CONTENT

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RATEN ICN at the 50th anniversary

Constantin Paunoiu, Serban Valeca, Daniela Diaconu, Dumitru Barbos
Institute for Nuclear Research (RATEN ICN) Pitesti, Romania
daniela.diaconu@nuclear.ro

Established 50 years ago with the mission to provide the technical and scientific support to the nuclear national programme, the Institute for Nuclear Research developed a large range of competencies and a complex experimental infrastructure.
The R&D activity of the Institute for Nuclear Research covers all important aspects of nuclear energy programme, mainly addressed to CANDU technology:
- development and application of reactor calculation models and safety assessments,
- evaluation of nuclear fuel performance,
- structural materials and components aging mechanisms and life-time assessment,
- design, development and testing of equipment for Cernavoda NPP
- plant life extension of the life of the power plant
- radioactive waste characterization, treatment-conditioning techniques and studies on low and medium active waste and spent fuel disposal.
- radioprotection and environment radioactivity measurements

Concrete contributions to the nuclear energy development in Romania have been materialized, among others, in:
- design, execution and commissioning of equipment the Units 1 and 2 of Cernavoda NPP such as the heat transmission reactivity monitoring system and the failed fuel location system for units U1 and U2;
- testing the fuel loading-unloading machine heads for U2,
- assistance to the commissioning of Units 1 and 2,
- investigation of CANDU irradiated fuel
- development of radiological characterization techniques of radioactive waste
- development of treatment and conditioning technologies specific to each type of radioactive waste (solid, liquid, gaseous) resulting from the Cernavoda NPP or from other domestic industrial waste producers, medicine, etc.
- site characterization for the near surface disposal of operational waste, performance and safety assessment.

Another research direction of the institute, continuously growing in the last 15 years, is devoted to the development of the advanced nuclear systems and SMRs. Priority is given to the Lead Fast Reactors technology (LFR), in the development of which ICN will host, construct and operate a complex and unique experimental infrastructure, including the ALFRED demonstrator. Testing of materials compatibility with the molten lead environment, modelling and simulation of neutron and thermo-hydraulic characteristics of the ALFRED demonstration reactor, preparation of the ALFRED licensing as well as the education and training of the necessary personnel are already underway to support to the ALFRED project implementation.
The experimental support infrastructure is devoted to the demonstration, testing and qualification of ALFRED materials and components methods, and verification and validation of codes and standards used in designing ALFRED reactor. The first two out of six components of this infrastructure, ATHENA and CHEMLAB, should be operational by the end of 2023.
The entire R&D activity is supported by a broad and dynamic international collaboration programme based on agreements, partnerships and contracts with research centers, European structures and international organisations.

I. NUCLEAR ENERGY

I.1. Advanced Nuclear Systems and SMRs

I.1.1. ALFRED Revised Configuration

Marco Caramello*[1]*, Michele Frignani*[1]*, *Mariano Tarantino*[2]*, Giacomo Grass*[2]
[1]*Ansaldo Nucleare S.p.A., Genova, Italy,* [2]*ENEA, Italy*
Marco.caramello@ann.ansaldoenergia.com

Over the last 20 years, research and development on fast reactors and in particular on Generation IV concepts cooled by heavy liquid metals has made significant advancements, also thanks to a broad support by national and international funds.
Looking at the European context, the technology benefitted from approximately 145 M€ of investments, increasing the technology readiness level throughout various areas including safety and resilience, neutronics and thermal hydraulics. One of the projects currently pursued in Europe and supported by numerous research centres, private companies and member states is ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator), born around 10 years ago within the Euratom FP7 “LEADER” project. Following the conclusion of the project, Romania declared its support in ALFRED development by proposing itself as hosting country for the demonstrator and at the same time the FALCON consortium (Fostering ALFRED Construction) was set out, consisting of Ansaldo Nucleare, ENEA and RATEN-ICN. In recent years, the consortium has further revised the reactor concept to solve some thermal hydraulic issues leveraging the increase of technology know-how. The changes were supported by a new deployment strategy based on different power stages, appointing ALFRED as the prototype of an SMR-type LFR, adding to the formal role as the demonstrator. As a result of this, a new configuration of ALFRED's reactor coolant system has been developed, in which the number of components has been greatly reduced, from 8 to 3 steam generators and pumps, adding flexibility in space management on the reactor cover to install an additional emergency heat removal system and to simplify connection with the main auxiliary systems of the plant. In this paper the new configuration of the ALFRED reactor coolant system is presented, with a detail on the main innovations and modifications, system simplifications and perspectives.

I.1.2. ALFRED Licensing Preparatory Stage

***Mirela Nitoi*[1]*, Michele Frignani*[2]*, Marco Caramello*[2]*, Giacomo Grasso*[3]**
[1]*RATEN ICN, Romania;* [2]*Ansaldo Nucleare, Italy;* [3]*ENEA, Italy*
mirela.nitoi@nuclear.ro

ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator), as the demonstrator of the lead-cooled fast reactor technology (LFR) is considered the necessary first step in the European LFR roadmap, to bridge the gap between the research achievements and the commercial application. In order to ease the licensing process and shorten the deployment schedule, its design will be based on robust technical options with high readiness level, as well as on increased safety margins.
An international Consortium Fostering ALFRED Construction (FALCON) has been established in 2013, gathering European organizations who aim to implement ALFRED as the prototype of a viable LFR technology, in the Small Modular Reactor (SMR) segment, by 2035-2040.
When the ALFRED implementation in Romania was decided, the existing Romanian regulation framework hadn’t been explicitly applied yet to advanced reactor systems, and demonstrator-type reactors. Indeed, in past applications, the acceptance by the Romanian Regulatory Body (CNCAN) of a proposed design had only occurred upon issuance of the construction license.
To enlarge on one hand the applicability of the existing regulation and to include on the other hand the specificities arising when considering innovative technologies, both parties, CNCAN and FALCON Consortium, convened about the benefit of adding a preparatory phase to the authorization process (in anticipation to the classical licensing process). Following this agreement,

the ALFRED licensing process will consist of two main stages: a pre-licensing phase, followed by the well-known, traditional licensing phase.
The paper will present the activities performed on licensing preparation, under the PRO-ALFRED project umbrella, highlighting the main conclusions.
The steps and elements of the envisaged project licensing process will drive towards recommendations to achieve an efficient ALFRED deployment.

I.1.3. Design of SEALER-55, a small LFR for on-grid power production

Janne Wallenius
LeadCold, Stockholm, Sweden
janne@leadcold.com

This paper presents an optimised design of LeadCold's 55 MWe SEALER design for on-grid power production. By use of (U,Hf)N rod sections in the inner core, a flat power and flux power profile is achieved, allowing to extend the residence time of the fuel to 25 full power years, while respecting a radiation dose damage limit of 120 dpa. In addition, revisions of dip-cooler and RVACs design are presented. A dip-cooler capacity of 5 MW is obtained by use of a concept previously suggested for ALFRED. Actuation of the dip-coolers in case of an un-protected loss of heat-sink will ensure that temperatures in the primary system do not exceed those of nominal operation. The RVACs design of SEALER now permits to achieve passively actuated natural convection of air for cooling of the guard vessel during un-protected loss-of-heat sink events, with system temperatures remaining below the rapid creep limit of the cladding. Moreover, a novel method for overlay weld protection of 15-15Ti cladding tubes with alumina forming Fe-10Cr-4Al-RE has been developed, permitting to reduce the time required and associated costs for manufacture of the fuel. Finally, an R&D programme aiming at operating an electrically heated prototype of SEALER in Sweden in 2024 and a research/demonstration reactor in 2030 is presented.

I.1.4. Conceptual Design for RDI and Licensing Infrastructure in Support of ALFRED Implementation

M. Constantin[1], G. Grasso[2], M. Tarantino[2], M. Caramello[3], M. Frignani[3], I. Turcu[1], M. Hororoi[1], V. Cojocaru[1], D. Diaconu[1], M. Nitoi[1], M. Apostol[1], D. Gugiu[1], A. Alemberti[3], P. Agostini[2]
[1]RATEN ICN, Pitesti, Romania, [2]ENEA, Bologna, Italy, 3ANSALDO Nucleare, Genova, Italy
marin.constantin@nuclear.ro

ALFRED is the demonstrator of Lead Fast Reactor (LFR) technology. It is planned to be hosted on Mioveni nuclear platform, Romania. ALFRED will demonstrate the economic and technical viability of LFR technology. In the licensing phase a broad spectrum of experimental activities is aimed to demonstrate the capacity for a complete control of the phenomena, to qualify materials, components and equipment, and also to validate and verify the tools and methods.
A set of six large experimental facilities (ATHENA, HELENA2, ELF, ChemLab, HandsON, Meltin'Pot) are in different stages of preparation to be built as support of the licensing process. Beyond the licensing, the identification of valuable solutions for the open issues of the LFR technology (such as corrosion, erosion and degradation of the structural materials, behaviour of cladding and coatings, behaviour of pump and other critical components in endurance regime, severe accidents scenarios and phenomenology, fuel assembly thermal-hydraulics, pool thermal-hydraulics, chemistry of the coolant and cover gas, etc) will be targeted to support the sustainability of the experimental infrastructure.
An open access is envisaged. It will be based on project competition and it is aimed to create the synergies with other research centres and to contribute to the acceleration of the implementation of the most valuable ideas. A coordination Hub is planned to serve as management unit for the

infrastructure and ALFRED demonstrator, and for the open access management. At the same time, the Hub will be equipped with educational labs and logistics for the education and training activities.
The paper presents the design requirements developed mainly in the framework of PRO-ALFRED project and the evolution of the concepts toward the implementation. For each experimental facility the key features together with the vision on the implementation are presented. The functionality of each facility is individually described and also the integrated operation is explained taking into consideration the siting aspects, the utilities and common services to be developed.

I.1.5. PLOFA in a HLM pool facility: long term cooling experiment

Pierdomenico Lorusso[1], Giacomo Grasso[1], Ivan Di Piazza[1], Francesco Lodi[1], Daniele Martelli[1], Mariano Tarantino[1], Marco Caramello[2], Gabriele Firpo[2], Michele Frignani[2]
[1]ENEA, Italian National Agency for New Technologies, Energy and Sustainable Economic Development, [2]Ansaldo Nucleare S.p.A., Genova, Italy
pierdomenico.lorusso@enea.it

Since 2000s, the development of Generation IV fast reactors cooled by Heavy Liquid Metals (HLMs) has been the focus of several research activities and projects, many of which co-funded by the European Commission and involving a large number of the most important agencies and institutes operating in the nuclear field.
One of the key points of HLMs regards their good neutronic and thermo-physical properties, which allow to design cores with a high pitch-to-diameter ratio. In terms of passive safety, with a properly designed configuration, it is possible to increase the system capability to remove the decay power in the natural circulation regime with a consequent reduction of the active safety systems involvement. Such safety-related aspect has been experimentally investigated at the ENEA Brasimone Research Centre, in the framework of the Euratom H2020 SESAME, an EU co-funded project dedicated to the development of liquid metal-cooled fast reactors.
An experimental campaign reproducing Protected Loss of Flow Accident (PLOFA) scenarios has been executed on CIRCE, a lead-bismuth eutectic cooled pool-type facility reproducing in relevant scale the main components of HLM-cooled fast reactors. In particular, a test section named HERO has been installed in the CIRCE main vessel, including a steam generator with double wall bayonet tubes scaled 1:1 in length with respect to the one envisioned for the ALFRED reactor. The tests reproduce the loss of primary coolant flow, while the power supplied by the heating source decreases according to a characteristic decay heat curve. The feedwater in the secondary system is regulated to operate the main steam generator as a Decay Heat Removal (DHR) system.
This paper presents the PLOFA transient reproducing the worst case, where the feedwater supply to the steam generator is suddenly stopped, simulating in such a way the full loss of the heat sink (no DHR). The main phenomena occurring during the transition from forced to natural circulation (e.g., temperature peaks in the heating source, mixed convection and stratification in the main pool) are presented and discussed. As main outcome, the experiment shows that, despite the loss of the forced circulation regime in the primary loop and the full loss of the heat sink, the entire system is still capable to operate safely, assuring an effective long-term cooling, as long as the thermal heat losses from the main vessel balance the decay power supplied by the heating source.

I.1.6. Control System Development for Liquid Lead Testing Installation

Alexandru NIȚU, Laurențiu AIOANEI, Andrei VÂLCU, Vasile RADU, Marian HOROROI, Viorel IONESCU, Livia STOICA, Denisa TOMA, Valentin OLARU
Institute for Nuclear Research, Pitești, Romania
alexandru.nitu@nuclear.ro

The development of ALFRED Generation IV demonstrator reactor involves the carefully selection of the structural materials able to withstand the harsh environment of liquid lead. The compatibility of the candidate structural materials with the liquid lead is considered one of the critical issues in the development of the LFR reactors. It is known the contact between the liquid lead and the

structural materials may produce a severe deterioration of the mechanical properties of the structural materials. In comparison to the current Generation III commercial reactors, the materials for internal components and fuel cladding of the Generation IV reactors will be operated at elevated temperatures, in a corrosive and high radiation environment. Therefore, there is a mandatory requirement to test and to model the behaviour of the candidate materials in the specific environment of LFR reactors. This paper describes the first lead testing facility developed by the RATEN ICN, called LIquid LEad Testing INstallation (LILETIN). The facility has been designed to perform tensile tests in a liquid metal crucible configuration. It allows the following operational parameters: maximum temperature 5000C, the volume of liquid lead 0.9liter, and the strain rates controller in static lead conditions. In this first stage of the LILETIN development, the experimental results characterize the tensile behaviour of the tested specimens in the liquid lead environment without the oxygen control. The important issue related to the monitoring and controlling the dissolved oxygen is the subject of the paper content. In this context has been decided to use the electrochemical oxygen pump (EOP). This method allows both removal and addition of oxygen to the crucible containing liquid lead, depending on the polarity of the applied electrical potential. In order to be able to implement this method of the regulating the oxygen concentration on the LILETIN installation, some improvements were achieved and they are described in the paper.

I.1.7. Competence Building Actions for ALFRED Infrastructure

M. Apostol[1], M. Constantin[1], D. Diaconu[1], D. Chirlesan[2], S. C. Valeca[2], S. Fianu[2], G. Plaiasu[2], D. Dupleac[3], M. Tarantino[4], G. Firpo[5]

[1]Institute for Nuclear Research (RATEN ICN) Pitești, Romania, [2]University of Pitesti (UPIT), Romania, [3]University Politehnica of Bucharest (UPB), Romania, [4]ENEA, Italy, [5]ANSALDO Nucleare

minodora.apostol@nuclear.ro

An advanced research infrastructure is envisaged to be built in Romania, on the Mioveni nuclear platform, in the frame of the implementation of Advanced Lead Fast Reactor European Demonstrator (ALFRED). It consists in the ALFRED demonstrator itself, as well as six experimental facilities (ATHENA, HELENA2, ELF, ChemLab, HandsOn, Meltin'Pot), and the coordination Hub. The experimental facilities are dedicated to support the activities for testing and qualifying materials & components, for validation and verification of numerical tools and methodologies, and also to identify valuable solutions for the open issues of the Lead Fast Reactors (LFR) technology. The implementation of ALFRED demonstrator in Romania introduces a set of obligations and responsibilities both in terms of adequate preparatory activities (such as the development of experimental basis, the development and validation of computational tools), and also in ensuring the full range of competencies associated with the LFR technology. Important steps were performed in the PRO ALFRED project (September 2019 - November 2020), coordinated by RATEN ICN in the frame of the national RDI Programme. One of its specific objectives was to develop methods for building the needed competences related to the ALFRED infrastructure requirements as well as to implement specific training actions. To achieve this objective, a partnership was established, bringing together two Romanian universities (University of Pitesti – UPIT and University Politehnica of Bucharest - UPB), as the main suppliers of technical personnel for the nuclear field in Romania. The European support of ENEA and ANSALDO Nucleare, Italy, as members of FALCON Consortium, was also available. A dedicated competence building work package in the PRO ALFRED project covered numerous aspects of the preparation of the training activities able to ensure the Human Resources (HR) needs. All of them contributed to the elaboration of a suitable HR strategy. The paper will present the main activities targeting this objective, together with the most important outcomes.

I.1.8. The OPEN100 Project: The World's First Open-Source Platform For The Design, Construction, And Financing Of Nuclear Power Plants

Adam Zuckerman
Energy Impact Center, Washington, DC, United States
adam@energyimpactcenter.org

Nuclear energy is the only technology capable of generating the quantities of low-carbon electricity needed to meet the world's growing energy demand while accounting for its own lifecycle carbon emissions. However, we need far more nuclear power than currently exists to achieve widely accepted international climate goals — one hundred times more. Achieving this vision will require the construction of nuclear plants that are more affordable and competitive. Despite being the single largest source of low-carbon electricity generation in the United States, over time the trend toward building ever larger and more complex reactors has made nuclear energy uncompetitive. The OPEN100 Project, the world's first open-sourced platform for the design, construction, and financing of nuclear power plants, demonstrates that new nuclear developments can be deployed using existing and well-known technology without prohibitive costs, resulting in a framework to develop realistic clean energy alternatives to fossil fuels. To address challenges commonly associated with new nuclear development, the Energy Impact Center conducted a comprehensive review of the nuclear industry. Informed by 1,500+ interviews with subject matter experts around the world, we confirmed that the nuclear industry is not restrained by technology or public opinion. Rather, we identified that challenges to new nuclear deployment is most often linked to cost overruns driven by escalating size and unnecessary complexity. Instead of building bigger or more complex projects, cost reductions will be achieved by returning to a simpler, more streamlined process that adopts today's best practices in construction. A focus on standardization and speed of delivery can course correct the nuclear industry. The OPEN100 Project offers developers around the world three foundational tenants for successful nuclear development: engineering schematics, construction schedule, and detailed economic analysis. Offering a reference design, it leverages the engineering behind the most successful nuclear energy deployments in history to create the foundation for a new generation of power plants that are easier and more cost-effective to build. Characteristics include:

- Proven Technology: Pressurized Water Reactor - UO2 pellets at sub-5% enrichment;
- Rapid Construction Period: 18-24 months;
- Flexible Siting and Low Water Requirements: Due to direct air-cooling condensers;
- Variable Power Output: 50-150MWe.

The OPEN 100 Project is an international, collaborative, community-driven effort. Supported by an engineering team in Atlanta, Georgia (USA), the Energy Impact Center formed strategic collaborations enabling OPEN100 to incorporate institutional knowledge from the best-in-class laboratories, power plant developers, and nuclear industry specialists around the world. A list of collaborators is available on the OPEN100 Project's website: www.open-100.com. The free and publicly-accessible nature of OPEN100's reference plant encourages the growth of a robust developer industry, one that relies on a set of common standards that taps into an existing supply chain and enables fast-tracked licensing. In this way, it can provide the world with a clear pathway to a sustainable, low cost, zero-carbon future.

I.1.9. Conceptual Design of an Innovative Molten Metal Micro Modular Reactor (4M)

B. Aliyari, O. Noori-kalkhoran, M. Rahgoshay
Department of Nuclear Engineering, Faculty of Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran, Faculty of Nuclear Engineering, Shahid Beheshti University, Tehran, Iran,
Babak.aliyari@gmail.com

Considering the new energy market context including the competition between nuclear energy and renewables, new types of nuclear reactors with improved economics and safety features are necessary. Small modular nuclear reactors (SMRs) have important advantages in term of the capital cost, return of investment, and construction time in contrast with traditional nuclear plants. SMRs have more flexibility in the electrical grid and higher inherent safety such as lower fuel inventory, and fission products (both important in severe accidents). Smaller reactor core, fully passive cooling systems, and lower neutron flux are other benefits of this reactor type. Very Small Modular Nuclear Reactors (VSMRs) are a subcategory of SMRs that are defined as reactors with a power capacity of lower than 15 MWe. VSMRs have all the advantages of SMRs in the cost, construction time, and inherent safety at a higher level, that makes them more attractive. These reactors are used for specific applications such as ship and spacecraft propulsions, floating power plant platforms, power supplies for remote places, and emergency power of the hospitals. In this paper, a thermal-neutronic conceptual design of a subcritical VSMR, core will be critical by use of a neutron generator, with modified inherent safety is presented. In this reactor, hereafter named “4M”, the heat is generated by uranium fission, extracted by using liquid Lead as primary coolant and Helium flow for the secondary circuit. Helium removes the heat from the Lead using heat exchangers, and electricity is generated by a Brayton cycle. This study focuses on the thermal-neutronic design of the primary loop while secondary coolant and heat exchanger were considered only as boundary conditions. Thermal-neutronic parameters of reactor core have been simulated by using MCNPX 2.6 code and ANSYS 18.2 software (CFX module). Simulation results confirmed the ability of the innovative 4M system to reach acceptable performances that are discussed in detail in the manuscript.

