTRACT 310

Development of Spent Fuel Transportable Storage Cask (4) Corrosion Test for Body Materials Containing the Propylene Glycol Water Solution

YOUJI SAKATA, KAZUO KAWAKAMI, SHINYA OHISHI

1. Introduction

Our designed spent fuel transportable storage cask uses propylene-glycol-water-solution(PG water) for the side neutron shielding material. The PG water is antifreeze liquid under low temperature condition. Boric acid is added to PG water to enhance shielding performance.

Because the body containing PG water is required to keep the structural integrity under hot condition during a long storage term,

PATRAM 2010 Posters : continued

the durability of the metal material used for body was confirmed by the stress corrosion cracking(SCC) tests ,and the corrosion potential test.

2. Outline of test
Materials that contact with PG water are stainless steel weld overlay (308,309) and two-phase stainless steel cladding (SUS329J4L).

Corrosion resistance of these materials were tested by SCC test using U-bend test specimens. The liquid solution was mixed formic acid and boric acid.The formic acid was added to PG water to simulate the degradation of the PG water during a long storage term Test period was 2 years and 6 months.

And, as another approach, these materials were tested on corrosion resistance by corrosion potential test.

The cyclic potentiodynamic curve was measured by potentiodynamic method to evaluate the repassivation behavior.

In addition, the test was conducted by using simulated test specimens with actual scale as-welded surface that might affect the pitting and crevice corrosions.

3. Test results
The SCC test results showed that there is no indication of SCC on the stainless cladding weld overlay and two-phase stainless cladding. Therefore, these materials were confirmed to be free from the possibility of SCC.(Refer to fig.1)

The results of corrosion potential test showed that these materials were free from the possibility of pitting and crevice corrosions. (Refer to fig.2)

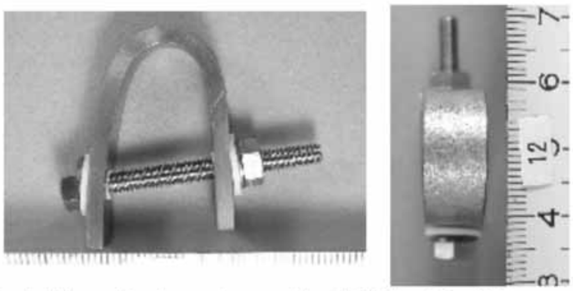


Fig.1 U-bend test specimen after SCC test for 2.5 years

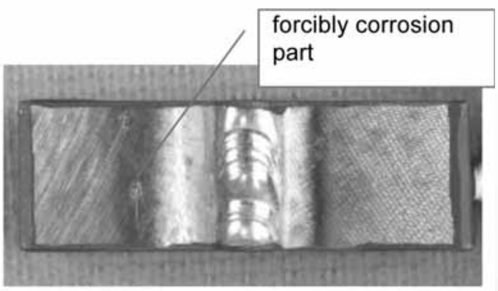


Fig.2 specimen after corrosion potential test

SECURITY

ABSTRACT 158

An Overview on Security and Safety in Transport of Radioactive Material in Romania

GHEORGHE VIERU

The transport of radioactive material (RAM) is an important part of the management of global nuclear security and safety regime. Each Member States (MS) is responsible for the security and safety of radioactive material during transport, since radioactive material is most vulnerable during transport. As IAEA stated, the vulnerability of a package during transport highlights the absolute need for adequate security in transport. Also, "regulating safety is a national responsibility since radiation risks may transcend national borders so international cooperation will serve to promote and enhance safety globally by exchange experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences" (IAEA Safety Standard TS-R-1).

The paper will present main aspects related to the transport of RAM, including NORM (Naturally Occurring Radioactive Material), the risk and safety assessments during transport, the adequate security measures taken. The routes of transport of RAM and NORM in Romania, the environmental impact and radiological consequences following a potential accident (event) during carried out this activity will be also presented, since the protection of the people and the environment are of primary importance and the essential goal to be achieved.

It is also to be noted that this paper considers the main results of the scientific research carried out by the author as CSI (Chief Scientific Investigator) for Romanian Research Contract concluded with IAEA Vienna.

ABSTACT 424

Making it Happen: The Current Status of US Origin and Gap Nuclear Fuel Removals

CHARLES E. MESSICK, JERALD L. TAYLOR

This paper provides a brief update on the National Nuclear Security Administration's Global Threat Reduction Initiative programs which return U.S.-origin foreign research reactor fuel and Gap nuclear materials. It will discuss recent program accomplishments, initiatives, and future activities. The goal of the two programs is to recover U.S.-origin and Gap nuclear materials that could be used by terrorists to make an improvised nuclear device.

PATRAM 2010 Posters : continued

PACKAGE TECHNOLOGY AND DESIGN

ABSTRACT 22

Early Testing of Drum Type Packagings – The Lewallen Report

ALLEN SMITH

The DOE Rocky Flats Plant developed a packaging for shipment of Pu in the early 1960â€™s, which became the DOT 6M specification package. These design concepts were employed in other early packagings. Extensive tests of these drum type packagings were performed at Savannah River Laboratory (now Savannah River National Laboratory) in 1969 and 1970. The results of these tests were reported in â€œDrum and Board-Type Insulation Overpacks of Shipping Packages for Radioactive Materialsâ€ , by E. E. Lewallen. This report was foundational to design of subsequent drum type RAM packaging. Design guidance developed as part of this study is referenced and incorporated into currently certified packagings. This paper summarizes this important early study of drum type packagings.

ABSTRACT 29

Long-term Aging Study of Polyurethane Foam and Wood

SHARON WILLIAMSON, JAY FISCHER

This paper describes results from a long-term aging study of LAST-A-FOAM® Polyurethane Foam and multiple wood species in two storage conditions over ten years. Polyurethane foam and wood is used in transportation and impact mitigation applications that have certificates for decades of use. This on-going study shows the material aging characteristics and their fitness for long-term use in these applications.

The storage conditions are (1) extreme environmental exposure of foam and wood in a subterranean concrete vault with an open grate at the top and (2) protected exposure of polyvinyl film wrapped polyurethane specimens in an outside, weather-tight, stainless steel enclosure at Portland General Electricâ€™s Trojan long-term spent fuel storage facility to mimic conditions of the ISFSI impact limiter. Material from both storage conditions is removed and tested at intervals, and results are compared with previous tests.

ABSTRACT 30

CASTOR® 1000 – a new cask for VVER-1000 fuel assemblies

FELIX THOMAS, THOMAS FUNKE, BERNHARD KUHNE

GNS have many years of experience in the development and manufacture of specialist storage and transport casks for different spent nuclear fuels. This experience enabled GNS to develop CASTOR® casks for successful storage of spent fuel from Russian reactors in Germany and the Czech Republic.

CASTOR® casks are licensed for both storage and transportation and have been in continuous use since 1983.

The casks used in Germany and the Czech Republic are the CASTOR® 440/84 type designed to store fuel elements from the Russian VVER 440 type reactor. Based on the success of these casks GNS have developed the CASTOR® 1000 cask designed for fuel assemblies from Russian VVER 1000 reactors with high burn-up and a maximum fuel rod enrichment of up to 5 wt-% U-235.

The CASTOR® 1000 cask shares many design features with the CASTOR® 440/84 cask. The primary features are:

The cask body consists of a cylindrical thick-walled one piece casting, including the bottom, made of ductile cast iron (DCI). Externally, circumferential fins are machined in the body to improve the heat removal. For neutron moderation, polyethylene filled, axial bore holes are distributed uniformly in the cask wall. Trunnions are attached to the cask body top and bottom for handling.

The lid system consists of primary and secondary lids to provide the double barrier system required to fulfil the long-term storage criteria. Both lids are independently sealed by metal-O-rings and fastened by screws.

Fuel elements are stored in a purpose built basket designed with individual compartments which is inserted into the cask. The basket is manufactured using steel and borated materials for structural integrity and neutron absorption to ensure sub-criticality. Heat removal is achieved using aluminium plates.

Corrosion protection of the cask cavity and the sealing surfaces is made by nickel coating. The outer surface of the cask, including the surface of the secondary lid and the fins, is protected by a multilayer paint system, which can be decontaminated easily.

The paper deals with the development, design and manufacturing challenges

ABSTRACT 31

9977 and 9978 Shipping Packages – Design, Criticality and Shielding Consideration

DEBDAS BISWAS, RAYMOND REED, GLENN ABRAMCZYK

Savannah River National Laboratory (SRNL) developed two new, Type B, state-of-the-art, general purpose, fissile material shipping containers. They are designated as 9977 and 9978 and are mainly used as replacements for the U.S. DOT specification 6M container which were phased out in September 30, 2008 due to non-compliance with current requirements of 10 CFR 71. DOT 6M Specification Packages were used extensively for the transport of Type B quantities of fissile radioactive materials (uranium and plutonium metals and compounds) since the 1960â€™s. The 9977 and 9978 shipping packages were designed as a cost-effective, user-friendly replacement. The packages accommodate plutonium (< or =4.4 kg), uranium (< or =13.5 kg) and other special nuclear materials in bulk quantities and in many forms with capabilities far exceeding those of 6M.

Allowed package contents were determined accounting for nuclear criticality, radiation shielding, and decay heat rate. The Criticality Safety Index (CSI) for the package is 1.0. Radiation shielding analyses demonstrated that the 9977 and 9978 comply with the federal regulations for non-exclusive use.

The 9977/9978 General Purpose Fissile Package (GPFP) is a robust single containment package, capable of transporting plutonium and uranium metals and oxides. The packages are designed to ship radioactive contents in several configurations; Radioisotope

PATRAM 2010 Posters : continued

Thermoelectric Generators (RTGs), nested food-pack cans, site specific containers, and DOE-STD-3013 containers. These packages provide a high degree of single containment. The single containment 9977/9978 design meets the latest revision of 10 CFR Part 71, which eliminated the requirement for double containment for plutonium shipment. Each shipping package includes a 35-gallon stainless steel outer drum, insulation, a drum liner, and a single containment vessel (CV). The 9977 includes a 6-inch ID CV while the 9978 includes a 5-inch ID CV.

A nuclear criticality safety and the dose rate evaluation demonstrate the safe configurations of the new shipping container for plutonium and uranium metal/oxide loading under various conditions for the Safety Analysis Report for Packaging (SARP). The evaluation is in compliance with the performance requirements of 10 CFR 71.55 and 71.59 for criticality and 10 CFR 71.47 for shielding using the validated SCALE and MCNP codes.

ABSTRACT 35

Development and Study of New Shielding Materials

ALEXANDR MASLOV, YURY METELKIN, VLADISLAV ORLOV, OLEG YUFEROV, VICTOR SERGEEV

Investigations carried out during last five years were directed to development and study of new shielding materials with higher absorbing ability of gamma radiation. Materials developed are aimed to be used in transport, storage and disposal containers of spent nuclear fuel and radioactive waste. As a result we have got two new materials, in which high ability to absorb the gamma radiation is provided due to presence of the uranium dioxide in their composition. There are the high-density radiation shielding composition (concrete) with density of 6.5 g/cm3 and the high-density CERMET (uranium dioxide and stainless steel) with density of 8.3 g/cm3. Application of the new shielding materials for casks will provide high efficiency of gamma and neutron protection due to high density and high content of oxygen in their compositions.

ABSTRACT 42

Mechanical Assessment Within Type B Packages Approval - the Application of Static and Dynamic Calculation Approaches

STEFFEN KOMANN, MARTIN NEUMANN, VIKTOR BALLHEIMER, FRANK WILLE, MIKE WEBER

This paper demonstrates exemplarily how numerical and experimental approaches can be combined reasonably in mechanical assessment of package integrity according to the IAEA regulations. The paper will also concentrate on the question how static mechanical approaches can be applied, and what their problems are in relation to dynamic calculation approaches.

