CONTENT

OPENING SESSION

IAEA NPTDS Activities on Non-Electric Applications of Nuclear Energy

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Most trajectories to net zero require massive amounts of low-carbon electricity but decarbonising the power sector will not be sufficient, as the power sector represent only 40% of the emissions. There is also a need for massive amounts of low carbon heat and low carbon fuels, as well as the technologies to produce them, in order to support the "hard to abate sectors", those that cannot be electrified (such as steel industry, cement manufacturing, heavy duty transport, buildings heating). Nuclear energy is also a source of low carbon heat and can also be used to produce low carbon fuels such as hydrogen. Capitalizing on nuclear heat is virtually untapped potential up to now since currently 1% of nuclear heat (that is not used for generating electricity) has been used for non-electric applications.

The interest in non-electric applications of nuclear energy is growing, driven by a series of factors, including environment, economics, security of energy supply and in addition, the emerging reactor technologies – the advanced reactors and small modular reactors - can be optimized to be used for non-electric applications, for single or multi-purpose use – generating commodities like hydrogen or potable water, together or without electricity generation for the grid. The optimization is based on criteria such as cost of products, return of investment and energy efficiency. The new reactor technologies and their use for non-electric applications come with additional challenges, as their deployment requires a "level playing field" in terms of policies, support to innovation, and financing, as well as clear market signals to favour low carbon. Technology readiness and economics will be key to advance their deployment.

Since currently it is estimated that nearly half of the emissions reductions to 2050 come from technologies that are not yet commercialized, there is a stringent need to accelerate demonstration and bring the advance reactor technologies to the market.

The International Atomic Energy Agency has been developing and updating regularly specific tools to support non-electric applications, that are available for free download from the IAEA website:

- DEEP or The Desalination Economic Evaluation Program is a software that can be used for performance and cost evaluation of various power cogeneration configurations for desalination.
- DE-TOP or Desalination Thermodynamic Optimization Program is a tool for the thermodynamic analysis and optimization of nuclear cogeneration systems (currently with options for nuclear desalination and district heating applications).
- HEEP or the Hydrogen Economic Evaluation Program can be used to assess the economics of hydrogen production using nuclear energy.

I. NUCLEAR ENERGY

I.1. Advanced Nuclear Systems and SMRs

I.1.1. Romanian Regulatory Framework for Small and Modular Reactors (SMR)

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The paper describes the Romanian Nuclear regulatory framework for the safety of nuclear installations and indicates the compatibility with new nuclear projects, in particular Small Modular Reactors (SMR). The paper indicates the current regulatory framework, regulations, applicable for design, sitting, construction, commissioning and operation for any nuclear installation in Romania. Further, the paper stresses out the major challenges in applying the existing regulatory requirements to new facilities.
The regulatory framework in Romania in the nuclear field is generally technology-neutral and is prepared for the licensing of Small and Modular Reactors (SMR). The regulations issued by CNCAN are aligned with the European standards and legislation, as well as with the International Atomic Energy Agency, IAEA, safety standards. Regarding the regulatory review and assessment and authorisation processes, CNCAN may make use of the regulatory reviews performed in countries that have licensed the same or similar nuclear installations. In such cases, CNCAN would seek assurance that the regulatory framework and review process in the respective countries are stricter or at least as thorough as the regulatory framework and review process in Romania and the safety goals and objectives are compatible. The steps of the licensing process are the same for all type of nuclear installations, CNCAN uses a graded approach that considers, among other criteria, the complexity of the installation, the technology used and the experience of the licensee.
Applying the regulatory requirements using a graded approach is a major challenge to all regulators, independent of the activities regulated. The paper concludes that the current regulatory framework on nuclear installations existing in Romania is applicable and adjustable to all types of nuclear projects, such as Small and Modular Reactors (SMR), as long as the possible alternative technical solutions established in the project shall demonstrate compliance with the nuclear safety objective.