I.1.10. Neutronic Study of ALFRED Small Modular Reactor Core Using MONTE CARLO Simulation

Saeed Zare Ganjaroodi, Mohammad Rahgoshay
Faculty of Engineering Science and Research Branch, Islamic Azad University, Tehran, Iran
szareganjaroodi@yahoo.com

Small Modular Reactors are an innovative approach to a new generation of nuclear power plants, which gained a growing interest in many countries during recent years. The purpose of small modular reactors is to develop, design and construct innovative small nuclear power plants with high economic competitiveness and higher level of safety. This type of reactors could be produced as separated modules in the factory, being further assembled on the power plant site. Lead-Cooled Fast Reactor (LFR) is one of the six advanced 4th generation reactors. These systems have excellent material handling capabilities due to the use of a fast neutron spectrum, and they could use a closed fuel cycle to convert more efficiently the enriched uranium. The ALFRED reactor design investigated in this work has a 125 MW electric power being cooled by lead. ALFRED design solutions allow components to be removed from the Reactor Vessel to facilitate inspection, maintenance and, replacement. In the present study neutronic parameters of the ALFRED reactor core such as effective multiplication factor, excess reactivity and neutron flux distribution are investigated using Monte Carlo codes. In this paper, the reactor core is simulated by both SuperMC3.2.0 and MCNPX2.7.0 codes. In this code, the KCODE card was used for critical source calculations, and 700000 particles with 200 cycles were considered. Finally, the results were benchmarked with references to show an appropriate consistency. Results illustrated that the calculated excess reactivity is around 78.05 (mK) when 10% of the normal control rods are into the core. Also, the insertion of the control rods into the core can move the maximum flux height of about 40% to the end.

I.1.11. Polonium Evaporation in LFR and an Experimental Device Proposal for Testing Absorbent Materials

Ana Maria IVAN, Daniela GUGIU
RATEN ICN Pitesti, Romania
ana.ivan@nuclear.ro daniela.gugiu@nuclear.ro

In all pool-type fast reactors, a cover gas is maintained above the coolant free level in order to accommodate the thermal expansions of the coolant and the level differences due to the

differential pressures in the primary circuit, as well as to reduce the environmental effects on the reactor cover, where the penetrations of the main components are realized. Its chemistry control involves not only the removal of impurities, but also the chemical analysis of any trace elements for the detection and location of any failure occurred to a fuel rod, determining the release of fission products in the reactor coolant.
Trace amounts of hazardous radionuclides will volatilise into the cover gas above the liquid metal in the Lead cooled fast reactors. The radioactivity transported with the cover gas has to be removed before exhaust gases are released in the environment. For this purpose, it is useful to identify the volatile hazardous radionuclides transported in the cover gas above the liquid lead ALFRED demonstrator, to estimate their quantities during normal operation and finally to study the performance of getter materials that can bind the volatile radioactivity and hence remove it from the gas.
The investigation is based on the hazardous volatile radionuclides inventory in ALFRED reactor at EOC (End-of-Cycle) averaged over all 171 fuel subassemblies and in the whole lead coolant that have been evaluated using MCNPX and FISPACT codes in the framework of FP7 – LEADER project.
One of the volatile elements considered as most critical is Po. The paper presents the differences regarding the Polonium adhesion characteristics among different absorbent materials like, stainless steel, noble metals, quartz wool, silver or coated charcoal, materials that can be used as filtering medium for the gas purification system of ALFRED reactor.
The main outcome of the work consists in a polonium trap design proposal able to test different absorbent material. The trap can be mounted on the MELTIN'POT facility for tests with irradiated molten lead. MELTIN'POT is a facility meant for testing fuel-cladding-coolant and to estimate the transport and deposition of the fission products in representative accidental conditions, so as to support the design of accident management provisions.

I.1.12. Strengths and Challenges in using Passive Systems in Advanced Nuclear Reactors

I. V. Gondac, E. Stoica, M. Nitoi
Institute for Nuclear Research, PO Box 78, 0300 Pitesti, Romania,
ileana.gondac@nuclear.ro

There is an increased interest for the "passive systems" topic, highlighted by the various activities and initiatives that have been undertaken worldwide notably by IAEA, NEA or in frame of Euratom projects. The utilization on large scale of passive systems in nuclear power facilities and the evaluation of their performances represent a broad research topic, approached lately by the international scientific community.
Current NPP fleet already uses passive systems to some extent from which some level of operating experience can be gained. WENRA report related to Regulatory Aspects of Passive Systems, specifies that "both innovative and evolutionary new nuclear power plant (NPP) designs propose to rely more heavily on passive systems to fulfil several safety functions".
The purpose of this work was to develop an useful working framework for identifying strengths and challenges in using passive systems in advanced nuclear reactors. The paper presents an analysis highlighting pro and con main points on the use of passive systems in advanced reactors, including the discussion of the results.
Basic design principles, some considerations regarding the operating experience of such systems are presented. The requirements for qualification of passive shutdown systems have been elaborated. Main difficulties associated with assessing the performance and reliability of such systems in the context of nuclear safety are also discussed.
The particular case of ALFRED reactor was presented. Inherent safety characteristics, passive and engineered safety features to achieve safety functions for ALFRED reactor were identified.
By identification of main strength aspects and of challenges regarding the use of passive systems, a better image about the feasibility of using these systems in advanced nuclear reactors was obtained.

I.2. Nuclear Safety & Severe Accidents

I.2.1. NOR-ROM Project for the Enhancement of Nuclear Safety and Security in Romania

Madalina Coca, Andreea Ungureanu, Madalina Ionita, Iulia Jianu
National Commission for Nuclear Activities Control (CNCAN), Bucharest, Romania
madalina.coca@cncan.ro, andreea.ungureanu@cncan.ro, madalina.ionita@cncan.ro, iulia.jianu@cncan.ro

This paper introduces a new project, "Enhancement of Nuclear Safety and Security in Romania – Improvement of Disaster Resilience and Preparedness for Radiological and Nuclear Events" that has started in 2019, as a continuation of the project implemented in 2013-2017. This project is being carried out in partnership with the Norwegian Radiation and Nuclear Safety Authority (DSA) and the International Atomic Energy Agency (IAEA), the national partners in Romania being CNCAN, as project promoter, together with the General Inspectorate of the Romanian Gendarmerie, the General Inspectorate of the Romanian Police, the General Inspectorate of the Romanian Border Police, General Inspectorate for Emergency Situations and the Ministry of Internal Affairs (MAI). The main objectives to be achieved in the project include alignment of the national framework and regulatory practices with the latest international standards and European Union legislation in the field of nuclear safety and protection against ionizing radiation; the implementation of recommendations from the international missions in Romania on nuclear safety, detection and response to events involving nuclear and radioactive materials not subject to regulatory control, cyber security for nuclear installations and training and intervention in case of emergency; implementation of several activities of the national action plan associated with the National Strategy for Nuclear Safety and Security; the implementation of the new responsibilities that CNCAN has in managing the nuclear emergency situations, as well as improving emergency preparedness and response by implementing lessons learned.
The results of this project as mentioned above will consist of the improvement of the regulatory, licensing and control framework through development / updating of 25 specific documents, such as regulations, good practices guidelines and internal procedures. CNCAN has issued in 2020 a new nuclear regulatory safety guide GSN-10, on time limited ageing analyses relevant to the ageing management for nuclear installations, as an outcome of the specific activities carried out through the Norway project. The issuance of the GSN-10 guide was necessary and appropriate for facilitating compliance by licensee, with the requirements regarding the ageing management for nuclear installations. Also, issuing the GSN-10 guide was necessary for supporting CNCAN review, assessment and inspection processes in the field of ageing management for nuclear installations.

I.2.2. Aspects of Safety Analysis for Romania's TRIGA

M. Mladin, G.A. Budriman,S.D. Dulugeac, C.G. Dobrinoiu
RATEN ICN Pitesti, Romania
mirea.mladin@nuclear.ro

Safety analysis for the TRIGA reactors operated by RATEN ICN Pitesti is illustrated. Excerpts from the Chapter 16- Safety Analysis of the latest revision of the Safety Analysis Report (SAR) for TRIGA Steady-State (SSR) are presented including a summary of: initiating events, simulation tools used, models created, and some of the safety analysis results.
The experimental devices associated with SSR and with ACPR (Annular Core Pulsing Reactor) are mentioned, and two safety cases for licensing purposes are presented:

- Safety Analysis for LoopA (SSR), actually the reactor physics and thermal-hydraulic for the most severe postulated accident scenarios at LoopA accommodating six advanced CANDU fuel specimens inside the Testing Section (TS) – TS Rupture and Hot Leg Rupture, both entailing loss of pressurization with no safety injection and no reactor scram. A short description of the irradiation device is included as well as the nodalization schemes of the primary circuit of LoopA for the two scenarios. The results displayed are: pressure evolution, fuel temperature, fluid fractions in the zones of the TS and total flow through the break;

- Pulsed testing of fuel in ACPR: design calculations and safety analysis for the tests done by RATEN ICN Pitesti in the frame of the EU MAXSIMA project. The basics of the MAXSIMA capsule are shown. Our group's tasks consisted in determining the pulse rods configuration or reactivity insertion worth necessary to obtain the desired energy deposition inside the test fuel (UO_2 cladded in Ti-15/15) during the pulses. Also, the power distribution inside fuel and the temperature evolution inside the capsule. Finally, for licensing, a maximum accident was defined and the gaseous fission product inventory, release and radiological consequences in the reactor hall are calculated.

A few concluding remarks are given, about the cases presented and, also, general ones about the safety analysis activity of our group for TRIGA reactors.

I.2.3. Romanian Regulatory Efforts to Enhance Nuclear Safety Culture

Madalina Coca, Andreea Ungureanu, Madalina Ionita
National Commission for Nuclear Activities Control (CNCAN), Bucharest, Romania
madalina.coca@cncan.ro , andreea.ungureanu@cncan.ro, madalina.ionita@cncan.ro

The evaluation of safety management and safety culture is an integral part of all regulatory oversight activities for nuclear installations. All the routine regulatory reviews and inspections reveal aspects that have certain relevance to safety culture. Interactions with plant staff during the various inspection activities and meetings, as well as the daily observation by the resident inspectors, provide all the necessary elements for having an overall picture of the safety culture of the licensee.

In 2019, CNCAN has issued a regulatory guide on the development and assessment of nuclear safety culture, to facilitate the implementation of the regulatory requirements and to support the regulatory oversight activities of CNCAN in this particular area. The regulatory guide recommends that every licensee performs a self-assessment to determine its own model of organizational culture and identifies the elements that support nuclear safety in the categories of artefacts, espoused values and basic assumptions, building and maintaining a healthy nuclear safety culture.

The paper highlights the aspects relevant for the development and assessment of nuclear safety culture, considering the regulatory and operational experience and the development of international nuclear safety standards. It also presents the regulations, guidelines and procedures aimed at enhancing nuclear safety culture in the organizations in the nuclear sector, as well as at strengthening the regulatory capabilities in the area of review and assessment of nuclear safety culture.

I.2.4. Considerations Regarding the Usefulness of the Ageing Management Programs for the CANDU NPP Key Components' Life Extension

[1,3] Lucan Dumitra, [1]Diniasi Diana, [1]Tudorache Gabriela, [2,3]Jinescu Gheorghita
[1]Institute for Nuclear Research, Mioveni, Romania, [2]University Politehnica of Bucharest, Romania
[3]Technical Sciences Academy of Romania, Bucharest, Romania,
dumitra.lucan@nuclear.ro

Ageing management programs (AMPs) represent a structured set of activities oriented to the surveillance, control and mitigation of ageing effects which affect the Systems, Structures and Components (SSCs) comprised in the ageing management process scope. Management programs are based on different predictive, preventive and corrective maintenance practices, environmental qualification programs, periodic testing and surveillance of technical specifications, in service inspection programs, erosion-corrosion programs, etc., as well as any other specific activity which might be performed at the Nuclear Power Plant (NPP) with the same purposes.

Research and development tasks are needed with the aim to establish clear definitions and objectives for all the programs related to the NPP components lifetime, not only in the long term extrapolation of the component integrity and behaviour, but also in the development of new management strategies at

the plant, able to address organisational issues, asset management, human reliability and ageing issues, all at once, in a coordinated approach.
Plant life management (PLIM) plan is an action programme whose aim is to achieve the original design life without safety deterioration and to keep the possibility of the nuclear power plant license renewal open, for its Long Term Operation (LTO). In recent times, this concept applies to CANDU technology reactors, but in the past, it was referred to as ageing management plan, with a similar methodology. A PLIM plan must integrate and, if necessary, complement all the activities related to the assessment and control of the ageing mechanisms affecting passive and long term CANDU NPP key components relevant to safety.
The paper presents the steps of the refurbishment activity strictly necessary for a CANDU Nuclear Power Plant Life Extension and the principal activities related to steam generators (SGs) restoration program. The paper also presents a short description of the Cernavoda NPP Unit#1 refurbishment program that represents the largest investment project developed by Nuclearelectrica (SNN).

I.2.5. Containment Modeling of VVER-1000 Nuclear Power Plant in the Double Ended Cold Leg Accident

Mohammad Rahgoshay
Department of Nuclear Engineering, Faculty of Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran
m.rahgoshay@gmail.com

Since the containment is the last barrier for the radioactive materials in their path to the environment, the inside containment accidents are of a crucial importance in the nuclear power plants safety. In case of a Loss of Coolant Accident (LOCA), the Large Break Loss of Coolant Accident (LB-LOCA) initiated by a DECL (Double Ended Cold Leg) break or guillotine type of break in the cold leg is of special concern. During this accident type, large amount of fluid mass and energy from the primary loop are discharged to the containment through the break. Consequently, the pressure and temperature in different containment's zones are increasing. The paper approaches the case of VVER-1000 Russian pressurized water reactor during the double ended guillotine break accident in cold leg by numerical simulations based on MELCOR and CONTAIN codes. Some important thermal-hydraulic parameters, such as pressure and temperature distributions in the containment's volumes are obtained also by numerical solving of the survival equations including the mass survival, momentum survival and energy survival. The finite difference method is used for numerical schema. After that, the results of these three methods (MELCOR code results, CONTAIN code results, and solving of the survival equations) are compared with the results presented, for the DECL accident, in the FSAR (Final Safety Analysis Report). In this paper, pressure and temperature distribution inside different volumes of VVER-1000 reactor containment (this VVER-1000 operates in Bushehr) due to DECL accident using MELCOR and CONTAIN codes and programming are compared with FSAR of VVER-1000 before melting. The results of FSAR have divided the inside of the containment to 23 sub-volumes and for most sub-volumes inside the containment, the results from the MELCOR and CONTAIN codes and programming are relatively consistent with the FSAR results and in a number of sub-volumes, there are differences. It is important that the results obtained from MELCOR, CONTAIN codes and programming are close to each other. Because FSAR, does not specify exactly the type of models and how to distribute the fluid and many other issues, in this paper, using three methods, the results are extracted and compared with FSAR to answer these questions. Also, the paper discusses the reasons for those sub-volumes that differ from FSAR results and whether the FSAR results are really right for them.

I.2.6. Investigation of the Molten Core Equations for In-Vessel Retention Strategy

Mohammad Rahgoshay[1], Abdolhossein Fereydoon[2], Zahra Ghilavizadeh[1], Mitra Athari[1]
[1]Department of Nuclear Engineering, Faculty of Engineering, Science and Research Branch, Islamic Azad university, Tehran, Iran, [2]Faculty of Mechanical Engineering, Semnan University, Semnan, IRAN
m.rahgoshay@gmail.com

Severe accidents leading to the core melting represent a key issue in nuclear safety domain. For water cooled reactors, it is important to maintain the integrity of the reactor pressure vessel (RPV) and to prevent the release of radioactive material to the containment and environment. In this context, many efforts and research have been done, including the development of computational codes, such as MELCOR and CONTAIN, as well as the performing of thermal-hydraulics calculations after the core melting. In this regard, one of the tools, considered in advanced reactors design to maintain the RPV integrity and the molten material inside it, is the use of In-Vessel Retention (IVR) strategy. Important factors in this strategy are represented by the heat generation conditions and heat transfer of corium after complete melting of the core and the reactor structures. The purpose of this paper is to investigate the governing equations of the molten layers, based on the equations available in literature, in order to investigate the integrity of the PWR RPV walls structure. In the paper, the governing equations of the molten layers are solved for the two-layers model. Molten materials are placed in two layers after the completely melt of the core. Therefore, the oxide materials in the lower head form a ceramic pool and light metals in the upper part of this pool are laid in the metallic layer. In this investigation, it is also assumed that the space between the ceramic pool and the RPV walls, as well as the light metal layer, is surrounded by a crust. Since convection heat transfer has a large share in these conditions, determining the heat transfer coefficient is one of the most important factors of this investigation. In the paper, using the existing equations and models, available in literature to analyse the mathematical model of IVR strategy, the heat transfer coefficients for different layers are determined and then the equations are solved based on them. Because in the equations of the mathematical model suggested by the literature available for the AP1000, the lower head angle in the ceramic pool area plays a key role in heat flux, the results have been investigated for different angles. In order to understand how angles of lower head have effect on the heat fluxes, the calculations are performed by MATLAB, for various angles. By comparing these results with those available in the literature, it can be seen there is a significant positive correlation between increasing of angles and heat flux in ceramic pool area.

I.2.7. Latest Romanian Regulatory Developments in Nuclear Safety

Madalina Coca, Andreea Ungureanu , Madalina Ionita
National Commission for Nuclear Activities Control (CNCAN), Bucharest, Romania
madalina.coca@cncan.ro , andreea.ungureanu@cncan.ro , madalina.ionita@cncan.ro

The paper presents the most recent regulatory developments in Romania, in the area of nuclear safety. New regulations and regulatory guides have been issued by the National Commission for Nuclear Activities Control, CNCAN, in the period 2018 – 2020, such as nuclear safety requirements on surveillance, maintenance, testing and in-service inspections for nuclear installations (NSN-16), on the licensing of the nuclear installations (NSN-22), on deterministic safety analyses for nuclear installations (NSN-24), on the decisional transparency in the licensing process for nuclear installations (NSN-25), on interfaces between nuclear safety and nuclear security (NSN-26), on the application of standards for the assessment and continuous improvement of nuclear safety for nuclear power plants (NSN-27). Several new regulations have been issued also regarding the management of emergency situations. Between 2018-2020, CNCAN has issued also several nuclear regulatory safety guides addressing the following aspects: the fulfilment of the overall nuclear safety objective set in the fundamental nuclear safety requirements for nuclear installations (GSN-03), the preparation of nuclear installations refurbishment (GSN-07), the restarting of nuclear installations after unplanned shutdowns (GSN-08),

the development and assessment of nuclear safety culture (GSN-09) and the time limited ageing analyses used for supporting ageing management programs for nuclear installations (GSN-10).
Work is ongoing to develop new regulations on the configuration management for nuclear installations, on electrical systems and on control and instrumentation systems for nuclear installations and on human factors engineering.
The paper highlights the most important aspects introduced by the new regulations and guides, taking account of the regulatory and operational experience and the development of international nuclear safety standards. It also presents the measures implemented for strengthening the regulatory capabilities for nuclear safety assessment and inspection of nuclear installations and related activities.

I.2.8. A Passive Safety System for CANDU6

Nita Iulian Pavel, **Nitulescu Luminita, *Pancef Rodica
**RATEN CITON, Magurele, Romania*
***RATEN ICN, Pitesti, Romania*
nitai@router.citon.ro

After Fukushima accident the safety enhancement of the new nuclear systems technology and the existing fleet of nuclear power plants (NPPs) was required. An important item is to increase the time window opportunity for plant operator to establish an alternate heat sink in case of Station Black Out (SBO) accident. In this paper, the efforts made by the RATEN in the frame of H2020 PIACE projects in order to implement a passive safety system in a complete active safety design plant as CANDU 6 project, are presented. In this project, RATEN is in charge of the engineering design and computational modelling aspects, required to integrate a passive safety system in the existing CANDU 6 project. In order to design a passive safety system, a three-days SBO accident was assumed to occur at the Cernavoda NPP, a CANDU 6 type reactor. An Isolation Condenser (IC) system able to transport the total energy produced in reactor core due to decay heat was designed and modelled. The engineering design solutions were made by RATEN CITON and the thermal-hydraulic analysis was performed by RATEN ICN using RELAP5 computer code, to confirm the natural circulation both in the secondary and primary circuit, during the SBO accident and heat transfer capability of IC with and without noncondensable gases. The passive safety system design consists in four (4x33%) closed loop circuits, one for each steam generator. Each loop has an IC design to transport 0.66% of nominal thermal power of the reactor. In order to avoid a quick reactor cooldown, the system provides four noncondensable gas tanks (one for each IC), connected to the outlet of IC, provided for reducing the IC heat flux simultaneously with reactor core residual heat decrease. The design concept was adapted to the CANDU 6 reactor power and the specific layout of Cernavoda site, starting from the ALFRED (the demonstrator of the lead fast reactors technology) passive system, to increase the plant operator window response time form 23 hours (current situation) to more than 72 hours, in order to establish a new heat sink for reactor.