Due to the local character of the interaction between the puncture bar and the cask body it is possible to develop a dynamic numerical model for the 1m puncture drop which allows an appropriate simulation of the interaction area. Results from existing experimental drop tests with prototype or small scale cask models can be used for verification and validation of applied analysis codes and models. The link between analysis and experimental drop testing is described exemplarily by considering a regulatory 1m puncture bar drop test onto the cask body of a recently approved German HLW transport package.

For the 9m drop test of the package it is difficult to develop a dynamic numerical model of the package due to the complexity of the interaction between cask body, impact limiters and unyielding target. Dynamic calculations require an extensive verification with experimental results. The simulation of a 9m drop of a package with impact limiters is thereby often more complex than the simulation of a 1 m puncture drop onto the cask body. A different approximation method can be applied for the consideration of dynamic effects on the impact loading of the package. In a first step maximum impact force and rigid body deceleration of the cask body during the impact process can be calculated with simplified numerical tools. This rigid body deceleration can subsequently be applied on a verified static numerical model. Dynamic effects, which can not be covered by the static numerical analysis have therefore to be considered by using an additional dynamic factor. The paper describes this approach exemplarily for a 9 m horizontal drop of a typical spent fuel cask design.

ABSTRACT 50

Handling and Packing of Damaged Spent Fuel

RONNY ZIEHM, GEROLD SIMON, SASDCHA BECHTEL

Damaged spent nuclear fuel has been stored in a separate area of the spent fuel pools at NPP Ignalina, Lithuania, near the reactor core. With the schedule of the plant shut down, these damaged fuels cannot remain in the pond and must be prepared for intermediate storage, a period of about 50 years, such as the intact fuels and for transportation to other storage facilities or reprocessing plants. Often a classification is necessary to distinguish between the different types of defects, so subsequently the appropriate method and techniques can be selected.

The developed concept and techniques for the performance will be demonstrated for the damaged fuels at NPP Ignalina in Lithuania. During the operation in the reactor, the gas leaked fuels are separated and stored in special single storage containers. Most of these fuels have only small cracks so they must not be encapsulated in special cartridges.

With the classification system, fuels are separated when there is a suspicion of possible fuel release which must be considered in the safety analysis. The tight fuels are stored directly in containers, i.e. CONSTOR® RBMK 1500/M2 from GNS, without special overpacks. The fuels which are suspicious of fuel leaking are packed in special cartridges and then stored in the CONSTOR®.

The layout of these cartridges must consider the handling in the pond, the long storage period without corrosion and radiolysis as well as the withstanding of accidental transportation situations.

Special provisions are needed for heavily damaged fuel. The heavily damaged fuel assemblies will be transferred to the working area, where they are treated underwater in a special working tray which is equipped with tools for cutting, sawing, EDM etc. For cleaning up the potential released fuel in the working area, a debris collection system is used. All the released fuel is collected inside metal filters which can be stored via cartridges in casks when fissile material is suspected.

The presented concept is not only a special application for NPP Ignalina, it is developed more generally to solve these problems with damaged fuel at other nuclear power plants applying these proven techniques.

PATRAM 2010 Posters : continued

ABSTRACT 60

New Construction Material on a Base of Depleted Uranium Dioxide for Radioactive Wastes and SNF Multipurpose Casks

VITALY GOTOVCHIKOV, VICTOR SEREDENKO, DMITRY KOLEGOV, SERGEI KOLEGOV, OLESYA TOKAREVA

The production technology of high-performance radiation-protective material (CERMET) on the basis of granulated depleted uranium dioxide (UO2) and steel was developed.

High specific density value of CERMET (9.2-9.4 t/m3) along with equal distribution of neutron absorbers stipulate for outstanding shielding properties, which have determined the main application field of the material – manufacturing of multipurpose casks for RAW and SNF.

By now the following R&D works have been conducted:

- Thermodynamics analysis of U- system within the temperature range of 300 – 4000 K in different gaseous mediums, resulted in determination of temperature stability limits for uranium oxides of different composition.
- Test for adjustment of UO2 melting technology in high-frequency induction furnace with cold crucible (d=120 mm), resulted in generation of melted UO2 samples with density close to theoretical.
- X-ray phase analysis, confirming the possibility of melted UO2 generation from U3O8 by means of reduction smelting. This will allow utilization of U3O8 (product of depleted uranium hexafluoride conversion) as raw material for the production of melted and granulated depleted UO2.
- Tests for UO2 alloying with neutron sorbates (gadolinium oxide or other components) in the time of melting.
- Construction of a pilot plant destined for the adjustment of the technology and shifting it to continuous operation.
- Test production of cast CERMET samples containing up to 50 wt% of UO2 by means of diffusion casting. Generated samples of different configuration were investigated in terms of microstructure and physical-chemical characteristics.

The above mentioned steps revealed the major advantages of CERMET with respect to multipurpose casks for SNF manufacturing:

- Enhanced neutron and gamma-radiation protection;
- Minimization of walls thickness and weight of a CERMET cask;
- Application of CERMET casks under more severe conditions, inapplicable for casks of traditional materials (steel, concrete, cast iron).
- Preservation of CERMET casks integrity under extreme conditions (fire, accidents, acts of terror);
- Possibility of efficient utilization CERMET casks for long-term underground storage;
- Economically efficient recycling of depleted uranium and low-radioactive steel amounts stored.

ABSTRACT 99

Development of Extruded Borated Aluminum Material for Basket of Transport/Storage Casks

JUN SHIMOJO, KEISUKE UMEHARA, AKITO OISHI, HIROAKI TANIUCHI, TADASHI NAKAYAMA

Basket material of transport and storage cask is required to have enough structural strength under transport and storage conditions to maintain safety functions such as sub-criticality and heat removal. The material is also preferable to be light to reduce the weight of the component because it is very important to improve the efficiency of transport and storage by increasing the number of fuel assemblies to be loaded in a cask. Aluminum is the suitable material for basket due to its low density and high thermal conductivity resulting in weight reduction and better heat trasnffering capability.

The aluminum plate with 1% borated, 1%B-A6061-T651 was developed and registered to the code case of Rules on Transport / Storage Packagings for Spent Nuclear Fuel issued by the Japanese Society of Mechanical Engineers. In addition to this material, extruded borated aluminum material has been developed. New material, which is about 1% borated A3004 aluminum alloy with Mn and Mg added rather more than those of standard alloy, guarantees high strength in elevated temperature, and is excellent especially in designing basket of PWR fuels because it can offer various cross-sectional shape by extruding.

Though boron content is limited up to around 1% to maintain material properties, such as thermal conductivity and elongation, as same as base alloy, the newly developed aluminum alloy has high neutron absorbing performance because of its high content of boron-10 by using enriched boron and the thermal conductive property is as same as that of base aluminum alloy since boron content is limited up to around 1%. From the point of view for design requirements for the cask basket, mechanical properties such as the long term creep behavior, strength in elevated temperature, effects of aging during storage, etc. were evaluated. The material exhibited excellent mechanical properties in elevated temperature, and demonstrated the Larson Miller Parameter of more than 12000 which is equivalent to 60 years storage.

Transnuclear's design transport and storage casks will be designed based on these data. These evaluated results are discussed in the paper.

ABSTRACT 102

Type B Cask Evaluation Using Advanced FEA Technology

MIRZA BAIG

A radioactive Type B cask has been designed and analyzed using the latest finite element analysis (FEA) technology. The stainless steel/lead/stainless steel wall cask, which is protected by polyethylene impact limiters, has been qualified for the normal conditions of transport and hypothetical accident test conditions of U.S. regulation 10 CFR 71 using a combination of explicit and implicit finite element software packages. The use of contact element technology for modeling the composite wall allows incorporation of the thermal effects more accurately than it was possible in the past with approximate analyses. The analytical process results in a safe, economic, and efficient design of the cask.

PATRAM 2010 Posters : continued

ABSTRACT 107

Technical and Economical Impacts on Transport and Storage of Over 5% Uranium Enriched Fuel for Next-Generation Light Water Reactor

TOMOFUMI YAMAMOTO, KOUJI HIRAIWA, HIDEO SONEDA, TATSUHIRO YOSHIZU

The Next-Generation Light Water Reactor development program was launched in Japan in April 2008. The primary objective of the program is to cope with the need to replace existing nuclear power plants in Japan after 2030. The reactors to be developed are also expected to be a global standard design. Several innovative features are envisioned, including a reactor core system with uranium enrichment above 5%, a seismic isolation system, the use of long-life materials and innovative water chemistry, innovative construction techniques, safety systems with the best mix of passive and active concepts, and innovative digital technologies to further enhance reactor safety, reliability and economics. In the first 3 years, a plant design concept with these innovative features is established and the effectiveness of the program is reevaluated. The major part of the program will be completed in 2015.

Next-generation LWRs should support a longer operation cycle of 24 months (vs. 13 months with conventional reactors in Japan), which will achieve the world's best capacity factor. For efficient and economical operation with this longer operation cycle, reactors must use fuel with a uranium enrichment level greater than 5% to increase the burnup from the conventional level of around 50 GWd/t to around 70 GWd/t. This will have the additional benefit of reducing the spent fuel volume by roughly 30-40%.

The use of fuel with over 5% enriched uranium, is not yet commercially practical anywhere in the world. Therefore, before moving on full-scale development activities, it is vital to study the feasibility and validity of associated technologies.

A quantitative assessment was conducted to clarify economical advantages of the use of fuel with over 5% enriched uranium including transport and storage of radioactive materials such as UF6, fresh fuel and spent fuel. Technical and regulatory issues in the transport and storage field were also addressed.

ABSTRACT 194

On the Environmental Isolation and Seismic Resistance Characteristics of the HI-STORM 100U Underground Fuel Storage System

K.P. SINGH, C.W. BULLARD, J. ZHAI, W.S. WOODWARD

In December 2009, the HI-STORM 100U became the first underground ventilated MPC storage system in the industry certified by the USNRC (Docket No. 72-1014). The obviously beneficial attributes of the 100U system, namely, a vanishingly small site boundary dose, incomparably robust resistance to missiles (such as a crashing aircraft) and inconspicuous profile have been disseminated widely in the industry literature. This paper seeks to add to the body of published knowledge on HI-STORM 100U in two areas:

i. The design features that enable the HI-STORM 100U to maintain complete isolation of the spent fuel from the environment under all operation modes, including natural disasters such as hurricanes, fire, and tsunami.

ii. The methodology to quantify the response of a HI-STORM 100U ISFSI under an earthquake using LS-DYNA. This methodology is derived from the HI-STORM 100U FSAR (that supports the system license) and ongoing work by the authors to demonstrate the underground systems innate structural capability to withstand the most severe earthquakes that have been postulated for any nuclear plant in the world. The paper contains a summary of the Design Basis seismic model, developed by parametric studies on LS-DYNA, which provides a reliable means to predict the seismic response of the 100U system and to quantify its structural margins of safety.

ABSTRACT 197

Experience With MIDUS™, a Type B(U) 99Mo Transportation Package

BRANDON THOMAS, BLANJAAR FRANK

There is a worldwide shortage of 99Mo, which decays to 99mTc, the most widely used radioactive isotope for medical diagnostic studies for the heart, kidneys, lungs, liver, spleen, and bone, among others. Mallinckrodt Medical, a major supplier of 99Mo, replaced its fleet of 99Mo transport casks with a new Type B(U) package designed and manufactured by EnergySolutions, and licensed by the U.S. NRC. The packages have been in service for two years, and an operational database is now available. This paper discusses the particular challenges for the MIDUS™ design and licensing, and how the field performance of MIDUS™ has compared with design objectives.

