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From Knowledge Transfer to Competence Building in Euratom: from Joint FP7 Projects to Horizon 2020 Programmes

Georges Van Goethem
Innovation in nuclear systems, and Education and Training
EC DG Research and Innovation K.4 / Fission, 1049 – Brussels
Georges.Van-Goethem@ec.europa.eu

One of the main goals of the Euratom research and training programmes is to contribute to the sustainability of nuclear energy by generating knowledge (research) and developing competencies (training). Euratom training programs, in particular, aim to continuously improve and disseminate the nuclear safety culture, in compliance with the Euratom Treaty (1957). For this purpose, special care is taken of the new structure of nuclear industry and regulation in the EU, i.e.:

- increasingly multidisciplinary and international character of the nuclear sector, as a consequence of the privatisation of the energy market in the EU; marked trend towards outsourcing of activities; possible shortage of skilled professionals; new risk governance approach for decision making (including need for the public to understand before accepting)
- marked trend towards longer time scales (from cradle to grave may exceed 100 years); movement of competent workforces across the EU; emphasis on continuous professional development (CPD) to ensure the highest levels of nuclear competences (drawing the lessons from Fukushima); as a consequence, new approaches for human resource management.

Also worth mentioning are the changes in the EU governance framework in higher education and training (E&T) that reflect EU's modernisation agenda (e.g. increasing emphasis on accountability and performance management, trend towards cost-sharing policies, especially under Horizon 2020):

- impact of new European tools for E&T (e.g. ECTS is formally implemented in most EU countries); supporting and steering role by the EC in the *European Credit System for Vocational Education and Training* (ECVET); new approach to VET following the demography and the new culture of the learners (e.g. so-called "X", "Y" and "Z" generations).

As a consequence, one of the new challenges for knowledge transfer and competence building in the EU is to align the requirements related to borderless mobility and lifelong learning with a new type of E&T schemes, for example, by producing, wherever appropriate, *European Passports* in synergy with the stakeholders. Specifically, borderless mobility will be possible only when mutual recognition of competences amongst the Member States of the EU becomes a reality, thereby ensuring, in particular, a single market for services. Lifelong learning in the EU, in particular, requires a new type of CPD, based on the definition of specific qualifications in terms of learning outcomes (knowledge, skills, and attitudes) and on the transfer, recognition and accumulation of those learning outcomes.

Currently there are seven Euratom FP-7 projects of the EFTS type (Euratom Fission Training Scheme): they will be described in the paper, using the above ECVET tools (see also website of the ENEN association: http://www.enen-assoc.org/). They are examples of Euratom responses to the need of a new CPD approach for specific qualifications and competencies in selected domains:

- TRASNUSAFE: health physics sector (e.g., ALARA principle)
- ENEN III Training schemes : nuclear systems suppliers (e.g. GEN III - IV)
- ENETRAP II: safety authorities (e.g., Radiation Protection Expert)
- PETRUS II: radwaste agencies (e.g., repository and systems design)
- CINCH: nuclear and radio-chemistry (e.g. chemistry of nuclear fuel cycle)
- CORONA: VVER personnel (e.g. special assignment training tools)
- EURECA!: super-critical water reactor (e.g. safety of Generation IV).

Lead Cooled Reactors – Material Issues

Alfons Weisenburger
KIT – Karlsruhe Institute of Technology
Alfons.weisenburger@kit.edu

Lead cooled nuclear systems are nowadays investigated and designed as one option for future GENIV reactors. Beside the subcritical system MYRRHA cooled with PbBi that is under design phase at SCK-CEN, the EU – project LEADER aims to design a small lead cooled fast reactor named ALFRED as a Demonstrator for a future commercial system.
This talk will reflect the material challenges in operating such reactor types. The material requirements, the actual selected options and the status of knowledge will be given. A special focus will be on the compatibility of selected reactor materials with Pb alloy coolants and strategies to mitigate these effects. Solubility of steel constituencies in the liquid metal is one of the issues to be considered. Especially austenitic steels suffer due to the high solubility of Ni in Pb alloys from severe dissolution attack at temperatures above 500°C even in oxidizing conditions. Oxide scales that form on steel surfaces in contact with liquid metal are one option to increase the operation field of such materials. However, the use of ferritic/martensitic steels is mainly limited to temperatures below 500°C due to the formation of thick oxide scale that can hinder heat removal significantly. Al surface alloying using pulsed electron beams is one possible method able to ensure the use of steels for higher temperatures and longer operation times. Additional topics briefly discussed will be influence of PbBi on creep rupture of T91 type steel and fretting corrosion of steels in Pb.

The Allegro Project / European Project of Fast Breeder Reactor

Peter Liska, Branislav Hatala
VUJE, a. s., Trnava, Slovakia
peter.liska@vuje.sk

In the Gas Fast Reactor development plan, ALLEGRO is the first necessary step towards the electricity generating prototype GFR. The ALLEGRO start of operation is planned after 2020. The international collaboration is emphasized since the development of ALLEGRO is now part of the Strategic Research Agenda of the European Sustainable Nuclear Energy Technology Platform.
ALLEGRO is a low power Gas Cooled Fast Reactor studied in the European framework. It is a loop type, non electricity generating reactor. Its power is about 75 MWth. Several objectives are assigned to ALLEGRO. At first, it will demonstrate the viability of the GFR reactor system, no reactor of this type having been built in the past. Most of the GFR architecture, materials and components features are considered at reduced scale in ALLEGRO, excluding the energy

conversion system. ALLEGRO will rely on the same safety options as the reactor system. In addition,
the ALLEGRO core will allow the progressive qualification of the GFR ceramic fuel, with the possibility to load some ceramic carbide or nitride sub-assemblies in a first MOX core.

The Advanced Lead Fast Reactor European Demonstrator: Status of the Design

Alessandro Alemberti
Ansaldo Nucleare, Corso Perrone 25, 16152, Genova, Italy
Alessandro.Alemberti@ann.ansaldo.it

The Generation IV International Forum (GIF) member countries identified the six most promising advanced reactor systems and related fuel cycle as well as the R&D needed to establish the feasibility and performance capabilities of the next generation nuclear energy systems known as Generation IV.
Among the promising reactor technologies for fast reactors (Sodium and Lead Fast Reactors) being considered by the GIF, the LFR has been identified as a technology with great potential to meet the goals of increased safety, improved economics for electricity production reduced nuclear wastes for disposal, and increased proliferation resistance.
Ansaldo Nucleare, with its past experience on fast reactors, is promoting research and development of a pure Lead cooled fast reactor as the coordinator of the LEADER project (Lead-cooled European Advanced DEmonstration Reactor) funded by the European Commission in the frame of the seventh Framework Program.
The project aims the development of a Lead Fast Reactor Industrial size plant to a conceptual level and of a LFR technology scaled demonstrator (ALFRED).
Ansaldo Nucleare has in charge the design of the main components/systems (such as the Reactor Vessel, the Steam Generators, the Primary Pumps, the Decay Heat Removal System) for both the LFR plant and the demonstrator.
The paper presents a summary of the project, with a particular reference to the design status of the LFR plant demonstrator called ALFRED (Advanced Lead Fast Reactor European Demonstrator).
The objectives of the activities carried out for ALFRED are:

- define the main suitable characteristic and design guidelines for the facility;
- design a scaled demonstrator fully representative of the industrial size reactor;
- use components/technologies already available in the short term to be able to proceed in the near future to a detailed design followed by the construction phase;
- evaluate safety aspects and perform a preliminary safety analysis;
- minimise the cost of the demonstrator.

Moreover the demonstrator shall confirm that the newly developed and adopted materials, both structural material and innovative fuel material, are able to sustain high fast neutron fluxes and high temperatures.
The authors acknowledge the European Commission for funding the ELSY and LEADER (website, www.leader-FP7.eu) projects in its 6th and 7th Framework Programs. Acknowledgment is also due to all the colleagues of the participant organizations for their contributions in many different topics.

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IAEA Activities on Nuclear Fuels

Uddharan Basak, Gary R. Dyck and Victor Inozemtsev
Division of Nuclear Fuel Cycle and Waste Technology
International Atomic Energy Agency, Vienna, Austria
u.basak@iaea.org

The objectives of Nuclear Fuel Cycle and Material Section (NFC&MS) of the International Atomic Energy Agency (IAEA) are to enhance and further strengthen the capabilities of interested Member States for policy making, strategic planning, technology development and implementation of safe, reliable, economically efficient, proliferation resistant, environmentally sound and secure nuclear fuel cycle programmes. In order to meet these objectives, NFC&MS plans and implements various activities related to nuclear fuel and fuel cycle for thermal and fast reactors. Major activities in the section involve nuclear raw materials, power reactor fuel engineering, spent fuel management, advanced fuel cycle, codes & databases.
The nuclear raw materials group under (programme B1) has the activities related to uranium & thorium resources and integrated nuclear fuel cycle information system (iNFCIS). The power reactor fuel engineering group (programme B2) plans and implements the programme on the technology & performance of current and advanced fuel, fuel rod & fuel assembly, core components & primary circuit materials, water chemistry technologies, post irradiation examination methodologies etc. The activities on spent fuel management (programme B3) and advanced fuel cycles (programme B4) deal with the various issues related to long term storage of spent nuclear fuel and advanced fuel cycle options for the operating as well as the future advanced reactors. The advanced nuclear fuels and fuel cycle options mainly cover the activities such as advanced fuels for water cooled reactors & fast reactors, reuse options for reprocessed uranium, thorium fuel cycle, fuel and fuel cycle for high temperature gas cooled reactors (HTGRs), fuel cycle options for small and medium size reactors (SMRs) with long core life, partitioning & transmutation, minor actinide properties database etc.

Highlights of Research on Containment Phenomena within the Severe Accidents Research Network of Excellence

Ivo Kljenak [1], Ahmed Bentaib [2], Jeanne Malet [2], Renaud Meignen [2]
[1]*Jozef Stefan Institute, Slovenia*
[2] *Institut de Radioprotection et de Sûreté Nucléaire, France*
ivo.kljenak@ijs.si

Within the European Severe Accident Research Network of Excellence (SARNET), more than 50 organisations from Europe, as well as the U.S.A., Canada, South Korea and India cooperate on the research of open issues concerning phenomena that might occur during severe accidents in nuclear power plants. The collaborative work allows organisations to share their knowledge as well as benefit from each other's experience. In particular, the network offers a convenient way to organise benchmark exercises, in which organisations simulate experiments with different codes and compare their results. In the present paper,

some highlights of research within SARNET on phenomena that might occur in the reactor containment during a severe accident are presented.
During a severe accident, hydrogen might be generated due to uncovery of the reactor core and subsequent Zircaloy oxidation. The occurrence of high local hydrogen concentrations in the containment could lead to hydrogen combustion, which could threaten the containment integrity. Containment sprays may be used as a mitigation system to promote atmosphere mixing and prevent high local hydrogen concentrations, as well as to depressurize the containment. Within SARNET, a benchmark on an elementary heat and mass transfer experiment was organised. The experiment consisted in observing the behaviour of individual droplets. The purpose of the benchmark was to compare the performance of heat and mass transfer modelling within the spray models incorporated in different computer codes.
The modelling of hydrogen combustion has been also investigated through the organisation of benchmarks on experiments performed in ENACCEF experimental facility. The facility is located at the Centre National de la Recherche Scientifique in Orléans (France). It consists of a vertical cylindrical "acceleration" tube and a cylindrical dome. A gaseous hydrogen-air mixture is ignited at the tube bottom and the ensuing upward flame propagation is observed. Within SARNET, organisations simulate the experiments using different computer codes and the calculated results are compared.
If the cooling of the reactor core is not successful and the core melts, causing reactor pressure vessel failure, consequently spilling of the core melt into the reactor cavity filled with coolant, and the so-called "ex-vessel fuel-coolant interaction" occurs, which might lead to a steam explosion. Within SARNET, analytical activities on this topic include modelling of premixing of molten core and coolant, triggering of steam explosions, and steam explosions proper. Models of specific phenomena are developed, incorporated in computer codes, and their influence on the overall fuel-coolant interaction phenomenon is analysed.

Perspectives of Nuclear Energy in Canada following the Fukushima Event

Holly L. Anderson [1], Joanne Ball [2]
[1] CANDU Owners' Group, Toronto, Ontario, Canada
[2] Atomic Energy of Canada Ltd., Chalk River, Ontario, Canada
Holly.anderson@candu.org

This paper provides a perspective of Canada's response during the year since the tragic event which impacted the Fukushima Daiichi plant in Japan on March 11, 2011. The perspective is based on reviews undertaken in Canada, in consultation with other countries, and is presented with the kind permission of the Canadian utilities operating nuclear plants, and the Canadian Nuclear Safety Commission (CNSC).
Following the event, the World Association of Nuclear Operators (WANO) and the CNSC requested utilities operating nuclear plants in Canada to document the ability of their plants to address Beyond Design Basis Accidents (BDBA). The four Canadian utilities which operate nuclear plants are: Ontario Power Generation and Bruce Power (both in Ontario); Hydro Quebec and New Brunswick Power. The Canadian nuclear fleet comprises twenty CANadian Deuterium Natural Uranium (CANDU) units designed by Atomic Energy of Canada (AECL) Ltd., and varying in approximate output from 500 to 900 MWe.
The Canadian utilities co-ordinated their responses to WANO and the CNSC through a CANDU Industry Integration Team (CIIT), facilitated by the CANDU Owners Group (COG). The team included representatives from AECL, and from other countries operating

CANDU plants, to ensure all participants stayed aligned and work in a collaborative manner to resolve issues and benefit from the reviews and subsequent follow-up. The responses to WANO and the CNSC documented the basis for continued safe operation of CANDU plants and committed to ongoing assessments and safety improvements, which have now either been completed, are in progress, or planned.
During the past year, the Canadian nuclear industry and the CNSC have engaged the public in both information and licensing sessions, and public opinion on nuclear energy continues to be favorable in Canada. During this period, key licensing approvals have been issued for both continued operation, as well as for plant refurbishment. New build of two units at the Darlington Nuclear plant in Ontario is strongly supported by the local community and is pending a decision by the shareholder.

Research on the Transport and Chemistry of Fission Products in Primary Circuit and Containment Conditions at VTT

Teemu Kärkelä[1], Jarmo Kalilainen[1], Pekka Rantanen[1], Ari Auvinen[1], Jorma Jokiniemi[1,2]
[1]VTT Technical Research Centre of Finland, P.O. Box 1000, FI-02044 VTT, Finland
[2]University of Eastern Finland, P.O. Box 1627, FI-70211 Kuopio, Finland
Teemu.Karkela@vtt.fi

The behaviour of fission products (FPs) during a severe nuclear power plant accident has been studied at VTT for the past 25 years. The focus has been on the transport and chemistry of aerosols in the primary circuit and the containment. In the past containment aerosols have been studied e.g. with intermediate scale facilities AHMED and VICTORIA. In primary circuit the studies have included resuspension and revaporisation phenomena as well as retention of FPs in steam generators. In recent years the main interest has been on ruthenium and iodine, because of their high radiotoxicity and possibility to form gaseous compounds.
Research has included e.g. studies on the transport and speciation of ruthenium at a temperature gradient ranging from 1700 K to 400 K. The deposition profile of Ru was measured with the help of radiotracer (Ru-103). It was found that a significant fraction of ruthenium may transport as gaseous RuO_4. With a help of CFD modelling the retention process of gaseous ruthenium species could be described in details.
FPs deposited on primary circuit surfaces may release back into the gas phase, even after a long period from the beginning of a severe accident. According to studies with EXSI-PC facility when using CsI as a precursor with Ag, B_2O_3 or MoO_3 the release of gaseous iodine even at 400℃ was high. The effect of primary circuit surfaces on the transport of FPs is not considered in current severe accident codes.
Gaseous iodine reacts with air radiolysis products, such as ozone, in the gas phase of containment. The radiolytical oxidation of elemental and organic iodine was investigated using EXSI-CONT facility. The formation of iodine oxide aerosol particles in air by UV(c) radiation was detected on-line in the experiments. The main gaseous reaction products from the radiolytical oxidation of CH_3I were methanol and formaldehyde. During a severe accident, a part of the nucleated iodine oxide particles in the atmosphere will deposit on the various surfaces of containment. It was found that desorption of iodine from the particles deposited on painted surface was enhanced by gamma radiation.

Advances in Fuel Modeling and Design/Licensing Methodologies by Improved Knowledge and Uncertainty Quantification–AREVA Contributions

Viorel Ioan Arimescu
AREVA NP Inc., Richland, WA, 99354, USA
ioan.arimescu@areva.com

Statistical methods involving direct applications of the Monte Carlo sampling technique have been developed in recent years as an alternative to the traditional conservative bounding calculations for fuel rod design and licensing analyses.
The LWR best-estimate methodology for mechanical licensing analyses (first of its kind) that was recently developed by AREVA NP and, which was approved by the US-NRC for BWR applications, is presented herein.
In a nutshell, the new approach consists in quantifying the uncertainties of the relevant analysis outcomes, while the previous methods relied on conservative bounding assumptions and models, in order to compensate for the lack of either phenomenological knowledge or experimental data. In addition, a critical requirement is employing a best-estimate fuel performance code that is well qualified on a comprehensive experimental database, which allows adequate quantification of code model uncertainties.
The basis for the methodology is an uncertainty propagation method. Employing a best-estimate fuel code and a proper definition of uncertainties associated with all significant input variables, which are then propagated through the code, the uncertainties of relevant output parameters are then determined.
Non-parametric order statistics is used to estimate a certain quantile of the distribution of the outputs of interest, which is then compared to pre-set licensing acceptance limit. This technique has the advantage of being less time-consuming than other methods and is also independent of the number of input variables.
Finally, the outline of the methodology is presented focusing on the manner in which the uncertainties of the different input parameters are accounted for. Special attention is given to the verification/validation of the methodology and a simplified numerical example is presented, which also demonstrates the level of conservatism associated with the statistical procedure itself.

Nuclear Energy Scenario for Electricity Generation and Water Desalination in Egypt

Mohammad N. H. Comsan
Egyptian Nuclear Physics Association, Cairo, Egypt
comsanmn@hotmail.com

Egypt is a fast growing country with 78.9 million population and annual per capita installed power 0.286 MW as of July 2008. Moderate to mature population and economic growth trends forecast population and annual per capita installed power to reach 111 million and 0.63 MW by 2032 and 128 millions at and 1.02 MW by 2052, respectively. Taking these trends in consideration, installed electricity generation capacity are forecasted at 70 GW by 2032 and 132 GW by 2052, as compared to the 2008 installed power of 22.6 GW. Meeting these demands is almost impossible using known limited national fossil fuel reserves. Current electricity generation policy exhausts about 65% of country's total fossil production. Crude oil reserves are expected

to deplete by 2012, while gas reserves will be overstrained starting from 2030. A major policy shift towards the use of non-fossil resources is to be adopted.
Moreover, Egypt lies in a semi-arid to arid region where most of its renewable fresh water is transported by the Nile. Because of that Egypt is in severe shortage of potable water. The fast rate in population growth during the past few decades causes a severe adverse social, economic and environmental impact. The most significant phenomenon is the sharp decline in agricultural land and water per capita. It is anticipated that by the year 2030 water per capita will drop sharply, thus approaching the water poverty limit. Egypt's future strategy is based on a demand management approach, water savings and conservation projects: introduce a users' participation policy, enforce public awareness programs, inhibit sources of water pollution, enhance the use of non-conventional water resources, especially water recycling, reuse of drainage water, treated industrial and sewage effluents, rainfall harvesting and sea water desalination.
Aiming at securing its energy demand on continuous bases, Egypt electricity generation plans considers the addition of five NPPs using 1 GWe each by the years 2016 – 2027. Some of these reactors are to be situated at the El-Dabaa site on the Mediterranean Sea coast, 160 km west to Alexandria. The site was chosen based on extensive site characterization and evaluation studies during the period 1976-1986 according to International Atomic Energy standards and procedures.
Analysis and forecast of Egypt's energy resources and needs till 2050 shows the inability to depend on national oil and gas reserves for electricity generation that meets estimated targets at that time. Efficient utilization of energy resources regarding consumption, production and exports/imports requires a major policy shift towards the use of non-fossil techniques for electricity generation. Although wind and solar power can be used efficiently on local scales, yet constraints characteristic of large scale utilization of these renewables show they cannot be used for large scale continuous base load electricity. Resources of hydropower are expected to be utilized completely by 2022. Hence an electricity generation strategy based on gas/coal and nuclear options is suggested. The strategy is based on gradual introduction of nuclear power starting from 2018 and of coal fired plants from 2032. A nuclear share of ~ 13% of installed power is targeted by 2052. The suggested mix is based on careful choice of fossil, nuclear, hydro, wind and solar power and is believed to be most appropriate to meet Egypt's energy demand till 2052.