I.1.2. The future of Nuclear in Canada: the SMR revolution

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Today, as Canada pursues a major initiative to decarbonise its economy, the need for new generation capacity to meet ever growing electricity demand has created new opportunities for nuclear new-build. On December 2nd, 2021 OPG announced that it had selected General Electric (GE) - Hitachi as its technology partner for the Darlington New Nuclear Project (DNNP) with the construction of the first grid-scale Small Modular Reactor (SMR) in the Western World, the BWRX-300. The Darlington site is already licensed for new build and site preparation is already underway with in-service expected by the end of the decade. A successful project will open the door for the construction of many more units in Canada. Currently the provinces of Alberta, Saskatchewan, Ontario and New Brunswick are actively involved in evaluation of the option of building SMRs in their jurisdictions. Building on the Canadian Roadmap for Small Modular Reactors the four provinces published a Strategic Plan for the Deployment of Small Modular Reactors in March 2022.
OPG is also actively involved in the development of small scale SMRs for remote communities and industrial applications. Global First Power was established in partnership with Ultra Safe Nuclear Corporation (USNC) and is pursuing the construction of a demonstration Micro Modular Reactor (MMR) project at the Chalk River Laboratories in Canada.

This presentation provides an overview of the Canadian SMR program as viewed from the perspective of Ontario Power Generation and its subsidiary Laurentis Energy Partners (Laurentis). Laurentis is able to leverage the 8 years of SMR planning and learnings undertaken to date by OPG to assist other countries and operators on their SMR journeys and is already involved in multiple projects in Canada and Europe.

I.1.3. The SEALER Programme to Commercialize LFRs in Sweden

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LeadCold develops SEALER-55, a small modular lead-cooled reactor, intended for on-grid commercial power production. In order to demonstrate the technology a joint venture between LeadCold and Uniper has been established (Swedish Modular Reactors AB), having the purpose to design, licence and construct a lead-cooled research and demonstration reactor in Oskarshamn for operation in 2030.
In support of this objective, an R&D programme is carried out, with funding from private and public sources. In this contribution I describe the components of this programme, including the ongoing SUNRISE, ASGARD, BOREALIS, SOLSTICE, and FREDMANS projects. These feature, inter alia, development of laser welding techniques, liquid metal embrittlement, fretting, and irradiation testing of alumina forming steels, a steam generator manufacturing test, the construction and operation of a lead-pump test facility as well as an electrically heated LFR prototype, and qualification of methods for synthesis of uranium nitride fuel.
Moreover, I present the main design characteristics of the SEALER-E facility, the SEALER-D and SEALER-55 reactor units, and the associated timeline for building nuclear fuel and modular reactor manufacturing plants.
Finally, I provide a cost estimate of the programme and outline a business plan for commercial deployment of standardised SEALER-55 units in Sweden and elsewhere.

I.1.4. CIRCE-THETIS Facility CFD simulation: Steady State and Transient Compliance

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CIRCE (CIRColazione Eutectico) is a large scale LBE pool-type facility operated by ENEA (Italy), supporting the development of the Liquid Metal Fast Reactors (LMFR), in particular of ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator) design, representative of the LFR GEN-IV power plants concept.
During several H2020 past projects, various experimental campaigns have been performed in the CIRCE facility, supported by thermal-hydraulics numerical models, in order to provide data for pool thermal-hydraulics and to generate databases for CFD models validation.
The ongoing H2020 PATRICIA project foresees the installation in the CIRCE facility of a new test section, named THETIS. This configuration includes an innovative Helical-Coil Steam Generator (HCSG) and a new main circulation mechanical pump. Investigation of the transition from forced to natural circulation and between natural circulation modes under different and competitive heat removal mechanisms is planned in the experimental campaign. One of the tasks in PATRICIA is to build a CFD model of the test section able to simulate both steady states and experimental transients.
The complete CFD model of the CIRCE-THETIS test section is described in this paper. The main components, including the external insulations and the vessel cooling system are illustrated. The numerical strategy is described. The influence of the thermal radiation wherever relevant is highlighted. Different thermal boundary conditions are implemented. The model is brought to a

satisfactory steady state condition. The nominal steady-state pre-test simulation numerical results are presented. The heat losses are evaluated under fixed thermal boundary conditions.