MIDUS™ was designed to have approximately twice the capacity as its predecessor, but with the same outside envelope dimensions. Depleted uranium shielding was necessary in order to meet size and capacity requirements. Bulk shielding was not the only design challenge, since the MIDUS™ payload was in liquid form. After a 2002 incident involving elevated radiation levels on a package containing 192Ir, the NRC staff recommended special mitigation capabilities for MIDUS™. Unlike many payloads, high specific activity liquids could penetrate into the thin gap between the cask shield plug and body, bypassing much of the package's shielding. The paper describes the MIDUS™ shielding design, how the mitigation shielding was accommodated, and compares predicted and measured dose rates.

Due to the short (66h) half-life of 99Mo, the schedule for packaging and shipping MIDUS™ is critical. Every hour lost to delays on the front end amounts to a 1% loss of product, affecting both patients and medical staff awaiting procedures. MIDUS™ was designed for reliable, efficient operations. The paper reviews the MIDUS™ operational performance including typical package duty cycles, process time for

ABSTRACT 210

Evaluation of Hydrogen Yield in Spent Fuel Transport Package

MASANORI EBIHARA, YOSHIYUKI FUJITA

1. Introduction

During a transport of spent fuels in a wet type package such as NFT type transport package, there is a possibility that hydrogen gas is yielded from coolant (water in the package) radiolysis. Up to now, the hydrogen gas concentration in a cavity inside the package has been measured two times, both of which fell well below the hydrogen concentration lower limit of flammability: 4%. We introduce this data as a valuable reference concerning radiolysis of water in wet type cask.

2. Measurement of hydrogen gas concentration

(1) Type of package for measurement

The types of package for measurement were the NFT-14P type package for the PWR fuel transport and the NFT-38B type package for the BWR fuel. The contents are spent fuels with leak-tight cladding. Each gamma Irradiation dose is 844Gy/h and 640Gy/h. Neutron dose is negligible (less than 10e-5 times).

(2) Sampling of gas inside package cavity

Internal gas of the NFT type transport package sampled when the spent fuels were loaded off from the package at the pond in JNFL Rokkasho reprocessing plant.

First, the package was hooded with sampling hood in the pond. When the lid of the package was opened, the gas inside the package cavity was sampled through the sampling line from the hood.

(3) Measurement

The hydrogen gas concentration was measured with a gas detecting tube. If the concentration measured exceeded measurement range of the gas detecting tube, the sampling gas was diluted and then measured.

3. Evaluation of concentration of hydrogen gas

Each result of measurement for the NFT-14P and the NFT-38B (1.24% and 0.43%) fell well below the hydrogen concentration lower limit of flammability: 4%.

Moreover, these measurement results were at the level of the calculated values based on the literature data under a similar radioactive source environment.

The maximum calculated value of hydrogen concentration is below 1.96%, if the contents are the maximum in the safety analysis reports of NFT-14P and NFT-38B.

ABSTRACT 221

Radionuclides Concentrations in the Ocean at the Hypothetical Release from Submerged Type-B Package

DAISUKE TSUMUNE, TAKAKI TSUBONO, TOSHIARI SAEGUSA

Radioactive materials in Type B package have been transported safely on the sea under the IMO and the IAEA standards. Environmental impact assessments have been made by assuming that a Type B package might be sunk into the sea to gain the supplementary public acceptance for these transports since 1970s.

Evaluations of the value of radionuclide concentrations and their distributions are important in this assessment. Recent state-of-the-arts ocean circulation models can estimate realistic distributions of the radionuclides concentrations. Use of the state-of-the-arts model is more effective to gain the public acceptance than the previous one. In addition, the realistic estimated distributions of the radionuclides concentrations provide useful information for making an emergency plan at the submerge accident.

Recent progress of ocean general circulation models are rapid in the research field of global warming projection partly due to the increasing of computer resources. We employed the state-of-the-arts ocean models, the Parallel Ocean Program (POP) and the Regional Ocean Model System (ROMS) for coastal and global area, respectively. These model's resolutions are higher than the previous models. Although previous regional model simulates only for Japan Sea, this regional model simulates all around Japan.

We simulated background radionuclide concentrations by radioactive fallouts to validate the methods. Fallouts were input into ocean by nuclear weapon test since 1945 and mainly in 1960s. And their concentrations in the ocean have been measured to keep monitoring the artificial contaminations. Database of observed radioactive fallouts is useful to compare with simulated results in both coastal and global areas. Simulated results were in good agreement with observation. The representation by this model is better than the previous one.

And then, we simulated radionuclides concentrations at the hypothetical release from a submerged MOX fuel package off the Fukushima (200m depth, Fig. 1) and near Japan (2250m depth, Fig. 2). Simulated concentrations in the coastal and global area were quit smaller than the background concentration by the fallouts.

Fig. 1. Tracer concentration by the ROMS in the surface layer at the case of continuous release off Fukushima at the depth of 200m. × shows the release point.

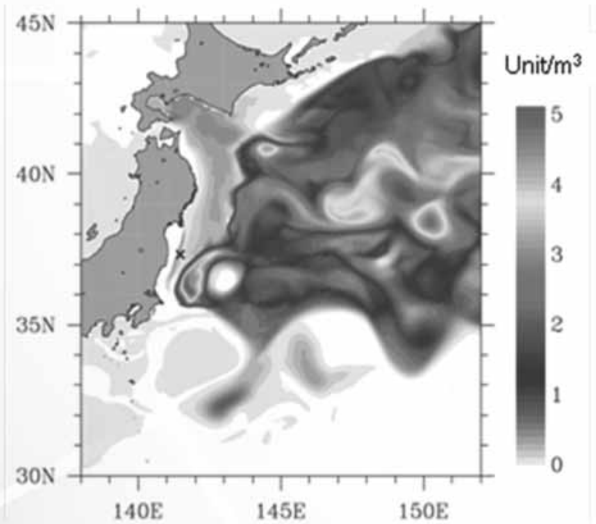
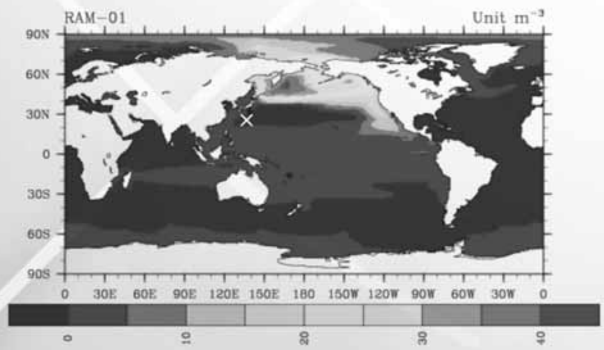


Fig. 2. Tracer concentration by the POP in the surface layer at the case of continuous release at the Izu-Ogasawara Trench at the depth of 2250m. × shows the release point.



PATRAM 2010 Posters : continued

ABSTRACT 227

Radioactive Waste and Spent Fuel Packaging and Transport Overview

SAM DARBY

The purpose of this paper is to advertise the activities of the World Nuclear Transport Institute (WNTI) in promoting the safe and efficient transport of radioactive waste and spent fuel materials. The WNTI has formed a working group - the "Waste and Spent Fuel Transport Working Group" (WTWG) - with the principal aims of:

- identifying issues with the potential to adversely affect the safety or efficiency of radioactive waste and spent fuel transports,
- using the knowledge and experience of its members to obtain a full understanding of these issues, and to develop an industry position,
- disseminating the learning to shippers and regulators.

The paper will briefly explain the history, ethos and current status of the WNTI. Mainly, the paper will describe WTWG activities and current issues for the industry; these will be developed in a number of specialist papers presented by other WNTI members at PATRAM 2010.

ABSTRACT 246

Development and application of a new type A, 10-ft (AIR) or 20-ft Box and OHT container for transport and storage of surface-contaminated equipment (class 7)

DETLEF MUGGE, DORIS DORFLER

AREVA NP as well as the nuclear business in general are very concerned about transport safety issues.

Container d.o.o, in cooperation with AREVA NP, has been developing, manufacturing and supplying special transport and storage containers for surface-contaminated objects since 2004. The containers are being tested and approved according to international applicable standards and regulations, IAEA Safety Standard IP-2 & Type A, ISO 1496/1, CSC, UIC, ADR, IMDG, RINA, TIR .

Special variations of type A containers (a 10-ft Box / AIR container and a 10-ft OHT / AIR container) have been engineered. This gives AREVA NP the advantage to be more effective, flexible and faster for technical services not only in Europe but also in overseas countries.

The heavy-duty constructions include advanced safety characteristics required for type A classified containers. In comparison to standard containers, these new types have a lower height that allows them to be brought on board of airplanes in Conformity with IATA instructions Chapter 5.0.4. With these features, the containers comply with all requirements for transports by aircraft (MD11).

These container types are not only the most flexible solution today but also the most viable transport packaging for nowadays applications.

Technical data for the 10-ft Box and OHT Type A / AIR containers

Max. gross weight 10-ft BOX 12,000 kg
Max. gross weight 10-ft OHT 12,000 kg
Max. load Box 10,050 kg
Max. load OHT 9,900 kg
Empty weight Box 1,950 kg
Empty weight OHT 2,100 kg
Length Box/OHT 2991 mm 2991 mm
Width Box/OHT 2438 mm 2438 mm
Height Box/OHT 2298 mm 2298 mm

ABSTRACT 280

A New Code of Practice on the Design, Manufacture, Approval and Operation of ISO Freight Containers for use as Industrial Packages Type 2

ROBERT MILLINGTON, BILL SIEVWRIGHT

The Transport Container Standardisation Committee (TCSC) is a UK nuclear industry body whose main function is to maintain and develop codes of practice relating to radioactive materials transport. Their role is 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. The Committee is comprised of fourteen companies who are all involved in the transport of radioactive materials.

The IAEA Regulations for the Safe Transport of Radioactive Material, Safety Standard Series No TS-R-1, permits the use of freight containers as Industrial Packages Type 2 (IP-2) for transporting radioactive material categorised as SCO or LSA provided that a number of specific requirements are complied with. However, to comply with these requirements ISO freight containers may need to be purpose designed to qualify as IP-2 packages. It is unlikely that standard, commercial, ISO freight containers would satisfy the regulatory requirements unless it can be shown that the material form is such that the requirement to "prevent loss or dispersal of the radioactive contents" can be satisfied.

TCSC have developed a new code of practice, TCSC 1090, which gives guidance on the requirements for the design, manufacture, testing, approval, operation and maintenance of ISO freight containers for use as IP-2 packages, which are appropriate for their intended use whilst satisfying the requirements of the IAEA Regulations for the Safe Transport of Radioactive Material.
The document represents good practice and takes the form of recommendations to achieve compliance with the Regulations for the Safe Transport of Radioactive Material.

This paper presents a summary of the code of practice, TCSC 1090.

PATRAM 2010 Posters : continued

ABSTRACT 284

Heterogeneity Versus Homogeneity Effects in Neutron Absorbers Used in Transport Packages.

MICHELLE NUTTALL

Many fissile material transport packages incorporate boron as a neutron poison (predominantly as a boron-metal matrix composite (MMC)) to maintain criticality safety. The MMC contains a pre specified proportion of Boron "homogeneously" distributed throughout the metal matrix.

Uncertainty arises as to the meaning of "homogeneous", in the context of providing the neutron absorbing properties assumed in the safety case for a package and the potential effect on criticality.

During criticality analyses it is usual for homogeneous materials to be assumed as uniform, without irregularities, with equal properties in all directions, at the atomic level. Since the boronated materials are "alloys", the constituents are not chemically combined but finely mixed, with the boron particles, of various sizes, viewable via a microscope. This signifies that at the atomic level the material is not homogeneous, but a heterogenous mixture with size and distribution of boron within the MMC being not strictly uniform.

Depending on variation in boron size and distribution, the neutron absorption capability of the MMC could be reduced, with consequential reduction in criticality safety margins.