NEWLANCER: Enlarging NMS Participation in Euratom Programme

Daniela Diaconu, Marin Constantin
Institute for Nuclear Research Pitesti, Romania
daniela.diaconu@nuclear.ro

Despite the EC initiatives to facilitate and attract a large participation of New Member States in the Euratom programme, their involvement in terms of number of projects and budget is still low. In the beginning of the last decade, the IAEA noticed that the East European nuclear R&D institutes had failed to adapt to the changing science and technology paradigm, and consequently risk "lagging behind" in management, structure, planning, and funding, all of which directly affects the sustainability of the institution. EC found also a low participation in the Euratom research programme. It is therefore necessary to assist top managers and senior scientists in nuclear research, to improve their management practices and improve access to national and international funding opportunities.
In this regard, NEWLANCER brings together nuclear research and academia from New and Old Member States in a complex and dynamic multi-level network to identify the most

efficient and effective solutions leading to a better integration of the NMS research and education at European level. This network will be the interface with the institutional management and national research authorities on one side, and to the European structures framing nuclear energy research strategies at the other side, aiming to make more visible the specialists point of view on the research needs and priorities. This network will explore current research capabilities in the NMS, will identify opportunities for future project proposals, and will produce working plans for a long-time cooperation and an advanced cohesion at regional level. This network will come up with proposals raised at institutional level, aligned to national strategies and distilled into a regional perspective.
NEWLANCER will also analyze the NMS participation in Euratom Framework Programmes, and compare and contrast this with the actual research potential of nuclear R&D organizations, taking into consideration the European strategic research agendas and energy policies. This comprehensive review will give a clear picture of the obstacles and difficulties that must be overcome. A catalogue will list the NMS existing research skills and infrastructure that should be further promoted.
Best practices and recommendations extracted from the Euratom success stories will reflect the findings of all project partners, both specialists and managers and will be addressed to a large spectrum of end-users interested in better integration of the research potential across EU. These will be applied inside the project in a pilot exercise for the elaboration of a new project proposal intended to integrate in the new built consortium organizations not yet-involved in Euratom projects either from NMS or OMS with low level of participation.
NEWLANCER will pave the way for a sustainable participation of the research institutes and universities from New Member States in nuclear energy research as framed by the agenda defined by European policies and initiatives such as SET Plan, SNETP, IGD-TP, ESNII, EERA, MELODI.

Status of National Participation in the INPRO Project of IAEA

Eugen Bancheş[1], Cristina Alice Margeanu[2]
[1] Nuclear Agency & Radioactive Waste, Romania
[2] Institute for Nuclear Research Pitesti, Romania
eugen.banches@nuclearagency

The INPRO Project (International Project on Innovative Nuclear Reactors and Fuel Cycles) started in 2000 by IAEA Member States (MS) and International Organizations (IO) interested in contributing to the development of innovative nuclear reactors and fuel cycles, proliferation resistant and able to contribute to an appropriate energy mix for society global sustainability in the 21st Century, has achieved the maturity stage in 2011.
INPRO Methodology for the "sustainability" of Nuclear Energy Systems (NES) assessment and NES assessment tools (Nuclear Energy System Assessment - NESA) were developed and validated by MS and IO contributions "in-kind" during over 10 years of sustained dialogue and cooperation activities as INPRO biennial Action Plans, Dialogue Forum on Common User Considerations (CUC) for NES, Collaborative Projects (CP) and consultations in INPRO Steering Committee Meetings (INPRO SCMs).
The 2011 year was for the Romanian institutions an opportunity to become involved in identifying and defining national interest activities (training national specialists and application of the INPRO Methodology and tools in order to assess scenarios of NES innovative development at national level) in the Action Plan project INPRO 2012-2013 and to assess the resources available to join MS and IO contributing "in-kind" to the development of methodology and

instruments NESA (one of the sufficient voluntary contributions in order to confirm by the IAEA the INPRO Membership status to the interested MS level).

In 2011, AN&DR initiated the promotion of the Romania participation in INPRO membership, through a National INPRO Group, joined by INR Pitesti, SNN, ICSI Rm Valcea and SITON.

Three CPs, respectively PROSA, SYNERGIES and ROADMAPS, the INPRO Dialogue Forum and INPRO SCM were identified as points of immediate interest for Romania's involvement in INPRO membership with contributions "in-kind". Meanwhile, CP MBIR and Dialogue Forums on CUC for SMRs and on Long-term prospectus for nuclear energy (post Fukushima), remain of interest for national participation in INPRO with Observer status.

AN&DR has initiated a memorandum to the Prime Minister to obtain the approval for Romania's participation in the IAEA INPRO Project membership based on contributions "in-kind" in CPs and SCM, as are specified in the national commitments of institutions concerned.

AN&DR analysed its own prospective resources available to launch a NESA at national level with INPRO Methodology, in two priority areas, namely in the Radioactive Waste Management and in the Resistance to Proliferation. It will also be analyzed the interest and the prospective resources available to other authorities and national institutions interested to extend NESA at national level in the areas of Nuclear Safety, Physical Protection, Environmental Protection, Economic Efficiency and Development of Nuclear Infrastructure, in order to support the harmonization of the legal basis for the NES development.

AN&DR's main target is to use INPRO Methodology and the IAEA support to obtain a substantiate update of the National Strategy for Development of Nuclear Field, National Strategy on Medium and Long Term Management of Spent Nuclear Fuel and Radioactive Waste, including the Disposal and Decommissioning of Nuclear and Radiological Facilities, National Energy Strategy, and National Strategy on Nuclear Safety.

I. NUCLEAR ENERGY

I.1. Nuclear Safety and Severe Accidents

I.1.1. The CANDU Owner's Group Severe Accident Research and Development Program

Joanne Ball [1]***, Holly L. Anderson*** [2]
[1] Atomic Energy of Canada Ltd., Chalk River, Ontario, Canada
[2] CANDU Owners' Group, Toronto, Ontario, Canada
Ballj@aecl.ca

The tragic event which impacted the Fukushima Daiichi plant in Japan on March 11, 2011 resulted in a re-evaluation and review by the nuclear industry world wide of the ability of plants to address Beyond Design Basis Accidents (BDBA). The Canadian utilities coordinated their responses to the World Association of Nuclear Operators (WANO) and the Canadian Nuclear Safety Commission (CNSC) through a CANDU Industry Integration Team (CIIT), facilitated by the CANDU Owners Group (COG). The team included representatives from AECL, and from other countries operating CANDU plants, to ensure all participants stayed aligned and work in a collaborative manner to resolve issues and benefit from the reviews and subsequent follow-up.
The integration of industry efforts through COG to respond to Fukushima drew upon parallel R&D activities, also undertaken through COG, to address the uncertainties associated with Beyond Design Basis Events in CANDU reactors. The program was started in 2008/09, focusing on developing an understanding of phenomena that had previously been identified through a Phenomenon Identification and Ranking Table (PIRT) process as being poorly understood, and having a large impact on retention of severe accident progression. An additional element of the original program was to investigate the behaviour of Passive Autocatalytic Recombiners (PAR) under BDBA conditions.
Events at Fukushima have reinforced the importance of Severe Accident research, and several new activities have been introduced to the COG R&D program in light of the event. This paper discusses the COG R&D Severe Accident program, highlighting what has been learned, and outlining plans for future work.

I.1.2. Stress Tests for Cernavoda NPP - the Regulatory Perspective

Cantemir Ciurea - Ercau, Elena Dinca, Madalina Tronea
National Commission for Nuclear Activities Control, Bucharest, Romania
cantemir.ciurea@cncan.ro

Following the Fukushima Daiichi accident occurred in March 2011, the Romanian authorities and the nuclear industry have performed reassessments of nuclear safety and emergency preparedness arrangements and have started to implement improvements, in line with the international efforts in this direction. The safety reassessments conducted in response to the Fukushima accident included the "stress tests" review required by the European Council, in compliance with the specifications and criteria issued by the the European Commission, based

on the work done by the European Nuclear Safety Regulators' Group (ENSREG) and the Western European Nuclear Regulators' Association (WENRA).
A significant effort has been made by the licensee for Cernavoda Nuclear Power Plant (NPP) to respond to the lessons learned from the Fukushima accident in a timely manner. Several potential design and operational improvements have been identified and their implementation is in progress to further enhance the existing safety margins and reduce the risk from severe accidents.
The reviews and inspections performed by the National Commission for Nuclear Activities Control (CNCAN) confirmed the licensee is able to support all the claims made in their "stress test" report and that any issues and opportunities for improvement arising from the stress test will be adequately addressed. This conclusion is supported also by the results of the peer review performed by a team of experts coming from the European Commission and several nuclear safety authorities from all across Europe.
This paper presents the current status of the work performed by CNCAN and Cernavoda NPP in relation to the European "stress tests", including the results of the peer review and the actions taken or planned to improve safety.

I.1.3. CANDU Severe Accident Phenomena Simulated by ASTEC Code

Marin Constantin, Andrei Rizoiu, Minodora Apostol, Alina Constantin
Institute for Nuclear Research Pitesti, Romania
marin.constantin@nuclear.ro

Some efforts were performed in the last years in order to investigate ASTEC (Accident Source Term Evaluation Code) adaptability to CANDU type reactors. A part of the work was dedicated to analyze which modules are appropriate and which need adaptations. Another part consisted of the development of algorithms and methods in order to simulate core degradation process which is very different in CANDU compared with PWR type reactors.
The paper includes results and considerations about the following modules: SOPHAEROS, CPA, IODE, MEDICIS, ISODOP, DOSE, CESAR, DIVA/ICHARE. Some calculations were performed in stand-alone mode and others in coupled mode. The exploratory calculations proof that the code may be successfully used for simulating the main severe accident phenomena in CANDU such as fission products transport in the primary heat system and containment, thermalhydraulics, fission products' chemistry, fuel cladding oxidation by steam, molten core concrete interaction, etc. Therefore ASTEC code may be used for detailed investigation of some severe accident phenomena in CANDU type reactors. All the results presented in the paper are oriented to the source term phenomena simulation and final evaluation of the source term released into the environment.
The paper also presents the progress of the efforts in developing models for CANDU core degradation in order to cover all the accident phenomena.

I.1.4. Investigating the Relation between Common Cause Coupling Factors and Ageing

Mirela Nitoi [1,2], Jaroslav Holy [3], Irina Dinu [4]
[1]Institute for Nuclear Research Pitesti, Romania
[2] EC JRC- Institute for Energy, Petten, The Netherlands
[3]*Nuclear Research Institute Rez, Prague, Czech Republic*
[4]*Cernavoda NPP, Romania*
mirela.nitoi@nuclear.ro

Usually, probabilistic safety analyses (PSA) are using the assumption of the constant component failure rate, and the assumption that the probability associated to common cause failure will be constant in time. Still, as ageing phenomena could have some influence on components or systems performances, is probable that it could influence also the probability of occurrence for common cause failures (CCF).
This paper is devoted to investigation of the most important common cause failures initiating factors from point of view of relation to ageing phenomena. The connections between common cause failure potential and ageing phenomenon, as the influence of the individual prevention factors against CCF were evaluated.
A questionnaire has been developed on this topic and all the inputs provided were summarized, compared and commented.
The analysis has not been limited only to the list of negative factors that would increase the CCF potential, it has included also positive factors representing the prevention and correction measures applied with the aim to avoid occurrence and recurrence of CCF events, whether related to ageing or not.
Two main topics were discussed:
- the strength of relation between the individual CCF coupling factors (contributing to total CCF potential) and ageing phenomenon;
- the effect of CCF prevention measures on the coupling factors versus ageing relation (estimated for those CCF coupling factors, which have been evaluated as coincident with ageing phenomenon, with at least medium level of coincidence).

I.1.5. Calculation of Argon-41 Production and Transport in Romanian TRIGA Facility using CATHARE2

Daniela Mladin [1], Mirea Mladin [1], Alexandru Toma [1], Ilie Prisecaru [2]
[1] Institute for Nuclear Research Pitesti, Romania
[2] Power Engineering Faculty, University "Politehnica" of Bucharest, Romania
daniela.mladin@nuclear.ro

A TRIGA facility model with CATHARE2 was created previously for PSA support studies with externally calculated fission product sources, for ^{87}Kr, ^{133}Xe, ^{131}I and ^{137}Cs. In present paper, the model for the TRIGA facility is used for calculating the ^{41}Ar transport and evacuation. Argon activity at reactor stack is calculated for normal operation and compared to monitor readings.
As the coolant water passes through the reactor core, ^{41}Ar will be produced through the reaction: ^{40}Ar (n, γ)^{41}Ar, where the ^{40}Ar is dissolved in the water. The gas source is produced by outgasing of ^{41}Ar due to heatup of water when it passes through the reactor core. The properties of ^{41}Ar as a radio-chemical component in CATHARE2 are user-defined. Then, two sources of ^{41}Ar are defined at the height where the core is placed. The

first, in liquid phase, comprising initially all the ^{41}Ar concentration produced by the (n, γ) reaction in ^{40}Ar. The second source, gaseous is created by retrieving the ^{41}Ar in the pool water and decreasing it while the gaseous source increases. After some time the sources reach the equilibrium.
Using MCNP code, the ^{40}Ar concentration at saturation 20 °C was introduced in the water surrounding TRIGA fuel elements, over the active length of the fuel (55.88 cm), for all 29 fuel bundles in the core configuration. Requesting the (n, γ) reaction tally in ^{40}Ar we produced the ^{41}Arsource.
The paper presents a model for TRIGA reactor to be used in the calculation of the ^{41}Ar activity at stack in normal operation and comparison of results against experimental data.

I.1.6. The Flame Front Behaviour Following a Severe Accident in a Generic CANDU 6 Containment

Gabriela Radu[1], Ilie Prisecaru[2]
[1]Institute for Nuclear Research Pitesti, Romania
[2]Univesity Politehnica Bucuresti, Romania
gabriela.radu@nuclear.ro

During a severe accident in a nuclear power plant, oxidation of the metallic components of the reactor core will produce hydrogen. The radiolysis of water and the corrosion of metals in containment are additional sources of hydrogen. This could potentially lead to the formation of flammable mixtures of hydrogen/air/steam in reactor containments. Combustion of these mixtures could lead to pressure and temperature levels that may challenge the containment integrity. Hydrogen combustion can cause containment building failure by static or quasi-dynamic pressure loads (deflagration or diffusion flame) or dynamic pressure loads (detonation), equipment failure due to thermal or pressure effects, and missile generation. Deflagration (slow flame) and detonation (very fast flame) are two completely different modes of combustion, with different propagation mechanisms. As detonation leads to spatially non-uniform pressures that exceed the maximum pressure caused by deflagration, this combustion mode is of interest in reactor safety analyses. A hydrogen detonation can be developed by either direct initiation, which requires high energy source, or flame acceleration (a Deflagration to Detonation Transition, DDT).
This paper presents a study of the behaviour of the flame front following in vessel phase of a severe accident initiated by a Station BlackOut (SBO) in a generic CANDU 6 station containment. The calculations are done, using the CPA-Front model of the code ASTEC, with different sets of model parameters with the best fit on the experimental data. The aim is to investigate the influence of some varied input parameters for the flame front model with respect to pressure and temperature histories in different containment zones.

I.1.7. Development of Instrumentation and Control (C&I) Systems in NPP using Modern Techniques of Distributed Digital Control Systems

Magda Vlad, R. Popovici, Mircea Popescu
Subsidiary of Technology and Engineering for Nuclear Projects, Romania
vladmg@router.citon.ro

The present paper deals with theoretical aspects of advantages for using digital control, monitoring and diagnosis techniques that can be applied in CANDU Nuclear Power Plant (NPP).
Progress in electronics and information technology (IT) has created incentives to replace traditional analog instrumentation and control (C&I) systems in NPPs with digital C&I systems, i.e. systems based on computers and microprocessors. Digital systems offer higher reliability, better plant performance and additional diagnostic capabilities. Analog systems will gradually become obsolete in the general IT shift to digital technology.
Instrumentation and control systems based on digital technology have many merits, compared with those of analog type, as follows: performing of control functions upgrading, reduction of systems components, design of circuits capable to realize complicated logic judgment, and displaying graphical information in order to easy understand process parameters such as flow rates, pressures, and operational states of pumps and valves.
The Human Systems Interface (HSI) in the control room means the translation of plant information into required operator actions, being critical towards the execution of a safe operation. Computerizing a HSI means to incorporate features such as computerized procedures, digital displays, touch-screen interfaces, pointing devices and large-screen overview displays. Computerization allows more tasks to be done by operators sitting at their workstations without moving outside the control room.
Because of these advantages and circumstances, the digital instrumentation and control systems utilization cannot be avoided and rather should be promoted.
Theoretical aspects outlined in this paper meet the requirements of nuclear safety, increase efficiency in operation and decrease radiation exposure to the operational personnel. These solutions can be implemented in Cernavoda NPP Units 1 and 2 in the modernization programs, or in the future projects for Units 3 and 4.

I.1.8. Thermal-hydraulic Analyze for Recirculated Cooling Water System from Cernavoda NPP Unit 1

Melania Bigu, Rodica Pancef, Nicoleta Anghel, Iulian Pavel Nita
Subsidiary of Technology and Engineering for Nuclear Projects, Romania
nitai@router.citon.ro

The paper presents the hydraulic analyze made for Recirculated Cooling Water System (RCWS). The work was required by increasing of 2 consumers: new nominal flow for Moderator Heat exchanger and new heat exchangers for Blow down system. More, there were some inconsistencies between operating manual isolating valves on load shedding procedure and design manual requirement for some systems (i.e. Main pumps motor cooling and D_2O tower flowrate required for loss of class IV regimes).
The paper introduces a new consumer for Turbine Building part of the system namely "Urban Heating". The system was tested for operating to several stationary regimes as follow:
- Nominal regime

- Reactor trip operating on class IV
- LOCA (loss of cooling agent) operating on class IV
- Reactor trip with loss of class IV
- LOCA with loss of class IV

For all above mentined regimes one studied capability of the RCWS to supply flowrate to all system consumers and to keep the pumps and by-pass lines in normal operating condition.

I.1.9. The Quasistatic Bending of a Bernoulli-Euler Beam – Some Generalized Analytical Models

Cezar Doca
Institute for Nuclear Research Pitesti, Romania
cezar.doca@nuclear.ro

Addressing the problem of the quasistatic bending of homogenous straight beams (bars), the paper presents, from simple to complex, the analytical models for a total of 19 general situations, individualized by the combination of: (a) the absence / presence of an elastic base (bed, support), (b) the absence / presence of a compressive/tensile axial force, (c) beam with constant / variable moment of inertia, and (d) beam with constant / variable distributed load.
More exactly, the generalized hypostasis vary from the simple case of a “beam without elastic base (bed, support), without axial force, with constant moment of inertia and with constant distributed load” to the complex case of a “beam supported on elastic base (bed, support), with tensile axial force, with variable moment of inertia and with variable distributed load”.
The generality of these (mathematically both beautiful and powerful) models enables their application in the study of the quasistatic bending of any mechanical structure equivalent to a homogeneous straight beam, i.e. a Bernoulli-Euler beam, examples for a Nuclear Power Plant being: fuel elements, pressure tubes, pipes, structural beams etc.
All the analytical solutions were verified with the *Wolfram MATHEMATICA®* software.

I.1.10. The Influence of Heat Generated by the Zirconium-Steam Reaction on Thermalhydraulic Behaviour of CANDU 6 Core

Marieta Serbanel
Institute for Nuclear Research Pitesti, Romania
marieta.serbanel@nuclear.ro

Two sources of heat generation are considered in the heat transfer calculations for CANDU 6 core using CATHENA code. These are the heat generated by the fuel pellet and heat generated as a result of the zirconium-steam reaction at high temperatures.
The heat generated by the zirconium-steam reaction can be controlled by the phenomenon of steam starvation. The inclusion of this behaviour in a CATHENA model is controlled by the "H2-TO-NC" or "STEAM-STARVATION" numeric options. These numeric options are mutually exclusive (i.e., only one of them can be specified) since each employs different methods for tracking the H2 produced and the steam available to support the zirconium-steam reaction. If these options are used and all the steam within a thermalhydraulic node is predicted to be consumed by the zirconium-steam reaction, then

the reaction rate is limited such that all of the available steam is consumed. No reaction limiting due to H2 blanketing of the zirconium surfaces is assumed in the reaction calculations.
Effect of heat generated by zirconium-steam reaction on thermal-hydraulic behaviour of CANDU 6 reactor has been assessed for the RIH 35% LB LOCA. Also, the phenomenon of steam starvation using the both numeric options has been evaluated in this paper.
CATHENA models for a PHTS circuit and fuel channel analyses have been developed for a CANDU 6 reactor loaded with NU37 fuel bundles, and the demonstration analyses results for a RIH 35 % large break LOCA are presented.
The results of NU37 core thermal-hydraulic behaviour during RIH 35% LB LOCA are discussed comparatively with the results of NU37 core thermal-hydraulic behaviour with zirconium-steam reaction.
Generally speaking, the CATHENA circuit analyses results presented in this paper fall within the foreseeable expectations with reasonable trends.