I.3. Nuclear reactors and nuclear fuels

I.3.1. New Synthetic Route for the Production of Mixed Oxides as Transmutation Fuels

Karin Popa, Jean-François Vigier, Olaf Walter, Daniel Freis
European Commission, Joint Research Centre (JRC), P.O. Box 2340, D-76125 Karlsruhe, Germany
karin.popa@ec.europa.eu

Generation IV reactors have the potential to play a significant role in the transmutation of minor actinides (such as Americium) contained in the spent fuel resulting from the operation of current and past power reactors. In this context, the assessment of advanced preparation routes of MA-containing nuclear fuels is essential. Liquid synthesis routes allow the production of homogeneous oxide fuels, for single or multiple fuel recycling scenarios, sometimes avoiding separation of the actinides during reprocessing.
The study aimed to demonstrate the feasibility of producing a solid solution of the target composition $U_{0.75}Pu_{0.20}Am_{0.05}O_2$ by using a method based on decomposition of mixed-oxalate in hot compressed water. The XRD and microscopic characterisations of the reaction product indicated that it was nano-crystalline (as expected from the applied mild reaction conditions), single-phase and homogeneous. The lattice parameter obtained was in good agreement with what was estimated for the given composition. After sintering, the material had a high crystallinity, which resulted in sharp diffraction peaks in the X-ray analysis. Under moisturised Ar/H_2 sintering atmosphere the material was stoichiometric, $U_{0.75}Pu_{0.20}Am_{0.05}O_{2.00}$. In a dry Ar/H_2 sintering atmosphere, the lattice parameter obtained was higher compared to the stoichiometric one, in agreement with the formation of sub-stoichiometric oxide, $U_{0.75}Pu_{0.20}Am_{0.05}O_{2-x}$. The preliminary results show a moderate swelling with time due to the alpha self-irradiation.
The present results demonstrate the feasibility of the hot compressed water decomposition method to incorporate Americium in oxide solid solutions in concentrations relevant for transmutation of MA.

I.3.2. Derivation and Solution of Fractional Neutron Transport Equation in Finite Participating Slab Media with Quadratic Scattering

M. Sallah[1,2] and A. Elgarayhi[1]
[1] Physics Department, Faculty of Science, Mansoura University 35516, Egypt, [2] Higher Institute for Engineering and Technology, New Damietta, Egypt
msallahd@mans.edu.eg

The space-angular fractional neutron transport equation is derived in the present work to describe the neutrons transport in finite clumpy reactors with quadratic scattering. Diverse methods of mathematical modelling have been used to characterize the neutron transport in the nuclear reactors. For chaotic media characterized by fractional structure, a stringent mathematical model has to be applied for neutron transport analysis. Neutron exposure induces temperature dependent embrittlement of the reactor background materials and lead to changing in their properties. The problems arise when the environmental properties of these materials with which the neutron interact are fractional in their space or position. The passage to fractional derivatives should start with Boltzmann transport equation, which can describe the anomalous transport of neutrons within processes of scattering, fission, and absorption. The generalized fractional form under more realistic conditions can be solved using Laplace transformation method. The space-fractional transport equation is approximated using the Pomraning-Eddington approximation to obtain two space-fractional differential equations in terms of neutron density and neutron flux. Furthermore, these resultant equations are coupled into a space-fractional diffusion-like equation to be solved in terms of Mittag-Leffler functions. The scattering is considered as quadratic scattering. Numerical results are presented graphically to show the effect of the fractional order in addition to the effect of particle-transport properties on the interesting physical parameters, such as: reflectivity, transmissivity, neutron energy, and net neutron flux.

I.3.3. Evaluation of spent fuel inventory and radioactivity for CANDU type fuel bundles containing different U-based compositions

C.A. Mărgeanu
Institute for Nuclear Research (RATEN ICN) Piteşti, Romania
cristina.margeanu@nuclear.ro

The rapid growth of the global demand for energy leads to an increase in the nuclear energy role and importance based on its ability to make a major contribution in solving the issues related to the availability of energy resources, climate changes, quality of the air and energy security. Nuclear energy could become one of the main actors in meeting the demand for energy, in terms of finding the most suitable solutions for the issues associated with radioactive waste management, nuclear safety, economic competitiveness, sustainable development and proliferation resistance. In terms of nuclear energy sustainability, recycling represents a promising option for improving the efficiency in using the natural resources and reduction of radioactive waste accumulation. Recovered Uranium (RU) from the spent fuel produced by Light Water Reactors operation could be considered a cheap source of optimum enrichment for CANDU fuel fabrication. Another alternative could be the use of the Natural Uranium Equivalent (NUE), an innovative fuel, which is a mixture of RU and Depleted Uranium (DU) in specific weight ratios. The paper investigated spent fuel inventory and radioactivity at the End-of-Irradiation for some U-based fuels, as follows: Natural Uranium (NU), Slightly Enriched Uranium (SEU), RU and NUE. The proposed analysis was performed using two fuel bundle projects, namely: the 37-element CANDU standard fuel bundle and the 43-element CANDU New 43 (CN43) fuel bundle developed by Nuclear Fuel Performances Group at RATEN ICN. Fuel irradiation was simulated by using ORIGEN-S code included in SCALE6 programme package up to burn-up degrees of 7 MWd/kgHE (NU and NUE fuels) and 11 MWd/kgHE (SEU and RU fuels), respectively. In all fuel bundle elements, identical fuel composition was assumed. In present work, the following spent fuel parameters were evaluated: radioactivity (total and in fuel components - actinides and fission products), fissile material inventory, and inventory of isotopes interesting for proliferation resistance. Preliminary analyses were carried out to select the most representative compositions for SEU, RU and NUE fuels. A comparison of the spent fuel parameters was performed both for the same fuel bundle (CN43) with different fuel composition, and for different fuel bundles with the same fuel composition (NU).

I.3.4. Reactor Physics Aspects Associated to the Conversion of a CANDU Reactor Core Fuelled with Natural Uranium to NUE Fuel

Iosif Prodea
Institute for Nuclear Research (RATEN ICN) Piteşti, Romania
iosif.prodea@nuclear.ro

The flexibility of CANDU reactor to accommodate advanced nuclear fuel designs has been proved along the years by many studies. The Natural Uranium Equivalent (NUE) fuel is the option envisaged in this study to be used in a generic CANDU power reactor. The considered NUE nuclear fuel sort is a mixture of 21% Depleted Uranium (DU) and Recycled Uranium (RU) as balance. The using of DU and RU is environmental friendly because these are very abundant by-products coming from Light Water Reactors (LWR) enrichment and reprocessing programs. Moreover, LWR is nowadays the most widespread kind of nuclear power technology in the world and its by-product amounts continue to accumulate. The paper's idea is to gradually convert an actual Natural Uranium (NU) fuelled core at equilibrium state to NUE fuel. In this respect, the NU nuclear cross sections have been changed to those of NUE at the moment of refuelling start. The Reactor Physics aspects pursued during transition from NU to NUE fuel and presented as results consist in: frequency of refuelling, fuel consumption, average core burnup along with safety parameters like maximum channel and bundle powers. The paper's conclusions state that such a

core conversion from NU to NUE fuel is possible to be achieved shaping the imposed NU power envelope and subsequently leading to slightly improvements in the fuel burnup.

I.3.5. Thorium Fuel Bundle Behavior Under Irradiation Conditions. A Comparative Study

A.C. Răduț[1], I. Prisecaru[2], R.M. Roman[1], E. Matei[1]
[1]Institute for Nuclear Research (RATEN ICN), Pitesti, Romania, [2]Politehnica University, Bucharest, Romania
cristina.radut@nuclear.ro

Actual trends in energy production emphasize the need for sustainability and implementation of low-carbon emission technologies. Regarding this matter, nuclear industry plays a key role by means of clean energy production and efficient utilization of mineral resources. In order to meet the future energy production challenges, an imperative need to develop advanced fuel cycles arises. The aim of this paper was to perform thermo-mechanical calculations for Thorium-based fuel used in a CANDU 600 type reactor. Moreover, recent studies indicate that fuel bundles with different enrichment configuration have the potential to achieve a higher burn-up. In this regard, the study was conducted towards investigation of disparity of heterogeneous fuel bundle counter to the widely regarded homogenous type. For the heterogeneous bundle case, the fuel composition has different fissile concentration ordered in the following configuration: the central element contains only ThO_2, the inner and intermediate rings contain 60 % ThO_2 and 40% SEU, while for the outer ring a fuel composition with 80 % ThO_2 and 20% SEU was used. For the homogenous fuel bundle case, a mixture of 77 % ThO_2 and 23 % SEU was considered. A single fuel bundle geometry has been used for both considered fuels, namely the T37 fuel bundle project. The neutronic and thermo-mechanical fuel behaviour investigations were performed using the lattice code DIREN together with the thermo-mechanical codes TRANSURANUS, ELESIM-TORIU and ROFEM-TORIU, respectively. The following key parameters were taken into account for analysis: temperature, fission gas release, internal pressure and sheath plastic strain. The paper's conclusions underline the burnup and operation improvements brought by the utilization of Thorium mixed oxide placed in a heterogeneous configuration, with 30 % increase in fuel discharged burnup, being directly linked to the decrease of discharged fuel bundles. Furthermore, the code by code analysis results sustain a good agreement, this being suggested by slight but expected differences regarding the evaluation of analysed parameters.

I.3.6. Application of Monte Carlo Techniques to Spent Fuel Shielding Calculations

N. Elbassiony[1], C.A. Mărgeanu[2], M. Sallah[1], A. Elgarayhi[1]
[1]Physics Department, Faculty of Sciences, Mansoura University, Egypt, [2]Institute for Nuclear Research (RATEN ICN) Piteşti, Romania
cristina.margeanu@nuclear.ro

Shielding calculations represent an essential element of the nuclear safety, considering the difficulties and challenges that may occur during the spent fuel manipulation, transport and storage, both for protection of human health and impact on the environmental. The paper aims to apply the Monte Carlo techniques for spent fuel shielding calculations. The spent fuel analysis starts at the moment of its discharge from the reactor, and after a defined period of cooling the radiation dose rates at the spent fuel transport are calculated. Two types of fuel, namely natural Uranium fuel and slight enriched uranium (SEU) fuel, in a CANDU standard fuel bundle with 37 fuel elements have been considered. The fuel burnup was simulated by means of the ORIGEN-S code in order to obtain the radioactive inventory and the photons source characterizing the spent fuel. For the spent fuel transport, a generic stainless steel shipping cask type B was considered; the radiation doses at the cask wall and in air up to 2m distance from the shipping cask have been calculated by using Monte Carlo MORSE-SGC code. Both ORIGEN-S burnup code and MORSE-

SGC code for shielding calculations are included in SCALE5.1 programs package. Considering the nuclear safety aspects, the spent fuel is necessary to be kept in temporary wet cooling storage for at least 6 months. During the cooling period, the spent fuel temperature decreases, and the fission products disintegration lead to reduction of the spent fuel radioactivity at levels allowing its safe manipulation. The estimated dose rates were low, their values sustaining a safe manipulation of the spent fuel shipping cask; however, the dose rates characterizing SEU fuel were about 40% higher than those obtained for natural Uranium fuel.

I.3.7. Thermohydraulic Parameters Analysis of VVER-1000 Reactor Core using 3DTH Program Design Based on Unorganized Networking

Saeed Zare Ganjaroodi, Mohammad Rahgoshay
Faculty of Engineering Science and Research Branch, Islamic Azad university, Tehran, Iran
m.rahgoshay@gmail.com

To analyse the thermohydraulic behaviour of the nuclear reactor core, it is necessary to model the complex geometry of the core and consider the relationship between the various components, such as fuel rods and fuel assemblies. Nonlinear equations including mass, energy and momentum survival in fluid are solved to calculate various thermohydraulic parameters including temperature, pressure, density, enthalpy, cross flow, and then coupled these with energy equation in fuel rod and obtain distribution temperature in fuel and clad. In this paper, the advanced 3DTH program is designed using the unorganized networking method based on Delaney triangulation. Due to the importance and accuracy in numbering of fuel rods, identifying adjacent sub-channels in checking their effects in solving equations of mass, momentum and energy, in this program, an innovative method of unorganized networking generation in computational fluid dynamics (CFD) is used. The 3DTH program is a complete thermohydraulic program for modelling and calculating the thermohydraulic parameters of the core, that solves the energy, mass, and momentum equations for fluid and fuel by using the finite difference method. This program is able to analyse and calculate the effect of thermal expansion on fuel rods and the effect of burn-up on the coefficient of thermal conductivity in fuel. On the other hand, this program has a variety of gap analysis models between fuel and fuel rods, including the Calza-Bini model and the gap model presented in the RELAP-5 code. In order to perform high-precision calculations, various types of corrected and updated critical heat flux (CHF) relationships, the effect of fuel porosity on its thermal conductivity has been considered in 3DTH program. In this study, the core of VVER-1000 reactor is modelled by 3DTH advanced program and COBRA-EN code to calculate the thermohydraulic parameters including coolant temperature and density, fuel temperature with axial and radial intervals mesh, core transfer coefficient and thermal expansion parameter of fuel rods. Results showed that the axial coolant temperature changes in the hottest fuel assembly from about 291 °C to 332 °C. Also, the maximum fuel temperature in this fuel assembly reaches about 1207 °C in the centre of fuel, which occurs in the fourth axial intervals, at a height of about 220 cm. Finally, the results obtained by using 3DTH program were benchmarked with COBRA-EN thermohydraulic code and the reactor FSAR (Final Safety Analysis Report) for comparison and validation to show an appropriate consistency.

I.3.8. TRIGA-SSR Thermal Column Neutronic Characterization by Foil Activation Method

Marius Nehoianu[1], 2Serban Constantin Valeca[1,2], Laurentiu Aioanei[1], Andreea Serbanescu[1]
[1]Institute for Nuclear Research Pitesti, Romania, [2]University of Pitesti, Romania
marius.nehoianu@nuclear.ro

Knowledge of neutron flux densities is needed both for the reactor control and for the use of neutron fields or neutron beams.

This paper proposes the neutron field characterisation of an irradiation site created into the graphite thermal column of the TRIGA steady state reactor operated by RATEN ICN Pitesti. Neutron flux and spectrum have been measured by using multiple foil activation method followed by neutron flux and spectrum determination by unfolding. The measured flux densities are reported to the Self-Powered Neutron Detector (SPND) Silver monitor placed into the Beryllium reflector (in the reactor core margin), near to the thermal column. Foil activation detectors, bare and Cadmium covered, dedicated for thermal neutron spectra, were used. Experimental corrections for neutron self-shielding as well as for gamma activity measurements have been applied. For unfolding, two guess spectra were chosen, as follows: a MCNP computation performed for the former thermal column configuration, and a spectrum for graphite generated by using the Cadmium ratio measured for the Gold reaction, respectively. The irradiation site is placed relatively near to the reactor core. This is why the fast neutron contribution evaluation has been also an important issue.
The unfolding results, reported to 10 MW reactor operation power, showed a thermal neutron flux density of 2 1011n/cm2s, with a fast neutron contribution less than 1%, that is not affecting the final results. The Cadmium ratio for the Gold reaction, as neutron spectrum thermalization indicator, has a measured value of 52, indicating the presence of an epithermal component.

I.4. Nuclear Technology and Materials

I.4.1. Using basic research to improve the simulation of MOX fuel behaviour

Marjorie Bertolus
CEA, DEs, IRESNE, Centre de Cadarache, 13108 Saint-Paul-lez-Durance Cedex, France

Nuclear fuel constitutes an essential component of the performance and safety of nuclear reactors. It is composed of complex materials with specific properties and is subjected under irradiation to a large number of diverse but interconnected phenomena. A better understanding of the properties of fuel materials and of the mechanisms underlying their changes under irradiation is key to the development of more accurate and predictive codes for the simulation of fuel elements.
An efficient approach to improve the understanding of fuel behaviour is to complement the examination of neutron-irradiated materials by a basic research approach combining separate effect experiments on model, surrogate or ion-irradiated materials and multiscale modelling from the atomic to the mesoscopic scale.
We will present the approach developed on (U,Pu)O2 fuels in the INSPYRE H2020 project and illustrate the results obtained on significant operational issues and on the development of constitutive laws for the behaviour of nuclear fuels.
INSPYRE has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 754329. This project is part of the research activities portfolio of the Joint Programme on Nuclear Materials.

I.4.2. Study to assess the ductile fracture in Zr-2.5%Nb by J-integral with the Finite Element Method using the Gurson-Tvergaard-Needleman Model

Vasile RADU, Livia STOICA, Alexandru NITU, Denisa TOMA, Valentin OLARU
RATEN ICN, Pitesti, Romania
Corresponding author e-mail: vasile.radu@nuclear.ro

The paper describes the ductile fracture assessment of the CANDU pressure tube, Zr-25.5%Nb alloy, employing fracture mechanics parameter, *J*-integral, based on the Gurson-Tvergaard-Needleman (GTN) model. The finite element modelling is performed with the fracture mechanics software FEACrack.
The main objective is to model the behaviour of the CANDU pressure tubes (Zr-2.5%Nb alloy) for both crack initiation and its propagation through the wall thickness. The GTN model allows modelling the cracking phenomenon with the help of some specific parameters. These parameters were obtained from an iterative process for Compact Tension (CT) specimen by comparison of the experimental results with those from finite element method analyses. The CT specimen under analysis has the material properties of Zr-2.5%Nb alloy. To perform crack initiation and propagation, the model takes into account the extinction of finite elements from the crack front. The fact is performed by de-cohesion when the achievement of the critical fraction of voids in the area is reached. The extinction of the finite elements on the front of the advancing crack happens when the material cohesion achieves the critical fraction of voids. Thus, the fracture toughness J_{IC} values obtained from the FEACrack model and fracture CT experimental test are compared and the results are in good agreement.
The results will improve the approaching of delayed hydride cracking (DHC) in the CANDU pressure tubes using the GTN model with a finite element model.

I.4.3. Research Activities in The Frame of National Research Programme "NPP Circuits Chemistry"

M. Fulger, M. Tunaru, M. Mihalache, D. Diniasi, A. Tudose, D.Petrescu, C.Ionescu, L.Popa
Institute for Nuclear Research Pitesti, Romania
manuela.fulger@nuclear.ro

The main CANDU NPP systems, such as primary heat transport, secondary heat transport and their auxiliaries, are large, complex, expensive and would require extensive outages to repair or replace their metal components when affected by corrosion and dirt. Water chemistry also plays a dominant role in the normal operation of CANDU thermal transport systems and in life cycle management. Ensuring that these systems and components operate reliably and efficiently on long term often requires solutions/recommends to degradation problems that may include changes in the chemistry of the fluids used in the plant or the replacement of materials that are more suitable for application.
The National Research Programme "NPP Circuit Chemistry" is mainly a technical and scientific support programme for Cernavoda NPP and major objectives are focused on: assessing the integrity of metal components in the primary and secondary circuits of a NPP, studies on nuclear materials with improved properties, studies on water chemistry and the use of suitable inhibitors, development of laboratory techniques useful in the analysis and diagnosis of degraded components in nuclear installations, development of solutions to minimize corrosion and extend the life of components. In addition, research has expanded on the corrosion behavior of nuclear grade materials candidate to the construction of GEN IV reactors (LFR and SCWR).
The aim of this paper is to present some of the results obtained from studies on: corrosion processes that can occur under simulated conditions of the primary and secondary circuit of the CANDU reactor, transport and deposition of corrosion products in the primary and secondary circuits, analysis of corroded components using advanced techniques, corrosion behavior in supercritical water of materials with improved properties. In the same context, the devices used in testing, the techniques applied and the technical solutions adopted to minimize/prevent degradation phenomena will be presented.

I.4.4. Study on the Development, Implementation and Analysis of Software Programs for the Acquisition and Control of Working Parameters for the Experimental Test Facility in Liquid Lead Environment

Denisa TOMA, Alexandru NIȚU, Laurențiu AIOANEI, Andrei VILCU, Livia STOICA, Alexandru FLOREA, Viorel IONESCU, Valentin OLARU
Institute for Nuclear Research, Pitești, Romania
denisa.toma@nuclear.ro

LIquid LEad Testing INstallation (LILETIN) is an experimental facility for thermo-mechanical testing in Heavy Liquid Metal (HLM) environment, conceived, designed and manufactured within the RATEN ICN. It is designed to investigate the mechanical properties of structural materials for Lead Fast Reactor (LFR) advanced nuclear systems.
LabVIEW engineering software is a programming environment for the development of automation programs for testing, monitoring and acquisition parameters of interest during the experimental tests carried out within the Thermomechanical and Microstructural Properties group of our department. By using LabVIEW, the quality of the results obtained in the thermo-mechanical testing of the metallic materials for the candidate structural components in Generation IV is considered to increase. This aspect is proven by the comparison between the experimental results obtained and the results from the external partners.
The paper presents the software developed for the liquid lead environment testing facility. The software produces on the front panel three states in the acquisition process: the first one is the one for initialization, in which the connections with Arduino and Quantum X are verified, the second one for the set-up, it is dedicated for the introduction of the initial data (the test parameters) by the operator, and the third stage is the acquisition, where it provides real-time information on the test

parameters. Once the data acquisition is started, a file is created in which the values of the parameters of interest are written, in our case being: force, elongation and temperature concerning time, but also the oxygen concentration. This file can be processed with dedicated software for data processing (.xls, .cvs, etc.).