During the recent manufacture of a MOX fuel transport package, which included a boron carbide (B4C) – aluminium alloy, material quality tests were performed to examine the structure of the material.

Although the tests confirmed the size and distribution of B4C within the MMC to be such that it could be classified as "homogeneous", supporting calculations were completed to determine potential effects on criticality of a heterogeneous v homogeneous neutron absorbing material.

To determine the change in neutron absorbing properties of the MMC due to atomic versus microscopic assumptions, the Monte-Carlo neutronics code MONK was used to extensively examine the effects of a heterogeneous MMC with a boron particlulate of various sizes and proportions compared to a homogeneous boron distribution within the material.

The paper presents, from a criticality viewpoint, the effects of heterogeneity v homogeneity for boronated poisons in a particular fissile material transport package. It also emphasises the benefits of the utilisation of software/computing developments during the calculational process, which enable wide-ranging surveys over many variables to be completed quickly and efficiently.

ABSTRACT 301

High Temperature Impact Limiter Composite Foam

MITCHELL JOHNSON, GLENN STROM

In the construction of impact limiters a number of materials are used as the energy absorbing material. Polyurethane foam has been used for more than 40 years. Other materials such as cellulose fiber, honeycomb, metal fabrications, wood, cork or other foamed materials have several drawbacks over polyurethane foams including: availability, uniformity, cost and uniaxial protection and poor fire resistance. In this paper, a new type of foam has been developed for use as a high temperature machinable or pour-in-place foam. It shows similar stress-strain curve shapes as polyurethane foams, but the is more brittle at temperatures below 150 °F. This is a composite foam with a heat distortion temperature over 350° F as compared to polyurethanes foams 270 °F. Typical urethane foams lose about 1.8 psi/°F up to 200 °F where as this composite foam only shows a loss of 0.8 psi/°F to 300 °F. This foam also has intumescent properties, that is, the volume increases when exposed to a high temperature heat source. This is important to as the foam will help seal out the gases and oxygen to help prevent smoldering fires and protect the payload. Dynamic impact and static compressive properties for both pour in place and machined foams will be presented versus operating temperatures ranging from -40 to 350 °F. The foams behavior when exposed to fire will also be discussed.

ABSTRACT 312

Transport Safety Justification of the Sealed Radionuclide Sources with Expired Assigned Lifetime

VLADIMIR ERSHOV, SERGEY ERMAKOV, VASILIY TEBIN

Sealed radionuclide sources (SRS) after use are transported to reposal or to utilization. Assigned lifetime (ALT) of SRS at moment of such transportation has often expired, and safety justifications used for transportation of fresh SRS, can't be accepted or at least are not sufficient ones.

There are special problems for SRS designed as special form radioactive material (SFRM), and transported in type A package. If we have not conformity evidence of such SRS to SFRM, we shall need to use expensive type B package and to pass through all procedures of conformity evaluation and approval for such packages. Corresponding certificate of approval which confirms conformity of such SRS design to SFRM requirements usually concerns to transportation of fresh SRS or only during ALT period. It needs to take into account that as a rule such certification based on results of tests of fresh SRS models, that does not submit any data about safety margin and margin for future as well.

There are some results of safety justifications for number of SRS designs that are employed in Russia after exploration and expire of ALT presented in paper.

In fact only possible way to justify conformity of such SRS to transport requirements is calculation methods. The conditions of SRS exploration, corrosion velocity of capsules materials, radiogenic helium accumulation (for alpha-radionuclides), and other characteristics affected to parameters of spent SRS are taken into account for determining conditions of such SRS. Finite element method, based on applied "ANSYS" code, was used for justification of stress and fire-resisting parameters of such SRS.

Computed values of strains in elements, which not exceeded limit with margin ratio were used as conformity criteria. Probabilities methods may be used as criteria for safety justification as well because such criteria used often for SRS exploration. Probabilities parameters are calculated by using ratio of real and limit stresses.

PATRAM 2010 Posters : continued

ABSTRACT 315

Design of Shock Absorbing Structures Based on Impact Test Data of Materials

SANG HOON LEE, WU SEOK CHOI, KI SEOG SEO

An appropriate application of impact limiters is of great importance to the design of a transport package which meets the structural safety requirements stipulated in the regulations. The design of impact limiters is not a simple task due to the difficulty of modeling the filler materials such as wood, polyurethane foam, honeycomb, etc. In this research, we characterize the dynamic stress-crush characteristics of the commonly used filler materials experimentally and apply the data to the design of impact limiters. We design a test equipment which can generate impact energy corresponding to the impact with a velocity of 13.4 m/s with variable impact masses. Several material models such as piecewise linear plasticity model, crushable foams are tested to model the filler materials in finite element analyses and calibrated to give the best correlation with the test data. Thus obtained material model is used in the design of impact limiter for the hot cell transport cask designed by Korea Atomic Energy Research Institute.

ABSTRACT 320

Thermal Analysis of Hot Cell Transportation Cask

JU-CHAN LEE, KYUNG-SIK BANG, WOO-SEOK CHOI, SANG-HOON LEE, KI-SEOG SEO

A hot cell transportation cask has been developed to safely transport spent nuclear fuels and high level radioactive wastes generated from the pyroprocess. The cask can transport 20 kg of PWR spent fuel rodcuts

or 14 kg of ceramic waste generated from the electro-winning process of pyroprocess. The cask is classified with type B(U)F package in accordance with the requirements of IAEA and domestic atomic regulations. Therefore, it should maintain its safe status under the hypothetical accident conditions as well as normal transport conditions. The regulations require that the cask should maintain the shielding, thermal and structural integrities to release no radioactive material. In this study, thermal analyses of the hot cell transportation have been carried to verify that thermal performance of the cask complies with the requirement. Thermal analyses have been performed by using the FLUENT code. The decay heat from the spent fuel and ceramic waste are considered with 32 W and 92 W, respectively. Initially, the cask is assumed to be operating at steady state under normal transport conditions with an ambient temperature of 38 and maximum insolation. It is then exposed to a 30 minutes, 800 , fully engulfing fire having a flame emissivity of 0.9 and the cask surface is assumed to have a thermal absorptivity of 0.8.

As the results of thermal analysis, the maximum temperatures of shielding and the O-ring seal materials were lower than their allowable values. Therefore, it is found that the thermal integrity of the cask will be maintained under normal and accident conditions. The results of thermal analysis will be used as basic data for license of the cask.

ABSTRACT 321

Conceptual Design of Transport and Storage System for Future Pyroprocess R&D Facility

KI-SEOG SEO

The Korea government reviewed and confirmed a "Long-Term Development Plan for Future Nuclear Energy System" at the 255th Atomic Energy Commission meeting held on December 22, 2008. The plan outlines the major R&D milestones associated with pyroprocessing. The objective of pyroprocessing R&D plan in Republic of Korea is to develop pyroprocessing technology for reducing spent fuel volume and providing Sodium Fast Reactor(SFR) fuel by electrochemical process. A pyroprocessing R&D facility, related transport and storage system will be provided by 2016 to realize an integrated pyroprocessing system. The recovered materials and waste forms will be generated from pyroprocess. Uranium and TRU material for SFR fuel will be recovered and several kinds of waste form will be generated as ceramic waste, glass waste, fly ash waste and metal waste. To obtain the design data of transport and storage system of pyroprocessing R&D facility, the characteristics of the recovered materials and waste forms had been analyzed. These characteristics included a radioactivity, decay heat and mass. A conceptual design of transport and storage system was performed for pyroprocessing R&D facility.

ABSTRACT 329

Finite Element Simulation of a Full-Scale 9-Meter Free Drop Test of a Spent Nuclear Fuel Transportation Cask

JOSE PIRES, DANIEL HUANG

This paper presents results of finite element simulations of a 9-meter free side drop test of a prototype CONSTOR® V/TC spent nuclear fuel transportation (SNFT) cask. The 181-ton prototype test specimen was manufactured by the Gesellschaft für Nuklear-Service mbH (GNS) of Germany and was tested by Germany's Federal Institute for Materials Research and Testing (BAM) in 2004. The analyses reported here were conducted by staff of the United States Nuclear Regulatory Commission (USNRC) under a cooperative research agreement between the USNRC and BAM. The drop test was analyzed using the commercial, dynamic, nonlinear, explicit finite element program LSDYNA. Results from this study can be useful to reaffirm the staff and industry practices for the design and review of SNFT casks.

The paper provides a detailed description of the finite element model of the cask as well as a comparison of analyses results to test data. This description includes a description of the types of elements used, contact definition assumptions, modeling of the bolts preload, and assumptions regarding the modeling of the wood in the impact limiters, which was modeled using the honeycomb material model in LSDYNA. The paper also provides an assessment of effects of modeling assumptions such as friction characteristics in the contact definitions and material properties for the wood in the impact limiters. All the analyses took advantage of symmetry with the more refined finite element mesh using about 530000 elements to model half of the cask. Analyses results include rigid body cask accelerations in the direction of the drop, filtered acceleration time-histories at specific

PATRAM 2010 Posters : continued

locations near the lid and base of the cask as well as strain time-histories at selected cask locations and at the primary lid bolts. Calculated acceleration time-histories near the lid and base, filtered using 100 Hz and 200 Hz low-pass filters, are consistent with similarly filtered recorded acceleration time-histories.

DISCLAIMER NOTICE – The findings and opinions expressed in this paper are those of the authors, and do not necessarily reflect the views of the USNRC.

ABSTRACT 331

DALMA 25: A Packaging for Enriched Uranium Aqueous Solutions

OSCAR NOVARA, FERNANDO ORLANDO, MARIANO FLORES, PABLO MAIORANA, CRISTIAN ORTIZ, FERNANDO PALAS

Due to the need to safely transport uranium solutions between front-end nuclear facilities, the Argentine National Atomic Energy Commission (CNEA, for its initials in Spanish) undertook the development and licensing of packaging appropriate to that kind of radioactive material.

This material is typically a uranyl nitrate acid aqueous solution with 19.75% enriched uranium in U-235 isotope. Both the maximum uranium mass and concentration are within the limits of what is considered a "safe load" from the point of view of criticality safety.

The packaging, called DALMA 25, uses as the primary container a high-density poliethylene (HDPE) 25-litre flask, commercially available for use in chemical laboratories but also used in the same front-end nuclear facilities to handle this solution. The flask is inserted inside a stainless steel barrel and immobilised with the help of specially designed parts. Inside the barrel, the flask is completely surrounded by a layer of chemical-spill absorbing material. The barrell is closed and sealed by means of a bolted stainless steel lid with an elasthomeric ring. This barrel, together with its internal components, makes up the containment system of the design, meant to prevent the radioactive content from leaking in hypothetical transport conditions.

The impact limiter system basically consists of a hood and a lid which fits into it, completely covering the containment system. Both the hood and lid are made of stainless steel plates and filled with an expanded polystyrene (EPS) foam lining thick enough to absorb mechanical energy.

For handling purposes, this cylindrical package is vertically placed on a pallet. The design is completed with rigging elements.

Prototypes are built to conduct all the tests for Type A packages plus 9-metre drop tests specific to liquid radioactive content, according to IAEA transport rules, to prove that the material transported does not leak out of the designed containment system.

The current paper describes the design and the validation process, presenting as well an analysis of the results of the corresponding tests.

ABSTRACT 346

Matrjoschka-Conception of Transportation and Storage Cask

FRIEDHELM H. TIMPERT, WOLF J. EULER

In light of the current security environment, besides the security of nuclear power plants more and more emphasis is paid to the nuclear waste management. In Germany in the last decades more and more attention has been drawn to the packaging and transportation of radioactive materials. While an elaborated set of rules and regulations formed by the Atomic Energy Regulatory Board internationally and nationally exists additional efforts have to be made to limit radiation exposure to the environment and to plant employees and the public especially due to the fact that in the near future high burn up of approximately 60 GWd/t (PWR) and 65 GWd/t (BWR) will become the standard. Additionally to a higher radiation shielding cask the package operations and maintenance should be optimized by higher safety stock.