I.1.11. Identification of Components Susceptible to Ageing by Probabilistic Techniques and Quantification of Ageing Effects

Minodora Apostol
Institute for Nuclear Research Pitesti, Romania
minodora.apostol@nuclear.ro

The probabilistic techniques, the analysis of operating experience and expert judgments represent important methods used for selection and prioritization of safety significant component susceptible to ageing.
The risk importance due to ageing can be evaluated by Probabilistic Safety Assessment (PSA) techniques, even if the ageing is not taken into account in PSA analyses. Two risk importance measures are recognized in these kinds of analyses: Risk Achievement Worth (RAW) and Fussel-Vesely (F-V) Importance. In order to obtain a single ranked list of events, RAW and F-V importance are evaluated by means of Smith Weighted Importance Measures (SWIM) index. The index allows to analyst to combine information from F-V list of events with the RAW list of events. Using a weighting factor selected by the user, a single ranked list of ageing-important components can be obtained.
The paper presents the ranked list of basic events for a top event considered in Cernavoda NPP level 1 Probabilistic Safety Evaluation study. The importance measures RAW and F-V are calculated using EDFT, a computer code developed in Institute for Nuclear Research (INR) Pitesti. The scaled risk threshold in the SWIM derivation has been chosen to be 0.005.
For quantification of ageing effects of the components selected with SWIM index in estimation of top event unavailability, the linear ageing model is used. The top event unavailability without ageing consideration and with ageing for age of components at 10, 20, 30 and 40 years will be presented.

I.2. Nuclear Reactors and Gen. IV

I.2.1. Long-Lived Small LMFR. Core Design Challenges

***Georgios Glinatsis*[1], *Daniela Gugiu*[2]**
[1]ENEA, Bologna, Italy, [2]INR, Pitesti, Romania
georgios.glinatsis@enea.it

In the recent years a variety of small (~ 300 MWe) and simpler reactors for electricity generation from nuclear power and for heat processing have been proposed. This interest is based on a desirable capital cost reduction and on the goal to provide energy from nuclear power away from large grid and/or large infrastructure systems, to minimize the environmental impact, to reduce the transport cost of fossil fuels and the energetic dependency.
Integrated systems, compact configurations, long-lived cores, etc. under various technologies: LWR, FNR and graphite-moderated HTR, are developed. The first one has the lowest technological risk, but the second can be smaller, simpler and with longer operation before refueling. The better neutron economy of the FNRs allows the use of the energy potential of uranium, while their high power density facilitates the development of compact reactor.
Compact cores allow a facile transportation and modularity, while long-lived compact cores allow also the improvement of the proliferation resistance behavior. In the same time the long-lived cores size is conditioned firstly by the burn up reactivity loss and it can be also conditioned by the fuel burn up limit and the neutron damage limits for both fuel and cladding.
In order to meet the goals of the long life core option, an appropriate approach to define the size of the core has been proposed. The impact on the criticality level and on the neutron performances for FNR with long life core is discussed. The neutron parameters dependency on the fuel irradiation cycle length has been also highlighted.
The feasibility and the success of the long-lived cores are strongly dependent on the new materials availability and on the high accuracy of the nuclear data.

I.2.2. Neutronic Analysis of the Structural Materials used in Heavy Liquid Cooled Fast Reactors

***Daniela Gugiu* [1], *Carlo Petrovich* [2]**
[1] INR, Institute for Nuclear Research Pitesti, Romania
[2] ENEA, via Martiri di Monte Sole 4, IT - 40129 Bologna, Italy
daniela.gugiu@nuclear.ro

One of the main constraints in the design of the Heavy Liquid Cooled Fast Reactors regards the selection of the structural materials that could undergo changes of their mechanical and physical properties because of the hard neutron spectrum irradiation. These changes are related to various factors such as the intensity and the energy of the radiation, the irradiation time, the temperature or the type of the material.
Besides the evaluation of the radiation damage of the reactor components through the computation of the standard damage parameter for the nuclear materials, the DPA (displacements/atom), attention should be also paid to the burnup and transmutation of the materials envisaged to be used in the harsh environment of the lead-cooled fast reactors.
Additionally, the neutron activation of the reactor internals should be evaluated during the design stage of the reactor because they represent an important source of gamma

radiation that should be taken into account in the shielding design. Moreover, the isotopic inventories of the structural materials are required for their classification as waste.
A preliminary evaluation of the hydrogen and proton production in the lead will be also presented; their concentrations are of interest for the oxygen control in the lead coolant. All the calculations are performed for the actual design of a lead cooled type reactor using as main computational tools the Monte Carlo transport code MCNPX 2.6.0 and the isotope inventory code FISPACT 2005.

I.2.3. Study of Polonium Removal Methods for Lead and Lead Bismuth Cooled Fast Reactors

Ramona-Maria Mihaiu, Viorica Maria Nistor
Institute for Nuclear Research Pitesti, Romania
ramona.mihaiu@nuclear.ro

The *GIF Technology Roadmap* identified the Lead-cooled Fast Reactor (LFR) as a technology with great potential to meet the small-unit electricity needs of remote sites while also offering advantages as a large system for grid-connected power stations. The lead-cooled fast reactor is cooled by either lead (Pb) or a lead-bismuth alloy (Pb-Bi), and features a fast neutron spectrum and a closed fuel cycle. Pure lead coolant has a few advantages over the Pb-Bi alloy. It provides better corrosion resistance, and the creation of the radioactive polonium isotope Po-210 is ~ 10000 times smaller in pure lead than in the lead-bismuth coolant. One of the important issues associated with using lead-bismuth as a reactor coolant is the radiological hazard due to activation of the lead-bismuth in such a reactor.
In a typical lead-bismuth cooled reactor the primary system is sealed and segregated from the secondary system by the steam generator. As a result, polonium is retained in the lead-bismuth eutectic during normal operating conditions and can cause problems only if coolant leakage occurs. However, some polonium migrates to the cover gas in the reactor plenum and will diffuse outside the primary system when the reactor is opened for refueling or maintenance.
The main purpose of this paper is to analyze the polonium issue in LFR and LBE-Cooled Fast Reactors and to identify and describe the methods/techniques for polonium removal, taking into account the considerable Russian experience in dealing with polonium-related issues that has been gained since more than 30 years (where several submarines were equipped with LBE-cooled nuclear reactors). In time, a polonium technology was developed including special polonium filters for air cleaning, polonium-adsorbing adhesive films for decontamination of large surfaces, special respirators, and pressurized suits for maintenance of contaminated areas.
The paper will present a history of first published polonium extraction methods and techniques and will analyze the main technologies also taken into consideration by the important international laboratories today: *Polonium Hydride Stripping*, *PbPo Distillation*, *Alkaline Extraction, Electro-deposition (or electroplating)*, *Formation of Rare-Earth Polonides*.

I.2.4. Using Low Void Reactivity Bundles in CANDU® Reactors

Andrei Rizoiu, Grigore Horhoianu
Institute for Nuclear Research Pitesti, Romania
andrei.rizoiu@nuclear.ro

Recovered Uranium (RU) fuel is intended to be used in CANDU reactors, given the important amount of slightly enriched Uranium (~0.96% w/o U235) that might be provided by the spent LWR fuel recovery plants. Though this fuel has a far too small U235 enrichment to be used in LWR's, it can be still used to fuel CANDU reactors.

The positive void reactivity is a major concern of any CANDU safety assessment, therefore reducing it is a long-time challenge for the bundle design and performance. Using the 43-element CANFLEX®-like design together with a certain amount of burnable poison (e.g. Dysprosium) dissolved in the central element may lead to significantly reducing the void reactivity.

The purpose of this paper is to investigate the effects of using 43-element RU bundles containing up to 7% Dy dissolved in the central element. The RU in this particular element was replaced by natural (NU) or depleted (DU) Uranium.

The expected outcomes of these design improvements are: higher exit burnup, smooth/uniform radial bundle power distribution and reduced void reactivity.

The cell calculations were performed using the transport codes WIMS and DRAGON, as well as a 69-group updated nuclear data library.

I.2.5. Effects of Low Void Reactivity Fuel Using on Core Integral Parameters in a CANDU 6 Reactor

Iosif Prodea
Institute for Nuclear Research Pitesti, Romania
iosif.prodea@nuclear.ro

The paper continues earlier studies started to find out the influence of Low Void Reactivity Fuel (LVRF) using on the average discharge burnup (ADB), refuelling starting point and on the Coolant Void Reactivity (CVR) in a CANDU 6 nuclear reactor. Previous simulations performed using enhancements added to the 3D diffusion code DIREN allowed for the possibility to simulate automatic refuelling operations up to 700 Full Power Days (FPD) for the CANDU 6 standard fuel design and about 1000 FPD for Recovered Uranium (RU-43) fuel design.

In this paper some configurations of RU-43 fuel bundle design with burnable absorbers (BA) in the central element (CE) and 0.96% enrichment were taken into account in order to find out their suitability to be used in CANDU 6 time-average and core burnup calculations. Dysprosium was chosen as burnable absorber and placed in decreasing amounts in the CE of the bundle (from 2 to 7 wt%).

Time-average calculations have been firstly performed to obtain critical cores, symmetric Zone Control Units (ZCU) power distributions and maximum channel powers close to 6500 kW.

The voiding effect simulated by uniform reducing of the coolant density revealed that the CVR decreases by Dy concentration increasing from 15.7 to 1.2 mk (WIMS calculations, no leakage) and from 13.7 to 6.7 mk through the core calculations. Despite of safety benefits in reducing CVR, the presence of absorbers in the CE also leads to penalties in ADB evaluated by DIREN time-average calculations. These ADB values range from 13.2 to about 4 MWd/kgU, according to Dy concentration increasing.

The possibility of performing automatic refuelling operations in actual DIREN code version is also investigated and presented in the paper.

I.2.6. Study of Spent Fuel Parameters for Low Void Reactivity Bundles in CANDU Reactors

Cristina Alice Margeanu, Andrei Rizoiu, Gheorghe Olteanu
Institute for Nuclear Research Pitesti, Romania
cristina.margeanu@nuclear.ro

The positive void reactivity is a major concern and an important criticism addressed to operating CANDU reactors. Atomic Energy of Canada Limited (AECL) is developing the Low Void Reactivity Fuel (LVRF) to improve the safety margins of the reactors and allow them to operate at their design capacity. The new fuel containing a neutron absorber (usually Dysprosium) and slightly enriched uranium is intended to reduce the void reactivity coefficient and the size of the resulting power pulse in a large break LOCA, which in turn provides greater assurance that fuel channel damage will be avoided.
The paper performs a study of spent fuel parameters (radioactivity, thermal power and gamma energy) at the end of irradiation (EOI) for the RU43-LVR fuel bundle (43 elements advanced LVR Recovered Uranium fuel bundle, developed in INR Pitesti and similar to Canadian CANFLEX-LVRF bundle) by considering various central element compositions and exit burnups. RU43-LVR fuel bundle elements (except for the central element) contain RU pellets, 0.96 wt% in ^{235}U.
The following central element compositions have been used: RU (0.96 wt% in ^{235}U), natural Uranium (NU) or depleted Uranium (DU) (0.2 wt% in ^{235}U) oxides, the dissolved neutron absorber being Dy (up to 7 wt%).
Fuel exit burnups have been provided by previous DRAGON and WIMS-D5B multigroup transport lattice codes calculations. Fuel elements irradiation was modelled by assuming constant power for entire irradiation period, the considered power being 45 kW/kg HE. In order to simulate LVRF bundles burnup ORIGEN-S code, included in SCALE6 programs package, was used. The spectral neutron cross-sections weighting factors (used as input in ORIGEN-S data) were given by lattice codes calculations. The paper provides EOI spent fuel parameters comparison both for different central element compositions and a specific composition but with different Dy concentrations, by taking into account the considered exit burnups.

I.2.7. Enhancing Nuclear Instrumentation: A Step Toward an Increased Control, Reliability and Accident Risk Mitigation at Cernavodă NPP

Laurențiu Ciufu, Mircea Popescu
Subsidiary of Technology and Engineering for Nuclear Projects, Romania
ciuful@router.citon.ro

The various issues and concerns surrounding Cernavoda NPP have created the need for more creative instrumentation and control (I&C) solutions. The needs have also opened the door for new I&C related products that reach out well beyond the traditional plant control loop. These products have been made possible by recent advances in I&C and digital technology. They include digital I&C, computerized calibration tools, testing and maintenance tools, real time process surveillance and diagnostic systems, engineering

analysis and simulation tools, and more advanced fault tolerant control systems that can allow the plant to operate until it can be conveniently shutdown for repairs. The advanced I&C concept starts from the base of proven CANDU reactor technology coupled with demonstrated new features in order to enhance economics, safety, operability and maintenance. Hence, in this paper, is presented an approach which consists of a particular emphasis on enhancing the Cernavoda's nuclear instrumentation in order to implement a key strategy to increase control, reliability and accident risk mitigation. NPP's I&C systems must be accurate to properly sense and communicate the process variables and reasonably fast to provide timely display, adjustment, and protection against upsets in both the main plant and its ancillary systems. This paper describes new I&C technical solutions, which theoretically can be implemented in actual operating Cernavoda NPP Units 1&2 and also future 3&4 Units, in order to improve the plant's automation, having the main effect to increase the efficiency of the actual operating mode. The new I&C systems described in this paper represent a key factor in meeting the operating requirements, as well as the nuclear safety objectives, which can significantly enhance the nuclear performance goals.

I.2.8. Destructive Examination of Irradiated CANDU Fuel Elements

Silviu Ionescu, Octavian Uta, Cristian Gentea, Marin Mincu, Marcel Parvan, Laurentiu Dinu
Institute for Nuclear Research Pitesti, Romania
silviu.ionescu@nuclear.ro

The object of this work is the behaviour of CANDU fuel elements under power cycling conditions. The tests were run in the 14 MW (th) TRIGA-SSR (Steady State Reactor) reactor from Institute for Nuclear Research (INR) Pitesti.
Zircaloy-4 is the material used for CANDU fuel sheath. The importance of studying its behaviour results from the fact that the mechanical properties of the CANDU fuel sheath suffer modifications during normal and abnormal operation. In the nuclear reactor the fuel elements endure dimensional and structural changes as well as cladding oxidation, hydriding and corrosion. These changes can lead to defects and even to the loss of integrity of the cladding.
During the irradiation, the fuel elements suffer dimensional and structural changes, and also modifications of the cladding surface aspect, as result of corrosion and mechanical processes. This can lead to defects and even the integrity of the fuel element can be affected
This paper presents the results of examinations performed in the Post Irradiation Examination Laboratory (PIEL) from INR Pitesti, on samples from a fuel element irradiated in TRIGA-SSR reactor, namely: dimensional and macrostructure characterization; microstructure characterization by metallographic analyses; determination of mechanical properties; fracture surface analysis by Scanning Electron Microscopy (SEM).
A full set of non-destructive and destructive examinations concerning the integrity, dimensional changes, oxidation, hydriding and mechanical properties of the cladding was performed. The obtained results are typical for CANDU 6-type fuel.
The obtained data could be used to evaluate the security, reliability and nuclear fuel performance, and for CANDU fuel improvement.

I.2.9. Fuel Cladding Materials for Lead-Cooled Fast Reactors

Madalina Stanciulescu, Dumitru Ohai, Alice Dinu, Ramona Mihaiu
Institute for Nuclear Research Pitesti, Romania
mada.stanciulescu@nuclear.ro

The Lead-cooled Fast Reactor (LFR) is a technology with great potential to meet the Generation IV goals. A significant challenge facing this technology is the selection and development of structural and cladding materials suitable for long-term use in the highly corrosive liquid metal environment.
The main purpose of this paper is to present the most studied materials for fuel cladding and to make a comparison between them, so that we can conclude which is the preferred candidate for LFR systems. Cladding material must be compatible with metal or nitride fuel, corrosion resistant in coolant agents (Pb/LBE), and have adequate strength, ductility, toughness, and dimensional stability over the operating temperature range and to doses up to 200 dpa. The primary candidates for cladding materials in the lower temperature LFRs are the ferritic-martensitic steels (T9, EM12, HT9, HT91, HCM12A, T91, NF616, E911, NF12, SAVE12, etc.)
Qualification of any of these materials requires establishment of acceptable performances for usage under the extreme conditions of LFR. However, to achieve the high potential aimed for the advanced reactor system concepts, a significant amount of R&D is yet needed in the areas of materials and coolant chemistry control.

I.2.10. Neutron Activation Analysis Cross Sections Library Upgrading

Iuliana Visan, Csaba Roth
Institute for Nuclear Research Pitesti, Romania
iuliana.visan@nuclear.ro

In the present work, an attempt to improve the cross section library for neutron activation analysis (NAA) is made. In this context, a different procedure has been developed for cross sections arising from the Evaluated Nuclear Data File libraries (available as free on the internet), for which the unsolved resonance region is given in a very fine energy mesh (up to tens of thousands points).
The used method consists in two steps:
1) Firstly, the initial cross sections are collapsed assuming a 1/E neutron energy distribution for the resonance region. The collapsing is such made the resonance cross section results are given in SAND-II resonance energy grid. For this, it was developed an own source-code written in MICROSOFT VISUAL BASIC and FORTRAN programming languages;
2) Secondly, the obtained results are then processed with interpolation/ extrapolation cross sections CSTAPE program which generates the cross section library files written in SAND-II energy grid (10^{-10} – 18 MeV on 621 energy groups). Sensitivity studies regarding the influence of the deviation from the 1/E neutron energy distribution are also carried out.
Finally, for some representative reactions, the obtained results are given as tabulated and also in graphical form, in comparison with initial data collected from Evaluated Nuclear Data File libraries.

I.2.11. CFD Simulation of an Air-Based Heat Exchanger in a Heavy Liquid Metal Test Loop

Eduard Pauna, Alexandru Catana, Mircea Ioan
Institute for Nuclear Research Pitesti, Romania
eduard.pauna@nuclear.ro

In this paper, a Computation Fluid Dynamics (CFD) simulation was performed in order to find flow conditions and heat transfer balance for a Heat Exchanger (HX), an important component of a Heavy Liquid Metal (HLM) structural material test loop. Cooling fluid is air (forced/natural circulation) at atmospheric pressure and heat source is a vertical bundle of pipes. Hot HLM circulates inside the pipes at 400 ^{o}C.
A power ranging from 3,000 - 10,000 W has to be carried out from the pipes' walls.
The design concept, geometrical dimensions and shape, specifically for a Heavy Liquid Metal Test Loop, is presented. The operating conditions of fluid flow thermo-physical properties of melted lead, stainless steel and air as well as initial and boundary conditions are taking into account. The local flow conditions are achieved by using a CFD typical chain of steps which was performed starting from preprocessing (geometry, mesh and boundary conditions), through solver and post-processing. Open Source platform (Salome-Meca geometry and mesh modules, the Code_Saturne solver, Paraview and Visit for post-processing) was used as computational toolkit.
Three dimensional thermal-hydraulic distributions of the temperature, pressure and velocity parameters were obtained with high degree of accuracy and detail.
CFD analysis results provide valuable data regarding the thermal-hydraulic operating conditions of the Heat Exchanger. The results will be used for improving the HX design parameters.

I.2.12. MCNP Calculation of Burn-up Credit Criticality and Isotopic Composition Benchmark

Georgiana Budriman, Mirea Mladin, Ştefan Dulugeac
Institute for Nuclear Research Pitesti, Romania
georgiana.budriman@nuclear.ro

The purpose of this paper is to compare the actinides and fission products' concentrations obtained from burn-up calculations done with MCNPX, with calculated concentrations and with chemical assay data (used to obtain the isotopic concentrations, mg/g fuel, for ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{133}Cs, ^{135}Cs, ^{145}Nd, ^{149}Sm, ^{150}Sm, ^{152}Sm and ^{153}Eu) from an international benchmark.
It will be made a study about the MCNPX code sensitivity to observe how it depends of number of burn-up steps. Thus, initially, 8 burn-up steps will be used, and then 19 and finally 34 burn-up steps. The calculations will be made for a simple square PWR cell at a 27.35 GW/MtU burn-up. The fresh fuel rod contains UO_2 (3 at% ^{235}U enrichment).
The neutron multiplication factor and the concentrations will be compared with the average of the results obtained by 16 organisations from 11 countries. To compare with experimental data participants used different codes and methods based on transport and diffusion theory; they used also Monte Carlo codes.
Nuclear data had different sources: different versions of Evaluated Nuclear Data Files (ENDF/B), Japan Evaluated Nuclear Data Libraries (JENDL) and Joint Evaluated Files (JEF). Nuclear data used in this paper for calculations with MCNPX will be from ENDF/B VII.