I.4.5. A Critical Review of the Liquid Metal Embrittlement Models as Support for Generation IV Materials Damage Analysis

Livia STOICA, Vasile RADU, Alexandru NITU, Denisa TOMA, Valentin OLARU
RATEN ICN Pitesti, Romania
livia.stoica@nuclear.ro

In the research field for the Generation IV nuclear reactors, to manage the structural integrity of the nuclear components under representative environment conditions represents a demanding task. In the open literature, the degradation mechanisms of the mechanical properties due to liquid metal contact were mainly classified as liquid metal embrittlement (LME) and liquid metal-assisted damage (LMAD). Important features of those degradation mechanisms involve a physic-chemical and mechanical process, and therefore the interpretation of which is essentially supported the wetting concept. There is a large kind of mechanisms that are proposed to model LME. The subsequent definition for the liquid metal embrittlement is accepted: LME is the loss of ductility of a normally ductile metal or metallic alloy when stressed in touch with a liquid metal that may lead to brittle fracture. To date, there are still gaps in the understanding of LME and there is still no agreement on the best assessment methods and design is essentially based on experimental correlations. Therefore, the paper aims to make a critical review of liquid metal embrittlement models as support for Generation IV materials damage analysis. Taking under consideration the resistance of stainless steel to LME, modelling can be instructive to the overall understanding of LME resistance, but also to use it in the practical applications associated with assessing the structural integrity for the systems in Generation IV reactors.

I.4.6. State of the Art Solutions for the Acquisition and Processing of Data from Nuclear and Industrial Installations

Cristian Costea, Mihai Arva
Institute for Nuclear Research, Pitesti, Romania
cristian.costea@nuclear.ro

The paper presents concepts from the theory of automation, acquisition and processing (by analog to digital conversion) of signals and concepts of PLC programming and high-level object-oriented programming. The paper aims to highlight the latest generation solutions related to the acquisition and processing of signals from instrumentation devices (sensors and transducers) used in process installations in the industrial and nuclear environment. The paper deals with issues related to the implementation of acquisition solutions using a comparative analysis between different development environments (Simatic Step7 and LabView). In order to highlight the characteristics between the two environments, case studies with experimental applications are presented. The paper includes the integration of the two environments using the unified OPC protocol, in this way being able to develop complex cross-platform applications. The paper also presents local human-machine interface solutions using dedicated HMI mode and global using process computers. The proposed solutions materialized with experimental results produced after the acquisition, processing and storage of process data as a result of the heat treatment applied for the LEU type nuclear fuel.

I.4.7. Review of Predictive Control Methods Used in Nuclear Domain

C. IVAN*, M. ARVA, C. GUŢĂ
Institute for Nuclear Research Piteşti, România
cosmin.mihail.ivan@gmail.com

This paper presents methods of predictive control used in the nuclear field, starting from elementary structures of automatic control and continuing with the integration of regulators in distribution systems.
Solutions for integrated implementation of modern methods of control are also presented, via the use of distributing control systems and the possibility of visualizing and utilizing process parameters in real time in a production management system (ERP - Enterprise Resource Planning for instance). At a regulating/supervising level, software of the HMI (Human Machine Interface) type is used, SCADA (Supervisory Control and Data Acquisition) which ensure the monitoring and control of the process as well as the recording in the database of the data received from the automation equipment.
In order to improve the control loops in nuclear plants, a new control law was studied for improving the stability and safety of the plant operation. Algorithms applicable in nuclear power plants have been described, such as Multivariable Integrated Model Predictive Control (MIPC) and Multivariable Predictive Control (MPC).
The paper introduces a case study where MIPC and PID (Proportional Integral Derivative) techniques were used in controlling turbine and steam generator. For complex nonlinear systems, the MIPC can be designed with DLF (Distribution of Local Fields) neuronal networks and dynamic model CARIMA (Control Auto-Regressive Integral Moving-Average). A block sketch is introduced for MIPC nuclear power plants.
This research also describes simulations which show that the rotation speed of turbine and outlet pressure of steam generator under the multivariable integrated model predictive control is faster, steady and has smaller overshoot than under the PID control, when the power load of nuclear power plant is changed.
The results of the simulations show that the multivariable integrated model predictive controller provides superior outcomes in nuclear energy plants compared to PID controllers. Thus, according to the obtained results, MIPC is a viable solution in cases of charge variation in nuclear energy plant installations.

I.4.8. Case Study on Oven Heat Transfer Process Modeling used for He Heat Treatment of Low Enriched Uranium Fuel Elements

Mihai-Catalin Arva[1], Hoarcă Ioan-Cristian[2], Andrei Vîlcu[1]
[1]Institute for Nuclear Research, Pitesti, Romania, [2]National Research and Development Institute for Cryogenics and Isotopic Technologies, Rm. Valcea, Romania
mihai.arva@nuclear.ro

Due to the increase of the computational power and the available computing resources, over time, several companies have been appeared on the market, capable of offering software analysis, modelling and simulation techniques which facilitate the technological development. They are used in many important fields such as academic, industrial or research and development (R&D). In the nuclear field, these techniques are useful for the development of several industrial equipment. This paper presents the steps required to perform a model of heat transfer for a He atmosphere treatment oven, located at Institute for Nuclear Research, using a simulation and modelling software platform, as well as the results obtained after the simulation of the proposed model. Modelling the heat transfer using the finite element technique (FEM) of the heat treatment oven can be used for both quantitative and behavioural estimations of several parameters: estimation of heat dissipation in the oven structure, evaluation of the yield of the heating element (electrical resistance), evolution over time of the temperature in the treatment area, temperature charts in different areas, etc. The paper presents the novelties in the field of modelling, being relevant to the

scientific field, and also describes through a logical chain, the steps needed to make a detailed model, specific to the engineering field. The obtained results after the simulation of this case study are important for the evaluation of the real thermodynamic behaviour and for the efficient optimization of the heat losses of the oven in the heat treatment process

I.4.9. Temperature Control in the Nuclear Fuel Heat Treatment Installation Using Programmable Logic Controllers

Mihai-Catalin Arva, Cristian Costea, C.G. Serban, Cosmin Ivan, Cornelia Guta
Institute for Nuclear Research, Pitesti, Romania
mihai.arva@nuclear.ro

This paper presents fundamental issues of systems theory and automatic regulation applied in the control-command process of heating installation that is used for the heat treatment within the manufacturing chain of the Low Enriched Uranium (LEU) type nuclear fuel at Institute for Nuclear Research. A process analysis and modelling of the system output response to the temperature regulation parameters were performed and based on the values obtained, the system output response was obtained. Since the temperature control system implemented in the installation was based on the control using classic thermo regulators, in this paper a new method of temperature control using programmable logic controllers (PLC) from Siemens S7-1200 family was proposed. In order to justify the choice of a suitable solution regarding the instrumentation devices, in this paper the aspects of temperature measurement methods using both thermocouple and variable thermo resistors instrumentation devices (RTD-PT100) are highlighted. The paper presents the proposed software solution, the chosen architecture and the human-machine interface (HMI), implemented in the Simatic HMI Touch KTP400 module. A validation using an experimental model of the proposed solution is also presented, thus making a complete evaluation of the control-command system. At the end of the paper, based on the experimental results obtained, a comparative analysis of the oven temperature stabilization graphs is presented between: numerical modelling (theoretical), classical thermo regulator system, and Siemens PID control system.

I.4.10. Influence of Thermal Aging on Different Concrete Classes Used in Nuclear Constructions

Irina Sturzeanu
Institute for Nuclear Research, Campului Street, No.1, Mioveni, Arges, Romania
irina.paraschiv@nuclear.ro

The activities for evaluating the behaviour of concrete structures, under different conditions that can occur both in the normal operation of a nuclear reactor and under conditions that exceed the normal limits, represents a basis for understanding the aging phenomenon and for knowing the effects that it can have on the durability and to ensure the required safety level.
Since certain concrete structures are classified as Category I and have an important role on safety, it is necessary to know how they will respond to different stresses (both under normal and accident conditions) that may occur during the lifetime of the plant.
Exposure to high temperatures is one of the degradation factors of concrete structures and its behaviour is difficult to characterize, because the thermal properties of the concrete are more complex than for other materials, not only because it is a composite material and the properties of its constituents are different, but also because its properties depend on humidity and porosity.
The exposure of concrete to high temperatures affects its mechanical and physical properties; due to the change of properties over time and depending on the conditions in service, it is necessary to evaluate the effects of the degradation factors on concrete and its performances.
This paper aims to evaluate the influence of different temperature levels that may occur in the operation of a CANDU type nuclear power plant, on the properties of the most often concrete

classes used in nuclear constructions. For this, C32/40, C25/30, C18/22.5 concrete samples were subjected to three accelerated thermal aging treatments for ~ 1 year. At the end of the treatment period, it was studied the impact of the temperatures on the mechanical properties that provide informations on concrete durability: compressive strength, permeability, modulus of elasticity and also the density changes that occur in concrete samples.

I.4.11. Simulation Model of Induction Heating for Brazing Instrumented Passages from the Nuclear Fuel Irradiation Device Upper Area, Using BNi-7 Filler Alloy

MEDIA M., CIOBANU N. T., VALECA S. C.
Institute for Nuclear Research, Pitești, România
marius.media@nuclear.ro

The paper presents a study used for vacuum practical brazing experiments of the thermocouple passage from upper plug area of the LEU (low enriched uranium) nuclear fuel irradiation experimental device, using BNi-7 filler alloy. The temperature of brazing for this filler material is above 980 °C. The initial design of the inductor like inductor shape, turns number, positioning in relation to the parts to be heated and the influence of relevant parameters of the induction heating on the temperature distribution in the heated parts were investigated on the base of numerical simulation of induction heating process. The numerical model for study was performed with FLUX2D software which uses the finite elements method for electromagnetic and thermal calculation, for induction heating study of the metallic parts assembly from upper plug area.
The pieces of interest whose surfaces must be joined by inductive brazing are made of Inconel 600 metal alloy. All materials from this study model are paramagnetic materials (μ~1) and have not the phase transformation at Currie point. The assembled parts are fixed inside a vacuum chamber (quartz tube) and the inductor is placed on the outside of the brazing chamber, the advantage being the shortening of the medium frequency power lines from the adapter transformer to the inductor. Also, it wasn’t necessary to make sealed passes of the water-cooled electric pipes of the inductor to the inside of the vacuum brazing chamber. The power source of induction brazing installation at INR Pitesti, is a motor – generator assembly which provides a constant frequency of 10 kHz. The results obtained from the simulation of the heating process were: the values of the current through the inductor coil and the heating parts, the distribution of the density of the magnetic flux, the map of the distribution of the current density, respectively the distribution of the temperature field in the whole of the heating parts. An important aspect of interest is the estimation of temperature difference between measurements made on the outer surface of the metal plug and the temperature values in the joining region of the brazing parts. The measurements are performed on the outer surface of the upper stopper in the physical brazing process. Parameters of induction brazing were designed using numerical simulation of high-frequency induction heating.

I.4.12. Acquisition System for Leak Testing of Fuel Cladding End Cap Welds

A.F. Florea, C. Ivan, I. Pirvu
Institute for Nuclear Research Pitesti, Romania
alexandru.florea@nuclear.ro
This paper describes an acquisition system method for testing the end cap welds of Zircaloy fuel cladding for leak. A graphical interface has been developed using the LabView software application for data acquisition, from a LEU (Low Enrich Uranium) fuel element testing facility, located on RATEN ICN platform. An operator interface for graphical value monitoring it was also designed and tested.
In order to achieve a user interface able to provide the efficient visualization of system useful information (errors, system status, completion of certain actions), the implementation of control functions on the pressure equipment was imposed. It is desired to control the operating mode of the pressurized enclosure, avoiding in this way the occurrence of any operations problems. Lastly, it is desired to acquire and decode the process data, coming from installation sensors and to store them for

online analysis. The HMI (Human Machine Interface) allows the visualization of configuration system values during the real time data acquisition.
The values for the process are read and displayed in real time. The maximum measurement and display error related to analog input lines is below 2% and the response time on each analog channel is 1 second. The system was built around an industrial PLC (Programmable Logic Computer), model Adam 4000 series, and LabVIEW application.

I.4.13. Intergranular Corrosion Testing of 304L Stainless Steel

Sabin Calin, Diana Diniasi
Institute for Nuclear Research Pitesti, Romania
sabin.calin@nuclear.ro

Corrosion resistance of austenitic stainless steel is influenced more by chromium than any other alloying element which may be present. Chromium is more reactive even than iron. High level of reactivity of chromium is, in fact, the main reason for corrosion resistance. The effect of chromium in corrosion resistance of stainless steels, is forming of a protective oxide film, through reaction of chromium with oxygen. In reducing conditions, corrosion resistance is given by this chromium oxide film.
Austenitic stainless steel, though, have a problem during solution annealing in the temperature range of 500-800°C, due to the formation of chromium carbide along grain boundaries, which causes chromium depleted areas in the metal.
In this paper it was evaluated the susceptibility to intergranular corrosion of some sensitized samples of 304L austenitic stainless steel. The samples have been thermal treated at 650°C for 1, 25, 50 and 100 hours, followed by water quench.
For evaluating the degree of sensitization (DOS) of the samples have been applied a destructive acid test and a non-destructive double loop electrochemical reactivation method.
Double loop electrochemical reactivation method provides quantitative data on the sensitization degree to intergranular corrosion.
The microstructure was evaluated using an optical microscope and classified according to ASTM A262 (Oxalic Acid Etch Test for Classification of Etch Structure of Austenitic Stainless Steels).
Evaluation of corrosion rate have been carried out comparatively by visual, microscopic examinations and weight loss of the steel. The corrosion rates for the samples with a sensitized structure were higher than for control sample.

I.4.14. The Non-Destructive EvaluationTechnique for Power Cable Ageing

Dumitru Puiu[1], Bogdan Corbescu[1], Costin Cepisca[2]
[1]Institute for Nuclear Research, Pitesti, Romania, [2]University Politehnica Bucharest, Romania
e-mail: dumitru.puiu@nuclear.ro

The development of cable condition monitoring methods has made a considerable progress over the last few years. It is stiff to be able to compare data obtained from different test laboratories; these may be reacted of tests conditions. Each plant operator may apply his own method and compare the results with his own reference data. However, it would then be necessary for each plant to generate own baseline data. On this line of insulation material degradation assessment for power cables due to ageing, Institute for Nuclear Research started some activities for development indentation technique.
The indenter modulus (IM) is an parameter associated with the specific compressive stiffness of a polymeric cable material. This parameter has been shown to be very sensitive to degradation resulting from thermal ageing and/or irradiation for a variety of materials tested to date (PVC, XLPE, EPR, PE).

Indentation is one of the few non-destructive and mainly non-intrusive cable CM methods currently available that is also widely used. The IM measurement has been incorporated into a recently developed portable indenter. Therefore, this technique can be used for on-site measurements.
Have been sampled 5 cable sections with 4 m long, HXH FE 180 E90; 3x 25 mm type. Samples have been thermal accelerating aged (by Loule Lenz effect); the equivalent operation time in NPP being of 0, 10, 20, 30 and 40 years.
To carry out a measurement, the instrument must clamp around the cable jacket or insulation to be measured, and the probe only penetrates the surface of the test material a few hundred microns.
The portable equipment has as component part a measurement and data acquisition system.. IM increase with insulation material degradation.
The results are useful to identify, model and manage the power cable material ageing phenomenon in the NPP. There is an issue to provide the non-aged initial samples of power cable manufactured many years ago for laboratory tests.

I.4.15. Gamma Scanning Examination of a Low Enriched Uranium Fuel Element Irradiated in the TRIGA SSR Reactor

Ovidiu ICHIM, Octavian UTA, Madalin SAVU, Adrian GANESCU
Institutul Cercetări Nucleare Piteşti – ICN, Mioveni, Romania
ovidiu.ichim@nuclear.ro

The paper aims to validate the gamma scanning technique for calculating the activity of gamma emitting fission products, existing in the low enriched uranium fuel elements, irradiated in the TRIGA SSR nuclear reactor. The first part of the paper presents the measuring equipment used and its characteristics. The second part presents the calibration method of the measuring equipment and the standard sources used, all from the endowment of the Post-Irradiation Examination Laboratory, respectively the way in which the computed tomographic reconstruction is done. The aforementioned steps make it possible to acquire the required data for the axial, radial profiles and computed tomography reconstructions for both the standard source and the studied irradiated fuel elements. Finally, the obtained data are analyzed and the gamma scanning technique is validated as a non-destructive method used for: identification of fission products present in low enriched uranium fuel elements irradiated in the TRIGA SSR nuclear reactor, construction of axial and radial distributions of fission products, section construction by computed tomography reconstruction, determination of the number of nuclei and activity of fission products of interest in irradiated fuel elements.

II. ENVIRONMENTAL PROTECTION

II.1. Radioactive Waste Management

II.1.1. Status of Waste Acceptance Criteria for Predisposal and Disposal in Europe

Chris De Bock[1], Liz Harvey[2], Crina Bucur[3]
[1]ONDRAF/NIRAS, Brussels, Belgium, [2]Galson Sciences Limited, Oakham, Rutland, United Kingdom, [3]Institute for Nuclear Research Pitesti, Romania
C.DeBock@nirond.be

Waste Acceptance Criteria (WAC) represent sets of quantitative or qualitative requirements that may include, for example, restrictions on the dose rate, activity concentration or total activity of particular radionuclides in the waste, or specifications concerning the waste form or packaging of the waste. To be accepted at different facilities throughout the waste management life-cycle (from pre-treatment to final disposal) radioactive wastes have to meet specific WAC. These may be site/facility-specific or generic (i.e. not linked to a particular site or facility).

The term "WAC" is typically used in relation to the reception of waste packages at storage and disposal facilities. Nevertheless, corresponding conditions, requirements or specifications (the terminology in use varies from country to country) also apply for other pre-disposal stages in order to optimise the safe and efficient handling and minimisation of the waste over the whole life-cycle of the radioactive waste. Furthermore, criteria defined for pre-disposal stages are intended to ensure that the waste packages produced meet the WAC for final disposal.

"Waste management routes in Europe from cradle to grave" (ROUTES) is one of thirteen work packages (WPs) being conducted as part of the European Joint Programme on Radioactive Waste Management (EURAD). It aims to provide an opportunity to share experience and knowledge on waste management routes between interested organisations, to identify safety-relevant issues and their R&D needs associated with the waste management routes and to describe and compare the different approaches to characterisation, treatment and conditioning and to long-term waste management routes, and identify opportunities for collaboration between Member-States.

This paper summarises the use of WAC in Member-States and some Associated Countries, focusing on the use of WAC as a management tool across the waste life-cycle. The commonalities and differences between national approaches to develop and apply WAC are discussed. The analysis presented is primarily based on national responses to a questionnaire circulated at the start of the ROUTES work programme. The associated report presents details of national waste acceptance approaches not previously available in the public domain.

With regard to pre-treatment, treatment and conditioning, and storage, most countries have finalised WAC that are in use for one or more facilities and are applicable to various waste classifications, reflecting the widespread nature of such radioactive waste management practices and the value of WAC as a management tool.

IAEA transport regulations, European Council Directives and international agreements concerning the carriage of dangerous goods are referred to as WAC for radioactive waste transport between nuclear licensed sites in most European countries. These are often reflected in national legislation, sometimes, with extensions or additions to reflect the national context.

There is much more variation in the status of WAC for disposal, including considerable variation for different waste classifications within individual countries, largely depending on the status of existing and planned disposal routes for different waste classes.

The project leading to this application has received funding from the European Union's Horizon 2020 research and innovation programme under grant agreement No 847593."