The possible points of malfunction of today's cask-(B)-types ask for the innovation of a new proven technology. This so-called Matrjoschka-Typ-B-System (which is already patented in USA and Russia) is a new developed technology following the recommendations of IAEA regulations. The basic idea is a container-in-a-container technology with quantifiable and shock-absorbing spring devices for loading case absorption at special high impacts. Mainly floated reinforced concrete fills the space between the sandwich cask design and gives to the system the highest safety. The heat removal up to 100 kW by convector design of the inner cask is guaranteed by gravitation. The Matrjoschka-concept fulfils both the internationally valid IAEA criteria for transportation and the requirements for long-term intermediate storage in US, Canada and various European countries even at temperature ≥ -50 °C.

The paper will describe the details of the new concept and the properties of the designed technique, the material selection and the calculations for drop test, percussion test and bending test.

A research study at Magdeburg Technical University has proven the advantage of this new technology.

ABSTRACT 421

ENGINEERING CHALLENGES IN THE MECHANICAL DESIGN OF A NEW SHIELDED SHIPPING CASK FOR VITRIFIED WASTE PRODUCTS

RK GUPTA, SP PATIL, DS SANDHANSHIVE, AK SINGH, KM SINGH

Shipping casks for Vitrified Waste Product (VWP) are designed, built, and maintained to ensure utmost safety and conformance to regulatory requirements which are essential for handling and transport in public domain. This is achieved by proper sizing for static and dynamic stability, detailed quality surveillance during manufacture, physical testing of models for design validation and selection of prime movers for transfer. Actual transport is carefully planned in cognizance of logistics of handling and interim storage in engineered cooling vaults for several decades prior to final disposal.

The closed fuel cycle adopted in our Nuclear Programme is extremely sensitive to safety and regulatory issues. But while Vitrified Waste Packages are produced in Trombay, Tarapur and Kalpakkam,

the interim storage site with engineered cooling under surveillance is available only in Tarapur. Thus, cross country road transport of VWP Casks from Trombay (130 Kilometers) and Kalpakkam (1200 kilometers) need careful site-specific considerations and mandatory compliance with safety guidelines.

Five years ago, such a transport cask was designed and built for Trombay plant which, at that time, produced vitrified waste packages with activity limited to 25000 curies per canister. While the cask is still being used for shipment of VWP from Trombay to Tarapur, the plant now requires a new cask for shipping packages for fresh batches of HLW having radioactivity levels close to 250,000 curies per canister. This paper describes the design of such a cask and challenges associated with the restrictions of using existing handling facilities for loading and transporting. Authors realize that all aspects of an acceptable final design require extensive involvement of experts from several disciplines and no single agency can do this work in isolation. But the contents of this paper are limited to describing critical design issues from the mechanical engineers' point of view. Changes made in the old design have been brought out in respect of fabrication, physical testing, modifications of existing intra-facility-transfers at Trombay and unloading and handling at Tarapur. An additional design consideration is horizontal mounting of the new cask on transport trailer as against vertical mounting currently in practice. Needless to say, design considerations of the previous instance would also be touched upon, albeit briefly. The authors have assumed that data made available to them on package size, weight, shielding and regulatory requirement are accurate and similar to the previous design.

TRANSPORT OPERATIONS

ABSTRACT 49

Analysis on Doses Received by Workers Transporting Radiopharmaceuticals in Spain

VICTORIA ACENA, ENGRACIA RUBIO, FERNANDO ZAMORA

Introduction

The work presents a detailed analysis of doses received by workers from 2003 to 2008 in the transport of radioactive material in Spain, in particular in the transport by road of radiopharmaceuticals.

The analysis also identifies the cause of doses, inform about the measures adopted in the period to reach a reduction of doses and present additional measures considered necessaries to achieve further reductions.

Analysis

The assessment confirms that the sector with higher doses is in the transport by road of radiopharmaceuticals, mainly due to operations involving generators of Mo/Tc. In Spain there are few carriers of radioactive material and the number of workers exposed is low. In consequence, collective dose is not significant but individual doses are high, especially in workers participating in loading-unloading operations.

The reduction of these doses has been considered priority by the competent authority (Consejo de Seguridad Nuclear) that had adopted different measures. The implementation of Radiation Protection Programs and particular ALARA measures adopted by carriers has lead to an improvement of the situation.

However, the present level of doses may decrease with new measures carried out by carriers, but fundamentally thanks to measures adopted by other participants in the chain of transport of radioactive material as manufactures, consigners and consignees

Conclusions

Although the doses received by workers in the transport of radiophamaceuticals in Spain are below annual limits they are significantly high in particular cases. Doses may be reduced by carriers adopting ALARA measures; however, a major implication of manufactures, consigners and consignees may permit a more important reduction of doses.

ABSTRACT 111

Cernavoda NPP Transport Activities – Operational Experience

IOANA ELISABETA DAIAN, ION POPESCU

This paper presents Cernavoda NPP transport activities operational experience accumulated during Cernavoda NPP – Unit 1 and Unit 2 commissioning and operation. Romania has one nuclear power plant, CNE Cernavoda , equipped with five PHWR - CANDU-6 Canadian type reactors with a 705 MW(e) gross capacity each, in different implementation stages. Unit 1 and 2 are in commercial operation since December 1996, respectively November, 2007.

Performed transport activities cover a wide area of materials transported, from routinely shipments of standard ra-dioactive sources for laboratories, fresh and spent fuel, radioactive waste, and specific shipments for maintenance activities during outages periods; these shipments deal with various types of equipment like fuel channels inspection equipment, sludge lancing equipment, scrap samples from fuel channels, pipes shipped for various types of analy-ses in Romania and abroad; some of these shipments involve multimodal transport operations with associated ra-diation protection and emergency preparedness.

Spent fuel is stored within the Intermediate Dry Spent Fuel Storage Facility (IDSFSF). IDSFSF is a MACSTOR sys-tem designed for storing spent fuel cooled for a period of six years consisting of storage modules located outdoors in the storage site and equipment operated at the spent fuel storage bay for preparing the spent fuel for dry storage.

The IDSFSF was designed to accommodate the spent fuel generated from 30 years operation of the 2 CANDU units, respectively 324.000 fuel bundles. The first module of IDSFSF was put in operation in 2003, the second and the third modules are in operation since 2006.

The spent fuel is transferred to dry storage from the storage bay with a trailer, in a transfer flask, within the Cer-navoda site exclusion boundary.

After pretreatment (collection, segregation, decontamination) and treatment (compaction or shredding, as appropri-ate) the solid wastes are confined in 220L stainless steel drums and transported to the Solid Radioactive Waste In-terim Storage Facility; this facility is located within the inner security fence of the plant site and is designed for stor-age of low and intermediate wastes produced by operation of CNE Cernavoda Unit1 and Unit 2, except spent res-ins, reactivity control rods and spent fuel.

PATRAM 2010 Posters : continued

ABSTRACT 140

ALARA Principle in Transport Classified "Not Under Exclusive Use"

EDUARDO GERULIS, FABIO SUZUKI

The transport of radioactive material, TRM, is an activity that can cause dose of ionizing radiation to workers that perform their shipment. These workers may be exposed to this field type to be take the exposure values measures from the load to be transported. However, with a right interpretation of the recommendations to TRM we can reduce these doses by applying the principle of radioprotection, as low as reasonably achievable, ALARA.

The measures are taken in the packed surface and from one meter these to classify them. Among other things, the classification of packaged sets if the carriage shall be classified "under exclusive use" or not. The transport "under exclusive use" is one in which a single sender performs the transport. In this case the configuration of the load is maintained until the recipient.

The sum value of the transport index, TI, to vehicles in the transport classified as "under exclusive use" is not more than 50 and the TI value of any single packed is not more than 10. The maximum dose rate theoretical at two feet from the surface of a vehicle that shipment one TI equal to 50 is considered equivalent to 0.1 mSv/h on accepted model.

Formulary for monitoring cargo and vehicle, available as a model on the recommendations of TRM, are used to transport both classifications: "under exclusive use" and "not under exclusive use" and have blanks to fill rate dose measurements of the two meters of the surface of the vehicle. However, these measures, the two meters, should be carried only for transportation "under exclusive use" because for the transport "not under exclusive use" these values are at most equal to 0.1 mSv/h, as model.

Thus workers who perform the TMR "not under exclusive use" need not be exposed to the radiation field to obtain such measures, because these values are already known. This paper discusses this interpretation and provides an estimate of dose received in the transport of radiopharmaceuticals, in a Brazilian plant, if obtain such measures is performed.

ABSTRACT 154

Development of Monitoring System for Land Transport Conditions of Nuclear Materials

KIYOAKI YAMAMOTO, WATARU YUASA, SHINICHI UCHIDA, SHOICHI INOSE, SHIGEO FUJIWARA

It's very important to monitor land transport conditions continually in order to implement the safe and smooth transport of nuclear materials, especially to quickly obtain the accurate visual information and carry out emergency response swiftly and properly in the event of transport accident.

Plutonium Fuel Development Center of Japan Atomic Energy Agency(JAEA) developed the system for real-time monitoring of transport conditions of nuclear materials, and applied it to transport operation of JAEA.

The monitoring system mainly consists of system for monitoring location of convoy system comprising several vehicles and system for monitoring image of the transport conditions.

The location monitoring system adopts satellite wave and ground wave communication methods. The location information from GPS is transmitted to ground station from transport monitoring equipment installed on vehicle of convoy system via the satellite, and transmitted to the Transport Control Center (TCC) of JAEA periodically through telephone line. The location information from GPS is also directly transmitted to ground station from transport monitoring equipment by ground wave and transmitted to TCC through telephone line. In both cases, lo-cation information is shown on a monitoring panel in TCC.

Image monitoring system is one for transmitting motion picture taken by cameras equipped on the roof of vehi-cle of convoy system to TCC through ground wave and showing it on a monitoring panel in TCC. It's also possible to transmit image to TCC with this system, which is taken by the portable camera for emergency response. The camera is taken out of transport monitoring equipment, installed on the proper place, and remotely controlled from TCC.

For the development of the transport conditions monitoring system, following requirements were considered tak-ing into account the use in emergency response and long-term transport,

- Compactness to fit limited space on the vehicles
- Applicability for conditions in vibration and temperature change anticipated in transport operation.
- Long-term stability and redundancy of the power supply

The developed system is very useful not only for monitoring normal transport conditions but also for planning emergency response program because it can transmit accurate visual information to TCC in occurrence of trans-port accident.

ABSTRACT 199

Measurement Experiment of Radiation Dose by Moved Radioactive Source to Apply the Spent Fuel Transportation Condition by Sea

SUHONG LEE, SANGWON SHIN, JAEMIN LEE, SEONGHYOUN YOON, KIYEOUL SEONG

There are many mathematical models and the relevant assessment programs that evaluate the effect of radiation exposure by radioactive material. But the most basic evaluation of the effect focuses the effect of it by the radiation source on fixed location. The general public is exposed by direct or indirect exposure when they contact the contaminated materials and are located in contaminated grounds. When radioactive wastes or materials are transferred to interim storage facility or another nuclear facilities, the evaluation about subjects of radiation exposure in transportation routes of the applicable ones should be assessed by fixed radioactive source, not moved radioactive source. But when radioactive source is moved, it is difficult to apply general exposure mathematical models to radiation exposure. Accordingly, the mathematical models are presented to evaluate the effect of radiation exposure about moved radioactive source with integral concept according to space segmentation. But it requires that the effect evaluation of radiation by moved radioactive source through verification experiment that copies the similar conditions is assessed because the various variables that cannot be evaluated in the mathematical models under the actual conditions. Therefore, In this study, I measures of radiation exposure dose according to the characteristics of mobility by each nuclide making a experimental apparatus of the conditions that copy moved radioactive materials with the moving distance of 2-way direction, 50m. I made the experimental apparatus and set up its rail to minimize reduction and acceleration sections so that the operating speed is realized. I made the actuator reach speeds of maximum 50km/h. Also, I did the system that can control itself automatically so that it can be operated in the most suitable form for the transportation conditions. As for the experimental conditions, I executed the test according to difference of distance between radioactive sources and measuring equipments and speeds of them assuming the actuator as a means that transport the radioactive material and doing the measuring equipment as the subjects of radiation exposed person. Also, I evaluated the effect of radiation exposure between nuclides with same conditions executing the same tests about various ones.