I.2.13. CFD analysis for some components of an HLM material testing loop

Alexandru Catana, Eduard Pauna, Mircea Ioan
Institute for Nuclear Research Pitesti, Romania
alexandru.catana@nuclear.ro

A Heavy Liquid Metal (HLM) structural material test loop is proposed for design and construction at the Institute for Nuclear Research (INR) Pitesti, Romania. The main goal of the loop is structural material test behavior in HLM environment, corrosion phenomena and oxygen concentration control. Secondary, the thermal-hydraulic performances of some components are to be tested and then validated. Finally yet importantly, the loop will be a valuable tool for some numerical simulation methodology validation.
For design purpose, the detailed and accurate knowledge is necessary. HLM structural material test loop have some particular components: pumps, pipes, flow meters, T-shapes, elbows, testing vessel, adjustable valves, *heat exchanger*, pre-heater etc.
The paper objective is to obtain a detailed and accurate knowledge for flow conditions in pipes, T-shapes, elbows, and flow meters, particularly the pressure loss, by using advanced Computational Fluid Dynamics (CFD) tools.
Each component has a specific contribution to fluid flow condition and for heat transfer and balance. Among all parameters the pressure loss for each component, and finally for the whole loop, is of a particular importance for establishing a proper pump operating power. The main component of loop is structural material testing vessel in which the testing samples will reside. The desired testing conditions are obtained by ensuring proper mass flow, temperature and local velocities and therefore accurate flow meters are necessary to measure the mass flow and hence the velocity. Flow control is also necessary for a proper oxygen concentration control.
The primary result of this work is therefore the so-called calibration curve for different differential flow meters types and secondary the pressure drop for each of the analyzed component. The results of CFD analysis will be directly applied in the design process of all components and finally the whole test facility.

I.3. Nuclear Technologies and Materials

I.3.1. Investigation of Thermally Grown Oxides on FeCrAl Alloys Exposed to Heavy Liquid Metals

Adrian Jianu, Alfons Weisenburger, A. Heinzel, R. Fetzer, M. DelGiacco, W. An, G. Müller
Karlsruhe Institute of Technology, Institute of Pulsed Power and Microwave Technology, 76344 Eggenstein-Leopoldshafen, Germany
adrian.jianu@kit.edu

Heavy liquid metals (HLM), namely lead and lead-bismuth alloys, are considered as working fluids for advanced nuclear reactors due to their thermal and neutronic properties. Concerns related to the use of HLMs arise from the materials compatibility, in terms of corrosion and mechanical resistance.

The protection of steels, exposed to oxygen-containing HLMs, is guaranteed when alumina scales are able to grow on their surface. However, the development of a thin, continuous, dense, stable, adherent Al_2O_3 layer, which provides an effective diffusion barrier that protects an Al-containing steel from corrosion in HLM, requires critical Al levels (above ~7 wt%), generally leading to unacceptable mechanical properties. Therefore, the solution for this problem consists in alloying steel in the near surface with aluminum. Cr addition to a binary Fe–Al alloy is able to induce the formation of thin Al_2O_3 scales at significantly lower Al content, due to so called “third element” effect. However, the most suitable Al/Cr ratio to achieve the best corrosion resistance is still under investigation.

In the temperature range envisaged for HLM coolants (400-700℃) and depending on the Cr content and oxygen activity, the oxide layer may be composed of stable α-Al_2O_3 or of metastable polymorphs (γ, θ). The formation of metastable alumina phases could be detrimental for corrosion resistance in HLMs because: (i) the growth rate of metastable alumina phases is generally higher than that of α–Al_2O_3, with, as consequence, a huge Al consumption, detrimental for the material resistance after long time exposure, and (ii) the change in volume during the transformation of metastable forms into α–form (contraction of around 14%) can generate stresses in the oxide layer and can weaken its adherence to the underlying substrate, leading to spallation.

The Al/Cr ratio and the role played by the metastable alumina polymorphs on the corrosion resistance of materials exposed to HLMs have to be exactly defined.

Results of the oxide scales investigation, grown at 400-600℃ on FeCrAl alloys with low Al concentration, exposed to oxygen-containing liquid lead, will be presented. Model FeCrAl alloys with 4-8 wt% Al concentration and 6-16 wt% Cr concentration were prepared by vacuum arc melting. For characterization, grazing incidence synchrotron X-ray diffraction, scanning electron microscopy, energy dispersive X-ray spectroscopy and X-ray photoelectron spectroscopy were employed. Alumina stability domains at different temperatures, function of Al concentration and Al/Cr ratios, were defined. This work aim was to establish the best coating compositions for protection of structural steels exposed to liquid lead alloys.

I.3.2. Proposal of Laser Iion Beam Accelerator for Inertial Nuclear Fusion

***Florea Scarlat*[1,2], *Anca Scarisoreanu*[2]**
[1]*Valahia State University of Targoviste, Targoviste, Romania*
[2]*National Institute for Laser, Plasma and Radiation Physics - INFLPR,*
Bucharest – Magurele, Romania
scarlat.f@gmail.com

The inertial nuclear fusion with laser beams, relativistic electron beams, ion beams, micro-particle beams and superconducting projectiles has been and is still analytically investigated and numerically calculated by various authors along years and nowadays. The necessary parameters of such beams are the energy ranging between 10^5 and 10^6 J, power from 10^{14} to 10^{15} W and the spot size on the target from 0.1 cm^2 to 1 cm^2.
Starting from the record laser peak power of 1.25 PW and radiation peak intensity of 10^{20} W/cm^2 produced at LLNR using the chirped pulse amplification (CPA) laser technology as well as from ELI Nuclear Physics - laser system, 3 APPOLON 10 PW (150J/15fs) (http://www.extreme-light-infrastructure.eu/index.php) proposed to be realized, this paper presents the principle and the configuration of a compact ion accelerator with optical laser in an ultra-relativistic regime, for the inertial nuclear fusion.
The accelerator operation principle is based on the interaction of a laser beam with plasma. Plasma is an ideal medium for the acceleration of particles because it may stand longitudinal electric fields of high values (several GV/m), approximately four orders of magnitude greater than the ones obtained with RF cavity (10 MV/m). Plasma allows the conversion of the electromagnetic field of the laser radiation into plasma waves which can capture and accelerate the charged particles. Moreover, the main system parameters of the accelerator are also presented.

I.3.3. Study on Tribocorossion Mechanism of Two Model Alloys

***Cristian Ionescu*[1,2], *Gabriela Plaiasu*[1], *Marioara Abrudeanu*[1], *Pierre. Ponthiaux*[2], *Sanda Vulpe*[1], *Radu Vasile*[1], *Florina Constantin*[1]**
[1] *University of Pitesti, Materials Science and Engineering Laboratory, Romania*
[2] *Ecole Centrale Paris, Grande Voie des Vignes, 92295 Châtenay-Malabry, France*
abrudeanu@upit.ro , gabriela.plaiasu@upit.ro

Tribocorrosion is described as an irreversible process, which involves mechanical and chemical / electrochemical interactions between surfaces in relative motion, in the presence of a corrosive environment. The aim of this work was to quantitative and qualitative compare the total material loss of two Ni - Cr model alloys after tribocorrosion tests performed for different normal loads. Tribocorrosion of two Ni – Cr model alloys was investigated for different contact pressures in a solution of boric acid and lithium hydroxide at room temperature. The total material loss generated by the protective oxide layer deterioration due to combined chemical / electrochemical and mechanical interactions was assessed and compared from quantitative and qualitative point of view for these two Ni – Cr model alloys. The sliding track morphology was investigated by SEM and profilometric measurements after sliding tests. No zirconium particles coming from the pin were found. A local particles analysis by EDS revealed Cr, O and C contents higher than in the surrounding matrix. Abrasive particles could consist in Cr oxides or/and Cr carbides. Cr oxide particles could be identified by SEM – EDX analysis on the alloys samples surface after polishing. The influence of corrosion on the wear track morphology is not clearly

visible on the micrographs, even at the lowest normal force. Despite the very low value of the contact pressure at a normal force of 1 N, mechanical wear by abrasion has the largest contribution in total wear alloys surface before, during and after sliding tests. Before starting the sliding tests, a protective oxide layer formed at the alloy surface, after 3 h of immersion in the electrolyte. This passive film has better protective properties for the Ni – 30Cr model alloy than for the Ni – 15Cr alloy, due to higher Cr content. When the sliding started, a degradation of the passive film was observed. The passive film deterioration and the resulting electrochemical wear component were quantitatively evaluated. The total volumetric material loss in the sliding track was measured. For both Ni – Cr model alloys the total wear increases with the normal force. The corrosive (electrochemical) wear slightly increases as a result of the active area increase. The material loss due to active area corrosion in the sliding track is lower for the Ni - 30Cr model alloy, because of higher Cr content. The main difference between the two model alloys is given by this component of the total wear.

I.3.4. Thermophysical Properties of UZrH Fuel between 300 and 1400 K

Tiberiu Meleg, Mariea Deaconu, Iulia Dumitrescu
Institute for Nuclear Research Pitesti, Romania
tiberiu.meleg@nuclear.ro

The purpose of this work is to perform measurements of heat capacity, thermal expansion and thermal conductivity for UZrH fuel, having 45%U, at temperatures ranging between 300 and 1400 K. The methods used were, as follows: the differential scanning calorimetry for the specific heat capacity, dilatometric measurements for linear expansion, and differential thermal analysis for thermal diffusivity, respectively. For all measurements dynamic argon atmosphere were used at 1100 mbar pressure, in thermal transient conditions, at heating and cooling rates ranging between 0.0833 and 0.833 Ks^{-1}.
The measurements results on the cool-down regions show, in all cases, two distinct behaviours, at temperatures corresponding to the δ (350÷700K) and the β structures of the zirconium hydride (700÷1,355 K). The specific heat follows a linear law in the δ region, and an exponential one for the β structure. The values of the density changes obtained from the thermal expansion measurement, exhibit a linear decrease for both structures. Starting from the equations of heat capacity, density and those of the thermal diffusivity, the equation of thermal conductivity of UZrH was founded. The thermal conductivity, for a large range of temperatures (400÷1,100 K), was founded to be temperature independent.

I.3.5. Corrosion of the CANDU Steam Generator Tubing

Dumitra Lucan, Lucian Velciu, Maria Mihalache, Alice Dinu, Diana Astefanesei, Marian Pantiru
Institute for Nuclear Research Pitesti,Romania
dumitra.lucan@nuclear.ro

Particles enter a steam generator primarily as corrosion products generated within the steam cycle. Best choice always is to minimize the transport of corrosion products into the steam generator. This is the reason to control the corrosion product transport within the steam supply system. But over time, even with careful system cycle water chemistry control, corrosion products will tend to accumulate in the steam generator. They eventually deposit on surfaces,

especially on high-heat-flux surfaces and in low-velocity regions. There is also evidence that some deposition occurs in areas of flow restrictions. During operational thermal transients and hydraulic shock, deposits can break away from these surfaces and re-enter the fluid stream. During these "crud bursts," particulate concentrations in the steam generator can climb dramatically.
The steam generator tubes can fail by a large number of corrosion mechanisms due to the concentration of the impurities and deposits formation. In any corrosion failure, proper analytical practices and techniques will provide valuable information which, if properly understood, will lead to identification of the failure mode. Once the failure mode has been established, a review of the background history and service conditions will aid in the procurement of sensible corrective measures. Obtaining proper background information and performing the appropriate analyses, combined with an understanding of the material behaviour as well as pertinent experience, will produce the best chance of obtaining the accurate conclusions in any failure investigation.
This paper details the examination methods used to properly determining the root cause of a failure, and includes optical and scanning electron microscopy illustrations and RDX spectra of the most common corrosion mechanisms, including general corrosion, pitting, crevice corrosion, corrosion fatigue and intergranular corrosion.

I.3.6. Crystallographic Texture of Zircaloy-4 Claddings

Alice Dinu, Madalina Stanciulescu
Institute for Nuclear Research Pitesti, Romania
alice.dinu@nuclear.ro

The crystallographic texture of re-crystallised Zircaloy-4 was investigated using an X'Pert Pro MPD difractometer working in Bragg-Brentano geometry. The movement of the samples is achieved following a spiral with a step of 5^0 per rotation up to a tilt of 85^0 in Ψ. In the same time, the sample achieves a complete rotation in ϕ (360^0). It emphasised the characteristic texture to the recrystallised Zircaloy-4 claddings: the bimodal distribution with the basal poles oriented at 30^0 toward the radial direction, in R-T plane.
The pole figures obtained in measuring geometry used in this paper fairly match good with those obtained with the combined reflection/transmission X-ray diffraction. It was emphasised the bimodal distribution of the basal poles in the R-T plan, with the maximum intensity at 30^0 towards the radial direction.
The prismatic poles (101.0) are oriented in A-T plan at 65^0 towards the radial direction; while the prismatic poles (112.0) are oriented in the A-R plan at 75^0 towards radial direction.
The pyramidal poles (101.1) are oriented in R-T plane at 30^0 towards radial direction and the prismatic poles (112.2) are oriented in A-R planes at 50^0 towards radial direction.
The pole figures obtained in measuring geometry used in this paper fairly match good with those obtained with the combined reflection/transmission X-ray diffraction.

I.3.7. Determination of Gas Flow Equilibrium Parameters due to the Mix of Two Gases with Different Thermo-Dynamical Characteristics

Sorin Gherghinescu, Ionuț Spiridon, Gheorghe Popescu
Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
sorin@icsi.ro

In this study, the thermo-dynamic characteristics of gases leak rate through a system have been studied. Different pressures versus time scenarios have been applied to a model system. Nitrogen and industrial air were used to perform the pressurized leak rate test. The thermal conductance influence was measured from both pressure-decay experiments and from constant pressure-flow experiments using gases at variable temperatures due the length of the gas supply piping. Results from the two type gas tests were compared and communicated in this paper. In the evaluation of real components, subassemblies, or systems, conductance and also the thermo-dynamical flow of the gas trough piping system of can be estimated from a simple pressure-decay measurement. Obtained values can be used to estimate the effect of thermal conductance on the leak rate, to minimize the construction complexity of the piping and also to improve the insulation factor of the refrigeration system. In cryogenic systems is important to determine the leakage of gas for the cooling power regulation and automation system. In the cryogenic distillation process gas purity and thermal regime are very important design features. This study establishes the parameters of the thermal control system and gas purity. By plotting discharge curves can be established the influence of change of parameters in the system and can locate the leakage occurred by measuring the corresponding flow. Discharge curves were plotted on an area covering the normal running of the plant. The experimental results obtained demonstrate the importance of continuous monitoring of flow in nuclear installations on the one hand due to ensuring the safety in operation and on the other hand to control the process parameters.

I.3.8. Gas Leak Localization by Calculation of the Local Leak Rate for an Interconnected System in a Vacuumed Cold-box

Ionut Spiridon, Sorin Gherghinescu, Gheorghe Popescu
Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
ispiridon@icsi.ro

Byproduct of electricity production by nuclear power plants, Tritium, is considered to be the main fuel source for fusion reactors. The tritiated water obtained in the fission reactors is transformed with the help of TRF's in pure tritium gas. This is further used as primary source in the fuel cycle for the fusion reactors. The technical assembly of the TRF is regarded as a complex system of specialized equipment consisting in sensors, detectors, valves and piping arrangements. Working pressure and temperatures of the TRF system define the types of sensors, the vacuum technology and especially the possible operational safety issues. Leaks in the systems represent one of the main safety issues in a TRF. Environmental and work safety regulations state that if the leak from a system, in a closed or open space, exceeds some threshold value it must be reported. Tritium has a special regulation because of its radioactive impact.
This paper describes a calculation procedure intending to demonstrate that the leaks in the system could be localized and estimated. Using a least squares technique in a spreadsheet, parameters of a nonlinear equation can be determined for a localized leak in an enclosure, like a cold-box, for example. The calculations were based on a leak test

using nitrogen and also industrial air. The pressure loss in the system was modelled by evaluating the leak through an orifice with specified open surface. This coefficient of opened surface was then used to calculate the loss of tritium in the system. The first step in the test was to fill the system. The pressure in the system was recorded as a function of time. This loss in pressure was described as proportional to the orifice coefficient.

I.3.9. A Reliable Measurement Technique of Heavy Water as Nuclear Material

Iuliana Piciorea, Gabriela Raducan, Ioan Stefanescu, Tatiana Oprea
Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
iulianapiciorea@yahoo.com

The aim of this study is to analyse the reliability of the Fourier Transform InfraRed spectroscopy method used in measurement of deuterium concentration from heavy water.
Heavy water is successfully used as moderator and coolant in nuclear reactors CANDU type. Therefore there has been a strong demand for a reliable detection system to measure the deuterium concentration in heavy water. An InfraRed spectrometer was used to build a calibration curve, using 9 secondary heavy water standards.
To estimate the uncertainty of the measurement method we had to take into account the possible sources of uncertainty which could be measured. The estimated uncertainty value is low compared with the uncertainty associated to the vibrational densimetry method, so this method is very reliable and it can be used in the range of deuterium concentrations higher than 99% D_2O mass.
FT-IR spectroscopy is a non-invasive and non-destructive technique without using any chemical reagents. Also, the calibration curve is very stable and it can be used to quantify the sample spectra as long as a method to verify it is established.
This analytical technique is suitable to measure the deuterium concentration of heavy water which is used in nuclear reactors and needs a very good accuracy measurement.

I.3.10. Behavior and Characteristics of PT/C/PTFE Catalyst used in H_2-H_2O Isotopic Exchange Process

Gheorghe Ionita, Ionut Spiridon, Anisia Bornea, Ioan Iordache
Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
ionita@icsi.ro

In order to check and to prove the technology for tritium removal from heavy water produce by the Cernavoda NPP, an experimental pilot plant for deuterium and tritium separation (ExpTRF), based on cryogenic distillation coupled with catalytic exchange between hydrogen and liquid water, has been built at ICIT Rm-Valcea and tested within an comprehensive program for many years. One of the key points in the functionality of the mentioned technology it's represented by the efficiency of tritium transfer process from tritiated water mixture to simple hydrogen/deuterium water and tritium gas. This process needs a special catalyst with high activity and hydrophobicity. Based on its hydrophobicity, the catalyst structure repels the liquid water and allows the access of water vapour and hydrogen to the active catalytic centres, proving high catalytic activity and performances stability on time. To increase the efficiency of isotopic exchange process, the catalyst is used in so called "catalytic mixed packing" consisting of in alternating layers of catalyst and

hydrophilic materials. On hydrophilic materials an efficient isotopic transfer between water vapour and liquid water take place, similar to the water distillation process.
At the present time, the ExpTRF is shut down for the refrigeration power upgrade of the cryogenic distillation unit. The isotopic separation module is also upgraded and the old catalytic mixed packing is replaced with a more compact new packing.
The paper presents the behaviour and main characteristics of platinum hydrophobic catalyst (Pt/C/PTFE) used in the old liquid phase catalytic exchange column (LPCE) of the ExpTRF for several years of active operational condition. Surface area, pore volume, pore distribution as well as platinum area and platinum particle size for used (Pt/C/PTFE) catalyst have been measured and are reported in comparison with initial values (before using). The influence of mechanical impurities from reactants and from hydrophobic materials on the catalyst are presented and discussed. No significant modifications of activity and main characteristics have been observed. Future investigations on hydrophilic packing will be undertaken, too, to complete the final conclusions.