II.1.2. Radioactive waste characterization practices and gaps in Europe

Crina Bucur[1], Denise Ricard[2]
[1]Institute for Nuclear Research Pitesti, Mioveni, Romania, [2]National Agency for Radioactive Waste Management (ANDRA), Chatenay Malabry, France
crina.bucur@nuclear.ro

Safety of interim storage and final disposal of radioactive waste (RW) depends strongly on characterization and quality control of the waste. Unlike for raw waste, characterization of conditioned radioactive waste is more complex and needs specific non-destructive techniques and methodologies since the conditioned RW is typically embedded or surrounded by a matrix, may contain wastes coming from different primary sources and therefore the radiological spectrum might become more complex, and the radionuclides in the conditioned waste may be different by those contained in the row waste (e.g., due to incineration).
This paper presents the results obtained by the survey carried out under CHANCE project related to the currently used characterisation methodologies by European waste producers, disposal operators and waste management organisations, and by coupling these methodologies to the available waste acceptance criteria for the different disposal solutions.
No matter what categories of RW managed, the requirements for characterisation are driven by the waste acceptance criteria (WAC), either for storage or disposal facilities. The following parameters are usually required to be assessed through RW characterisation: specific activities for easy to measure radionuclides (ETM) and difficult to measure (DTM) ones, the fissile nuclides (to ensure sub-criticality in the repository), specific heat power (particularly for geological repositories), complexing and chelating agents, accelerators of leaching processes, organic substances, pyrophoric, flammable, explosive, corrosive or oxidizing materials. In some cases, nuclide vector (NV) compatibility and material vector compatibility has to be ensured and material vector (metallic, ceramics, rubble) has to be declared for some storage/disposal facilities.
Regarding the methods used for characterisation of conditioned RW, nuclide vector and scaling factor (SF) methods are widely used in many European countries for the characterisation of standardized RW streams, and direct information about the origin of waste streams under consideration is also used in many cases. Gamma spectrometry (open-geometry for the whole RW package, or segmented/collimated) is the widely non-destructive techniques used in the characterisation of conditioning RW. Some other countries are using dose rate conversion with approved NVs (so called “dose to Becquerel” methodology) to derive radionuclide activities, but this methodology require good knowledge of the origin and/or history of the waste, which can be problematic for legacy wastes. Spectroscopic techniques (alpha, beta and gamma spectrometry) are applied in the majority of the institutions involved in RW characterisation either for inspection or in the process of SF development. More advanced methods such as active and passive neutron measurements, calorimetry and X-ray inspection are used in some specific cases to identify the fertile and fissile materials, or to analyse the matrix and its density distribution.
As specific problematic issues for the characterisation of conditioned RW the proper characterization of the conditioned legacy waste packages is considered a general problem, as well as the determination of a viable source term in already conditioned RW and detection of declarable DTM isotopes (i.e. Sr-90) and sealed radioactive sources.
These results were obtained under CHANCE project, that has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 755371.

II.1.3. Study of Technetium Sorption on Pure Clay Mineral

Mirela OLTEANU, Crina BUCUR, Camelia ICHIM
Institute for Nuclear Research, P.O. Box 78, Pitesti, Romania
mirela.olteanu@nuclear.ro

Tc-99 is an important radionuclide for the long term environmental safety assessment of a disposal facility because it has a long half-life, a high mobility in TcO_4^- form, and is present in significant quantities in the radioactive waste. The redox/chemical conditions in the near field of the disposal facilities have a major influence on the chemical speciation of Tc, on its solubility, and its behaviour in the migration/retention processes.
The sorption behaviour of Tc in alkaline conditions, with and without addition of a reducing reagent, was studied in the "batch" sorption tests on pure clay mineral. Montmorillonite was investigated as pure clay mineral as it is the main component of the bentonite used as backfill and buffer in a geological repository. The batch sorption tests were carried out with a mineral: solution ratio of 50 g/l, alkaline pH (with pH adjusted with NaOH 0.1M solution), and Re as stabile chemical element with similar properties with Tc. The aliquots of the clay mineral were pre-equilibrated with synthetic montmorillonite pore water for 24 hours and after that a known volume of Re spiked solution was added. Another set of tests was prepared in similar way with addition of 0.1g of reducing reagent (Na2S2O4) that has the role to obtain and maintain, for a limited period of time, a reducing media in the mineral: solution systems. The concentration of Re in liquid fractions sampled at the specific periods of time was analysed by inductively coupled plasma emission spectrometry (ICP-OES). Two types of sorption tests were carried out: a kinetic sorption test performed to determine the time needed to reach the sorption equilibrium, and tests for deriving the sorption isotherms that were carried out with and without addition of reducing reagent. The sorption isotherms were obtained for Re initial concentration in contacting solution between 2.5 – 50mg/l, for tests performed without reducing reagent addition and between 0.5 – 15 mg/l, for tests with addition of reducing reagent.
Sorption kinetic test indicated that more than 30 days are needed to achieve the sorption equilibrium and a slightly increase of Re sorption with pH was observed: from 6 % for pH=9, to ~10% for pH=12. The addition of sodium dithionite increases the Re sorption at ~18%, with a decrease of redox potential value (Eh< 0 mV). The experimental data were fitted with linear sorption isotherm, both for the tests with addition of reducing reagent and for the tests without addition. Without reducing reagent in the test samples the distribution coefficient determined from the slope of the linear sorption isotherms is 1.6 cm3/g dry mineral, and the addition of reducing reagent increases the Re distribution coefficient to 4.25 cm3/g dry mineral.
This study was performed under FUTURE work package of EURAD project, funded by the European Commission within the Horizon 2020 Framework Programme under Grant Agreement no. 847593.

II.1.4. First Outcomes of the Uncertainty Management multi-Actor Network (UMAN)

Frank Lemy[1], Valéry Detilleux[1], Dirk-Alexander Becker[2], Daniela Diaconu[3], Astrid Göbel[4], Agniezka Strusińska-Correia[4], Julien Dewoghélaëre[5]
[1]Bel V, Brussels, Belgium, [2]GRS, Braunschweig, Germany, [3]RATEN, Pitesti, Romania, [4]BGE, Peine, Germany, [5] UTADIS, Paris, France
Corresponding author's e-mail: frank.lemy@belv.be

Programmes for the disposal of radioactive wastes are implemented in a series of phases, in which the scientific understanding of the disposal system, the design of the disposal facility and its safety case are progressively advanced. Such an approach provides a basis for decisions relating to the development, operation and closure of the facility, which need to be made in the presence of both irreducible and reducible uncertainties. It also allows a stepwise management of safety-relevant uncertainties inherent in the different phases of the programme and in the long-term behaviour of the disposal system.

The Council Directive 2011/70/EURATOM requires that transparency be provided by ensuring effective public information and opportunities for all stakeholders concerned to participate in the decision-making process. Therefore, choices made on the basis of limited information in early programme phases have to be duly justified and communicated. Several of these choices may then have to be confirmed during subsequent stages of the process (e.g. during site selection, construction or operation of the facility). Activities associated with the programme such as site characterisation, process modelling, safety assessment, etc. are also evolving, leading to new viewpoints and sometimes new uncertainties. At the end of the process, it should be demonstrated that remaining uncertainties do not undermine safety arguments. Hence, the management of uncertainties and associated stakeholder involvement represent key parts of successful programme planning as well as key issues when developing or reviewing the safety case of disposal facilities.

The main objectives of the Uncertainty Management multi-Actor Network (UMAN) are to provide an opportunity to different actors of Member States to share their experience and views on uncertainty management and to identify emerging needs associated with this topic that could be addressed in subsequent waves of the European Joint Programme on Radioactive Waste Management (EURAD). UMAN contributes also to the vision of EURAD by fostering mutual understanding and trust between different categories of Programme participants and actors (i.e. WMOs, TSOs, Research Entities and Civil Society Organisations).

UMAN includes the review of existing uncertainty management strategies, approaches and tools. Furthermore, existing knowledge and views on the identification, classification, characterisation and significance of uncertainties associated with specific topics are synthesized. A particular focus is put on the following topics: waste inventory and impact of predisposal steps, site and geosphere, human aspects, spent fuel and the near-field of geological disposal facilities. Then, possible options to manage these uncertainties are identified and discussed. Interactions between different types of actors including civil society and the understanding of their views are central to UMAN.

The different tasks and the process followed in UMAN to reach its different goals are described. The results achieved during the first two years of the programme are then used to illustrate the first outcomes of this process. Finally, some of the activities foreseen in the coming years in the framework of this initiative are presented.

II.1.5. Dealing with Low Level Waste When Planned Permanent Disposal Plans Are Delayed

John Saroudis
CNPSA Romania
John.saroudis@cnpsa.ro

More than 15 years ago Ontario Power Generation identified the need for a permanent low-level waste disposal facility to handle the waste generated by its 10 operating nuclear units. The solution, a low-level waste disposal facility located near the Bruce reactor site which would have handled all the waste generated by the operating units. In 2019 when the facility construction was to start the local community decided to no longer endorse the project and per previous agreements OPG agreed to no longer pursue the project at that location.

Finding alternative locations is a long-term undertaking. To responsibly manage its waste OPG reverted to a plan whereby waste quantities would be significantly reduced in order to maximise the use of existing waste storage facilities and allow time to develop a new permanent location.

This presentation describes the Hamilton Waste Sorting Research facility operated in collaboration with McMaster University in Hamilton, Ontario and providing valuable experience in the sorting, storage and disposal of low level nuclear power plant waste to significantly reduce its volume and maximize existing available facilities.

II.1.6. A Special Case for the Evaluation of Shielding Requirements for the Management of DSRS during Decommissioning of RADON Type Maišiagala Disposal Facility in Lithuania

Ernestas NARKŪNAS, Povilas POŠKAS, Audrius ŠIMONIS, Artūras ŠMAIŽYS
Lithuanian Energy Institute, Kaunas, Lithuania
ernestas.narkunas@lei.lt

Maišiagala radioactive waste disposal facility was built according to the typical project TP-4891 of the former Soviet Union and construction was completed in 1963. This facility is a RADON type disposal facility and is situated in the south-eastern part of the Republic of Lithuania, ~30 km north-west from capital Vilnius and ~7 km north-west from Maišiagala town. It was used for the disposal of institutional radioactive waste and disused sealed radioactive sources (DSRS) coming from various research and industry (except NPPs) applications. In 1989, the facility was closed and sealed. Since then, several upgrades of the facility have been performed, but it has been finally stated that the facility does not meet the criteria established for near surface disposal facilities and must be decommissioned.

During decommissioning, special attention should be paid to the management (retrieval, packaging and transportation) of two stainless steel tanks, which are located at the bottom of the disposal vault and were used for placement of DSRS. It should be noted that during the facility operation, the disposal of DSRS was not performed in a systematic way, i.e. typically, the disposed DSRS (usually with their biological shielding) were mixed with other radioactive waste; however, a certain amount of the DSRS (without biological shielding) was placed into the abovementioned special stainless steel tanks (10 l and 15 l capacity). These two tanks filled with DSRS will pose a higher radiological risk and will require special consideration during the decommissioning of Maišiagala disposal facility. It is foreseen that each of the tanks will be handled separately, i.e. each tank with DSRS will be retrieved intact, placed into an appropriate package and transported to the Ignalina NPP for further management.

Transportation rules require that the dose rate on the surface of a radioactive waste package has to be below 2 mSv/h and this value was set as a basis. Initially, the required shield thickness to achieve the set basis was evaluated assuming homogeneous source distribution within each tank and the results were used for workers dose assessment. However, the actual heterogeneous nature of the DSRS distribution inside the tanks could lead to the situation when particular DSRS with higher activity is next to the tank wall (thus increasing the dose rate) and this situation was also evaluated.

Therefore, this study is dedicated to the necessary shielding evaluation of two stainless steel tanks with DSRS in order to comply with radiation protection requirements for the handling and transportation of these tanks during the decommissioning of Maišiagala disposal facility.

II.1.7. Efficiency Calibration Methods of GeHP Detector for Gamma Ray Measurements from Solid Radioactive Waste

Cristina DIACONESCU, Maria MIHALACHE, Ionut FLOREA, Camelia ICHIM
Institute for Nuclear Research, P.O. Box 78, Pitesti, Romania
cristina.diaconescu@nuclear.ro

Gamma ray spectrometry is a non-destructive method applied in quantification and identification of the gamma emitting isotopes from the radioactive waste. One of the main challenges in quantitative gamma-ray spectrometry is to establish the detection efficiency for different energies, source-detector geometries, and composition of the samples or sources. In principle, there are three approaches to calculate the efficiency: experimental, numerical, and semi empirical.

The purpose of this paper is to obtain the detection efficiencies of GeHP detector used for measurement of gamma emitting radionuclides from solid radioactive samples using the three approaches previously mentioned.

Semi empirical approach is based on the efficiency measured for a simple geometry and source, which is adjusted using correction factors calculated from the attenuation coefficients that in turn depend on the sample density and composition.
The MCNPX code based on Monte Carlo method was used in the evaluation of detection efficiency in measurements of gamma rays emitted from solid samples in order to test the applicability of MCNPX code in gamma spectrometry.
The initial calibration was done with standard liquid source (100ml radioactive solution with known activities of gamma emitting radionuclides, in 250 ml plastic vial), at contact with the GeHP detector. The detection efficiencies determined by this simple source and geometry represent the base values for semi empirical and numerical simulation approaches.
Three solid simulated radioactive samples, one of sand and two of soil, with known radionuclide concentrations were used to calibrate the GeHP detector and to determine the detection efficiency.
To obtain accurate results it is necessary to have chemical composition of the samples and detector, but also the geometry detector - sample. The mass of each chemical element in the analysed samples were determined by SEM/EDS, and these data were used both in numerical simulation and in the semi empirical method applied to evaluate the detection efficiency. For the modeling approach the detector characteristics given by the manufacturer were used. The attenuation coefficients were established using XCOM (NIST Database) program.
The efficiency values obtained by the three approaches were compared and successfully applied in the activity measurements of radioactive solid samples with relative errors less than 15%.
The results obtained by testing and applying these three approaches were used in optimizing the gamma ray measurements.

II.2. Radioprotection & Air, Water and Soil Protection

II.2.1. Advances in Radioactive Decontamination via STRIPCOAT-HMR Research Project

D. PULPEA, G. TOADER, T. ROTARIU, A. MOLDOVAN, G.B. PULPEA, A. PODARU
Military Technical Academy`Ferdinand I`, Bucharest, Romania
daniela.pulpea@mta.ro

During the last ten years, our research team conducted multiple studies regarding the possibility of removing radioactive contaminants from different types of surfaces with the aid of strippable coatings. The main purpose of these coatings is to entrap hazardous materials inside their polymeric matrix. To avoid subsequent contamination, the coatings used for decontamination should be biodegradable. Consequently, our research work focused on replacing the conventional materials used for decontamination with a new type of polymeric coatings consisting of environmentally friendly materials. These ideas resulted from numerous extensive studies conjoined in multiple research papers, master's degree thesis and PhD thesis, all of them being part of three national research projects, including our ongoing research project StripCoat-HMR. This paper aims to highlight the research activities performed by our research team from Military Technical Academy „Ferdinand I" (ATMFI) in the field of decontamination of radioactive materials. The above mentioned ongoing research project, entitled „Strippable coatings for heavy metals and radionuclids decontamination (StripCoat-HMR)" presents an alternative to the existing commercial decontamination solutions bringing forward an innovative method of decontamination which implies the use of polymeric solutions that are non-toxic (can be prepared and used without risk) and biodegradable (does not generate additional waste after decontamination) and they can be easily prepared (by simply mixing the components), using inexpensive raw materials. This method of decontamination is efficient (ensures a high degree of decontamination) and fast (in a maximum of 24 hours the films can be easily detached, thus obtaining a clean surface).

II.2.2. Continuous Monitoring of Radon Activity Concentration for Workplaces

Ruxandra Cristina Săpoi, Margareta Cheresteș, Alexandra Nacu, Costinel Toma
Dositracker
ruxandra.sapoi@dositracker.com

Since the publication of the Council Directive 2013/59/Euratom and implementation of its requirements in the Romanian legislation in 2018, a series of recommendations have been published regarding the radon exposure in workplaces, including Radiation Protection 188 – Technical Recommendations for Monitoring Individuals for Occupational Intakes of Radionuclides (European Commission, 2018) and Radiation Protection 193 – Radon in Workplaces (European Commission, 2020).
In this paper the authors would like to emphasize the importance of continuous monitoring of radon activity concentration for workplaces. In order to better estimate the workers' exposure to radon, the concentration of radon activity during the occupancy times should be taken into account.
The results of the pro-bono studies for continuous radon activity concentration measurements at various workplaces in public and private institutes in Romania are also presented in this paper.

II.2.3 Study of the Self-Absorption Effect in Gross Alpha and Beta Activity Measurement by using GFPC Method

Valentina Neculae, Relu Dobrin, Cristian Dulama
Institute for Nuclear Research, Pitesti, Romania
valentina.neculae@nuclear.ro

Natural radioactivity is always present in the environment. Water, especially ground water, is not free of radioactive isotopes. Natural radionuclides, including potassium-40, and those belonging to

the thorium and uranium decay series, can be found in water as a result of either natural or technological processes involving naturally occurring radioactivity. Also, traces of natural origin radionuclides are normally found in all drinking water. Their concentrations vary from place to place because the radionuclides are released from rocks and minerals which form the aquifer through the erosion and dissolution process. Considering this, determination of the radioactivity of drinking water becomes one of the important parts of environmental studies. The process of identifying individual radionuclides in drinking water and determining their concentration is time-consuming and expensive. Thereby, it is more practical to use a screening technique, where the total radioactivity present in the form of alpha and beta emitters is firstly determined, without regard to the identity of specific radionuclides. Even if the gross alpha and beta measurement is one of the simplest radioanalytical procedures, this method has a major drawback given by the self-absorption of alpha and beta particles in the sample due to the thickness of the salts deposit.
The aim of this work was experimental determination of the self-absorption corrections for gross alpha and beta counting using a low background proportional counting system, in case of increased salt content in water samples. On that purpose a series of calibration sources with different thickness were prepared using known amount of radioactive solution containing the reference radionuclides, Am-241 for alpha measurements and Sr/Y-90 for beta measurements. The equipment used for this research work was a low background gross alpha and beta counter, model MPC 9300. The self-absorption curve was generated by determining the counting efficiency as a function of mass density of the source (in grams per square centimetre). The obtained results were checked and validated within the proficiency tests in which the Radiation Protection Laboratory has participated, this being a good way for assessing the analytical methods performances of a laboratory.

II.2.4. Optimization of soil and sediment sample preparation methods used for natural uranium content determination

Daniela Stanciu, Monica Valeca, Gina Zăvoianu
Institute of Nuclear Research Pitesti, Romania
elenadanielastanciu@yahoo.ro

Naturally occurring radionuclides represent a very important class of environmental contaminants. Uranium concentration levels in environmental samples are of great importance because uranium presents both a chemical and a radiological hazard to the environment.
The purpose of this paper is to develop methods for the most efficient digestion of organic compounds from soil and sediment samples, in order to determine the concentration of natural uranium by spectrophotometric method. The digestion techniques used in this paper are conventional digestion with strong acid or mixtures of strong acids with a strong oxidant ($HClO_4$, HF, HNO_3, with H_2O_2) and acid digestion in the microwave field using the microwave digestion system “Speedwave XPERT” from Berghof Products+ Instruments GmbH”.
In order to determine the optimal digestion method, several experimental tests were performed. An efficient digestion method is the one that ensures the highest possible recovery for the desired element.
Recovery experiments using a real environmental sample with a known Uranium concentration were performed. In this sample, it was added a certain Uranium amount from a certified reference material. Both samples (the real one and the one with the uranium addition) were analyzed and the recovery yield was calculated. Repeatability experiments involving repeated determinations on the same sample, under the same laboratory conditions and using the same method, were also performed and the uncertainty (standard deviation) was calculated.
The results showed that, in the case of the microwave digestion, a higher recovery yield was obtained as well, as a better repeatability, than in the case of conventional digestion.
At the same time, the microwave method allows a simultaneous digestion of more organic samples, it is faster and requires a smaller volume of acids. All these aspects demonstrate that microwave digestion is an efficient method of mineralization of environmental samples in order to determine the concentration of natural uranium.

II.2.5. Determination of Environmental Radioactivity in Manzala Lake Sediment Samples by Gamma-Ray Spectrometry

M. Mitwalli[1,2], C. Dulama[3], A. H. El-Farrash[1], D. Chirleşan[4], M. Sallah[1,2]
[1]Physics Department, Faculty of Science, Mansoura University, Mansoura, Egypt, [2]National Network of Nuclear Sciences, Academy of Scientific Research and Technology, NNNS-ASRT, Cairo, Egypt, [3]Institute of Nuclear Research RATEN-ICN, Piteşti, Romania, [4]Faculty of Science, Physical Education and Informatics, University of Piteşti,
110040 - Piteşti, Romania
meto_mms@yahoo.com

The present study aims to estimate the environmental radioactivity hazard for Manzala lake Egypt. Twenty sediment samples were collected from pre-determined locations, which involve deep springs and lakes. The radiometric measurements were done using Hyper Pure Germanium spectroscopy (HPGe). The gamma-emitting nuclides library has been designed to assess the activity concentration for radionuclides belonging to natural series of uranium and thorium, as Thallium-208, Bismuth-212, Lead-212, Bismuth-214, Lead-214, Radium-226, Actinium-228, Thorium-234, and Uranium-235. The measured overall average value of the mentioned radionuclides' concentration was 15 Bq/kg, while for Potassium-40 was 286 Bq/kg. The true coincidence summing corrections have been applied by using the Genie-2000 software, and LabSOCS calculated total efficiencies. Most of the resulted values are moderated that indicates normal levels in comparison with the international and worldwide reference values for similar environments. Some locations have significant radioactivity due to the excessive load of numerous polluted wastewaters, including sewage water and chemical fertilizers used in the surrounding farmland, which may lead to an accumulation of naturally occurring radioactive material. However, factories may dispose their industrial waste in the lake through many drains that pours into the lake, which may alter the composition of soil and rocks flooded by the waters in that region. Therefore, the present work can be used as reference study to detect any harmful radiation that would affect the humans and assess the changes in the radioactive background in the investigated area.