ABSTRACT 207

Conceptual Data Modeling for Development of Spent Fuel Transport Simulation Tools

SANGWON SHIN, SUHONG LEE, JAEMIN LEE, CHANGYEAL BAEG, JEONGHYOUN YOON

A country such as Korea, in the situation that the national management policy of spent fuels has not been determined, a number of flexible assumptions and scenarios should be considered for spent fuel transportation. It is necessary to analyze the scenarios and assumptions systematically to find out the best option to apply for the future general and/or specific transportation modes. The important input data such as the expected quantity that spent fuels will be generated and the yearly traffic, transport modes, size of vessels in marine transportation, transport system, cost evaluation, etc are required to develop a system that makes it possible to establish an optimized transport plans of spent fuels. In this respect, we analyzed the domestic and foreign status of transport of spent fuels, the relevant regulations, the basic components related to transport and methodology to analyze proper amount of spent fuels. A study of the conceptual data modeling to establish the spent fuel transport simulation tools has been carried out through development of the procedure and methodology with proper database. The development of the simulation tool consists of interface design that can improve graphic user interface for the evaluation. The basic protocol for entity of database and the relationship between individual parameters were defined focusing on job flow. The results of the conceptual data modeling can be applied for development of a transportation simulation tool.

ABSTRACT 243

Transport of Large Components as Special Arrangement Within Sweden on a Norwegian Vessel

SVERRE HORNKJOL, TONJE SEKSE, HELMUTH ZIKA

Three disused steam generators from the Ringhals Nuclear Power Plant were transported from the Ringhals site to Studsvik for dismantling and waste management. The weight of each steam generator was 300 tons.

For this transport a Norwegian ro/ro vessel was hired due to its ability to transport all three steam generators in one transport operation. This required approval of special arrangement both from Swedish and Norwegian authorities. In addition, a licence from the Norwegian Ministry of Health and Care Services for the transport of nuclear material onboard a Norwegian vessel was needed as the steam generators were to be considered as nuclear waste.

The process for the cooperation between the authorities in both countries and the process for getting a licence to ship nuclear waste onboard a Norwegian vessel is described together with the conditions applied. The results from the joint inspection of the loading of the steam generators on board the vessel are also presented.

ABSTRACT 258

The Radioactive Material Transport Users committee (RAMTUC)

CHARLIE CARRINGTON, MARK FLYNN

The Radioactive Material Transport Users Committee (RAMTUC) was formed in the 1980's and is the UK industry's forum for the development of strategy and policy regarding the regulations for the safe transport of radioactive material. RAMTUC represents both the nuclear and non-nuclear industry members who design manufacture consign load transport unload use or dispose of radioactive material within the UK and is a non-profit group that promotes radioactive transportation safety by providing seminars and classroom training as well as by participation in domestic regulatory activities in its promotion of the safe and efficient transportation of radioactive material.

The seminars have covered a number of subjects including emergency response, insurance, new regulatory proposals and new general updates from UK competent authority the Department for Transport. These have been aimed at the whole of the UK radioactive transport community from nuclear facilities to the small users of radioactive material.

The non regulatory training syllabus for the industry is held by RAMTUC, this is a three tier system and is administered by RAMTUC on behalf of the industry. RAMTUC have also been involved in the development of new National standards leading to future National Vocational Qualifications for the activities involved in transporting radioactive material.

RAMTUC employs consultants who represent UK industry in a number of the transport related working groups, focusing on both the UK and the international regulations covering all modes.

The poster presentation will provide PATRAM members with an insight into UK nuclear transport industries forum RAMTUC's current work and communication tools.

ABSTRACT 267

VRR from UK

CHRIS PURDY

For more than 30 years Sellafield has reprocessed spent nuclear fuel for overseas customers. The process separates the re-usable components, approximately 97%, from the waste (approx. 3%). The highly active component of this waste is concentrated, stored for a period of time and then converted into a solid glass form (vitrified) that is poured into purpose designed stainless steel canisters. Reprocessing contracts entered into since 1976 between the UK and

PATRAM 2010 Posters : continued

overseas utility customers have contained an option to return this waste to the country of origin and in 1986 HM Government took the decision to exercise these options.

This year International Nuclear Services and its subsidiary Pacific Nuclear Transport Limited, on behalf of the Nuclear Decommissioning Authority, the owners of the Sellafield Site in the UK, have commenced shipments of VRR to overseas customers.

This paper describes the preparation, packaging, marshalling and transport of the first consignments, to Japan and The Netherlands respectively utilising a type B package, moved by rail, sea and ultimately road to the final destination a purpose built long term storage facility in the country of origin.

ABSTRACT 279

Statistics on the Transport of Radioactive Material in Sweden and Doses to the Transport Workers.

THOMMY GODAS, BIRGITTA SVAHN, ERIK WELLEMAN

During 2008-2009 shipment data (number of packages and shipments, type of packages, TI, distances and exposure of personal) from transport of radioactive material outside the nuclear fuel cycle were collected and analysed in Sweden in accordance with IAEA Safety Requirements No.TS-R-1 (para.308) requiring periodic assessment regarding protection and safety for the transport workers. Shipment data was collected for transport of radioactive material for medical, scientific and general industrial applications.

The collected data show that road transport is the major shipping mode used in Sweden with an annual volume of 14000 non-excepted packages (mainly type A) with a total TI of nearly 5500. Domestic transport by air, rail and sea is negligible for domestic shipments of non-nuclear radioactive material within Sweden. The overall most frequent radioactive material shipped was radioisotopes for medical applications (>75%).

Exposure data for 23 workers (drivers and handlers) were assessed. Only background exposure dose were recorded on 75% of the TLDs used. However, dose rate measurements and electronic dosimeters used indicated that higher doses than registered on the TLD should be expected.

Multiplying the TI value with the distance gave a total amount of 750000 TI x km for annual transport of radioactive material outside the nuclear fuel cycle in Sweden, which implies long transport distances and many hours in the vehicles. Under these circumstances it was questioned whether or not the personal TLDs were placed in an optimum way on the body.

Further studies showed that nearly doubled dose was registered if the dosimeter is placed on the back compared to the conventional place at the front of the body. The study also showed that estimation of received doses for the purpose of planning radiation protection measures a sufficiently accurate estimation can be done from the knowledge of TI, working hours and the type of vehicles used.

ABSTRACT 290

Utilization of Proven Shipping Methods to Transport Nuclear Reactor Components

AARON WIENER

Abstract: Utilization of Proven Shipping Methods to Transport Nuclear Reactor Components

The safe, secure, and efficient transportation of nuclear reactor components can be affected by many aspects within the transportation system, geopolitical factors along the route, as well as customer need.

That's the challenge Global Transportation Systems, Inc. (GTS Group) faced in July 2009 when they were asked to ship a defective, surface-contaminated nuclear reactor coolant pump motor from Krsko, Slovenia to Waltz Mill, Pennsylvania. In order to meet customer requirements, GTS Group needed to secure the correct carrier; coordinate classification, certification and spec packing; comply with all European Union and United States Department of Transportation (DOT) regulations; and maintain a tight grip on our transit plan to eradicate the probability of costly delay.

Our presentation will describe how GTS Group utilized staff expertise and the cooperation of reliable partners and agents to arrange and coordinate successful delivery.

We will also discuss how project freight coordination best practices—proven through two decades as a global logistics industry leader—are adapted to support DGR and RAM handling. We will explain how GTS Group leverages:

- Operations Staff training and certification in IMDG regulations, and IATA DGR and Radioactive Materials Handling, and U.S. DOT Ground Transport to ensure conformity with NRC and all international transport regulations
- ISO 9001:2008 Certification and a companywide commitment to quality and reliability standards to meet and exceed customer expectations

Speaker: Aaron M. Wiener, Operations Specialist GTS Group

Mr. Wiener joined GTS Group in 2001. He manages and provides full service international import/export support and logistics for hazardous materials, air charters of various DGR, including "forbidden" and license controlled commodities. Mr. Wiener received a bachelor's degree in business management from Radford University in Radford, Virginia. He has completed IATA DGR and Radioactive Transport Training and U.S./TSA Air Carrier Security Training.

ABSTRACT 294

Hazardous Material Transportation Incident Data for Root Cause Analysis

THOMAS MCSWEENEY, AUTHUR GREENBERG, DANIEL BLOWER, MARK ABKOWITZ

This project, funded by the Transportation Research Board of the U S National Academy of Sciences, reviewed the data currently being collected on hazardous materials accidents by various governmental agencies. The information collected in each database was first reviewed for completeness and accuracy and then was compared with a comprehensive set of accident descriptors that were arranged into five discrete parameter sets: vehicle, driver, packaging, infrastructure, and situational. It was found that no one database

provided all the data listed in the comprehensive set of accident descriptors but a reasonably complete set could be obtained by joining datasets using common parameters such as date, time, U S DOT Number, and GIS coordinates of the accident location. At the present time the common parameters are often left blank or are inconsistently filled out, effectively defeating the joining of the databases for many accidents. It was found that the comprehensive dataset focuses on information describing the accident and the behavior of components following the accident. The causes are often factors leading up to the accident. The joined datasets can be used to identify a suite of accidents, perhaps 100, that may have a common cause, for more detailed investigation of contributing and root causes. The results of the detailed investigations could then be generalized back to the larger set of accidents. It was found that some carriers and some agencies, such as the U S National Transportation Safety Board, have used this approach to investigate the root causes of single accidents. The project found one instance where the Safety Board has successfully used the proposed approach to identify the contributing and root causes of a class of accidents, accidents at grade crossings with no active signals. The project team believes that the proposed approach, the joining of databases and performing more detailed investigation for a subset of the accidents, could be implemented by public agencies and industry to identify the contributing and root causes of accidents. If the approach was part of a continuing program, it should be possible to continuously improve the safety of hazardous material transportation.

ABSTRACT 295

Commercial Viability of Mixed Oxide Fuel Transport in the United States

FREDERICK YAPUNCICH, DOROTHY DAVIDSON, REMI BERA, MICHAEL VALENZANO

The commercial viability of a Mixed Oxide (MOX) fuel feedstock for United States (US) nuclear power stations is predicated on the US regulatory framework and the physical infrastructure of these plants. MOX fuel is a blend of uranium oxide and plutonium oxide. The commercial reactors in the United States currently rely on traditional fresh uranium feedstock. However, the international community has been utilizing reactor grade MOX fuel since 1972. A basic review of these fuel types in conjunction with the safety and security issues associated with the transport of this material is presented. An overview of various MOX fuel shipping casks is also provided.

Recommendations to optimize the use of MOX fuel in the United States are developed based on a comparison of the US transport regulatory culture and the international model. These security recommendations include privatization of certain aspects of the transport of MOX fuel and the harmonization of NRC/DOE classifications. Development of MOX transport systems amenable to the applicable MOX fuel fabrication plant and the respective utilities' fresh fuel assembly receipt infrastructure is necessary.