I.3.11. Corrosion Tests of In600 at the Specific Parameters of the Steam Generator Primary Circuit

Diana Astefanesei, Dumitra Lucan, Mihaela Silvia Stoica
Institute for Nuclear Research Pitesti, Romania
diana.astefanesei@nuclear.ro

For long period of steam generator (SG) operating, the heavy water conduces to materials degradation. Accordingly to this, the study of steam generator materials at the primary circuit specific parameters (t=310℃, p=110 atm, pH=10.5, O_2<20ppb) has a large requirement.
The goal of this paper is to obtain data about the In600 corrosion process in primary circuit conditions. With this purpose the samples were tested in CANDU SG primary circuit conditions and dynamic regime using the recirculating loop. After autoclaving, the samples were analyzed by electrochemical measurements, optical microscopy and X-ray diffraction. The results of testing program consist in the assessment of the composition, the structure of the oxide layers and the behaviour to corrosion (corrosion rate), in relationship with the testing period. The results will be presented as plots, tables, micrographies and X-ray emission spectra. The recirculating loop offers the possibility of testing the structural materials behavior in the conditions of primary or secondary circuit from a Nuclear Power Station at uniform and localized corrosion, reproducing the technical specifications of both circuits (temperature, pressure, pH, content of O_2 and H_2, if we made the choice). Our experimental results can be used for a future data base referring to corrosion processes of the main SG components.

I.3.12. Microstructural Characterization of Irradiated CANDU Pressure Tube Samples

Raluca Florentina Negrea, Octavian Uta, Marina Tatarascu, Diana Ilinca, Gabriela Gheorghe, Ovidiu Ichim
Institute for Nuclear Research Pitesti, Romania
raluca.negrea@nuclear.ro

Post-Irradiation Examination Laboratory (PIEL) of Institute for Nuclear Research Pitesti (INR) is equipped with means of handling, testing and analysis of irradiated nuclear fuel and structural materials.
The aim of this paper is to highlight flaking, roughness, surface irregularities and cracks in specimens of Zr-2.5%Nb taken from CANDU pressure tube, irradiated in ″capsule C5″ irradiation device and tensile tested in PIEL.
CANDU reactor pressure tube, made of Zr-2.5%Nb, is one of the most critical components due to the severe operating conditions within the reactor.
The PIEL of INR Pitesti equipments allow to perform metallographic and ceramographic analysis using the metallographic microscope LEICA TELATOM 4.
Metallographic examination consists in the ability to highlight the microscopic structure of the material acquired in the reactor during operation (structure, grain size, thickness of oxide layer, hydriding).
Metallographic analysis were performed on sample A tested at 70°C and on the sample B tested at 250°C; A and B samples were selected from the samples which have previously been tensile tested.
The fracture surface was microscopic analyzed along the dividing line, together with the appearance of hydrides in the material.
The results revealed by the post-irradiation examinations are used to obtain a database necessary for evaluating the behavior under irradiation of the materials made of Zr-2.5%Nb.

I.3.13. Metallographic Examination of Two Nickel-Based Alloys

Lucian Velciu, Diana Astefanesei, Laurețiu Popa, Alexandru Nitu
Institute for Nuclear Research Pitesti, Romania
lucian.velciu@nuclear.ro

In this paper we present the metallographic examination by optic microscopy of some samples debited of two nickel-based alloys (as received), so-called Inconel 617 (UNS N06617) and Haynes 230 (UNS N06230). Samples were prepared by metallographic methods (cutting, mounting, grinding, polishing, etching) and then examined for inclusions (as polished), microstructure (etched), micro-hardness and chemical composition using following devices: optical microscope Olympus GX 71, micro hardness (Vickers) by the automatic cycle OPL microdurometer and spectrometer type ARL ADVANTA'X IntelliPower™ Series.
Metallographic analysis of samples consisted in: Visual aspect and surface morphology, microstructure (average grain size), non-metallic inclusions and micro hardness determinations.
Average grain size were determined by linear interception method (Heyn-ASTM E 112) and reported as number G (ASTM Grain Size No).
Materials micro hardness (Vickers) was calculate with relationship of technical book of device (OPL-France).

Samples description, preparation and notation: the samples were mechanical cutting from #2 mm sheet “as received” (annealing) with the dimensions of plates 30x15x2 mm, then prepared by metallographic and notated thus In 617- for Inconel 617 alloy and A 230 - for Haynes 230 alloy; ST –cross section, SL-longitudinal section. Chemical composition is in accordance with ASME SB-168 and ASME SB 435 standards. Average grain size (G) is in accordance with ASTM SB-168/435 standards.
The results can serve to create a metallographic database with different typo-sizes candidate materials for the heat exchangers of the future nuclear reactors.

I.3.14. Crack Propagation Measurements on C-Shaped Specimens from CANDU Pressure Tube

Gabriela Gheorghe, Ion Man, Marin Mincu, Raluca Negrea
Institute for Nuclear Research Pitesti, Romania
gabriela.gheorghe@nuclear.ro

The pressure tubes are considered the most critical components of a PHWR reactor, being subjected to severe operating conditions. The absorption of hydrogen/ deuterium from the cooling agent or from the intertubular gas (present in the space between the pressure tube and the Calandria tube) increases the susceptibility for the initiation of the slow crack mechanism in hydride state.
To put into evidence Zr-2.5 % Nb susceptibility to the Delayed Hydride Cracking (DHC) phenomena, fracture mechanical tests are carried out. The so-called "C-shaped" samples are the most fitted ones with respect to the purpose of this work. In order to simulate the real absorption of hydrogen on the pressure tube during the irradiation, non-irradiated, manufactured C-shaped samples were hydrided.
A C-shaped sample hydrided in this way must have a notch, which will serve for the initiation of the DHC along the radial direction. A DM 1007 Dyna numerically controlled milling machine was used both for obtaining the C-shaped sample itself and for making the notch. The notch (a “V” profile) was realized using a plate of tungsten carbide with a special profile, with a tip radius of 0.015 mm and the angle between the flanks of 45°. The parameters characterizing the notch (the depth of the “V” profile, the angle between the flanks, the tip radius, and the diameter of the fixing holes) were measured using TESCAN MIRA II LMU CS equipment. This MIRA Scanning electron microscope (SEM) is a high-quality, fully PC-controlled equipment, equipped with a Shottky Field Emission electron gun designed for operation under high vacuum or variable pressure conditions. Such SEM equipment is very useful for the investigation of fracture surfaces and microstructure of irradiated or non-irradiated nuclear materials.
The C-shaped samples being the subject of this work can be further used for the measurement of mechanical parameters describing the initiation and propagation of delayed hydride cracking phenomenon in the radial direction of the CANDU pressure tubes.

I.3.15. The Evaluation of Organic Inhibitors for the Protection of Grade 2 Titanium Alloys in Chloride Media

Laurentiu Popa, Mariana Tunaru, Lucian Velciu
Institute for Nuclear Research Pitesti, ROMANIA
laurentiu.popa@nuclear.ro

Knowing the remarkable corrosion resistance and low cost of titanium alloys in comparison unalloyed steels and high nickel alloys, numerous components used in equipments from nuclear industry should be manufactured from titanium alloys. Now the titanium can also be considered a proven background material for chemical, petrochemical, power and marine industry.
Titanium alloys are often used in the thermal transfer technology especially in the heat exchangers cooled with sea water. However, it is not indicated its use in strong oxidant and any concentrated chloride environments. If the inorganic oxidant species are not present in a corrosive medium, it is necessary to add very small organic inhibitors quantities adequately to respective environment, with the aim to protect the titanium components surfaces.
The corrosion failures of titanium alloys are inevitably associated with the non-observance of environmental limits prescribed in corrosion handbooks, e.g., its use in solutions containing chloride salts.
The evaluation on grade 2 titanium alloys samples in solutions containing Cl^- (NaCl) in the presence of some organic inhibitors and in their absence, respectively, by using the potentiodynamic polarization method, was performed. On the basis of these experimental results, the corrosion rates were determined with the aim to establish their protection efficiencies.

I.3.16. The Use of MD-40 Biocide in the Inhibition of the Microbiologically Induced Corrosion in Raw Water

Mariana Tunaru, Lucian Velciu, Laurentiu Popa
Institute for Nuclear Research Pitesti, Romania
mariana.tunaru@nuclear.ro

The basic objectives of a successful cooling water treatment program are increased process efficiency, increased production output and maximized equipment service life. For a cooling water treatment program to achieve maximum effectiveness, it must address the problem areas of corrosion, deposition and microbiological contamination. Open recirculating cooling water systems are continuously infected with microorganisms. These microorganisms multiply rapidly in warm cooling waters and can lead to fouled heat exchangers, system metal loss due to microbiologically induced corrosion, degradation of cooling tower wood, and clogged screens and filters. Reduced flow rates, a reduction in heat exchanger capacity and the accumulation of microbiological growths on tower decks and tower fill are all signals of a severe microbiological problem.
This paper presents the results obtained regarding the testing of the MD-40 biocide efficiency in the annihilation or the diminution of the microbiologically induced corrosion of carbon steel -R525 B. The tests were performed both in untreated water and in treated water with biocide (25 ppm, 50 ppm, 100 ppm, 150 ppm).
For the samples behaviour evaluation in the two considered environments, we used potentiodynamic polarization method. The information obtained in a polarization

experiment is the current density as a function of the potential and by Tafel extrapolation corrosion rate is determined. The biocide treatment contributed to the corrosion rate decrease from 0.2189 mm/year (in raw water) to 0.0277mm/year (in water with 50 ppm biocide).
The samples surfaces were studied by metallographic technique, emphasizing less dense and the shallow pits for those tested in water with biocide.

I.3.17. The Development of Practical Techniques to Quantify the Ageing Degree for the Power Cables; Breaking Elongation Tests

Dumitru Puiu,Tiberiu Gyongyosi, Florin Voicu, Elena Dinu
Institute for Nuclear Research Pitesti, Romania
dumitru.puiu@nuclear.ro

For the ageing management of the power cables are required tests and analysis where through are developing the practical techniques to quantify the ageing degree characterizing the insulation materials.
The studies, tests and analysis performed on insulation test specimens from PVC power cable samples accelerated thermal aged having different ageing degree proved that the mechanical properties of electric insulation material tensile tested provide a warning of the imminent functional degradation. Thus, for a real ageing time of 52.1 years resulted in a diminution of 50 % initial elongation.
The breaking elongation test is a destructive one. To apply the results practically are required the non-destructive monitoring techniques of the PVC power cables jacket condition from a nuclear power plant and of those tested for breaking elongation and correlation of the results. The non-destructive monitoring techniques developed for the power cables condition consist of the compression modulus and physical properties determination for the cable jacket material by thermo-gravimetric measurements.
The obtained results are useful for modelling, detection and control of ageing phenomenon of the PVC power cable materials from nuclear power plant. Must be noted that have being an issue to provide the non-aged samples of these cables made many years ago.

I.3.18. Microstructure and Properties Induced by Surface Engineering Treatments Including Plasma Electrolytic Nitriding/ Carburizing/ Boriding of Iron Based Structural Materials

Elisabeta Coaca[1], Gheorghe Oncioiu[1], Silviu Florea[1], C. Lungu[2], Victor Andrei[1]
[1]Institute for Nuclear Research Pitesti, Romania
[2]National Institute of Laser, Plasma and Radiation Physics, Bucharest, Romania
eliabetaandrei@yahoo.com

The paper objective is the improvement of iron based structural nuclear materials properties by Plasma Electrolysis techniques.
Plasma Electrolysis techniques, including Plasma Electrolytic Saturation processes (PES) as Plasma Electrolytic Nitriding / Carburizing / Boriding (PEN/ PEC/ PEB) have a significant potential as methods for modifying the surfaces of engineering components in order to improve their in-service performance, useful working lifetimes, aesthetic appearance or economics of production.

The surface of austenitic stainless steels 304L and 316L was modified by complex surface treatments which include:
- plasma electrolytic carbo-nitriding, boro-nitriding, boro-carburizing and boro carbo-nitriding by means of Plasma electrolytic saturation (PES); the saturation of anodic surfaces with C, N, B (and combination of these) was performed using suitable electrolytes (aqueous solutions of inorganic acids, appropriate salts containing the desired elements and certain organic compounds);
- C and B deposition in Thermionic Vacuum Arc Plasma.

The coatings obtained in various experimental conditions have been investigated by means of electron spectroscopy, electrochemical techniques; the properties of the thin films have been correlated with the microstructure and the composition of the surface layers which are strongly dependents of the different regimes of diffusion treatments.
The preliminary results on duplex treatments which include PEN/PEC/PEB and TVA processes demonstrate that we can select the processing parameters for essential improvement of corrosion behaviour in some aggressive medium and high values of microhardness.

I.3.19. Study of the Ageing Behavior of Incoloy 800 HT and 304-L Steel Using Neutron Scattering Techniques

Ion Ionita[1], Pavol Mikula[2], Gyula Torok[3], Premysl Beran[2], Manuela Fulger[1], Bogdan Venescu[1]
[1] Institute for Nuclear Research Pitesti, Romania
[2] Neutron Physics Institute Rez, Czech Republic
[3] Institute for Solid, State Physics and Optics, Budapest, Hungary
ionionita@lycos.com

The Incolloy 800 HT is widely used in various types of heat treating furnaces, in power generation for steam generators tubing and high temperature heat exchangers in gas cooled nuclear reactors. Recently, this alloy starts to be studied as candidate material for fuel cladding in GEN IV reactors like supercritical water reactors (SCWR).
The 304-L alloy is an austenitic stainless steel with high ductility, low yield stress and relatively high ultimate tensile strength. This material has good corrosion resistance and is widely used in nuclear power plants.
Study of the ageing behaviour for the Incoloy 800HT and 304-L steel standing a heat treatment at high temperatures using both SANS (Small Angle Neutron Scattering) and neutron diffraction measurements was performed.
At nanoscale level in the case of Incoloy 800 HT, a phase transition is observed above 500 degrees while in the case of the 304-L steel such a transition is observed above 450 degrees. SANS measurements were performed at KFKI Budapest.
Diffraction measurements performed at NPI Rez did not show any phase transition with thermal treatment; only strain appearance was put in evidence. The dependence of temperature treatment strain is different for the two kinds of samples.
Both SANS and diffraction measurements are planned to be performed at LNF Dubna, in May, 2012, under a “round robin” exercise. Diffraction measurements could also be performed in INR Pitesti TRIGA Research Reactor, when it will restart to operate.

I.3.20. XPS Characterization of the Surface Structures Developed on the Surface-Engineered Materials for Advanced Nuclear Systems

Mariana Minca[1], Gheorghe Oncioiu[1], Cristian Lungu[2], Silviu Florea[1], Oana Rusu[1], Elisabeta Coaca[1], Victor Andrei[1]
[1]Institute for Nuclear Research, Pitesti, Romania
[2] National Institute of Laser, Plasma and Radiation Physics, Bucharest, Romania
mariana.minca@nuclear.ro

The nanostructured coatings offer a great potential for various applications due to their superior characteristics: high values of micro-hardness, better tribological properties, and great corrosion resistance in aggressive media. The results of the characterization of surface structures developed on structural nuclear materials by surface engineering techniques for improvement of the properties (corrosion resistance, hardness, and wear) are reported as follows:

- surface structures obtained by complex surface treatments which include plasma electrolytic carbo-nitriding, boro-nitriding, boro-carburizing and boro- carbo-nitriding by means of Plasma electrolytic saturation (PES); C and B deposition in Thermionic Vacuum Arc Plasma
- Be-C-W/CFC systems obtained by TVA, of interest to ITER Project.

X-ray photoelectron spectroscopy (XPS) was applied to analyse surface composition, chemical states and depth distribution of elements in the surface layers.
To establish chemical states of identified elements, a study of the chemical shift effects in the obtained spectra was carried out.
To analyze to greater depths than is practical using sputtering, ex-situ mechanical processes for removing material were used, (Angle lapping and Ball cratering techniques).
The preliminary results regarding duplex treatments, which include plasma electrolytic carbo-nitriding of austenitic stainless steel substrate and carbon deposition in Thermionic Vacuum Arc Plasma, demonstrate that we can select the process parameters for the essential improvement of corrosion behavior with some aggressive medium and high values of micro hardness.
In the case of selected surface treatment parameters, the XPS analysis of the interface carbon layer/ substrate showed a 50% higher sp^3 carbon bond in the carbonic deposited layer.

I.3.21. Inhibition of Mild Steel Corrosion in Simulated Pore Solution by Some Organic Inhibitors

Doina Stefanescu, Lucian Velciu, Marin Mincu
Institute for Nuclear Research Pitesti, Romania
doina.stefanescu@nuclear.ro

Reinforced concrete structures are largely used in the nuclear industry, including Nuclear Power Plants.
Steel embedded in concrete is normally in a passive state against corrosion due to a thin iron oxide layer that forms on the steel surface in the high alkaline environment of the concrete. However corrosion of reinforcing steel is the major cause of concrete structure deterioration and is mainly attributed to the presence of chloride ions near the concrete–steel interfaces.

One of the common corrosion prevention practices is to incorporate inhibitors in potentially corrosive environments.
The objective of this research was to evaluate the performances of two types of corrosion inhibitors in a concrete environment: ethanolamine and benzotriazole.
The inhibitors effectiveness was evaluated using metallographic, microscopy and SEM analyses. Electrochemical techniques i.e. potentiodinamyic polarization and electrochemical impedance (EIS) were also used to assess the inhibitors effect on the corrosion of steel reinforcements in concrete.
The data obtained from EIS tests were used to calculate the protection efficiency (*PE%)*.
Ethanolamine and benzotriazole provided a good protection for the steel in Simulated Concrete Pore (SCP) solutions. However, etahanolamine confer a better protection. The study indicates the applicability of these compounds for steel corrosion protection in reinforced concrete structures.

I.3.22. Electro-deposition as a Surface Engineering Solution for the Future Gen IV Reactors

Oana Alice Rusu, Mariana Mincă, Elisabeta Coacă, Gheorghe Oncioiu, Silviu Florea, Constantin Diaconu, Victor Andrei
Institute for Nuclear Research Piteşti, Romania
oana.rusu@nuclear.ro

The metal electrochemically synthesized coatings possess many advantages including high thermal and electrical conductivity, excellent mechanical property and high resistance when exposed to aggressive environments. For this reason, the electro-deposition is considered as an engineering solution to the problems that can be envisaged due to the extreme conditions to which different materials of nuclear plants are subjected.
Finding environmentally friendly alternatives to established electro-deposition processes, with the aim to design structural materials with better properties adequate for use in nuclear reactor environments, is still an important general challenge. Chloroaluminate based ionic liquids (ILs) have been proven to be the most promising media for industrial scale deposition suitable for anticorrosion purposes. ILs are a relatively new class of compounds characterized by high conductivity, extremely low vapour pressures, low viscosity, low toxicity, non-flammability, high thermal stability, wide electrochemical window and being liquid in a wide range of temperatures. Understanding the formation mechanism of the metal layer and the effects of the aggressive nuclear environments on the electro-deposited coating is a necessary pre-requisite to continue the research in the field.
The aim of this work is to review the methods used lately for electro-deposition of different metals (particularly aluminium) from ILs, on different steel substrates, in particular structural materials, the mechanisms find for the coating formation, the geometries and the chemical composition of the substrates obtained, the corrosion resistance of the coatings, the effects of different parameters like current density and plating time on the microstructure and thickness of the metal-deposited coatings. The information provided by this work is a valuable start point for continuation of the research in the field.
Our group is at the beginning of a research project concerning the development of new methods to improve the behaviour of structural nuclear materials in the operating environment by electrochemical surface engineering. For metal electro-depositions and corrosion tests we use a Corrosion Cell Kit model K0047 and a Flat Cell model K0235 connected to a Potentiostat model PARSTAT 2273-SYS. The chemical states of the films

can be identified by X-ray photoelectron spectroscopy (ESCALAB 250). Few preliminary experiments have been already performed.

I.3.23. Improving the Monitoring System for F/M Test Rig Heater

***Constantin Iorga*[1]*, Şerban Constantin Valeca*[2]**
[1] *Institute for Nuclear Research Pitesti, Romania,*
[2] *University from Pitesti, Pitesti, Romania,*
constantin.iorga@nuclear.ro

Out of pile tests are used to collect qualitative and quantitative data required for the characterization of the equipment and nuclear facilities behaviour during operation.
The fuelling machine testing rig available at Institute for Nuclear Research (INR) Pitesti is a complex installation that ensures the working conditions necessary for performing the functional tests and for demonstrating the machine's performance to operate in CANDU type reactors.
The thermo-mechanic heated loop (THL) of the testing rig ensures the thermo-hydraulic parameters in terms of pressure, heat and flow, similar to those in the reactor fuel channels.
In the testing stage of the fuelling machine (F/M), the THL's electric heater (P=2.1 MW) operates under conditions specific to a vessel with a pressure of 110 bar and ensures a working temperature of 307^{o}C. Taking into consideration the working conditions of high pressure and temperature and the modular mechanical structure of the heater, it is necessary to guarantee a high level of safety and protection during its exploitation.
This paper presents the activities required to elaborate a technical solution and for performing on-line monitoring and signalling of the electric heater (EH) during operation. The purpose of this system is to simultaneously control the working agent for each of the 12 EH modules, without influencing the operation logic of the recording and protection equipment available.
The system will allow the exploitation process optimization ensuring the increase in the protection level of the nuclear facilities and that of the safety level for the exploitation personnel.