II.2.6. Modeling the Distribution of Radioactive Material Released from the APR1400 Reactor in the Environment, by using Hysplit and ADMS

Mohammad Rahgoshay[1], Jalil Jafari[2], Farooq Ghaderinia[3]
[1]Department of Nuclear Engineering, Faculty of Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran, [2]Reactor and Nuclear Safety School, Nuclear Science and Technology Research Institute (NSTRI), [3]Department of Nuclear Engineering, Faculty of Engineering, Science and Research,Branch, Islamic Azad University, Tehran, Iran
m.rahgoshay@gmail.com

The APR 1400 is an Advanced PWR which has an electrical power of 1400 MW. The latest power plant equipped with this type of reactor is Barakeh power plant, which is located in the United Arab Emirates. This power plant has 4 units of APR1400 Reactor. In this paper, sever accident, with using MELCOR code, has been modelled. The desired accident scenario is SBO (Station Blackout). Eventually, with the destruction of the containment building due to the accident, radioactive materials leak into the environment. The amount and type of radioactive material leaked to the environment is calculated by the MELCOR code. Then results of the MELCOR code are used in HYSPLIT and ADMS software to model the dispersion of radioactive materials. The absorbed dose for different intervals, within 24 hours from the beginning of the release of radioactive materials in the environment has been calculated by HYSPLIT software. But ADMS software, the absorbed dose is calculated only up to a radius of 50 km, because this code can only perform high-precision modelling up to a radius of 50 km. Also, the route of distribution of radioactive materials up to 10 days after the start of release of radioactive materials in the environment has been calculated by HYSPLIT software.

III. SUSTAINABLE DEVELOPMENT

III.1. Education, Training and Knowledge Transfer

III.1.1. Experimental Study of the Fatigue Cracks Initiation on the Zr-2.5%Nb Pressure Tube Specimens

Ramona POPESCU[1], Serban Constantin VALECA[1], Viorel IONESCU[2]
*[1]UPIT Pitesti, Romania, [2]RATEN ICN Pitesti, Romania**
viorel.ionescu@nuclear.ro

This paper presents the results of the mechanical fatigue tests under various load spectra performed in RATEN ICN on the pressure tube specimens made from Zr-2.5%Nb. The mechanical fatigue tests consist of two cycling sequences having different mechanical loading amplitudes. The tests are performed on C-shape specimens with V-notch tip hydrides, prepared from un-irradiated pressure tube. The type V-notch simulates defects due to the interaction of the inside of pressure tube with the impurities of the thermal agent. The general test procedure includes two steps: hydride formation and fatigue tests. Hydride formation involves the hydrogen migration under stress gradient, hydrides reorientation and precipitation in the region of the stress spots. This operation is performed by thermal cycling the specimens under a constant load to achieve the accumulation and reorientation of the hydrides at notch-tip. The fatigue test is performed on individual specimen in the same tensile testing machine. During the test, the fracture stress on the notch-tip hydrides appear. When the fatigue crack initiation, the test is stoped and the number of cycles for every sequent is saved. Are defined the *"cycling sequences fraction"* as the ratio of cycles number for every sequent and the *"life time prediction function"* expressed in the number of cycles until the fatigue crack initiation. An analytical relationship that correlates the cycling sequence fraction with the mechanical stress amplitude applied for every cycling sequence and the hydrogen concentration is obtained. The results could be used for the evaluation of the life time and for the structural integrity analyses of pressure tubes in the CANDU fuel channels subject to periodic inspections.

III.1.2. Electrochemical Methods for Characterization of Oxides Grown on Austenitic Stainless Steel Components from CANDU Circuits

George - Sebastian Ciobanu[1], Dumitru Chirleșan[1], Aurelia Elena Tudose[2]
[1]University of Pitesti, Romania, [2]Institute for Nuclear Research Pitesti, Romania
sebastianciobanu12@yahoo.com

Aqueous media at high temperature and pressure, involves development of oxide films on the surface of the structural materials (carbon steels, stainless steels, zirconium and nickel alloys) from the primary circuit and the secondary circuit of heat transport. Corrosion resistance of alloys in high temperature water is related closely to the oxide films developed on their surfaces. It is essential to perform characterization of the oxide layers developed in water at high temperature, because the chemical composition and the oxide layer structure play an important role in understanding of corrosion mechanism. In this paper, the corrosion tests have been carried out on 304 L stainless steel samples by exposure in simulated conditions of the primary circuit of CANDU reactor (demineralized water solution with LiOH, pH = 10.5, at a temperature of 310ºC and a pressure of 9.8 MPa) and the secondary circuit of CANDU reactor (demineralized water solution with volatile amines, pH = 9.5 ÷ 9.7, at a temperature of 265ºC and a pressure of 5.1 MPa). After oxidation, the films were characterized by electrochemical impedance spectroscopy and Mott - Schottky methods. The results obtained by these methods are comparable and indicated that on the surface of the 304 L SS samples oxidized in both aqueous media, a duplex oxide layer was formed. The inner layer is more protective and the outer layer is more porous. Also, the effect of the working environments temperature was observed both from the values of the oxide layer thicknesses and

from the values of the polarization resistances which indicated lower corrosion rates for the oxidized samples in the specific solution of the primary circuit.

III.1.3. The Uranium Recovery from Secondary Products Resulting from the Fabrication of Experimental Fuel Elements

Valentina Roxana Dima[1], Monica Valeca[1], Mariana Postelnicu[2]
[1]University from Piteşti, [2]RATEN INR Piteşti România
mariana.postelnicu@nuclear.ro

As any industry, nuclear industry generates a diverse range of secondary products and radioactive wastes. The principle of recycling and reuse of secondary products is an important element in any waste minimization programme. A main objective of the TRIGA 14MW research reactor is the testing of experimental fuel elements and nuclear materials.
The execution of experimental fuel elements has as intermediate stage the manufacture of fuel rods. There is a stage of mechanical processing in the technological flow of fuel rods fabrication, from which are obtained secondary products, in the form of cuttings and slurry, which contain low enriched uranium.
The paper proposes a study on the recovery of the uranium from these secondary products, using the method of chemical processing, which involves several steps, finally obtaining a UO_2 powder of nuclear purity.
In the first step of the chemical processing, the cuttings are calcinated, resulting a mixture of uranium, zirconium and erbium oxides. This oxides mixture is dissolved in a 67% nitric acid solution. Only uranium oxide dissolves in nitric acid, the other oxides being insoluble in this acid. The next step is to purify the uranyl nitrate solution with an organic solvent. The pure solution of uranyl nitrate followed the ADU (Ammonium Diuranate) route to obtain the uranium in the form of UO_2 powder.
The results obtained after applying the method described above, showed a recovery yield of uranium, from the secondary products, of approximately 70%.
The obtained UO_2 powder can be used in the advanced experimental CANDU fuel fabrication and due to the high price of low enriched uranium, its recovery has an economic importance, too.

III.1.4. Diametral Compressive Testing Method by UO_2 Pellets

Nicoleta LIHOR[1], Monica VALECA[1], Vasile PITIGOI[2]
*[1]UPIT Pitesti, Romania; [2]RATEN ICN Pitesti, Romania**
vasile.pitigoi@nuclear.ro

In the study of brittle materials such as graphite, marble, stones and ceramic materials in general, conventional tensile tests have difficulties in preparing samples, testing devices and performing tests. That is why we opted for a simpler testing method, namely diametral testing method by UO_2 pellets.
For this, a new, convenient testing method has been proposed and used, for which the samples are easy to prepare and test.
To this purpose, circle punches were made with various ratios between the punch radius and the UO_2 pellet radius with values of 1,2, 1,5, 2 and ∞.
Diametral compressive tests were performed, after which the contact half-width between the punch and the pellet was measured by using a pressure-sensitive paper strip.
For the calculation of the tensile strength were used the test results for which the ratio $b / R \geq 0.27$, where the shear stress near the contact surface becomes very high, so the fracture occurs in the center.
In our case the values obtained by using the punch at which the ratio is $b / R = 1,2$.
The statistical analysis of the results concerning ceramic materials was performed by applying the appropriate Weibull Theory for these materials.

This testing method for determining the tensile strength by diametral compressive tests is a novelty in the field of testing ceramic materials, for which the performance of test samples is a problem difficult to solve.
The data obtained for the determined mechanical parameters are used in the design of fuel elements or in projects related to the use of ceramic components.

III.1.5. Investigation of the CANDU Steam Generator Tubing Material's Corrosion Behavior

Straut Ion – Adrian[1], Valeca Serban Constantin[1,2,3,4], Tudose Aurelia[1], Petrescu Daniel[1], Lucan Dumitra[2,3]
[1]University of Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research, Mioveni, Romania, [3]Technical Sciences Academy of Romania, Bucharest, Romania, [4]Academy of Romanian Scientists, Bucharest, Romania
dumitra.lucan@nuclear.ro

The paper presents an overview of experimental results of the research activities developed in INR's Nuclear Materials and Corrosion Department in the field of Steam Generator ageing management.
The impurities and corrosion products existing in the steam generator concentrate in the tube - tubesheet crevices. The chemical reactions that take place between the components of the concentrated solutions and the crevices materials bring about an aggressive environment. The presence of this environment and of tube - tubesheet crevices lead to localized corrosion and thus the same tubes cannot assure the heat transfer between the fluids from the primary circuit to the secondary one. The presence of the deposits into crevices accelerates the concentration process of the impurities. Crevice corrosion is one of the predominant forms of localized corrosion limiting the operation life of CANDU steam generator. Crevice corrosion involves a number of simultaneous and interacting operations, including mass transfer processes, production of metal ions within the crevice and hydrolysis reactions, resulting in a very aggressive solution from the point of view of corrosion. These intermediary corrosion processes are in a complex interdependence and they imply a number of important parameters, including both the crevice gap and depth.
The major goal of this paper was an experimental program focused on: the impurities concentration and deposits consolidation processes inside the steam generator tube – tubesheet crevices and the influence of these deposits on the CANDU steam generator's principal structural materials. The paper presents the results of optical and scanning electron microscopy examinations and the results of electrochemical measurements. The experimental results allowed us to establish the contribution of the impurities concentration and deposition on the evolution of the corrosion process and to the prediction of the effects of corrosion on the ageing process of the steam generator and the behavior of this key equipment under long term operation conditions.

III.1.6. Assessment of Advanced CANDU Nuclear Fuel Element Irradiated in TRIGA Research Reactor

A.A. BICA[1], S. VALECA[1], A.C. RADUT[2]
[1]University of Pitesti, Romania, [2]RATEN Institute for Nuclear Research, Pitesti, Romania
bicaanaalexandra@yahoo.com

Given the natural abundance of thorium resources (three times larger than those of natural uranium) one of the options used in advanced nuclear fuel cycles is the utilization of mixed oxide of thorium and uranium fuel. Numerous studies indicates that some of the $(Th,U)O_2$ properties, in comparison with those of UO_2, may contribute to the improvement of different fuel performance parameters, as follows: higher thermal conductivity, higher modulus of elasticity, higher melting temperature, higher resistance to corrosion when the fuel is exposed to reactor coolant, and also release of reduced amounts of fission gas during fuel irradiation.

The aim of this study is to evaluate, by means of ELESIM - TORIU computer code, the irradiation behaviour of an experimental fuel element, identified as A23, irradiated into the C1 capsule of TRIGA Research Reactor, owned and operated by RATEN ICN Pitesti, with a ramp power history. The experimental fuel element has reduced length and contains thorium dioxide fuel pellets with 5% UO_2 (90 % enriched in ^{235}U) and 9.7 g/cm^3 density.
The temperature-sensitive performance parameters taken into account in the current study are: fission gas release, sheath deformations and internal pressure. Also, the results provided by ELESIM - TORIU calculations were compared with the data obtained by non-destructive and destructive examination performed in the Hot Cells Nuclear Laboratories of RATEN ICN Pitesti.

III.1.7. Strontium Separation by Chromatographic Extraction Process for ^{90}Sr Assessment

M. ILINA[1], M. DIANU[2]
[1]University of Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
magdalena.dianu@nuclear.ro

The ^{90}Sr radioactive isotope is not a naturally occurring radionuclide and it is generated by human activities. Due to the behaviour of ^{90}Sr in various media (atmosphere, soil, water, plants, human body) it possesses a potential hazard to the environment and people if the waste containing this radionuclide is not properly disposed of.
The ^{90}Sr radioisotope is a pure β emitter (E_{max}= 546 keV) with a half-life of about 29 years. The most sensitive and widely used technique for ^{90}Sr quantification is liquid scintillation counting (LSC), which relies on the efficient radiochemical separation of the interest nuclide from other beta-emitting radionuclides.
The radio-chemical separation technique of ^{90}Sr is based on development and testing of the analytical methods applicable to natural isotopes in the form of Sr^{2+} species.
Chemical separation of strontium was developed by using the Eichrom® Sr Resin, which is mainly used for Sr^{2+} separation on the base an extraction system that presents selectivity for strontium. The extractant is a crown-ether [4,4(5)-di-t-butylcyclohexano-18-crown-6], whose macro-cyclical structure delimits intramolecular cavities in which Sr^{2+} ions are retained, forming complexes with a high stability constant. The strontium affinity for the resin increases with the nitric acid concentration, reaching a maximum value for a concentration of 8M HNO_3. Therefore, high decontamination factors of interfering elements are achieved by loading the sample in 8M HNO_3. Two methods for strontium separation were carried out: (a) a single chromatographic extraction process, and (b) double chromatographic extraction.
The performance parameters of separation process were assessed by inductively coupled plasma – optical emission spectrometric (ICP-OES) method. The results achieved by performing the two strontium separation methods, with the aim of Sr^{2+} concentration quantification in aqueous samples, reveal that double chromatographic extraction process represents an interferences free strontium separation method, obtaining very good decontamination efficiencies values. Therefore, this selective and quantitative separation method can be performed prior to ^{90}Sr activity assessment. The average chemical recovery yield, for each of the performed separation procedure, exceeded 95 %; therefore, these methods can be easily applied to samples with small activities of ^{90}Sr.

III.1.8. Quality Monitoring of TRIGA Reactor Primary Coolant by Analytical Control of Iron and Silica Content

Banu Cristina Elena[1], Enache Nicoleta Gabriela[2], Serban Constantin Valeca[1]
[1]University of Pitesti, Romania, [2]Institute for Nuclear Research Pitesti, Romania
enache.gabriela@nuclear.ro

The primary coolant of TRIGA Research Reactor is high quality demineralized water, which in addition is an auxiliary moderator and a biological shield.

The components of the cooling system are made of stainless steel and aluminium. These components are constantly exposed to water. This constant exposure makes the system potentially vulnerable to degradation processes including corrosion. Corrosion modes of attack are strongly dependent on the quality of the cooling water.
Demineralized water quality is maintained by the means of permanent purification. Purification is performed by circulating the water through an installation with two filtration stages: mechanic and ionic filtration. The purification system maintains low conductivity of the water to minimize corrosion of all reactor components, particularly the fuel elements, reduces radioactivity in the water by removing nearly all-particulate and soluble impurities, and maintains the optical clarity of the water.
Because some components of the primary cooling system are made of stainless steel, analytical determination of total iron content of demineralized water is crucial for system corrosion assessment, the increase in iron content leading to a higher corrosion of components and growthing of activation level by activation of corrosion products.
Demineralized water silica content is measured because it is crucial for determination of ionic exchange resin exhaustion (silica is the first anion who is released by anionic resin).
Quality monitoring of primary coolant by analytical control of iron and silica content is conducted on weekly basis by sampling water from reactor tank, and inlet and outlet of ionic filter.
The paper’s main objective is to present the evolution of the iron and silica content of the TRIGA reactor primary coolant and includes iron and silica analysis results. Based on the steady good results of iron analysis (low concentration of iron in coolant), we may conclude that corrosion of primary reactor system is minimal and by analysing of silica content evolution, we can know when ionic exchange resin is exhausted.

III.1.9. Control and Limitation of Radioactive Emissions in the Environment for a Nuclear Installation

Andreea IONITA, Monica VALECA, Roxana IORDACHE
UPIT Pitești, România; RATEN ICN Pitești, România
andreea.ionita98@yahoo.com

By limiting the release of radioactive effluents into the environment tone may ensure the protection of the health of the population and of the environment.
When during the normal operation of a nuclear installation, radioactive effluents, in liquid or gaseous form, are discharged into the environment, derived release limits, in terms of annual emissions of these effluents must be established to ensure compliance with the annual effective dose constraints.
The annual effective dose constraints refer to the sum of the annual dose due to external exposure, and the committed dose due to intake of radionuclides of a representative person from public taking into consideration all relevant exposure pathways associated with the releases into the environment of radioactive effluents from a nuclear installation.
First step in establishing the derived release limits is to evaluate the annual doses corresponding to the discharges for each radionuclide emitted, on each release pathway.
There are several pathways that human body may suffer irradiation from gaseous or liquid radioactive effluents discharged into the environment.
All of these pathways contribute to the effective equivalent dose due to incorporation (inhalation and ingestion) of radioactive materials and the equivalent dose due to external exposure (ground contamination, body surface contamination etc.).
This paper aimes to present the calculation model used to establish the derived release limits for the effluents resulted from routine activities on the RATEN-ICN site.
For all nuclear activities on the ICN platform, a dose constraint was established as one tenth of the limit for exposure of the population, i.e., 0.1 mSv/y.
This annual dose constraint was divided according to the type of effluent, 60% for liquid effluents, and 40% for gaseous effluents, and was further used as a basis to calculate derived release limits for radionuclides which are characteristic to the nuclear and radiologic installations from the site.

One may conclude that, if correctly established and monitored an effluent control system based on derived release limits may ensure that the operation of a nuclear facility or a nuclear installation does not pose threats to the environment or the population.

III.1.10. Establishing the Theoretical Limit of the Liquid Nitrogen Requirement for the Controlled Filling with Ice Plug of a Horizontal Pipe with Nominal Diameter 300 Crossed by Demineralized Water

Maria-Andreea STOICA[1], Șerban VALECA[1], Bogdan CORBESCU[2]
[1]University of Pitești, România, [2]Institute for Nuclear Research, Pitești, România
bogdan.corbescu@nuclear.ro

The technique of the large diameter horizontal pipes isolating with ice plug is used in the repair or replacement of components in hydraulic installations whose working agents are present in the liquid state. The main benefit of this method is the elimination of the need to interrupt the entire activity of the network, in order to reduce the duration of the intervention itself, thus eliminating the costs of transferring and storing liquid from the network loop. The ice plug formation inside the pipe is performed using a sleeve shape device, mounted outside of the pipe in the area chosen for sealing. An annular cavity is formed inside it in which liquid nitrogen is to be injected. The sleeve has a nitrogen vapor outlet to limit the pressure generated during the procedure. The completion, formation and maintenance of the ice plug as well as its length and thickness are subordinated by: the type of liquid conveyed, the working temperature of the liquid, the freezing point, the dimensions of the pipe segment subjected to the intervention, the initial temperature of the liquid, convection currents in the liquid and flow rate, distances to the first flow restrictions, ambient temperature, and the rate of nitrogen entering the sleeve. The procedure addresses the modeling of the phenomenon of the formation of an ice plug for stopping the flow of water through a horizontal pipe with Dn 300 taking into account the data obtained experimentally. The aim was to obtain, based on a built calculation model, some theoretical values regarding the volume of liquid nitrogen consumed.

III.1.11. Probabilistic Safety Assessment Applications in Support of the Nuclear Installations Operation

Birsan Georgian[1], Mita Farcasiu[2], Serban Valeca[1]
[1]University from Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
birsan.georgian@yahoo.com

Probabilistic Safety Assessment (PSA) studies provide a systematic and comprehensive analysis of complex installations (such as Nuclear Power Plant – NPP), in order to identify credible accident sequences and assess the probabilities of sequences occurrence, as well as the associated consequences. PSA represents an important tool which can be used to evaluate the risk impact of changes in installations design or operation. The obtained results can directly lead to an increased safety and reliability of the facilities, through changes of the project or operating procedures. This paper identifies solutions for the improvement of the nuclear installation's operation using PSA studies and their applications. A procedural task for accident sequences modelling is the systems modelling using fault tree (FT) analysis. FT is the most common method used for representing the failure logic of the systems. In order to demonstrate the possibility of the improvement of nuclear installations operation using the FT evaluation results and risk importance measures, a case study was performed for Boiler Level Control (BLC) station from CANDU NPP. For the system modelling, a FT was developed. This station maintains the level in each boiler at a specific setpoint. In analysis BLC program is required to provide Large Level Control Valves (LCV) close and Small LCVs open for the feedwater control to the boiler in case of reactor shutdown. The code used for qualitative and quantitative analysis is EDFT, part of PSAMAN computer programs package

developed in RATEN INR Pitesti. Reliability data used in the FT quantitative analysis are included in a generic database used by the EDFT code. Both failures and human errors (omission, commission) leading to the system unavailability were obtained after the FT quantitative evaluation. A study to prioritize the major contributors with the greatest impact on the system unavailability risk and safety was achieved by taking into consideration two risk importance measures: Risk Achievement Worth (RAW) and Fussel-Vesely Importance (FV). The quantitative acceptability criteria used for these importance measures (at component level) are as follows: RAW>2 and FV>0.005. The sensitivity studies to see the effects of the changes in components data (failure rate, test interval) are also performed. The main results are the list with important and critical components in the BLC station unavailability and the recommendations for the future actions. These actions could be: using better quality components, modifying maintenance policy, adjusting testing activities, and suggesting component redundancies. The obtained results demonstrate that PSA technique is a key tool in modeling the systems and installations and in improving their reliability and safety. The paper is elaborated in the frame of RATEN ICN and University of Pitesti agreement and has an educational purpose.