ABSTRACT 304

One Hundred Transportation of Nuclear Fuel Bundles - Experience Gained by Nuclear Fuel Plant Pitesti

TIBERIU IVANA, GHEORGHE EPURE

Nuclear Fuel Plant (FCN) is a facility that produces nuclear fuel bundles CANDU-6 type for CANDU nuclear power plant based on natural and depleted uranium. The transportation of nuclear fuel bundles from FCN to Cernavoda Nuclear Power Plant is performed by FCN authorised trucks (about 340 km between the two locations). FCN has the activity of transportation and the vehicles authorised by Romanian regulatory body National Commission for Nuclear Activities Control (CNCAN). In addition each transfer of nuclear material is authorised by CNCAN.

From 1996 to 2008, 100 nuclear fuel bundles transportations was performed by FCN Pitesti. In this moment about 15 nuclear fuel transportations were performed in each year. A transportation from Pitesti to Cernavoda consist regulary from 20 wooden crates with 720 nuclear fuel bundles, containing 15 tons of natural uranium dioxide corresponding with 14 tons of natural uranium. Each transport is escorted by qualified personnel from military troop. FCN has proper crew formed by responsible of transport, driver 1, driver 2, maintenance person, radioprotection technician. The persons involved have a strongly training on the domain. Randomly the transportation of nuclear fuel bundles is monitored by Romanian authority CNCAN. In this moment is intended to monitoring each transport of nuclear fuel bundles by satellite including this aspect in a large specific programme. All the activities regarding transportation is under quality assurance control and included in the integrated management system of FCN

ABSTRACT 317

Risk Assessment for Transport of Large Amount of Radioisotopes

WOON-KAP CHO

Risk assessments were performed for the transport of large amount of radioisotopes for industrial or medical applications using a specially designed type B transportation container, KRI-BGM. The container can carry maximum 370 TBq of solid Ir-192, 29.6 TBq of liquid Mo-99 and 37 TBq of liquid I-131 respectively. For the radiation risk assessment, transport of maximum activity of those radioisotopes was considered. Transportation route is selected from the production facility where radioisotopes are produced to the customer facility where radioisotopes are to be consumed. Transport distance is 300 km including highway and downtown area along the route. An ordinary cargo truck is used exclusively as the transportation conveyance. Radiation risks were estimated for incident free and accident condition of transportation using RADTRAN 5.6. Various parameters such as population density around transport route, weather condition, probability of specific accidents such as impact, fire were considered to evaluate the risk involved in the transport of radioisotopes. From the results of this study, the exclusive truck transport of radioisotopes using type B container showed low radiological risks with manageable safety and health consequences. This paper will provide the results of the risk assessment for truck transport of large amount of radioisotopes and present the radiation risks for the public and radiation workers involved.

PATRAM 2010 Posters : continued

ABSTRACT 319

Assessment of Accident Risk for Transport of Low and Intermediate Level Radioactive Waste

WOON-KAP CHO

In November 2005, Korean government designated Kyoung-Ju as a geological repository site for the disposal of low and intermediate level radioactive waste (LILW) and it is expected that the disposal facility for LILW will be operated from the beginning of 2013. As all nuclear power plant (NPP) sites where radioactive wastes are produced and temporarily stored and the disposal facility are located alongside the coast, the sea transport of LILW using dedicated ship is the most prospective transport mode. A dedicated ship, HJ Cheong-Jeong Nu Ri, was constructed in 2009 and it has maximum loading capacity of 1,520 steel waste drums which contain LILW generated from NPP operation. There are two types of DOT-17 type steel waste drums and each type has volume of 200 liter and 320 liter respectively. Two IP-2 type packages will be used to transport eight LILW drums of 200 liter and 300 liter. The sea transport of LILW will be carried out from three NPP sites, Kori, Uljin and Yonggwang to the disposal facility and the transport distances are 80 km, 200 km and 760 km respectively. Risk assessments for sea transport of LILW were performed using RADTRAN 5.6 and radiological risks for transport of LILW from the three NPP sites to the disposal facility were estimated and compared. This paper provides methods and results of the radiological risks assessment for the sea transport of LILW by the dedicated ship.

ABSTRACT 341

Spent Nuclear Fuel and Nuclear Waste Transport in the United States

AMY SEWARD

Spurred largely by concerns over climate change and the mandate to secure national energy goals, global interest in nuclear energy has surged over the past few years. The U.S. nuclear industry is in response positioning itself to compete with the largely nationalized nuclear industries of countries such as France, Russia and Japan in supplying emerging nuclear states with nuclear reactors. The capacity of states to offer the full range of goods and services for both the front and back end of the nuclear fuel cycle will be a major factor in the decisions of emerging nuclear states in choosing reactor vendors. The U.S. nuclear industry is at a disadvantage in that it is currently unable to offer the full range of such services. In contrast to states such as Russia, which offers spent nuclear fuel take back and final disposal for states to whom it supplies nuclear reactors, the U.S. has yet to resolve disposition of spent nuclear fuel and waste from its own reactors. President Obama recently initiated a comprehensive review of the United States' policies for managing the back end of the nuclear fuel cycle. The move is a significant not only in resolving the long-standing dispute between the U.S. nuclear industry and government, but in that it seeks to enhance the competitiveness of the U.S. nuclear industry. The potential scenario in which the United States offers back end services involves consideration of a number of legal, regulatory and logistic issues, including the safe and secure transportation by land and sea, both internationally and domestically. Such transport poses challenges from both security (i.e. nonproliferation) and safety perspectives. The proposed paper intends to evaluate domestic and international transportation infrastructure and requirements should the United States offer spent nuclear fuel and waste disposal services.

ABSTRACT 344

Study of Applicable Limits on Non-Fixed Surface Contamination for the Safe Transport of Radioactive Materials

MASAHIRO MUNAKATA, HIROKO TEZUKA

The regulatory requirements and controls for limiting the level of radioactive surface contamination on packages and conveyances have been in existence for some four decades. Recently, regulation limits on non-fixed surface contamination for transport of radioactive materials were evaluated in TECDOC-1449 [IAEA, 2005].

I derived the applicable limits for surface contamination by using a model for radiation exposure based on an investigation of actual conditions of the transport in Japan. Figure 1 shows the schematic view of the scenario for the transport of a spent fuel cask. I classified the transportation workers in eight groups for indoor and outdoor transportation work. Parameters for radiation exposure were defined based on the investigated actual conditions for each classified work. The radiation exposure pathways were considered as follows: the external exposure, internal exposure by inhalation/ingestion and skin exposure. The dose limits for surface contamination were derived for each 415 nuclides which were included in the IAEA Safety Standard [IAEA, TS-R-1].

Nuclide Ac -227 showed the maximum radiation dose on a package preparation and transport worker. Based on unit surface contamination (1Bq/ cm2), it was 0.48mSv/a while that of the model in TECDOC-1449 showed 9.20mSv/a. If 0.3mSv/a is used as a reference dose, the applicable surface contamination limit of Ac-227 becomes 0.63Bq/cm2 in this model and 0.03Bq/cm2 in the TECDOC model. The contamination limit of the TECDOC model is much lower than that of this model. The reason of this variation may be the difference of the used parameters (i.e. conservative or realistic) and of the inhalation exposure model for alpha radionuclides, which is the most important pathway in the dose evaluation for safe transport. For other radionuclides, the derived limits by this model have a safety margin of about 1 order of magnitude against to the regulation limits derived by the model in TECDOC-1449.

It can be concluded that the model in TECDOC-1449 is more conservative than the model in this study. Studying quantitative evaluations more should be needed for deriving reasonable limits of surface contamination.

This study was conducted as a project supported by the Japan Nuclear Energy Safety Organization.

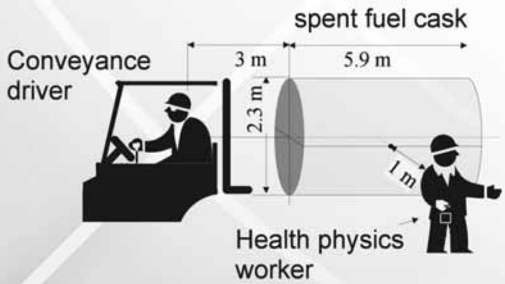


Fig. 1 Schematic view of scenario

PATRAM 2010 Posters : continued

ABSTRACT 356

Looking to the Future Transport of Nuclear Materials in Japan

YUKARI TANAKA

Following a basic national energy policy to establish the nuclear fuel cycle, construction of nuclear fuel cycle facilities of Japan Nuclear Fuel Limited (JNFL) in Rokkasho-mura, Aomori Prefecture has made steady progress with a uranium enrichment plant, a low level radioactive waste (LLW) disposal center, a high level radioactive waste (HLW) storage center and a spent fuel (SF) storage facility already in operation. A spent fuel reprocessing plant is about to start its commercial operation.

Nuclear Fuel Transport Co., Ltd. (NFT), as the only company in Japan that specializes in the transport of nuclear fuel materials, has engaged in transport business, since its establishment in 1973. NFT is mainly responsible for transporting SF and LLW generated from the nuclear power plants to the nuclear fuel cycle facilities. Over 4000 MTU of SF and 300,000 drums of LLW has been transported safely from nuclear power stations to the JNFL facilities.

According to the stable operation of nuclear power plants and progress of construction of Reprocessing Plant in Rokkasho, we will face a new stage of various and massive transport including Fresh Mix Oxcide fuel transport from Rokkasho and spent fuel transport to interim storage facility in Mutsu. Transport of radioactive waste generated from decommissioned nuclear power plants to a disposal site is also envisaged. We have to challenge many issues including the followings to make preparation for new business.

- Development of new packages
- Magnification of Rokkasho (Mutsu-Ogawara) port capabilities, etc.
- Further enhancement in safety and security of transport

This paper presents the summary of our experience and future nuclear fuel transport aspects in Japan together with our planned measures to cope with them of nuclear fuel materials transports in Japan.

ABSTRACT 420

Mox Fuel Transport Experience in Chubu Electric Power Co

YAMADA KATSUMI, TAKEUCHI YASUJI, TAKAHASHI KENJI, SEKIGUCHI RYUSUKE, KUMAZAKI TAKAHIRO

1. Introduction

We are planning to launch the project for utilizing plutonium uranium oxide (MOX) fuel in thermal reactors (referred as pluthermal) at Reactor No. 4 of our Hamaoka Nuclear Power Plant in FY 2010. Japan is poor in energy resources, but consumes large amounts of energy. To ensure a stable future energy supply, it is indispensable that we promote pluthermal power generation and other nuclear fuel cycling methods, which make effective use of uranium resources.

2. Process up to the Start of Use of mixed oxide fuel (Anticipated)

3. Experience of MOX fuel transport

The first shipment of MOX fuel from France to Hamaoka Nuclear Power Plant was successfully completed in May, 2009. The shipment was conducted with physical protection measures to comply with the international agreement on the security of nuclear fuel material, as well as with safety measures to comply with international transport regulations.

This was a great achievement for Chube Electric to take the first step to utilize plutonium for power generation by thermal reactor, and for Japan to improve Pu-balance and to justify our national fuel cycle policy to reprocess spent fuel.

Lessons learnt through the transport and handling of MOX fuel will be presented for improvement of future transport and handling procedures.

TRANSPORT REGULATIONS

ABSTRACT 289

Origin of Radioactive Material Regulations – A Historic Perspective

PETER VESCOVI

In 1936, the attention of transportation authorities in the United States was drawn to the fact that radioactive substances being sent by mail were emitting gamma radiation that was affecting undeveloped photographic films. Postal orders were issued, which remained in effect until 1949, excluding radioactive material from the mails. In the ensuing years, transport of large quantities of radioactive material for atomic weapons programs and frequent shipment of radioactive isotopes for use in hospitals and universities relied on conservative principles of over-shielding, which resulted in using packages with heavy lead shielding. The excessive cost of transportation of packages with the lead shielding brought pressure to develop new regulations that would permit lighter but still safe packaging. A Subcommittee on Shipment of Radioactive Substances was formed to advise the Bureau of Explosives on formulating Interstate Commerce Commission (ICC) regulations governing the packaging, stowage, and transport of radioactive materials.