I.3.24. Plugging of Demineralized Water Flow Into Horizontal High Diameter Pipeline (Dn 200 mm) with Artificial Ice Plug

***Tiberiu Gyongyosi*[1]*, Serban Constantin Valeca*[2]*, Valeriu Nicolae Panaitescu*[3]*, Ilie Prisecaru*[3]**
[1]*Institute for Nuclear Research Pitesti, Romania*
[2] *University of Pitesti, Romania*
[3] *University Politehnica of Bucharest, Romania*
tiberiu.gyongyosi@nuclear.ro

The local isolation of a pipeline section, placed horizontally in an installation, can be made with an ice plug resulted after a controlled process inside of pipeline. The technique is applied in order to execute repairs or replacements of components, without stopping and drain off the working agent from installation. This isolation represents a technological process developed continuous especially by the area-line firms, according to each application's requirements.

On principle, the technique is simple and assumes applying of a special built sleeve for each case. Assembled outside on the tube in the area chosen for flow plugging, it form an annular space with tube surface or in its right closeness. This space is allocated, in our case, to vaporisation of the liquid nitrogen. The heat and mass transfer, executing continuous between the tube wall and refrigerant ring shaped, encouraging the ice deposit in subsequent layers. For the first phase each technical application has to be treated experimentally, both regarding the ice plugging device and liquid nitrogen gauging during entire ice plugging process and stable keeping in the tube. The time process for stopping with ice plug of the liquid flow through the tube and the liquid nitrogen consumption depends on the geometrical configuration of the device and by the cooling room sizes. Choosing the overall sizes of the device and respectively, of the liquid nitrogen room sizes, are direct related by the nominal tube diameter value and by the liquid capacity affected.
The paper contains a constructive description of the experimental technological facilities insisting on the new design variant of the ice plugging device used for plugging of the demineralised water flow on the test section (Dn 200 mm). The test, the first results get and some conclusions are following.
The paper is dedicated to the specialists working in the research and technological engineering.

I.3.25. The Experimental Installation of the First Plug (M1- M.E. Version) using an Unconventional Plastic Deformation Technique

Tiberiu Gyongyosi[1], Ilie Prisecaru[2], Valeriu Nicolae Panaitescu[2]
[1] Institute for Nuclear Research Pitesti, Romania
[2] University Politehnica of Bucharest, Romania
tiberiu.gyongyosi@nuclear.ro

Evaluation of the steam generator tube wall thickness, performed with eddy- current technique, provides data regarding time evolution of the flaws depth and detection of new surface flaws, also. Each steam generator tube has to comply certainty with preset performance criteria during the operation, which leads to the decision to follow-on the operation, for maintenance or replacement. In other words, determination of the flaw evolution rate, knowing the operation condition (the parameters and the working fluid chemistry) gives the possibility for residual lifetime assessment of each inspected tube.
The steam generator has to keep the structural integrity during entire operation lifetime in normal and design basis event conditions. The tubes shall be inspected according to the standard requirements. It is thus performing an assessment of the structural integrity and of the leaks value, the results being compared with the specified performance criteria. The leak rate hasn't to lead to radiological overdose, the consequence required by the control body. The steam generator tube defective detected has to be plugged at both ends following its mechanical separation from the primary heat transport system. The steam generator defective tube isolation assumes plugs assembling which geometrical configuration owning to installing position at the tube end and to the plastic deformation technique applied. The used high-speed plastic deformation techniques develop locally neither the additional mechanical stresses nor structural changes in the deformed material.
The paper presents, after a brief introduction, some considerations regarding the techniques used for plugging of the steam generator defective tubes followed by a description of the plastic deformation technique by electro-hydraulic shock developed, a results analysis and conclusions.

The paper is dedicated to the specialists working in the research and technological engineering.

I.3.26. Investigation of Nuclear Structural Materials Mechanical Properties

Alexandru Nitu, Vasile Pitigoi
Institute for Nuclear Reserch Pitesti, Romania
alexandru.nitu@nuclear.ro

The emphasis on neutron economy in their design, allows CANDU reactors to utilize uranium resources more efficiently than any other commercial reactor. The use of zirconium alloys is an essential feature of the neutron-economic design, the CANDU fuel cladding being represented by thin-walled Zircaloy-4 fuel sheet.
Zirconium alloys are used in nuclear technology due to a very low absorption cross-section for thermal neutrons and good resistance against water corrosion. Therefore it is frequently used as cladding for nuclear fuel rods; in CANDU fuel rods case the Zircaloy-4 alloy has been chosen.
This paper aims to develop an analysis of the Zircaloy-4 cladding tubes mechanical properties by using tensile tests.
Initially, samples were mechanically cut by turning the Zircaloy-4 tubes SANDVIK in order to obtain rings of 11 mm height. After that, the rings were cut in a lengthwise direction and milled.
The above mentioned tests have been performed at both room temperature and at high temperatures. After testing, the fracture was highlighted by using optical microscopy, thus making a correlation between the obtained mechanical properties and the material structure observed by optical microscopy. For present investigations the Zeiss Observer A1m microscope was used.
The results will be used for future investigation of Zircaloy-4 cladding tubes mechanical properties and also to improve the existing database.

II. ENVIRONMENTAL PROTECTION

II.1. Radioactive Waste Management

II.1.1. Current Status of the Development of a New LILW Repository in Romania

Antonius Gheorghe-Sorescu, Ortenzia Niculae, Dan Corneliu Slavoaca
Nuclear Agency & Radioactive Waste, Bucharest, Romania
antonius.sorescu@andrad.ro

For waste generated by the operation and decommissioning of Cernavoda NPP (with 2 CANDU-6 type reactors in operation and another 2 proposed in the future), the revised National Radioactive Waste Management Strategy will include a new near surface type repository to be commissioned in 2019. The new repository will accommodate the Low and Intermediate Level Waste with short lived radionuclides and limited quantities of long lived radionuclides as accepted according to the results of the safety assessment for long term behaviour of waste and repository system.
Nuclear Agency for Radioactive Waste (AN&DR), the owner of the new near surface repository project, followed a comprehensive programme for site characterization and to develop the support technical and safety documentation necessary to obtain the siting license of the repository.
The paper presents the current status, issues and findings of the repository project, at the siting stage of the repository, which is a very important stage in the project development and the preparation of the new investment.

II.1.2. Recommendations for the Radioactive Waste Treatment and Conditioning Station at Cernavoda NPP

Stela Diaconu, Cristian Litescu
Nuclear Agency & Radioactive Waste, Romania
stela.diaconu@andrad.ro

Steps have been made towards the development of a new treatment and conditioning station for processing the radioactive waste generated by the operation, refurbishment and decommissioning of the Cernavoda NPP reactors, within an international project leaded by NUVIA Ltd.
The project reviewed, assessed and then identified international waste treatment technologies that could be used to treat the low and intermediate level waste streams generated at Cernavoda Nuclear Power Plant.
During the review, consideration was given to both emerging technologies and proven technologies, such as Nochar (super absorbent polymer), which is already under license by Romanian companies. The most favoured waste treatment option(s) for each waste stream was identified using Best Practicable Environmental Option (BPEO) assessment. Recommendations are presented also for the location and the conceptual design of the station, along with Preliminary Safety Assessment Report and Waste Acceptance Criteria.
The project took into account the views of many stakeholders, e.g. Nuclearelectrica S.A., Energonuclear, CNCAN and AN&DR; it will be further revised by all the actors in order to

respond to their latest waste management strategies and to solve, with economical and environmental efficiency, the necessity to have acceptable disposal packages for the final repository.

II.1.3. Romanian Approach to Implement a Deep Geological Repository for Radioactive Waste in Crystalline Rock Formations

Ruxandra Toma[1], Raluca Fako[1], Gheorghe Barariu[1], Ilie Prisecaru[2]
[1]Subsidiary of Technology and Engineering for Nuclear Projects, Romania
[2]University Politehnica of Bucharest, Faculty of Power Engineering, Romania
tomar@router.citon.ro

The Nuclear Agency and Radioactive Waste (AN&DR) is responsible and coordinates the national strategy on medium and long term regarding the management of spent fuel and radioactive waste, including the disposal and decommissioning of nuclear facilities. The Romanian National Strategy, approved in 2004, estimates the operation of a deep geological repository for spent fuel and high level waste to begin in 2055. Starting with the implementation of this strategy the Subsidiary of Technology and Engineering for Nuclear Objectives, Romania, completed a series of R&D studies regarding the spent fuel and long lived radioactive waste disposal and the deep geological repository concept.
The rock formations studied in Romania for implementing an underground research laboratory on the site and later a deep geological repository are: green shales (Hârşova – Dăieni – Rahman – Casimcea – Vulturi), granite (Vârful Pietrii – Masivul Tismana – Muntele Mare) or clay (probably, Socodor).
In Romania, crystalline rock formations represent a feasible option for spent nuclear fuel disposal. In recent studies a series of sites on the Romanian territory has been identified and research on the possibility of selecting a site in crystalline rocks has been made.
This paper presents the Romanian approach to implement a deep geological repository for radioactive waste in crystalline rock formations and re-evaluates the selected potential sites regarding the long term safety of the repository and the characterization of the crystalline rock formations.

II.1.4. A Material Science Perspective on Developing Concepts for Final Disposal of Spent Fuel in Romania – Unknown Data, Challenges and Constraints

Raluca Fako, Florin Sociu, Gheorghe Barariu, Radu Georgescu
Subsidiary of Technology and Engineering for Nuclear Projects, Romania
fakor@router.citon.ro

According to “5th National Report under the Convention on Nuclear Safety” for Romania (CNCAN) and “7th Situation Report on Radioactive Waste and Spent Fuel Management in the European Union” (EC, Brussels, 22.08.2011), Romania reported for the period 1985 – 2007 about 1,869 m^3 institutional wastes disposed at Baita Bihor and, also, at the end of 2007 a quantity of 1,072 t HM spent fuel in interim storage. Romania is in the stage of the development of conceptual design for a geological repository, with an unknown location, to be operational starting with 2055 and to accommodate about 20,000 t HM and other LILW/LL coming from operation, dismantling and refurbishment of Cernavoda NPP.
For deep geological repository sitting (in a crystalline formation or, respectively, in sedimentary host rock), in case of Romania, due to several constraints (geologic,

economic, demographic and so on), many apparently acceptable locations must be excluded from the selection process even if AIEA selection criteria for host rock are fulfilled. Also, must be considered the opportunity of spent fuel and long lived, low and intermediate level wastes co-localization.
In this context it is mandatory to predict possible long-term evolution of the container material to achieve input data for any scenarios with possible chemistry alterations of the nearfield and farfield environments. Must be developed analyses to demonstrate the existence or not of any impairment of the long-term safety of the multiple barriers system of the Deep Geologic Repository in Romania (DGMA).
The paper is reviewing several disposal concepts worldwide, stressing on presumed interfacial interaction between container material and deep geological environment, respectively on environmental constraints due to corrosion phenomena, radiation, microbial metabolic activities etc. and material specific constraints such as manufacturing route, container design, metallic materials ageing phenomena etc.
It is intended an integration of experimental results for a series of corrosion systems that is useful for a preliminary selection of candidate materials. Authors are emphasizing the importance of targeting both basic and applied research to ensure conformity of materials in real service conditions taking into account the long term safety of engineered barrier systems.

II.1.5. Testing the Capacity of Liquide Membranes for Extraction of Cs-137 and Co-60

Carmen Arsene
Institute for Nuclear Research Pitesti, Romania
carmen.arsene@nuclear.ro

Generated from the TRIGA Research Reactor operation, the β-γ radioactive waste contains Co-60, Co-57, Mn-54, Cr-51, Zn-65, arisen by neutron activation of the structure material, as well fission products as Cs-137, accidentally released. The waste is produced by decontamination of ion exchange filters used for decontamination of the water from reactor pool.
The aim of the present work was to test some emulsion liquid membrane (ELM) systems prepared in view of removing the radioactive cations from different types of wastes.
To remove Cs-137 and Co-60 nuclides from the radioactive waste generated by the TRIGA reactor, two types of emulsions were used, having the following compositions: Emulsion A - tributil phosphate, Span 80 (emulsifier), Aliquat 336 (as carrier), kerosene and $NaHCO_3$; Emulsion B - tributil phosphate, Span 80, kerosene and HNO_3.
In the experiments, different quantities of waste and emulsion, different mixing speed (300 or 500 rpm), pH values (8.5-13) and rest period (30 or 60 minutes) were used, the constant parameters being the temperature (25^0C), the ratio emulsion: waste (1:10) and the contacting time (20 minutes).
The capacity of emulsion to extract the radioactive cations was tested by passing the emulsion through the waste in more steps. The ratio emulsion / waste used in experiments was 1 : 10.
The extraction yields were very small in case of using the emulsion B. In case of experiments using emulsion A, the extraction yields for Cs-137 was about 17%, and for Co-60 of 95%.

II.1.6. Performance Assessment for Disposal of Irradiated Graphite from Research Reactors in a Near Surface Repository

Alina Constantin, Daniela Diaconu, Crina Bucur
Institute for Nuclear Research Pitesti, Romania
alina.constantin@nuclear.ro

The final disposal of irradiated graphite is a topical issue and urgently seeks for favourable solutions, but no firm decision has been taken so far concerning conditioning and packaging of graphitic waste although these materials impose major challenges due to big volumes and associated radionuclide inventory. Worldwide, there are more than a hundred of nuclear reactors containing poligranular graphite either as moderator or as neutron reflector and the most of them are located in United Kingdom, France or former Soviet Union. Moreover, different carbonaceous materials have been used for fuel protection or are part of reactor components.
In Romania, irradiated graphite does not exceed 10 tones and comes part from the thermal column of TRIGA SSR 14 MW operated by INR Pitesti and part from the decommissioning of VVSR reactor from IFIN-HH Bucharest. The total radionuclide inventory is low and mainly dominated by ^{3}H, ^{14}C, ^{152}Eu, ^{60}Co and ^{36}Cl, depending on the position in the thermal column and the irradiation history. However, an optimal solution for its final disposal is envisaged for the National Radioactive Waste Management Plan. An alternative may be the disposal in a near surface repository.
^{14}C and ^{36}Cl are radioisotopes which may exhibit long term impact, both with potential to be released from graphitic material. ^{14}C has an extremely complex behaviour in geological environment and may be partitioned between liquid, gaseous and solid phases, its composition being controlled by the carbonate chemistry, microbial and gas generating processes. ^{36}Cl behaves much more simpler, being potentially released only in liquid phase and with no significant interaction with materials expected in a geological disposal facility.
In this context, the paper proposes a GoldSim model for performance assessment of irradiated graphite disposal in a surface repository with characteristics borrowed from the Saligny site, considering only ^{14}C and ^{36}Cl radioisotopes. As there are still a lot of uncertainties and lack of some relevant input parameters, the model is rather a support on which adjustments are still to be made as soon as new data become available.

II.1.7. National Disposal Facility for Low and Intermediate Radioactive Waste

Svetomira Prodanova
State Enterprise Radioactive Waste, Sofia, Bulgaria
svetomira.prodanova@dprao.bg

The national *"Strategy on management of spent nuclear fuel and radioactive waste till* 2030", approved by the Bulgarian Council of Ministers on 5 January 2010, settles plan for establishment of National Disposal Facility (NDF) for Low and Intermediate Level Short-Lived (LIL-SL) Radioactive Waste.
“Radiana” site adjustment to the Kozloduy NPP is selected as site for construction of the National Disposal Facility. According to this plan the facility shall be fully operational by 2015. The NDF shall be modular type near surface multibarrier engineered disposal facility. State Enterprise Radioactive Waste (SERAW) is the entity legally responsible for site selection, design, construction, commissioning and future operation of NDF.

II.1.8. Separation and Determination of Lanthanides using High Performance Liquid Chromatography

Georgeta Cosmina Lazar, Marina Tatarascu, Aurel Benga
Institute for Nuclear Research Pitesti, Romania
cosmina.lazar@nuclear.ro

Radioactive liquid waste resulting from post-irradiation examination of spent nuclear fuel contains a mixture of radionuclides, alpha, beta and gamma emitters, including small amounts of lanthanides (rare earths). On the other hand, some isotopes of lanthanides are used as burn-up monitors. Under these circumstances, is of great interest to establish optimal conditions for rapid separation of lanthanide.
This paper presents tests on synthetic solutions containing lanthanides using dynamically modified Reverse Phase High Performance Liquid Chromatography with ion pair (RP-HPLC). The RP-HPLC system consists of a solvent delivery, an autosampler, an UV-VIS spectrophotometric detector, a reverse phase Symmetry C18 column and a post column delivery pump.
Experiments were conducted using a mixed standard solution containing 14 lanthanides (from the complete series of 15 lanthanides only promethium is missing) plus yttrium, scandium and thorium (which have similar chemical behaviour with lanthanide ions). The eluent mixture consists of different proportions of α-hidroxy isobutyric acid with sodium salt of n-octane sulphonic acid.
The RP-HPLC system was able to perform both qualitative identification and quantitative measurement of lanthanide ions, provided that at least 100 nanograms of each highlighted lanthanide was present in the synthetic solutions subjected to the analysis. The presented results were obtained in gradient mode, the paper containing chromatograms which illustrate the optimal conditions of the method.

II.1.9. Comparative Measurements of Alpha Sources Activity from Liquid Effluents and Wash/Decontamination Waters at Cernavoda NPP

Ovidiu Ichim, Marin Mincu, Gabriela Androne, Cosmina Lazar, Octavian Uta, Raluca Negrea, Marina Tatarascu, Diana Ilinca, Aurel Benga
Institute for Nuclear Research Pitesti, Romania
ovidiu.ichim@nuclear.ro

This paper analyzes the alpha sources obtained from liquid effluents and washing/decontamination waters at Cernavoda NPP. For each case, radiochemical separations were performed, followed by manufacturing of individual alpha sources of Uranium, Plutonium and Americium, respectively. Two methods were used to obtain these sources. First method involved a co-precipitation with cerium fluoride, using for separation a set of UTEVA and TRU chromatographic columns, whereas the second method involved a co-precipitation with neodymium fluoride, using for separation a set of TEVA and 2 TRU chromatographic columns. Both methods used ^{233}U, ^{242}Pu, respectively ^{243}Am spike solutions. Thus 12 alpha sources were obtained: 4 sources of Uranium, 4 sources of Plutonium and 4 sources of Americium, (in each case, there was a source obtained by each separation method both from liquid effluents and from washing/decontamination water, respectively).
For each case, the source total activity was measured, using two measurement systems: the dual alpha spectrometer ORTEC and the gross alpha counter MPC-2000-DP. The work aimed to compare the results obtained with the two measurement equipments on the above mentioned sets of samples.

For the sources prepared by first radiochemical separation method, the two measurement systems yielded results within ± 12% compared to each other, whereas for the sources prepared by the second radiochemical separation method, the relative variations of the corresponding measurements were within ± 21%.

II.1.10. Separation and Determination of Plutonium out of Liquid Waste from Post-Irradiation Examination of Nuclear Fuel

Marina Tatarascu, Cosmina Lazar, Diana Ilinca, Marin Mincu, Ovidiu Ichim, Raluca Negrea
Institute for Nuclear Research Pitesti, Romania
marina.tatarascu@yahoo.com

One of the techniques used for destructive characterization of spent nuclear fuel is alpha spectrometry. Alpha emitters such as Uranium, Plutonium, and Americium can be measured only by destructive methods.
The purpose of this paper is the separation of Plutonium out of liquid waste from post-irradiation examination of nuclear fuel and its qualitative and quantitative determination.
The sample (spent fuel) is brought into a liquid acid homogeneous form containing the alpha emitters. First step in order to separate the Pu fraction out of the fuel solution is done by passing the fuel solution through a DOWEX column, containing anion exchange resins. This eluated solution is further passed through two consecutive chromatographic columns, getting rid of impurities in this way. Finally, the Plutonium alpha source is prepared by co-precipitation, being deposited on a membrane, which is fastened on a stainless steel disk.
By using the procedure mentioned above, we prepared two Plutonium sources. One of them contains, next to the ^{242}Pu spike solution, the ^{238}Pu, ^{239}Pu, and ^{240}Pu isotopes. The second one was similar, except for the fact that the spike solution was not present. These sources are ready to be measured by alpha spectrometry, for the determination of their total activity and isotopic content.
The measurements have been performed by using two equipments: the gross alpha counter MPC 2000-B-DP and the ORTEC 576 alpha spectrometric system, with two individual measurement chambers. The calibration in energy and efficiency was done with standard alpha sources.
The alpha spectra of plutonium sources were presented, together with the total activities of the samples and, also, activities of individual interest Plutonium isotopes.