III.1.12. Utilization of Metallic Getters for the Separation Systems of the Tritium Resulted from Generation IV Nuclear Reactors

Mihaela Florea[1], Irina Sturzeanu[2], Monica Valeca[1]
[1]University from Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research, Campului Street, No.1, Mioveni, Arges, Romania
mihaela.florea@nuclear.ro

The research studies in the field of nuclear fusion and the fuel management for Generation IV nuclear reactors, indicate that the tritium release in the environment is expected to increase. Given the fact that tritium is present, in different quantities, in all nuclear energy production systems, it is important to elaborate strategies to mitigate the tritium release by finding methods to capture it.
An efficient approach for tritium capturing from the gas flow resulted from nuclear fission is the use of materials that absorb tritium. At the temperatures specific for the new generation of nuclear reactors, several metal hydrides based on Ti, Zr, La, Ce and Y, were taken into account to be used. These are reacting at high temperatures with the hydrogen isotopes, conducting to a stable product, which gives the possibility to recover the hydrogen isotopes.
This method named ‘metal getter’ technology can be used for detritiation of the inert gas flow and collecting the elementary tritium, in direct mode. This involves the utilization of a base material, which dissolves the hydrogen isotopes and forms metal-hydrogen phases or metal hydrides.
In this context, this paper presents the results obtained in the research activities performed until now, to develop mitigation technologies for tritium released, having as objective the minimization of tritium permeation. The experimental program aims to obtain information regarding the behaviour of Ti used as metal getter in tritium separation systems. For this purpose, the absorption/desorption characteristics were investigated, using the thermogravimetric method, at constant pressure.

III.1.13. Determination of Uranium Concentration in Biological Samples

Alexandra Georgiana Floarea[1], Monica Valeca[1], Rodica Magereanu[2]
[1]University of Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
ada.floarea@gmail.com, monica.valeca@nuclear.ro, rodica.magereanu@nuclear.ro

Determining the concentration of uranium in biological samples is a significant challenge due to the very low level at which it is found in this type of sample (urine from personnel exposed to work with ionizing radiation), as well as due to the particular chemical characteristics of material matrices in which they are found.

Chronic exposure to uranium leads to its deposition in a proportion of 90% in the lungs and lymph nodes, in macrophages and interstitial tissue. In the lung epithelium, it causes fibrosis lesions as well as neoplastic lesions and metaplastic changes. Renal radiotoxicity is manifested by renal damage: renal atrophy, renal paralysis, tubular atrophy.
The internal dose is determined by analyzing urine samples, which monitor the concentration of uranium, based on various dosing techniques. The frequency of sample measurement depends on the concentration of natural uranium in the urine at the last sample measured. As its concentration in urine increases, the interval between two biological sample supplies decreases.
The analysis of excretions or other biological samples can be used to detect uranium in the body and this type of analysis has the advantage that it does not require the presence of the individual during the measurement. Measuring uranium activity in urine is a useful technique for assessing long-term accumulation in the body, while providing the ability to verify assessments of internal contamination by other techniques. The radiochemical and analytical analysis procedures take time, so usually the result is obtained late, and a second identical sample cannot be obtained.
The most modern analytical techniques require, first, the preparation of a homogeneous, stable sample, in aqueous solution and possessing a known matrix. Referring to actinides, in general, for aqueous solutions, they may be present either in dissolved form or adsorbed on the surface of the suspended solids. In most cases, actinides can be solubilized and stabilized by acidification.
In order to determine the uranium in the urine, the urine samples from the exposed personnel were subjected to the mineralization procedure with HNO_3 and H_2O_2 followed by calcination, and subsequently, the residue is taken in HNO_3 solutions of various concentrations, depending on the protocol followed, for the passage of Uranium in solution. After preliminary processing, Uranium is subsequently determined by three optical methods of analysis, namely by UV-VIS spectrophotometry, fluorimetric or ICP-OES emission spectrometry.

III.1.14. TRIGA Reactor LEU Nuclear Fuel Monitoring

A.G. BICAN[1], S.C. VALECA[1] and A. RADULESCU[2]
[1]University of Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research Pitesti, Pitesti, Romania
andradageorgiana13@yahoo.com, serban.valeca@upit.ro, aronela.radulescu@nuclear.ro

This paper contains the results obtained after performing the dimensional measurements on several nuclear fuel elements from the TRIGA SSR (Steady State Reactor) core, in order to characterize the LEU nuclear fuel behavior to irradiation.
TRIGA Reactor owned and operated by RATEN ICN Pitesti is the only operating research reactor in Romania; it has a dual core design, the reactor pool accommodating a Steady State Reactor core designed to be operated at a maximum power of 14 MW and a pulsed ring core with a maximum pulse of 20000 MW. TRIGA SSR was intended to be used for nuclear fuel and structural materials testing, as well as for radioisotopes production.
The paper comprises brief information on the facility, the operation history and core features, some peculiar characteristics of the nuclear fuel used and also the requirements for ensuring the safe operation of the reactor.
Periodical examination of the TRIGA SSR nuclear fuel elements has to be accomplished to establish the specific long-term behavior of this fuel type, and also to assure the compliance with the provision and technical conditions of operation (LCO's). For the proposed analysis, 10 LEU fuel elements from the reactor core were selected.
The paper also contains a description of the non-destructive methods used for the underwater dimensional measurements (i.e. elongation and bowing of the nuclear fuel elements, as well as bowing orientation relative to fuel elements marks) which are performed with special pieces of equipment in the pool of the reactor.
From the analysis performed on the processed data for the elongation and bowing of the selected TRIGA fuel elements, it was found that the elements met the requirements regarding the limits and technical operating conditions of the reactor, for no element reaching the maximum allowed values.

III.1.15. Basic Characterisation of the Time Dynamics of a Loss-of- Cooling Accident in the Spent Fuel Pool of a CANDU NPP

M. Barbu[1], L. C. Dinu[2]
[1]University of Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
madalinabarbu55@gmail.com

The paper addresses the scenario of a loss-of-cooling accident at the spent fuel pool (SFP) of a CANDU NPP, evaluating the qualitative consequences of such an accident and estimating some critical parameters characterizing such an accident.
During the normal operation of a SFP, the cooling is done permanently by circulating the de-mineralized water through the dedicated system. In case of a loss of cooling accident (e.g. blackout of the power station on the site for a quite long time) the temperature of the water in the SFP rises continuously up to the boiling point. This situation is reached in a time interval $t_{boiling}$ with respect to the very starting moment of the accident. Further on, the level of the water in SFP starts to decrease continuously so that after a time $t_{dewatering}$ (considered also with respect to the moment when the accident occured) the most exposed spent fuel assemblies (those which are at the highest level, on the vertical axis, in SFP) begin to remain uncovered. Such a situation is considered of high risk, considering the further potential evolution of the accident if the loss of cooling cannot be reversed, either by re-establishing the normal functioning of SFP or by other external cooling means.
For a typical standard SFP, whose geometrical details are provided in the paper, $t_{boiling}$ = 3 days, whereas $t_{dewatering}$= 10.7 days, which means enough time is available for a practical intervention in order to limit the consequences of such an accident and to prevent the escalation of consequences (e.g. release of radioactive fission products in the atmosphere). The two critical parameters characterizing the loss of cooling accident were evaluated by a dedicated computer programme created to simulate the first phase of loss of cooling accident at SFP. This computer programme is written in Visual Basic 6.0. in order to obtain a friendly user tool.

III.1.16. Advanced Fuel Designs Influence on The Adjuster and Mechanical Rod Systems Performances in a CANDU Reactor

Diana Soare[1], Serban Constantin Valeca[1], Iosif Prodea[2]
[1]University of Piteşti, Romania, [2]Institute for Nuclear Research (RATEN ICN) Piteşti, Romania
d.elena49@yahoo.com

CANDU reactors can accommodate different nuclear fuel designs such as Natural Uranium (NU), Slightly Enriched Uranium(SEU), Recycled Uranium (RU), Mixed Oxide (MOX) fuel, Natural Uranium Equivalent (NEU) and Thorium-based fuel (Th). The NU, RU and Th fuels are envisaged in this study to be used in a generic CANDU power reactor in order to find out their influence on the performances of the adjuster (ADJ) and mechanical (MCA) rod systems. The RU fuel type has an enrichment of 0.96% in U235 and also contains tails of U234 and U236, while the Th based fuel is a mixture of 77% ThO_2 and 23% UO_2 with 9% SEU, the U/Th mass ratio being 0.3. The methodology used in the paper is based on the calculation of ADJ and MCA core reactivities when NU fuel is replaced by RU fuel or Th fuel, respectively. Lattice neutronic properties along with incremental cross sections were firstly calculated using the WIMS and DRAGON programs for every fuel kind. Then, core simulations with the DIREN program have been performed tracking some Commissioning configurations of Cernavoda NPP Unit 1, in which ADJ and MCA systems were calibrated based on Zone Control Units (ZCU) level variation. The results show that the ADJ bank core reactivities slightly decrease with respect to the fuel enrichment while those of MCA system present an easy increase in capability by the fuel enrichment increasing. The total ADJ and MCA reactivity amounts still remain close to their reference values reported in public literature, thus the ADJ and MCA systems performances are preserved.

III.1.17. Instrumentation of Devices used in Irradiation Experiments

A.G. Nuta[1], A. Amzoi[2], S. C. Valeca[1]
[1]University of Pitesti, Romania, [2]RATEN Institute for Nuclear Research, Pitesti, Romania
alex.nuta21@gmail.com

In the nuclear field, knowing precisely the temperature of the fuel as well as the temperature's evolution, gives the opportunity of evaluation and control of the nuclear reactor operating parameters. The instrumentation of devices represents an importance operation through which they are introduced in the areas of interest, the necessary means for controlling the reactor operation. Monitoring the temperature in different points of the nuclear installation involves the use of thermocouples with thin metal walls. Special conditions imposed on the areas crossed by these thermocouples between the measuring point and the connection for signal collection, require the use of special processes for joining, insulation and protection.
The paper deals with the instrumentation of a fuel rod used in the RATEN ICN TRIGA research reactor with the purpose of measuring the fuel temperature. The instrumentation of the fuel rod is done by introducing inside of it three K-type thermocouples, for a real-time temperature monitoring of the central area of the fuel pellet during the reactor's operation. Thermocouples are introduced in the fuel rod by the sealing plug and then by the upper plug.
Vacuum induction brazing represents the process through which the two sealed passages are made, first at the level of the upper plug, and second by the sealing plug of the instrumented fuel rod. The brazing operation takes place at 1080°C, temperature which is achieved by inductive heating. As filler, a Ni based alloy, in powder form is used. The process takes place in a cylindrical enclosure, made from quartz, in which a vacuum is created by means of a pumping system, to avoid oxidation of parts due to high temperature. The fuel rod instrumented with K-type thermocouples can be handled manually and can be introduced in each location of interest in the TRIGA type fuel box.
The value of the temperature inside the instrumented fuel rod represents an important characteristic parameter of the reactor core, because exceeding the upper limit of the fuel temperature increases the risk of the fuel rods damaging, which can lead to nuclear accidents.

III.1.18. Quality Control for Radiometric Measurements

Lixandru Mihai-Albert[1], Monica Valeca[1], Grațiela Tudor[2]
[1]University of Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
albert.lix98@yahoo.com; gratiela.tudor@nuclear.ro

The present paper aims to describe: the quality control method used to monitor the analytical performance of radiometric techniques and the means used for quality control applied to the measurement of gross alpha/beta activity, through gas-flow proportional counting.
Quality control must be an integral part of a quality assurance system and must include the entire analytical process from the entry of samples into the laboratory until the analysis report is issued.
The main statistical tools for quality control are the control charts. These are practical means for monitoring the evolution of the performance parameters.
In order to be able to check the detector's response under fixed measurement conditions, 40 measurements were made, using a Sr/Y-90 standard source with energy and counting rate close to those of the radionuclides in the samples, , as well as the background counting rate. For fixed measurement systems, the background counting rate shall be used as a measurement quality control parameter, indicating possible contamination or system failure.
The response of the detector can be checked daily or at the beginning and the end of a batch of samples. In our case, the verification is carried out daily, by short-term measurements of the test source of Sr/Y-90.
The test source shall be counted to achieve an uncertainty of less than 1% at 1σ and the control or tolerance limits shall be set at ±3σ or ±3%, according to the acceptance criteria of the quality control programme.

Warning and acceptance limits have been established for each monitored parameter.
The results of the monitoring were presented in the paper in the form of control charts. The applicability of control charts is based on the assumption that the data is subject to normal distribution.
To achieve the control chart, firstly a number of 40 values of a parameter are acquired, and the arithmetic mean is calculated , then an estimator of the standard deviation σ is determined, so that one can finally define the center-line, the control limits and the warning limits.
Data plotted against time is distributed on either side of the *CL* center-line which represents their mean value. The *UWL* and *LWL* warning limits are calculated by adding or subtracting a value equal to twice the estimator of the standard deviation to the CL value. The *UCL* and *LCL* control limits are calculated by adding or subtracting three times the value of the estimator for the standard deviation to the *CL* value.
Acceptance limits for quality control parameters are used more as control limits on control charts to determine whether further investigation or corrective action is required.
From the analysis of the control charts for all the monitored parameters it followed that no additional investigations or corrective measures are required, the measurement values falling within the limits of acceptance and control.
One can conclude that the measurement system has considerable stability which might guarantee the precision of future measurements.

III.1.19. Evaluation of Irradiated Environmental Samples in The TRIGA Reactor Using the Neutron Activation Analysis Method

George Bogdan SOARE[1], Adrian Florinel BUCSA[2]
[1]University of Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
soaregeorgeb@gmail.com

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.
In order to determine the chemical elements that have negative effects on human health, we used as indicators biomonitors (mosses) and soil samples. Ground moss 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. Soil is a good monitor of pollution by accumulation process of dangerous trace elements in its matrix.
The method used in the study is the neutron activation analysis (NAA-k_0) using k0 standardization. This method 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. 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's irradiation was conducted in K11 grid location of the TRIGA SSR 14 MW (Steady State Reactor). The study was made on 4 samples of ground moss from the environment (rough samples) and 4 samples of soil from Pitesti area.
The thermal neutron flux supplied by the TRIGA SSR 14 MW reactor in the rabbit location was fair enough for activating elements like manganese, potassium, bromine, europium, lanthanum, arsenic, scandium, antimony, iron.
The differences indicated 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 and soils as good biomonitors.

III.1.20. System Modelling in Nuclear Power Plants

***Madalina Breazu*[1], *Ileana Gondac*[2], *Cristina Constantinescu*[2], *Adriana Dutcec*[2], *Serban C. Valeca*[1]**
[1]University of Pitesti, [2]Institute for Nuclear Research Pitesti,
Ileana.gondac@nuclear.ro

This paper is elaborated in the frame of agreement between Institute for Nuclear Research Pitesti (RATEN ICN) and University of Pitesti. The purpose of the paper is an educational one and evaluates the Nuclear Power Plants (NPP) systems unavailability, using fault tree (FT) technique from Probabilistic Safety Assessment (PSA) methodology. Use of PSA leads to numerical estimates of the NPP risk, offering a consistent and integrated model of the plant. A FT analysis is used to find the credible ways in which an undesired event (top event) can occur. The FT is a graphical model of the various parallel and sequential combinations of faults that will result in the occurrence of the predefined top event.
The major goal of the paper is to show how the modelling of NPP systems by FT technique identifies the elements such as: basic events, failure modes, failure rates, components failure probabilities, minimal cut sets, system unavailability. A generic case study was performed for D_2O Supply System, from CANDU6 NPP, aiming to evaluate the contribution of the components failure to the system's unavailability. A qualitative and quantitative evaluation of D_2O Supply System was performed using the EDFT code, part of PSAMAN computer programs package, developed in RATEN ICN by the PSA team.
To achieve the paper's objective, the following elements are envisaged:

- the knowledge of terminology, notation, symbols and the FT model construction;
- the knowledge of purposes and methods of the FT analysis (qualitative, quantitative);

interpretation of the results and identifying the solutions for improving the D_2O Supply System availability.

III.1.21. Assessment of the Accidental Criticality for CANDU Fuel Bundles Containing U-based Fuels

***R. Joitescu*[1], *C.A. Mărgeanu*[2], *S.C. Valeca*[1]**
[1]University of Pitesti, Romania, [2]Institute for Nuclear Research (RATEN ICN) Pitesti, Romania
cristina.margeanu@nuclear.ro

Nuclear criticality safety relates to prevention and protection against the consequences of an uncontrolled nuclear fission chain reaction, or a criticality accident. Accidental criticality can result if a sufficient quantity of fissile material (e.g. U^{235}, Pu^{239}) is arranged into a critical configuration. The most adverse and potentially dangerous aspect of a criticality accident is the release of nuclear radiation that affects nearby personnel. The present paper aimed to study the accidental criticality of CANDU fuel bundles containing U-based fuels. The effective multiplication factor (k-eff) was evaluated for different potentially dangerous geometrical arrangements of stacked fuel bundles (containing fresh or spent fuel) accidentally immersed in H_2O or D_2O. The arrangements of fuel bundles included 2, 3, 5 or 6 stacked fuel bundles, no any damage of fuel bundles structural integrity being assumed, so we don't have to deal with a potential spreading of fresh or irradiated nuclear material. The criticality analysis has been performed for CANDU fuel bundles with 37 and 43 fuel elements. As regarding the fuel composition, three types of U-based fuels were considered, namely: natural uranium (NU), slightly enriched uranium (SEU), and recovered uranium (RU). The criticality coefficient assessment has been performed by using KENO VI Monte Carlo code. For the spent fuel analyses, fuel burnup was simulated by means of ORIGEN-S burnup code, up to 8 MWd/kg HE for NU fuel and up to 10 MWd/kg HE for SEU and RU fuels, respectively. Both codes are included in SCALE5.1 programmes package developed by Oak Ridge National Laboratory, USA. For fresh fuel scenarios, RU43 and SEU43 fuels were characterized by 10%-16% higher k-eff values comparatively with NU37 fuel. As regarding the spent fuel accidental criticality scenarios, k-eff values obtained for NU37 fuel were 2%-5% higher than those characterizing SEU43 and RU43 fuels; SEU43 fuel has 2%-3% higher k-eff values than RU43 fuel. Multiplication factor values were lower than 1 in all accidental criticality scenarios, except for the accidental immersion of fresh fuel bundles arrangements in D_2O considering infinite lattice option for neutron leakage treatment. For a single "row" of fuel bundles placed on a concrete platform and covered by a layer of water (the most realistic option for neutron leakage treatment) 6 stacked fuel bundles seem to be the most dangerous configuration.

III.1.22. Application of Microwave Energy to the Treatment of Radioactive Sludge

Elena Doinita MORLOVA[1], Monica VALECA[1], Elena DUMITRU[2]
[1]University from Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti, Romania
morlova.elena21@yahoo.com

The article deals with the possibilities of microwave (MW) energy utilization in the treatment of the radioactive sludge.
Radioactive waste in sludge form from Cernavoda NPP looks like sludge obtained from treating domestic water and therefore in this waste can exist colonies of pathogen microorganisms like Escherichia coli, Salmonella typhi, Pseudomonas aeruginosa and Trichinella spiralis. The lab analyses showed that the nuclear waste in sludge contains radioisotopes β-γ and Tritium, in concentrations that are in the range of low-level radioactive waste.
The tests were conducted on radioactive sludge. The frequency of microwaves was 2450 MHz. The sludge was exposed to MW irradiation at different power levels and for various durations.
The results demonstrated that the MW is a rapid and efficient method that can reduce the sludge volume between an 80 - 95 %. Microwaves can be used as a highly efficient method of treatment the radioactive sludge reaching temperatures necessary for destroying colonies of pathogen microorganisms.