At its inception this subcommittee recognized that to provide smooth and uninterrupted international transport by air, water, and ground, the regulations needed to be similar in all countries involved. The ICC regulations were adopted verbatim by Canada and appraised by other countries in the process of formulating their own regulations for shipment of radioactive materials. ICC regulations governed the transport of hazardous materials by classifying to broad categories of explosives and poisons. A new subcategory, “Class D Poison” was added for radioactive substances, and basic principles for the safe packaging of radioactive substances emerged based on the philosophy of reducing the gamma radiation to a manageable amount. These principles for the safe packaging and transport of radioactive material were presented in a report prepared in 1951 for the Committee on Nuclear Science Division of Physical Sciences National Research Council by Robley D. Evans, Chairman of the Subcommittee of Shipment of Radioactive Substance. This report was the first useful guide for those responsible for packaging and shipment of radioactive materials. This presentation will provide a historic perspective on the influence this report had on the origin of many requirements in the current regulations for the safe transport of radioactive materials.

PATRAM 2010 Posters : continued

ABSTRACT 291

Proposed New Exemptions Under the Canadian Packaging and Transport of Nuclear Substances Regulations

GENEVIEVE TANGUAY

In Canada, the Packaging and Transport of Nuclear Substances Regulations (PTNS Regulations) which make reference to the IAEA TS-R-1 Regulations provides exemptions similar to those define in the IAEA TS-R-1 Regulations.

Another Canadian regulation, the Nuclear Substances and Radiation Devices Regulations (NSRD Regulations), applies to the use of nuclear substances and provides exemptions from licensing requirements for small quantities of nuclear substances either incorporated in sealed sources qualifying as check sources or in devices containing sealed sources.

The Canadian Nuclear Safety Commission (CNSC) is planning to amend the PTNS Regulations to incorporate the 2009 Edition of the IAEA TS-R-1 Regulations. This opportunity will be used to include in the PTNS Regulations similar exemptions regarding check sources and devices containing sealed sources as found in the NSRD Regulations.

This paper will discuss the proposed exemptions from the transport and packaging regulations of check sources and devices containing small amounts of sealed sources, including the rationale for the proposed exemptions and their impact on Canadians.

ABSTRACT 318

Design Approval of Transportation Packages for Radioactive Materials

WOON-KAP CHO

In Korea, the governing regulation for the use of nuclear energy including safe transport of radioactive materials is the Atomic Energy Act (AEA). Transportation packages for radioactive materials should meet strict safety requirements specified in the AEA and subordinate regulations. Regulatory organizations concerning design approval of transportation packages are the Ministry of Education, Science and Technology (MEST) as regulatory authority and the Korea Institute of Nuclear Safety (KINS) as a regulatory expert body. The design approval of transportation packages prescribed in the AEA is applied to the type B, type C packages and packages for the fissionable material manufactured in Korea or imported from foreign countries. Applicants who want to get design approval for a package should submit the required documents for the safety review of package design. Necessary documents for the safety reviews include the design data including drawings, safety analysis report, quality assurance manual and procedures and package performance test plans. The safety features of transportation packages are being checked by multiple steps of regulation such as careful reviews in the design and the performance tests of the package. Several safety tests such as impact, percussion, heating and leak test are performed on the package according to the AEA and the international safety regulations such as IAEA TS-R-1. If the results of the safety review and tests comply with the safety requirements, the MEST issues the certificate of design approval for the package. The design approval of the transport package is effective for 5 years and the packages should be re-approved every 5 years to extend the expiration date of the design approval. More than 40 package designs for transport of radioactive materials have been certified and 26 certificates of design approval of transportation packages are currently effective. This paper provides Korea’s regulatory procedures for the design approval of transportation packages and brief summary of approved transportation packages.

ABSTRACT 333

The Safety Review on the Sea Transport for the LILW Using the Exclusive Vessel

YONG JAE KIM, DAE SIK YOOK

The Korea Radioactive Waste Management Coorporation(KRMC) is currently undertaking construction of the LILW disposal facility in accordance with the permit issued and will accomplish by 2012.

KRMC have a plan which transport the LILW by sea using a exclusive vessel because that the disposal facility and each nuclear power plant are located in the seashore.

The Korea Institute of Nuclear Safety(KINS) as an expert organization of nuclear safety regulation reviews to evaluate the safety of sea transportation for the LILW in advance in order to prevent any sea contamination and radiation hazard of the fisherman, the seashore resident, radiation workers from the LILW release.

The KINS investigates the laws, regulations, regulatory requirements, technical standards and guidelines concerning sea transportation for radioactive material and composes a safety review team.

The primary safety review items for the safety of sea transport are as follows;

A. The safety of the design and operation of radiation facilities established in the exclusive vessel

B. The safety of the design of packaging for the LILW transport

C. The adequacy of the radiation protection plan for sea transport

D. The adequacy of the emergency response preparedness plan during sea accident

E. The evaluation of radiological impact on the environment adjacent to the rout of sea transport.

And, The safety review team carries out the site inspection for radiation facilities established in the exclusive vessel.

Also, The team checks the operational safety through a training and simulation of sea transportation. Consequently as a safety review results, The KINS evaluates that the safety of sea transport using the exclusive vessel by KRMC could be secured.

ABSTRACT 379

The Schedules to the IAEA Transport Regulations and their Future Development

YONGHANG ZHAO

The “schedules” are well known for providing practical guidance on how to apply the IAEA Regulations in practice. This paper presents the contents of the schedules and the potential developments as an electronic product

PATRAM 2010 Posters : continued

ABSTRACT 387

The IAEA Regulations TS-R-1

JIM STEWART

The IAEA Regulations for the Safe Transport of Radioactive Material were first published in 1961. They have been implemented widely for all modes of transport in the world. This paper describes the Regulations and their underlying principles.

ABSTRACT 393

The IAEA Review / Revision Process

JIM STEWART

The IAEA has established a two year review cycle to ensure the Regulations for the Safe Transport of Radioactive Material remain up to date. In addition there are checks to ensure that the review only leads to a revisions where necessary in order to assure reasonable regulatory stability. This process and the feedback system that supports it is set out in this paper.

ABSTRACT 394

Assisting Competent Authorities in Assessing their Capabilities

JIM STEWART

Several tools have been developed by IAEA to assist competent authorities in assessing their capabilities. The best known of these is the TranSAS mission, however there are several other tools available. This paper sets out each of the tools.

NOTES

NOTES

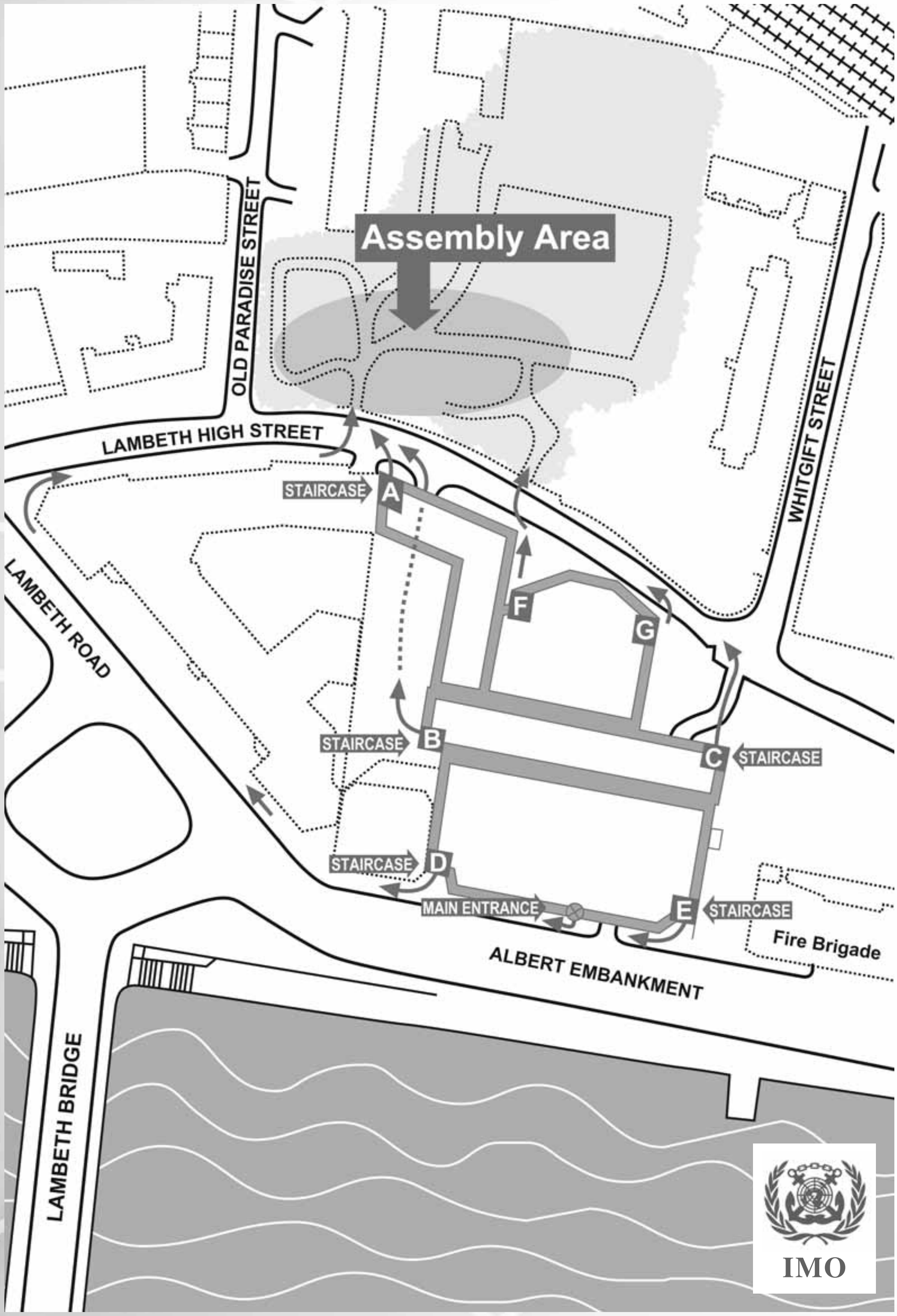
Fire and emergency brief for visitors

In an emergency, the fire and emergency alarms will be sounded followed by an instruction broadcast over the voice alarm system.

On hearing the audible alarm, evacuate the building immediately via designated emergency exits, which are clearly marked, and assemble in the assembly area in the park at the rear of the building.

A plan showing emergency exits and staircases and the assembly area is displayed to the left of this brief.

All visitors are requested to familiarize themselves with the building's emergency exits and the assembly area.



Competence for Nuclear Services

- Nuclear Casks
- Spent Fuel Management
- Waste Management
- Project Planning and Design
- Calculation Services
- Engineering and Decommissioning Services

GNS

GNS Gesellschaft für Nuklear-Service mbH
Hollestraße 7A · D-45127 Essen
Phone +49 (0)201 109-0 · Fax +49 (0)201 109-1100
www.gns.de

PATRAM Symposia are the only international conferences dedicated entirely to the exchange of information on the future of packaging and transport of radioactive materials and, therefore, presents a unique opportunity for the entire professional community, including industry, government, inter-governmental and research organisations to look to the horizon in relation to these issues.

For more details, e-mail or log onto:

info@**PATRAM**2010.ORG

WWW.**PATRAM**2010.ORG