II.1.11. The Safety Enhancement in Transport of Radioactive Material and the Problematic of Denial of Shipment

Viorica-Maria Nistor, Ramona-Maria Mihaiu
Institute for Nuclear Research Pitesti, Romania
viorica.nistor@nuclear.ro

Radioactive materials of all kinds need to be transported on national and international routes for use in industrial applications, public health, production of nuclear power and research purposes. Safety is on the top of priorities when shipping radioactive material, reasons for this kind of transport being governed by national and international regulations which are based on IAEA (International Atomic Energy Agency) Regulations for the Safe Transport of Radioactive Material - TS-R-1 Standard, that ensure high standard of safety.

The joint efforts of producers, the transport supply chain, users, regulators and other authorities have led to an exemplary record over last 50 years.
A number of factors may make it difficult to ship radioactive materials efficiently to their destination. Sometimes carriers and ports do not want to invest in the training or designated storage area required by regulations. Inconsistencies between countries in interpreting international regulations can make it impossible for consignors to fulfil all requirements. As a consequence of these difficulties, there are regions of the world that are cut off from essential radioactive materials. Recently there was noticed an increase of the numbers of instances of denial of shipment of radioactive material even when complying regulations. The increase in denial of shipment of radioactive material is generating both social and economic problems and needs to be addressed with a lot of attention and requires a quick solution. In response, IAEA has formed an International Steering Committee (ISC) on the Denial of Shipments of Radioactive Materials.
This paper describes the efforts made by IAEA to enhance the safety in transport of radioactive materials by all routes (road, rail, sea and inland waterway). Also, the paper examines the problematic of denial of shipments (the reasons for increasing denials of shipments of radioactive material, the main causes of the refusals and measures taken by IAEA addressing the issue of denial).

II.1.12. Separation of ^{99}Tc from Aqueous Liquid Samples

Mirela Olteanu, Crina Bucur, Maria Dragomir
Institute for Nuclear Research Pitesti, Romania
mirela.olteanu@nuclear.ro

^{99}Tc is an important radionuclide in nuclear fuel cycle because of the high ^{235}U fission yield (6%). It is an almost pure beta emitter (E_{max} = 294 keV) with a very long half-life (T1/2 = 2.11×10^5 y). The specific activity of ^{99}Tc is very low (0.63 Bq/ng); from this reason ^{99}Tc is difficult to be detected through direct measurement.
The separation and concentration steps are necessary in ^{99}Tc analysis. Among the ^{99}Tc conventional separation/concentration methods (including solvent extraction, ion exchange, precipitation co-precipitation), the extraction chromatography on resin with high selectivity for Tc (TEVA Resin) is frequently used today to separate and concentrate the ^{99}Tc from aqueous liquid sample.
The TEVA resin extraction chromatography was used in laboratory for assessment of Re recovery (Re as an inactive analogue for ^{99}Tc) from Cs and Sr in a simulated aqueous sample. The simulated aqueous sample is loaded on pre-packed columns of TEVA (Eichrom Industries, Inc.), $^{99}TcO^{4-}$ (ReO^{4-}) is strongly retained on TEVA. Spec resin (more than 98% from Re initial concentration). The ReO^{4-} retained on this resin is eluted with a small volume of concentrated solution of nitric acid (20 ml of 12M HNO_3). The final concentrated nitric acid solution is gently evaporated at half of initial volume and diluted with distilled water to 100 ml in a volumetric flask. The concentrations of Re, Cs and Sr was measured by inductively coupled plasma mass spectrometry (ICP-MS). More than 95% from the initial concentration of Re was recovered and the decontamination factor was comparable with data from literature.

II.1.13. Sorption Effect on the Radionuclides Mobility in Vadose Zone

Crina Bucur, Daniela Diaconu, Mirela Olteanu
Institute for Nuclear Research, Pitesti, Romania
crina.bucur@nuclear.ro

The low and intermediate level (LIL) waste generated from the Cernavoda Nuclear Power Plant operations will be disposed in a near surface repository on the Saligny site. Long term performance of this disposal option is significantly influenced by the radionuclide sorption on natural barriers (geological layers below the repository) and engineered ones (conditioning matrix, filling and liner materials, materials from the engineered foundation and so on).
Structural properties of the main geological formations of the Saligny site, the hydrogeological data, and also data on radionuclide migration and retardation processes in the engineered and natural barriers of the LIL waste repository system have been achieved in the past years in the frame of the Saligny site characterization program. Based on this site specific data, the radionuclides migration through the Saligny vadose zone was simulated and the radionuclide releases in the Beriasian aquifer were assessed. The source term was simulated using DUST-MS code while the radionuclides transport through vadose zone was simulated using the reactive transport module of FEHM code.
C-14 and Cs-137 are two radionuclides found in significant concentrations in the LIL waste to be disposed in the Saligny repository whose migration through vadose zone was simulated. For C-14, these simulations showed that even its sorption on the Saligny natural barriers is very low it has a favourable influence on the C-14 concentration that would be released in the Beriasian aquifer. The C-14 interactions with the cementitious materials and with the improved loess layer that will form the repository foundation have significant influence on these releases. On the contrary, Cs-137 is insignificantly sorbed in the cement matrices but it is highly sorbed in the geologic formations of the vadose zone. The simulations performed showed that this radionuclide will not reach the Beriasian aquifer not even in the most conservative situations.

II.1.14. Study of ^{3}H Efficiency Recovery from Cernavoda NPP Spent Ion Exchange Resins using Liquid Scintillation Analyzer

Liliana Bujoreanu, Costel-Danut Bujoreanu
Institute for Nuclear Research Pitesti, Romania
lvasii@yahoo.com

Radioactive resin wastes are produced during routine operations and maintenance activities at Nuclear Power Plant and contain a variety of contaminants including radionuclides, toxic, metals and chemicals. The principal radionuclides in spent resins include: ^{3}H, ^{14}C, ^{51}Cr, ^{55}Fe, ^{58}Co, ^{60}Co, ^{65}Zn, ^{89}Sr, ^{90}Sr, ^{95}Zr, ^{124}Sb, ^{125}Sb, ^{131}I, ^{134}Cs, ^{137}Cs and ^{144}Ce, respectively.
The paper presents the results obtained in the implementation and validation processes for a liquid scintillation spectrometric method used to measure the ^{3}H efficiency recovery from dissolved sample obtained as the result of spent resins combustion process.
The primary objective of all sample preparation methods for ^{3}H measurements is to obtain a stable homogeneous sample using sample oxidation method which is performed in automatic equipment –Model 307 Sample Oxidizer.
The second objective is to study ^{3}H efficiency recovery using liquid scintillation analyzer Tricarb 3110TR.

Methods for determining the presence of tritium in the ion exchange resins and the capability of measuring concentrations of tritium represent safety relevant information regarding the waste properties (chemical, physical and radiological properties).

II.2. Radioprotection

II.2.1. The Effect of Thermal Neutrons on Samples of Human Blood

Khalifa Saleh, Mohammed K. Alfakhar and Asma Abd Elrahman
University of Omar Almukhtar, College of Science, Physics Department,
Faculty of Science, El-Beida, Libya
khalifasaeid@yahoo.com

Radiation can be either useful or harmful when human are exposed to any kind of radiations. The effects of radiations on human blood samples were tested. In this study, blood was irradiated using a Ra-Be (α,n) thermal neutron source. The effects on the components of blood (Red blood corpuscles, white blood cells, and hemoglobin concentration) were examined and recorded.
The work undertaken focuses on the analysis of the blood sample after the subjects have been irradiated. Four cc (4cc) blood samples were gathered from 8 different volunteers (4 males and 4 females). After that, each sample was divided into two equal parts (1cc each) and placed in different polyethylene tubes, first one following to be used as a standard. The other sample was irradiated for six hour using the Ra-Be (α, n) thermal neutron source. Afterward, new samples were taken from some volunteers, and the process repeated with different lengths of exposure to the radiation (12, 18, and 24 hours, respectively).
It was found that the number of red blood corpuscles, white blood cells, and hemoglobin concentration decrease the longer the samples are exposed to thermal neutrons. This trend was consistent with each volunteer's samples, and came regardless of whether they were male or female.

II.2.3. Fault Detection and Isolation Modeling Methods Implemented in Gamma Dosimetric Network TRIGA Reactor

Jeni Talpalariu, Cornel Talpalariu, Matei Corina
Institute for Nuclear Research Pitesti, Romania
cornel.talpalariu@nuclear.ro

Upgrading works from Institute for Nuclear (INR) Pitesti TRIGA Research Reactor impose a new concept for integrated safety related systems, like fixed gamma dosimetric stations network.
The necessity to obtain a good measurement performances for gamma fields in all the monitored area, imposes to use a number of Programmable Logic Devices (30 -100 pcs of PLC) installed in local stations Two serial industrial networks (RS 485) and a UTP Ethernet connection are used for data and control communications. In such a complex network structure, the Fault Detection and Isolation (FDI) software modelling is inherently a complex one.
The paper presents dosimetric network configuration and performances, hardware and software structure considerations for Fault Detection and Isolation problem.
Mathematical models software implemented, for state estimator and fault detection, using the synthesis of detectors signals, inputs and command test vectors, connected to model-based predictions of outputs, perform real time fault detection when failures are presented.

In present paper, the dynamic channel recovery / reconfiguration and isolation software functions methods are considered.
We were concentrating on modeling methods implemented for safety, reliability and availability of every radiations dose measuring channel and for availability factor of the entire safety related network. We are also leveraging our efforts to investigate the safety improvement of maintenance operations for dosimetric measuring stations, placed in high radiations risk zone, using network remote real time fault detection and reconfiguration.

II.2.4. Experimental Tests for Separation of Actinides from Aqueous Samples

Cristian Dulama[1], Relu Dobrin[1], Alexandru Toma[1], Ruxandra Toma[2], Silvia Stoica[1]
[1]Institute for Nuclear Research Pitesti, Romania
[2]University "Politehnica" of Bucharest, Faculty of Power Engineering, Romania
cristian.dulama@nuclear.ro

Due to their long half-lives and high radiotoxicity, actinides represent an important hazard to the environment, especially when they are emitted in soluble form through the effluent pathway from nuclear installations. Measurement of actinides concentration is usually based on detection of their alpha particles emission which is a very sensitive process requiring preconcentration and separation of the analytes from the excess inactive matrix and other radioactive isotopes (fission products or natural radioactivity).
New separation techniques emerged following the development of chromatographic extraction resins based on organophosphorus complexants. The laboratory implementation of such a complex technique can be done only after some preliminary tests which shall aim to reveal both the real needs in terms of equipment and reactants and the critical points to be taken into consideration when applying the method as standard procedure.
The paper presents the experimental tests ran during the implementation phase of a sequential separation procedure for actinide determination in aqueous samples. The procedure is based on the use of UTEVA resin to preconcentrate tetravalent actinides from aqueous samples and separation of uranium and thorium, followed by the use of TRU resin to separate americium and plutonium. The measurement of the activity for each separation fraction is done by alpha particle spectrometry, sources being prepared by micro-coprecipitation with cerium fluoride. By using the alpha-LSC technique, the counting efficiency of the alpha spectrometric method was determined for the specific counting geometry used for this procedure.
Due to the high degree of complexity of the method, tests were performed by adding the steps in a backward manner and by tracing the analytical objectives with quantifiable parameters.
The separation yields for the analytes were determined following each separation step and the global separation yields were determined for the entire analytical process.
The results obtained as concerns the recoveries and counting efficiencies are in good agreement with data published by other authors showing that the procedure was well implemented and adequately applied.

II.2.5. Method of Uncertainty Evaluation in Dosimetric Measurements using Gum

Florea Scarlat[1,2], Eugenia Badita[2]
[1]Valahia State University of Targoviste, Targoviste, Romania
[2]National Institute for Laser, Plasma and Radiation Physics - INFLPR, Bucharest – Magurele, Romania
scarlat.f@gmail.com

The paper presents a practical method to evaluate and express the uncertainty of the dosimetric measurement result, as per ISO "Guide to the Expression of Uncertainty in Measurements" ("GUM") and ISO/IEC 17025 on "General Requirements for Competence of Calibration and Testing Laboratories".
The developed method consists of four steps. First step includes the definition of the measurand, the measurement process for the evaluation of the calibration result or of the measurement, the mathematical model of the measurand function of the input values, the estimate evaluation for the input values of the measurement process and the measurand estimate calculation by means of mathematical equations. The second step consists in the uncertainty evaluation for the result regarding the measurand value. It starts with the identification of all the uncertainty sources that contribute to the measurement uncertainty, the allocation of the probability density functions – Gauss (normal), rectangular (uniform), etc, for the standard uncertainties of the input values and next, the evaluation of the standard uncertainties for the input value estimates - the evaluation of Type A and Type B uncertainties. The third step consists of the application of the uncertainty propagation law for the calculation of the standard uncertainty of the measurand estimation. This step includes the sensitivity coefficients calculation for Type A and Type B uncertainties as well as the co-variants evaluation if the input values are correlated, the effective degrees of freedom, the coverage factor and the expanded uncertainty. Once the two components of the extended uncertainty of the measurement result are determined - the confidence interval and the confidence level, the fourth step consists of the result statement with the associated expanded uncertainty and the coverage factor applied. The dosimetric measurements results are traceable to PTB - Braunschweig. The method presented herein allows the comparison of internal calibration and testing lab measurements with the measurements conducted by external labs (outside one country) in order to verify the stated level of the performances.

II.2.6. The Determination of Environment Pollution using Neutron Activation Analysis – K_0 Method on Mosses

Adrian-Florinel Bucsa, Dumitru Barbos
Institute for Nuclear Research Pitesti, Romania
adrian.bucsa@nuclear.ro

The problem of toxic pollutant agents that exists in environment was and still is a main interest subject for nowadays health protection issues. Evaluating the impact of these agents on human health can influence major decisions that are to be taken by the authorities concerning the industrial activities developed in the inhabited areas or from the surrounding areas that directly impact the environment and inferentially the man. Precise measurements of toxic particles in the environment at the level of microelements are essential for exact and correct evaluations of the pollution degree.

In order to determine the chemical elements that have negative effects on human health, we used as vegetal indicator, the ground moss. This plant has the ability to retain in the tissue the chemical elements precipitated from the atmosphere, because it is missing the cuticle that would normally prevent the elements from penetrating the cell interior. In order to determine the elements' concentration in the samples, it was employed as analysis method the neutron activation (NAA-k_0) using k_0 standardization.

The neutron activation analysis is an analytic technique based on measuring the number and energy of gamma radiation emitted by the radioactive isotopes produced in the sample matrix by irradiation with thermal neutrons in a nuclear reactor. Usually, the sample matrix together with specific flow monitors, duplicates and standards for items of interest are irradiated for a selected period of time in a neutron flux in the core of the research reactor. After the irradiation and the specific radioactive decay, the energy spectrum of gamma rays is obtained by measuring the sample with a detection system for high-resolution gamma spectrometry.

The sample irradiation was processed in the TRIGA ACPR reactor in the rabbit (D10) location. The study was conducted on a total of four samples of ground moss from the environment (rough samples) that were processed, irradiated and studied and the results were recorded in a database.

The thermal neutron flux supplied by the ACPR reactor in above mentioned location was fare enough for activating elements like Mn, K, Br, Eu, La, As, Sc, Sb, Fe. During the entire irradiation period, the sample presence in the irradiation location did not disturb the average value of the thermal neutron flux.

The differences showed between the elements concentrations in the samples are actually because of the pollution more or less intense in the areas from which they were taken. Hence the quality of the ground moss as a good monitor.

From the analysis of the results obtained on these samples and following the comparison with the reference values, it was noticed a slight excess concentration for arsenic and antimony. These overhauls are smaller than 2 ppm. Nevertheless, the alert value is not reached. It is worth mentioning the fact that no sample showed elements like Cd, Co, Hg, Se, or Zn.

II.2.7. A Computer Program Method for Estimation of Entrance Skin Dose for some Individuals Undergoing X-ray Imaging

Taha Mohammed Taha[1,2], ***Califa Saeid Allehyani*** [2]

[1] *Radiation Protection Department, Nuclear Research Center, Atomic Energy Authority, P.O. Box 13759, Cairo, Egypt*

[2]*Physics Department, Faculty of Applied Sciences, Umm AlQuraa University,P.O. Box 715, Kingdom of Saudi Arabia*

tmfawwal@uqu.edu.sa

A computer program is designed based on practical measurements of entrance skin doses patients undergoing radiological examinations. Physical parameters such as field size, half value layer, backscatter factor, dose output, focal film distance, focal skin distance, normal operating conditions were taken into consideration for calculation of the entrance skin dose. It is measured by many techniques such as thermoluminescence dosimeters, ionization chambers.

TLD technique characterized by high precision and reproducibility of dose measurement is presented by addressing pre-readout annealing, group sorting, and dose evaluation. Fifty TLD chips were annealed for 1 hour at 400^0 C, followed by 2 h at 100^0 C. After exposure to constant dose from X-ray generator, a 0.6 cc Ionization chamber was located at surface of

water chest phantom that has dimensions 40 cm x 40 cm x 20 cm and connected with farmer dosemaster.
The Entrance Skin Dose was calculated using the generated software by changing the physical parameters and using the measured output doses. The obtained results were compared with the reference levels of International Atomic Energy Authority.
The constructed computer program provides an easy and more practical mean of estimating skin dose even before exposure. They also provide the easiest and cheapest technique that can be employed in any entrance skin dose measurement.

II.2.8. Determination of Gamma Spectrometric Activity in Selected Samples – Laboratory Intercomparison Exercise

Gheorghe Bulubasa [1], Cristian Dulama [2], Nicolae Bidică [1], Alina Miu [1], Mihaela Vladu [1], Alexandru Toma [2]
[1] Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
[2] Institute for Nuclear Research Pitesti, Romania
george.bulubasa@icsi.ro

The purpose of the present paper is to present the results obtained in 2011 for the determination of gamma spectrometric activity in different samples, during a laboratory intercomparison between the radioprotection laboratories from Institute for Nuclear Research (INR) Pitesti and Institute for Cryogenics and Isotope Technologies (ICIT) Rm. Valcea.
We used for the labs intercomparison three standard sources, with known radioactivity, and matrixes equivalent water, soil and zeolite, and also two samples with unknown gamma activity, in soil matrixes.
At LGABG – ICSI Rm. Valcea the samples were measured with a spectrometric system formed of HPGe ORTEC detector, DSPEC^{PLUS} analyser and a PC with Gammavision software installed, while at L5 INR Pitesti, the samples were measured with a spectrometric system formed of HPGe Canberra detector, InSpector type 1270 analyser and PC with Genie 2000 software.
The results of the intercomparison were inside the maximum extent allowed. In conclusion, the intercomparison was successful, establishing the fact that both laboratories, the most experienced (L5 – SCN), and also the new one (LGABG – ICSI), had good results and are able to perform an environmental monitoring program.

II.2.9. A Method for Low-Level Gross Alpha Activity Determination in Liquid Effluents

Relu Dobrin[1], Margarit Pavelescu[2], Cristian Dulama[1], Alexandru Toma[1], Silvia Stoica[1]
[1] Institute for Nuclear Research Pitesti, Romania
[2] Academy of Romanian Scientists, Bucharest, Romania
relu.dobrin@nuclear.ro

All nuclear facilities in nuclear fuel cycle release trace amounts of various radioactive materials into both the atmosphere and aquatic environment. To ensure that their operations have no significant radiological effects on the public and the environment around the sites, the regulatory requirements of CNCAN and EU have imposed maximum levels for these radioactive releases. Due to their long half-lives and high radio-toxicities, a

special attention is given to actinides (U, Np, Pu, Am, Cm), which are mostly alpha emitters. As required, detection limit of the method used to assess the total alpha activity concentration in liquid discharges should be better than 1 Bq/L. Traditionally, alpha particles have been quantitated with ZnS(Ag) scintillation detectors and gas flow proportional counters (GPC). Alpha spectroscopy has been performed by surface-barrier detectors and grid ionization chambers. All of these methods measure solid samples on planchets. Sample preparation is a difficult issue because of the self-absorption of alpha particle by the sample matrix or by the alpha particle itself. Several complicated and long procedures, including precipitation, filtration, chromatography, extraction, and stripping are required to separate the alpha emitting nuclides of interest from the sample matrix. In addition, evaporation, electrodeposition, and vacuum sublimation are necessary to prepare the samples for accurate analysis by one of these methods. These combined procedures are labor intensive and time consuming; meantime they result in low counting efficiencies, low recoveries, and poor reproducibility. In order to reduce the required time and effort to process and quantitate the alpha component in sample, liquid scintillation counting (LSC) combined with a special extraction resin (*Actinide Resin*) from Eichrom Inc. can be used.
The approach to the gross alpha activity measurement in liquid samples, which combines Eichrom's extraction technology with Packard's liquid scintillation analysis technology, is presented in this paper. The method is quick and allows for detection limits lower than 1 Bq/L (the value required by the regulations for minimum detection limit of the method used to assess the total alpha activity concentration in liquid discharges). Alpha activity concentrations of about 0.25 Bq/L were measured with accuracy better than 13% in only 60 minutes counting time. The minimum detectable concentration of 0.056 Bq/L can be achieved with a 60 minute counting time using 500 mL water aliquots. The sample preparation is faster and the detection limit is lower than in the case of traditional GPC method.