III.1.23. Instrumentation used in molten lead facilities

GIUIU ELENA CAMELIA[1], MUGUREL UNGUREANU[2], ȘERBAN VALECA[1]
[1]University from Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research, Pitesti,Romania
mindalac.cami@yahoo.ro, mugurel.ungureanu@nuclear.ro

The lead-cooled fast reactor (LFR) is one of the most promising generation IV (Gen IV) concepts being significantly safe, sustainable, economically competitive, and not-proliferant.
Romania, through RATEN-ICN, is part of the FALCON consortium, which proposed the construction of the ALFRED demonstrator and six support installations, having already prepared the technical documentation for the construction of the ATHENA (Advanced Thermo-Hydraulics Experiment for Nuclear Application) and ChemLab (Lead Chemistry Laboratory) experimental facilities.
The paper presents the instrumentation used in a molten lead research installation. To monitor and control the parameters of interest, sensors for oxygen concentration, pressure gauges, thermocouples, flow meters and level sensors are used. A section of the paper is dedicated to the LECOTELO (Lead Corrosion Test Loop) experimental facility which has been built at RATEN-ICN Pitesti with the purpose of corrosion / erosion testing of structural materials for Gen IV reactors in molten lead.
Also, the paper presents the data acquisition and measurement techniques, as well as the parameters that characterize the measuring devices.
After studying the different experimental facilities, as well as the tests performed in the LECOTELO facility, the following conclusions can be drawn:
- lead, due to its thermo-physical properties, is considered a good candidate as coolant for Gen IV reactors, its main disadvantage being the high corrosion rate of steels at high temperatures;
- for the safe operation of the molten lead installations, it is essential to use various measurement and control systems for the parameters of interest;

- for any liquid-metal facility, oxygen is certainly one of the most important chemical elements since it leads to coolant oxidation and coolant oxides deposition when oxygen is dissolved up to the solubility limit; such oxides affect coolant thermal-hydraulics, causing the plugging of the structures. Moreover, corrosion is influenced by oxygen dissolved in lead and a sufficient level of oxygen is usually preferred and required to have the formation of a protective oxide layer above steel surfaces.
To mitigate these issues, oxygen concentration in lead coolant has to be monitored and controlled using oxygen measurement and control systems.

III.1.24. Radiological monitoring of work spaces, tool for controlling workers exposure

***Dragan Maria Gabriela*[1]*, Monica Valeca*[1]*, Mircea Stefania Daniela*[2]**
[1]University of Pitesti, Pitesti, Romania, [2]Institute for Nuclear Research Pitesti, Mioveni, Romania
gabrielaadragan@yahoo.com, monica.valeca@upit.ro, stefania.mircea@nuclear.ro

This paper describes the main methods used in radiological monitoring of technological spaces, to ensure the radiological safety of professionally exposed workers.
The experimental part of the paper consists of the radiological characterization of the technological spaces in a nuclear facility. Radiological characterization is a complex process that includes measurements of contamination and dose rate in technological rooms with both portable equipment and fixed systems.
The radiological characterization of the technological spaces in a nuclear facility is an activity that allows the evaluation of the radioactive contamination in these spaces and implicitly the planning of the decontamination operations if the situation requires it.
For the evaluation of potentially contaminated surfaces in the technological rooms of the reactor building, direct and indirect measurement methods were used, employing portable equipment and smear sampling techniques which allowed for discrimination between fixed and loose components of radioactive contamination.
All these activities were carried out in compliance with the requirements and conditions regarding the radiological health and safety of the personnel, in accordance with the national and international legislation in force.
Following the radiological measurements of the surfaces in the technological rooms, a radiological mapping was performed, which allows an assessment of the areas at risk of contamination and radiation exposure.
In conclusion, an important issue of the nuclear activities is that due to the presence of radioactive materials, protective measures are to be taken to keep their spread under control. Thus, any nuclear facility must be provided with workspace monitoring systems and/or procedures.
Permanent monitoring of technological spaces is necessary to limit the radiation exposure of professionally exposed personnel.

III.1.25. Experimental study for characterization of the solutions obtained by mineralization of ionic resins marked with radiochemical tracers

Anghel Florica, Monica Valeca, Irina Zdru
Institute of Nuclear Research Pitesti, Romania
floricaneculae@yahoo.com; irina.zdru@nuclear.ro

Spent ion exchange resins replaced from the purification systems of CANDU reactors in operation at Cernavoda NPP represent an important category of radioactive waste. A good knowledge of the radionuclides content allows a proper selection of treatment, conditioning, storage and disposal of this waste category with a minimum radiological impact on humans and environment.
One of the methods for separation and purification of the alpha emitting radionuclides retained by ionic exchange process on ionic resins is the use of chromatographic columns for radionuclide separation and F_3Ce co-precipitation for preparing the sources to be measured by alpha

spectrometry. By this method, alpha sources whose activity can be measured using an Ortec spectrometer are obtained. To apply this separation method the ionic resin samples have to be converted into liquid samples with the organic content destroyed, usually achieved by acid digestion. By acid digestion the organic material is decomposed and the actinides are released into solution.
A microwave digestion system, modelMultiwave 3000 , Anton Paar, was used for resin samples mineralization. The mineralization process proved to be efficient, the ionic resins labelled with the radionuclides of interest (respectively ^{233}U, ^{243}Am, ^{242}Pu) were completely decomposed and the radionuclides in the resulted solutions were efficiently separated.
Practically, the mineralization process was achieved in three steps, using powerful oxidation reagents (HNO_3 and H_2O_2), with intermittent depressurization of the reaction vessels:

- 1 pre-mineralization of the sample with concentrated nitric acid at 190°C
- 2 mineralization of the sample at 220°C
- 3 advanced digestion of the sample with the addition of hydrogen peroxide 30%

This process leads to high recovery efficiency of the radionuclides.

III.1.26. Study of Radon concentration in drinking water from wells in the area of Mioveni town

Georgiana Vlad, Ovidiu Hirica, Monica Valeca
Institute of Nuclear Research Pitesti, Romania
elenadanielastanciu@yahoo.ro

The population exposure to nuclear radiation is largely due to natural radioactivity (radon exposure, external exposure to radiation of telluric and cosmic origin, ingestion of contaminated food and water). Radon contributes, a very large percentage, to the irradiation of the human body, the lung being the most affected organ.

The evaluation of radon content in groundwater is of interest in radiation hygiene studies.

Radon, from deep water, comes from the disintegration of uranium or radium deposits present in the geological formations transited by the water stream or that host the body of water from which they originate.

This paper presents a study of the concentration of radon in drinking water from deep wells located in the area of Mioveni. The study was carried out on 10 wells with drinking water intended for the population consumption, 7 located in the city, 2 in its nearby neighborhoods and one in Davidesti, a village situated North-West from the town. The drilling depths of the wells vary from 100 to 150 meters, and they are located at a considerable distance from one another.

The radon measurement was performed by alpha-LSC method, which is based on radon extraction in an organic scintillator, immiscible with water and on the determination of the alpha emission rate using an alpha/ beta counting analyzer, after the radioactive equilibrium is achieved between Rn-222 and its alpha emitting daughters (Po-218 and Po-214).

Following the measurement performed, the radon concentration in water from the ten sampling points was below 20 Bq/l, while the minimum detectable concentration was 0.5 Bq/l. According to the study one can conclude that the measured concentrations were well below the guideline of 100 Bq/l, established by the law.

III.1.27. Chemical Degradation Impact on Mechanical Strength of Mortars

Iulia Pietriş[1], Ionut Florea[2]
[1]University from Pitești, [2]Institute for Nuclear Research Pitesti, Romania
ionut.florea@nuclear.ro

Due to their mechanical and physical properties to act as a barrier and of the chemical ones as a selective binder for radioactive species, cement-based materials are some of the most often used materials in radioactive waste conditioning and surface disposal facilities.

As long as these materials are intact, the containment of the radionuclides within matrix material is assured. In the repository, which is not an isolated system, cement-based materials may undergo a variety of reactions because of infiltration and movement of solutes from the surroundings that could affect mechanical, physical or chemical properties of these materials and lead to a finite lifetime of the concrete structure.
Typical chemical degradation reactions are decalcification, carbonation, expansion generated by sulphate attack and dissolution and leaching of cement components, processes that will affect mechanical properties.
This paper presents the results of an experimental study carried out to investigate the mechanical behavior of prismatic specimens' mortars made from CEM II cement type subjected to the chemical degradation processes mentioned above.
Different investigations were performed on these specimens to assess the potential structural modification as well as the mechanical stress. These tests comprised of compression tests, measurement of the carbonation depth and expansion generated by external sulphate attack.
The results obtained allow the characterization of basic mechanical responses of the tested mortars and the identification of chemical degradation effects on the mechanical properties.
After 90 days, the CO_2 conditions seem to have not major effects on compression strength of the mortars, which means that the carbonation is an extremely slow process, but for those subjected to the sulphate attack it was observed a slight reduction in compressive strength.

III.1.28. Sociological Research Applied in the Development Stages of Nuclear Projects

Al Saadi Muntadher Ali Mohammed Jawad, Dumitru Chirlesan, Daniel Onofrei
UPIT, Romania, monte_ali@yahoo.com,
ICN, Romania, monica.valeca@nuclear.ro
ICN, Romania, daniel.onofrei@nuclear.ro

All over the world, public acceptance is a crucial factor in decisions on public and private investments in nuclear field. Therefore, it is important to understand the determinants of public acceptance of nuclear power and the driving factors behind individual perceptions.
In Romania, using field visits and interactive workshops, different levels of stakeholders acceptance and related driving factors to determine changes in perceptions are identified. Local economic wealth creation, lower cost of generating electricity, and low-carbon energy provision appear to have the strongest positive effect on nuclear power public acceptance.
A study should be designed to answer the research question being asked. A thorough evaluation of the literature can help the organisation to avoid the communication mistakes made in the past. Theoretically, research studies should become better and better with time as past mistakes are solved, and studies become more and more robust. However, this is generally not the case as in reality each study is a new and novel endeavour.
During the planning stages of the study, the potential benefits should be considered. The expected outcomes are strongly linked with the literature review, hypothesis, and rationale. An useful exercise is to plot a graph of the expected outcomes for each question. This also helps to identify the most appropriate statistical analysis of the prospective data.
This paper addresses a sociological perspective on the public information level and the degree of public acceptance regarding nuclear energy. Theoretical considerations are focused on these objectives, highlighting the specificity of research and how it could make a significant contribution to optimizing communication.
The practical aspects are related to the creation, analysis and evaluation of a questionnaire applied in a survey process.

III.1.29. Detecting Radioactivity Characteristic of a GEIGER-MÜLLER Counter Tube

A.D. Salaman[1], M. Valeca[1], A.F. Florea[2]
[1]University of Pitesti, [2]Institute for Nuclear Research Pitesti, Romania
e-mail: salamandenisa@yahoo.com

Part of the category of gas ionization detectors, a Geiger counter is an instrument for measuring radioactivity by detecting and counting ionizing particles. The device is widely used in applications such as radiation dosimetry, radiological protection, experimental physics, and the nuclear industry.
A Geiger counter consists of a Geiger–Müller tube (the sensing element which detects the radiation - two electrodes inserted in a glass or metal tube, filled with a noble gas at low pressure) and the processing electronics, which displays the result.
When a radiation passes through the counter, the excitation and ionization of the gas molecules takes place. If accelerated in the electric field, the ions and electrons formed can in turn produce secondary ionizations (the character of the internal discharge depends on the voltage applied to the counter).
In the case of the Geiger – Müller counter, the multiplication in gas of the charges by secondary ionizations occurs (the avalanche discharge). Since the operation of the Geiger – Müller counter is based on the existence of a high-intensity electric field, the avalanche discharge is intensifying and is accompanied by secondary avalanches. Thus, the voltage pulses that occur have a high amplitude (1 - 10 V or more) and can be counted directly without any prior amplification.
To illustrate the voltage / number of ionization characteristics of a Geiger – Müller counter, experimental measurements were performed, by monitoring the radiation background in different geographical regions.
The paper highlights the need for measurements of the radiation background present in the environment with high accuracy, using detection systems that use Geiger – Müller counter.

III.1.30. The Evolution of the Discharge Events During the Plasma Electrolytic Oxidation of Titanium

M. Stoica[1], S. Valeca[1], V. Ion[2], O. Rusu[2], A. Marin[2], E. Coacă[2]
[1]University of Piteşti, Romania, stoica_maria_victoria@yahoo.com
[2]Institute for Nuclear Research, Piteşti, Romania, vladut.ion@nuclear.ro

Plasma Electrolytic Oxidation (PEO) is an environmentally friendly technique used with great benefits to improve the surface properties of light alloys (Al, Mg, Ti, and Zr). The PEO coatings quality depends on the evolution of the discharge events during the PEO process. The aim of this work is to identify the influence of different process parameters on the evolution of the discharge events taking place during the PEO treatment of the Titanium alloy. The purpose of this paper is to acquire the knowledge and the skills necessary to use the PEO technique in order to improve the wear and corrosion resistance of Titanium grade 2 alloy for the next generation nuclear reactors.
Tests were conducted by controlling the following PEO process parameters: the duty cycle, the current density and the electrolyte composition. Experiments were performed in galvanostatic mode using a programmable DC power supply. The samples of Titanium grade 2 were connected to the positive pole of the power supply, while the negative pole was connected to the reaction vessel made of stainless steel. The *Voltage-time* anodization curves were recorded. The electrolyte used was aqueous solutions based on Na_2SiO_3 and KOH at different compositions.
On the recorded anodization curves one can distinguish the characteristic stages specific of a PEO process. The results obtained show that changes in the profile of the *Voltage-time* anodization curves are strongly dependent on the process parameters. By finding the optimal experimental conditions one can improve the coating qualities.

III.1.31. Independent Simulator for CANDU600 Fuel Handling System - Operating Procedures in Emergency Situations

Stefan C. GHEORGHE[1], Serban VALECA[1] and Darie PREDESCU[2]
[1]University from Pitesti, Romania, [2]Institute for Nuclear Research Pitesti, Romania
gstefancristian@yahoo.com, serban.valeca@nuclear.ro, predescudarie@gmail.com

This paper deals with the aspects on the software application development for the independent simulator for Fuel Handling Machine Head (F/M Head), by designing and implementing practical applications – technological jobs (sequences). This development, in the operating modes offered by the independent simulator, will require the application of procedures, specific to emergency situations in the operation of the Fuel Handling (F/H) System and also the development of a tool to evaluate the actions taken by the operator through:
- identification of procedures for the operation of F/H System in emergency situations whose application is possible by using the facilities offered by the experimental model for independent F/M Head simulator;
- elaboration of technological sequences / jobs whose development, in the operating modes offered by the independent simulator, leads to situations involving the procedures for F/H System operation in the previously identified emergency situations;
- software implementation of the identified situations;
- design of the tool for evaluating the learning process.
Fault scenarios will take into account important events in the operation of the F/H System in various uses of CANDU reactors and during acceptance tests for F/M Head 3 / RAM4, F/M Head 4 / RAM5 and RAM6, conducted at RATEN ICN Pitesti.

III.1.32. Management of the monitoring system for the professional exposed personnel from a nuclear installation

Ramona Ene, Monica Valeca, Gabriela Barbu
UPIT, Romania, ene.ramonaa@yahoo.com
ICN, Romania, monica.valeca@nuclear.ro
ICN, Romania, gabriela.barbu@nuclear.ro

Ionizing radiation is a specific risk associated with the operation in any nuclear installation and the fundamental objective of radiological safety is to protect people and the environment from the harmful effects of ionizing radiation. The operation of a nuclear installation, for experimental purposes or for the production of electricity and heat, is different from conventional industrial activities because it takes place in working conditions with radiological risks, whose management requires the use of special work practices to ensure personnel safety and of the installation.
These practices are the subject to operational radiation protection. The main purpose of radiation protection is to ensure an adequate level of radiation protection of exposed personnel and the environment against the harmful effects of radiation exposure, without unduly limiting the appropriate human actions that may be associated with such exposure.
In order to ensure an efficient control of the potential risks associated with the operation of nuclear installations, the Institute of Nuclear Research – Piteşti, Romania management team promotes the highest standards in the personnel health and safety policy and for the protection of the population and the environment.
The present paper aims to study and evaluate the monitoring system of professionally exposed personnel in a nuclear installation. For this proposed purpose, we would like to carry out an analysis of the dosimetric data for the proffesional exposed personnel from the TRIGA Reactor and the Post Irradiation Examination Laboratory (PIEL) Department and Radioactive Waste Treatment and Conditioning Plant (STDR) - controlled area, operating and maintenance personnel, where there is an equipment that can represent a source of radiation exposure.

Given the presented data, it has been shown that by applying the radiation protection principle and the measures to reduce the exposure to ionizing radiation, the radiation protection program applied in the ICN fulfills the purpose for which it was designed, that of ensuring an efficient radiation protection for its workers, and the activities performed on the platform are carried out according to the radiological security norms.

III.2. Research Infastructure

III.2.1. Parameters Configuration in the Vijeo Historian Database from the Tritium Experimental Separation Plant (PESTD)

Carmen Maria Moraru, Iulia Stefan, Ciprian Bucur, Ovidiu Balteanu, Nicolae Sofilca, Mihai Vijulie
National Institute of Research-Development for Cryogenic and Isotope Separation Technologies, Rm. Valcea, Romania
carmen.moraru@icsi.ro

Data base - Vijeo Historian enables to accurately store data for long-term reporting while also giving the option of visualizing and accessing the information through the Vijeo Historian portal, Microsoft Excel or Reporting Services.
Vijeo Historian improves production reporting and ad-hoc analysis by connecting, aggregating and presenting real-time information from multiple disparate systems throughout the Pilot plant for tritium separation (PESTD), allowing for making more informed and timely decisions. The person in charge of configuring the parameters from the Historian server, periodically imports the data schemes created in Citect, configures, publishes and creates the Excel and Favorites data templates. It also verifies the accuracy of the received data and proposes corrective measures, maintains the Historian Manager, creates alarms and trends and verifies the proper functioning of events and tasks and extracts the stored data using Excel and Web clients or automated reports and makes them available to the PESTD manager, according to the work provisions.
The purpose of the paper is to define and explain the steps to use and operate the Vijeo Historian data base from PESTD installation, in the Control Room.

III.2.2. Preliminary Investigation on the Thermal Energy Storage for Lead-Cooled Fast Reactors

MARIN ANDREI-ALEXANDRU,
Institute for Nuclear Research, Piteşti, România
Andrei.marin@nuclear.ro

The expected drastically changes in the energy mix determined by climate policy measures, especially the closing of fossil fuel-based plant and replacing by intermittent energy production, will generate much complexity for the balance between the production and the consumption of electricity.
Nuclear systems have a limited flexibility due to some technical limitations (thermal cycling of materials, poisoning with fission products, etc.) and to the economic constraints generated by the long period for the return on investment. In this context, finding energy storage solutions becomes a priority in the context of achieving climate goals.
The use of energy storage at a nuclear power plant could solve the load following difficulties. Technical solutions are available at the level of pilot installations, but their implementation needs to demonstrate the feasibility and the efficiency of the system. Among the existing solutions applicable to nuclear power plants, the thermal storage is preferable due to the avoidance of repeated conversions, from thermal to mechanical and then to electrical energy, which could have severe consequences on the global energy efficiency of the system.
In this paper a Thermal Energy Storage unit in metamorphic rocks, using synthetic oil as a transfer agent, has been proposed to be coupled with the nuclear reactor. Metamorphic rocks were chosen due to their thermal characteristics superior to other easy-to-use storage media.
The results of this paper represent basic elements for the subsequent investigations necessary for the technical and economic analysis of such an option for a LFR system similar to the Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED)

III.2.3. Embedded System for Testing and Calibration F/H Machine Testing Rig

A. Coman and D. Predescu
Institute for Nuclear Research Pitesti, Romania
alecsandra_coman@yahoo.com

CANDU 600 F/H Machine Head Testing Rig computer control system is a dynamic real-time distributed data processing system capable of supervising processes based on decentralized schemes. The software structure is represented by software products and operating procedures dedicated to either real-time process management or processing/transmission of data flows. Starting from the content of the project of design-implementation and risk-free exploitation of nuclear installations, the premises of software development reflected the ability to understand the real problems, the maximum possible modularization, the relative independence of each application at least from the perspective of accessing data resources and I/O operations, the flexibility of interfacing with the operator and the controlled process. The paper presents the work performed in order to develop a system for functional testing and calibration of F/H Machine Head Testing Rig subsystems, and had as main working directions the following: mathematical modelling and identification; designing a generic structure for the system; identifying the optimal implementation solution. Predictive analysis regarding the functioning of a device, technological process/complex system and its control or management according to a certain strategy (developed, possibly, based on an appreciable volume of information that must be examined and processed simultaneously depending on the correlation between them and their evolution in time) can be realized in the most economical way, by developing some initial simulation methods. The basic model of the complex device intended for calibration / testing of equipment from the F/H Machine Head Testing Rig includes the functional simulation for the considered subassemblies, based on the mathematical models comprising the transfer functions, respectively the linear analytical equations that characterize them functionally. The development of the modelling methods allows, as a first advantage, the performance of the predictive analysis regarding the operation of a device, subassembly or technological process and the evaluation, under minimum expenses, of the optimal adjustment parameters for the modelled physical system. The systems with functional simulation role for various processes had the main purpose of training operators, becoming increasingly attractive for design, optimization, and process diagnostics and for other purposes than training operators. A variety of applications can be detected, in which the simulators can bring substantial benefits such as: optimization of the operation, validation of the changes made to the process control software applications, development of strategies for emergency situations, testing and validation for changes made to the human-machine interface, development of support systems to assist operator's decisions by simulating the real-time response.