II.2.10. Participation of the CROWN Laboratory in the Intercomparison Exercises Organized by IAEA

Cristina Ciocîrlan, Cristian Dulama, Relu Dobrin, Alexandru Toma, Ovidiu Hirică, Silvia Stoica
Institute for Nuclear Research, Pitesti, Romania
cristina.ciocirlan@nuclear.ro

The CROWN Laboratory (Characterization of Radioactive Wastes and Nuclear Materials) is a functional entity in the Radiation Protection, Environment Protection and Civil Protection Laboratory and provides services regarding the radiological characterization of the materials from licensed nuclear practices in order to release them from the regulatory control or classification as radioactive wastes.
Since March 20th 2006 the Radiation Protection, Environment Protection and Civil Protection Laboratory was designated by the National Commission for Nuclear Activities Control (CNCAN) as notified testing laboratory.
As a member of ALMERA network (Analytical Laboratories for the Measurement of Environmental Radioactivity) of IAEA (International Agency for Atomic Energy), the CROWN Laboratory was invited to participate to intercomparison exercises organized by IAEA for radioactivity assessments of various environmental samples and reference materials, together with other national or international laboratories.
This paper shows the main results obtained by the laboratory at the most recent intercomparison exercises (IC) and proficiency tests (PT).

The exercises were organized on a regular basis, yearly having usually, two reporting stages: one rapid evaluation phase with a reporting deadline of three to seven days and a regular evaluation phase with a reporting deadline of a few months. The samples to be analysed covered a broad range of matrices from water, soil, phosphogypsum, vegetation to simulated air filters. Radionuclides of concern were both natural and artificial and the detection techniques required to perform analyses were: gross alpha/beta counting, gamma spectrometry, liquid scintillation counting and radiochemical separation with alpha spectrometry.
The results obtained by the laboratory in the PT and IC exercises confirmed our performances in the field of environmental radionuclides metrology and also, offered us the possibility to identify the areas which need further improvements.

II.2.11. Radiation Measurements Computerised System for Spent Nuclear Fuel Transport Monitoring

Cornel Talpalariu, Jeni Talpalariu
Institute for Nuclear Research Pitesti, Romania
cornel.talpalariu@nuclear.ro

The transfer of high activity nuclear spent fuel assembly, inside reactor containment building, between reactor core and transport container, or between container and long time storage facility, need a Fast Multisensor Dosimetric System, for spatial high level gamma-ray field's radiations monitoring. High level directive Gamma ray burst fields can be detected during effective transfer of spent or defective Fuel Element from active zone or from handling machine, to intermediary storage or to transport container. Therefore it is necessary to develop fixed or mobile multisensor tracking devices systems, Rad Hard and fast response skills, having spatial resolution and high rate acquisition time.
The above mentioned system can be also used for monitoring of high activity nuclear material handling and for detection and recovery of radioactive debris parts during transport operations.
The system configuration described in this work is based on a number of Rad Hard deploying mobile gamma ion chambers and serial wired network devices accomplish 100 milliseconds measuring /acquisition time, using a remote portable Laptop computer. High speed dosimetric measurements application software developed, and a colour printer, permit real time Gamma field instant data recording for analysis and evaluations.
In present paper the real time gamma burst diagrams recorded using a six mobile ion chambers system configuration, during a HEU spent fuel transfer from TRIGA reactor to a transport container, is presented. The works also includes the presentation of the software performance analysis and measurements results during performance tests and real applications.

II.3. Air, Water and Soil Protection

II.3.1. Effects of Alpha- and Beta- Emitting Radionuclides on Marine Bacteria

Nadezdha S. Kudryasheva, M.A. Alexandrova, T.V. Rozhko
Institute of Biophysics SB RAS, Akademgorodok, Krasnoyarsk, 660036 Russia Siberian Federal University, Krasnoyarsk, 660041, Russia
n_qdr@yahoo.com

Marine luminous bacteria serve as a convenient model for studing effects of ionizing radiation on living organisms. These bacteria are widely used as ecological bioassay for more than forty years. The bioluminescent (BL) bioassays are traditionally applied for monitoring of chemical toxicity, and not long ago we used them for the first time to monitor radiation toxicity in solutions of alpha- and beta-radionuclides. Main testing physiological parameter of the bioassays is BL intensity.
The purpose of the work was to study chronic effects of radionuclides on glowing of luminous bacteria *Photobacterium Phosphoreum*. Effects of model solutions of alpha-emitting nuclide ^{241}Am and beta-emitting nuclide tritium were studied. The bacteria were grown in nutrient media with addition of ^{241}Am (up to 7 kBq/L), ^{3}H-labled aminoacid valine, or tritiated water (up to 100 MBq/L). The ^{241}Am inhibited bacterial growth at all activities of the nutrient media. The tritium increased bacterial growth at activity < 30 MBq/L, and inhibited it at > 30 MBq/L. Bacteria were sampled at exponential and stationary stages of growth; BL time-course of the samples was studied and compared with that of a control (nonradioactive) sample.
Three stages were found in BL kinetics of the radioactive samples of ^{241}Am and tritium: (1) absence of the effect, (2) BL activation, and (3) BL inhibition. The BL activation reached 1,000 – 2,000%; it was attributed to hormesis phenomenon. No linearity in radioactivity – BL intensity dependences were found. All three BL kinetics stages were found in solutions of both ^{241}Am and tritium, i.e. the response of the cells was unified. The stages of BL time-course correspond to general regularity in responses of organisms to stress-factors: (1) identification of a stress-factor, (2) adaptive response/syndrome, (3) suppression of a physiological function.
Accumulation of ^{241}Am and tritium in cells and DNA was determined.

II.3.2. Possible Utilization of the Meat and Bone Meal as a Fuel

Felicia Bucura, Marius Constantinescu, Claudia Sandru
Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
marius@icsi.ro

The purpose of this work is to examine the possibilities of using Meat and Bone Meal (MBM) as fuel for energy production. The samples were provided by a plant that produces MBM. The paper describes some MBM characterization techniques, in order to sustain its energetic potential. In order to use MBM as a fuel, it is important to determine the fuel properties, namely: heating value, humidity, ash content; C, S, N, H determination. Also taking into account the environmental aspects, it was mandatory to determine heavy metals presence.
The most important quality of the fuel is the heating value, which is determined using a bomb calorimeter. We determined high calorific value, low calorific value and net calorific value for two different samples of MBM.

Humidity was determined at 70^0C for 8h, assuming a constant mass of the sample. After that the samples was dried again at 105^0C for 1h.
The elementary analysis of the samples was realized on EA Flash 2000 device. We used a reactor filled with copper oxide and electrolytic copper at 950^0C in order to perform the samples combustion. Also for a good separation of the gases resulted from MBM combustion, a porapack column was used. Ash content was determined at 815^0C for 1.5h.

II.3.3. Evaluating Macrominerals Content of Bone Meal Powder

Claudia Sandru, Felicia Bucura, Marius Constantinescu
Institute for Cryogenics and Isotope Technologies, Ramnicu Valcea, Romania
sandru@icsi.ro

Fertilizers are largely made from the waste products of slaughter houses, such as blood, bits of waste meat and other refuse, bones, hoofs, horns, and hair. Bone meal is used as a phosphate fertilizer type and is available in two types: raw and steamed. This organic residue contains substantial amounts of organic matter and nutritive elements such as N, P and Ca.
It is more brittle and can be ground into a powder form. Because the pace of releasing the nutrients is rather slow, bone meal powder makes an excellent natural time-release fertilizer classically employed to prepare the soil for planting.
The use of MBM as a fertilizer is comparable to the effect of soft rock phosphates and this, on normal agricultural soils, is hardly available for plants.
The present study is intended to provide data on the macromineral composition of two bone meal samples that will aid in the development of the applications described above.
Total concentrations of macrominerals in the bone meal were analyzed according the microwave aqua regia digestion, using Mars 5 Microwave System.
Major minerals levels in bone meal samples were analyzed by flame atomic absorption spectroscopy (Ca, Mg, Na, K, Mn, Zn, Fe) and UV-VIS absorption spectroscopy (P), at the levels mg/kg.

II.3.4. Bioaccumulation Assessment of Radionuclides in Aquatic Plants - Experimental Tests

Silvia Stoica, Monica Valeca, Alexandru Toma, Cristina Ciocîrlan, Gina Zăvoianu, Cristian Dulama, Sorin Mihai
Institute for Nuclear Research, Pitesti, Romania
silvia.stoica@nuclear.ro

Aquatic plant species are of paramount importance for water ecosystems, being the basis of the food chain. Various identified mechanisms can lead to biomass contamination with radionuclides present in the aquatic environment. Along with the micronutrients uptake, the superficial adsorption is one of the most important pathways of plants contamination.
The paper presents the results obtained in the experimental studies carried out in the Waste Water Treatment Plant, located within the Institute for Nuclear Research (INR) Pitesti, in order to describe the transfer of radionuclides in aquatic plants.
An overview of both the incorporation mechanisms and transfer pathways of radionuclides in aquatic organisms is presented as an introduction. The main properties of the plants used in experiments are also presented together with the characteristics of the environment in which the experimental studies were conducted.

Activity concentration of radionuclides in plants and water was determined by high resolution gamma-ray spectrometry.
The radiological characterization of the studied ecosystem established that four radionuclides (^{54}Mn, ^{60}Co, ^{137}Cs and natural uranium) can be used for the experimental study. These radionuclides are usually found in liquid effluents of INR Pitesti nuclear facilities.
The determined transfer parameters from water to aquatic plants showed that the incorporation represents the main mechanism of contamination.

III. SUSTAINABLE DEVELOPMENT

III.1. Polices and Strategies in Nuclear Research

III.1.1. The New Nuclear Technologies and Associated Latent Risk for Accidents

Gheorghe Florescu[1], Constantin Agapi[1], Valeriu Nicolae Panaitescu[2], Ioan-Bogdan Florescu[3]
[1]Institute for Nuclear Research Pitesti, Romania
[2]University Politehnica, Bucharest, Romania
[3] University of Pitesti, Romania
gheorghe.florescu@nuclear.ro

The new nuclear energy production technologies are designated to solve the actual consumer needs trying to eliminate the existing issues related to energy price and environmental pollution increasing. Several of the present technologies are presented to be modern, progressing, improved or new.
Nuclear Power Plants (NPPs) safety systems availability is necessary to be continuously monitored and supervised; in the same time the safety of the nuclear energy production needs surveillance. Actual development of the new technologies and discovery of new materials and efficient technological processes offer the opportunities for their appropriate implementation and use in the NPPs systems configuration and functioning/operation.
The new technologies, the actual scientific findings, and the international cooperation, offer the opportunities to mitigate the actual barriers to cumulate and use advanced energy production processes in order to get new energy sources and to build improved, reliable and safety power plants.
The paper presents the following issues:
• the state of the art in the level of description and promoting of new technologies for nuclear power systems development;
• actual trend on short or long term in presentation of the new technologies: meaning new, progressing, modern and improved technology in the nuclear area;
• criteria already used or that have to be used in classifying the new technologies or Structures, Systems and Components (SSCs) in the NPP sector;
• associated latent risk for accidents taking into account the hidden issues in the new NPPs design and nuclear energy production technologies;
• actual trend in NPPs development and required level of progress to comply with the necessary requirements.

III.2. Education, Continous Formation and Knowledge Transfer

III.2.1. The Public's Knowledge and Perception on the Nuclear Field SCN Pitesti Opinion Survey

Ileana Cretu, Eleonora Dragan, Maria Ionila, Mihaela Petra
Institute for Nuclear Research Pitesti, Romania
ileana.cretu@nuclear.ro

The accident at Fukushima Daichii Nuclear Power Plant that took place on 11th March 2011 has once again influenced negatively the general/wide public's opinion about nuclear energy. This has been the starting point of our work.
It seemed absolutely necessary to put together a questionnaire that would include questions concerning the sources of information for data about the nuclear field, the way people perceive nuclear power plants, the institute's image and the wide public's knowledge about the activities developed on the research platform at SCN Pitesti.
This paper aims to present the survey conducted under the title "The level of knowledge and public information on the activities carried at SCN Pitesti", but also to develop a strategy for promoting the institute's image in order to achieve the objectives set by SCN towards increasing the public's acceptance of the nuclear field in Romania.
The conclusions represent an X-ray examination of the paper's content and will highlight the main measures that need to be taken in the future to make sure that people's fears and concerns about the nuclear power change positively.

III.2.2. First Year Results of "Study Case: The Development and Implementation of the Knowledge Transfer and Preservation Process at Cernavoda NPP" Research Contract

Victoria Balaceanu[1], L. M. Georgescu[2], Cornelia Valache[2], Cristian Tiron[2]
[1]Institute for Nuclear Research Pitesti, ROMANIA
[2]Cernavoda NPP, Cernavoda, Romania
victoria.balaceanu@nuclear.ro

In December 2010, the International Atomic Energy Agency (IAEA) approved Contract no. 16373/RO entitled "Study Case: The Development and Implementation of the Knowledge Transfer and Preservation Process at Cernavoda NPP". This research contract is performed by the Institute for Nuclear Research (INR) Pitesti with the valuable collaboration of the Personnel Training and Authorization Department from Cernavoda NPP.
The above mentioned contract is part of the IAEA's Coordinated Research Project (CRP) entitled "Increasing NPP Performance through Process-oriented Knowledge Management Approach". The Member States that participate in this IAEA Project are: Bulgaria, China, Malaysia, Romania, Russia and United States.
The INR project objective is to develop and then to implement a coherent and consistent strategy for implementing the knowledge transfer and the preservation process for a dedicated operational process at Cernavoda NPP, employing a Process-oriented approach.
The present paper intends to highlight first year results of Contract 16373/RO. It must to be mentioned here completion of the visioning phase and first step in the planning phase. The visioning phase refers to a report containing a well-documented analysis of current

status of the Knowledge Management processes at Cernavoda NPP. First step in the planning phase consists in setting the appropriate plans for risk assessment and knowledge elicitation, preservation and transfer.
Based on all the knowledge gained in this activity, may be the most important conclusion is that the Process-oriented Knowledge Management (POKM) approach is more coherent and consistent than the basic Knowledge Management approach because it assures the improvement of the organizational performance and the safety based on the continuous interaction between the operational processes and Knowledge Management processes. At the same time it can be successfully applied in any nuclear and support organization, not only in NPPs.

III.2.3. New Approach in Radiation Protection Education and Training in Romania: RONEN Initiative

Olivia COMSA[1], Cornelia AVADANEI
[1]Subsidiary of Technology and Engineering for Nuclear Projects, Romania
[2]Nuclear Training Centre, "Horia Hulubei" National Institute of Research and Development for Physics and Nuclear Engineering (IFIN-HH), Bucharest, Magurele
comsao@router.citon.ro

Nuclear Training Centre (CPSDN) is developing, since 1970, the post secondary school and post university training of the personnel involved in the nuclear field and/or in the related areas. Organized since 1996 as a Department within the National Institute for Physics and Nuclear Engineering, CPSDN is contributing both to the professional specialization of personnel involved in nuclear research, and to the dissemination of the research results. CPSDN training programmes were permanently adjusted to the objective suggested by the beneficiaries/participants, taking into account their background and training necessities. With almost 50 years of training experience in the nuclear field, CPSDN is prepared to face the new challenges related to the compliance with European practices and increased concern on the safe development of nuclear applications. Nuclear Training Centre contributes to the implementation of ionising radiation applications by the development of Training Programs for:

- Radiation protection in use of nuclear gauges
- Radiation protection in use of radiological equipment for package control
- Radiation protection in diagnostic and interventional radiology
- Radiation protection in Nuclear Medicine
- Radiological Safety in Radiotherapy
- Radiological Safety in use of sealed / unsealed radiation sources
- Knowledge upgrade on Radiological Safety in use of sealed / unsealed radiation sources and radiation generators.
- Radiation safety in mining and processing of uranium and thorium ores
- Applications of radioisotopes and nuclear radiation sources (complex program)
- Decommissioning of nuclear facilities
- Transport of radioactive materials

To better respond to the new challenges of the EC developed by ENEN and SNETP, IGD-TP and MELODI platforms, CPSDN in cooperation with the most important nuclear institutions launched the initiative RONEN (**RO**manian **N**uclear Higher **E**ducation and Training **N**etwork)
RONEN is a joint initiative of important Romanian Universities, Nuclear Research Institutes, SME, NGO and Regulatory Body to coordinate their efforts in order to strengthen their institutional capability to assure a higher quality of nuclear education and

training according with the EU and European Nuclear Education and training Network requirements.
The main purpose of this initiative is to assist the RONEN consortium to define, based on Romanian and European documentation, a proposal regarding the strategy and future development programmes relating to nuclear E&T in Romania.
It also seeks to establish a national framework to ensure that there will be highly qualified staff to operate the new CANDU reactors currently under construction in Romania.
The partners who started this networking initiative are: IFIN-HH-National Institute for R&D in Physics and Nuclear Engineering-Horia HULUBEI, University Politehnica Bucharest (UPB), University of Bucharest, University Babes-Bolyai Cluj Napoca (UBB), University of Pitesti (UPIT), Institute for Nuclear Research Pitesti (ICN), S.C. DOZIMED S.R.L, S.C. ASCENDIA DESIGN S.R.L., Romanian "NUCLEAR ENERGY" Association (AREN), CNCAN, BNEN-Represented by SCK*CEN, Mol, Belgium.

III.3. International Cooperation

III.3.1. EU International Cooperation on Nuclear Safety and Climate Change

Cristian KANOVITS[1], Olivia COMSA[2]
[1]European Commission, Brussels, Belgium
[2] Subsidiary of Technology and Engineering for Nuclear Projects, Romania
Cristian.KANOVITS@ec.europa.eu

Nuclear safety and climate change are top priorities not only for nuclear community and environment experts, but also for the states and international organizations. UN, IAEA, OECD, EU launched specific international cooperation programs which address this topics. Following the unprecedented earthquake and tsunami that affected Japan and caused the nuclear crisis after Fukushima accident, the leaders of the international community tried to strengthen the international nuclear safety requirements.

IAEA asked the increase of the international community involvement in ensuring that nuclear power is safe and sustainable and asked for a global debate on the future of nuclear energy, as the nuclear accidents respect no borders and nuclear safety is an international common goal and a common responsibility. IAEA stressed also about the need to establish new standards of emergency preparedness and response, from the design of new facilities through construction and operation to their decommissioning, standards which must be an integral part of the international community's global strategy for sustainable development.

President of the European Commission Jose Barroso stated: "On nuclear safety, the EU will seek to commit its partners to highest safety standards, ambitious stress-testing of nuclear power plants and enhanced international cooperation".

The G8 leaders also exchanged thoughts on climate change and on the way they can reinvigorate ongoing climate change negotiations by underscoring the urgent need for a binding international framework for tackling climate change and the funds for communities to adapt to the climate change already happening.

Countries around the world have already begun to take increased risks from climate change into account in their nuclear safety protocols

Climate change means more incidents of freak and increasingly severe weather with directly impact on the development of nuclear power and the safety issues which are a matter of global public interest, not a national policy. It can impact on the electricity sector such as causing supplementary infrastructural needs or not allowing machinery operations at 100%.

The paper will present the EU policies, programmes and instruments in the field of nuclear safety, climate change in the frame of the international community's global strategy for sustainable development.

The EU Instrument for Nuclear Safety Co-operation (INSC) finances measures to support a higher level of nuclear safety, radiation protection and the application of efficient and effective safeguards of nuclear materials in third countries since 1 January 2007and its aim is to finance actions in the following priority areas.

The INSC, IfS, as well as the EU Energy and Climate Change Package will be presented as part of the EU international cooperation in nuclear safety and security and climate change.