I.1.5. Review of the Experimental Studies on the Behavior of Volatile Elements from Lead and LBE

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One issue of the lead technology is represented by the behaviour of the impurities produced in the nuclear system that can interfere with the normal operation, one of the most important processes involving impurities being the production by fission or activation of hazardous radioactive elements that volatilize from the liquid metal. Such processes have an important role in the assessment of radioactivity released under normal or accidental conditions, and thus in the licensing process of the nuclear facility. In the past years, a lot of efforts have been devoted to the thoroughly understanding of the fundamental processes and to obtain reliable physicochemical data that can be applied in the prediction of the release of radioactivity in normal or accidental conditions.
This paper presents a review of the most recent experimental studies on the behaviour of volatile elements with a focus on the most hazardous ones, such as polonium and mercury. In the experimental studies on evaporation of volatile elements from molten metals different relations of the Henry constants were obtained by different authors. These relationships have been used in the estimation of the vapour pressure of the volatile elements in the ALFRED reactor based on the evaluated molar fraction of Po and Hg in the whole lead mass after 5 years of irradiation (the fuel maximum residence time in the reactor). Moreover, the paper presents the research studies conducted in the past years on the adsorbent materials that can bind the Po and Hg in the cover gas above the molten metal and can be further used in the development of efficient filters.
Also, a perspective on the experimental infrastructure needed to investigate the main phenomena related to severe accidents in LFRs is provided.

I.1.6. Study on the Development of a System Concept for the Chemical Cleaning of Residual Lead from the Structural Materials Used in LFR Reactors

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The use of lead as a primary coolant is one of the most attractive options for LFR systems with several advantages in terms of safe operation, reliability, physical protection and security. An important issue would be the removal of residual lead from structural materials used in LFR reactors. Although initially, the material is in the liquid lead, when it is removed from the melt an amount of lead adheres to the surface of the material and solidifies instantly due to the temperature difference. The rapid hardening is due to the relatively high temperature at which the lead begins to solidify (<327°C) when the components are removed from the molten lead (450-500°C) and get in touch with the ambient temperature. The paper presents a preliminary investigation performed based on the specialized literature addressing the removal of the residual lead from the metallic surfaces after their extraction from the molten lead. A cleaning matrix containing the chemical composition of the cleaning solutions that can be used in these situations without affecting the metal surface has been established. Lead removal tests were performed on metal samples taken from the Instron machine used for mechanical testing in liquid lead. The tests performed allowed the selection of an optimal solution to be further used for lead cleaning. Moreover, the metallographic and SEM analyses confirmed the efficiency of the selected solution.

I.1.7. Conceptual Design and Layout of Air-Cooling Condensers (ACC) Required As Heat Sink For Alfred LFR Demonstrator Reactor

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The paper presents the conceptual design of Air-Cooling Condenser using latest technologies patented by Evapco-BLCT in 2018. In 2018, Evapco-BLCT from the USA, "Advanced Technology ACC" has brought to market the "no CoreTM" heat exchanger which represents a new, innovative technology that goes beyond the limits of heat transfer in dry cooling applications.
The "no CoreTM" heat exchanger (condenser) greatly improves heat transfer coefficients over a wide range of operating conditions to reduce surface area requirements and reduce material costs. The advanced ACC technology is centred on a modular design concept.
This new generation technical cooling solution has also been considered for conceptual design of the LFR ALFRED demonstration reactor secondary circuit cooling. The use of this cooling system can significantly speed up the approval of building permits for a power plant, as water use is no more an issue. Shortening a project schedule, for example by six months, can completely change the budget of a project leading to significant savings.
The proposed cooling system uses air-cooled condenser technology with a "V" shaped frame, in which the fans are located at the top of the ACC, and by introducing a deflector in the condenser, a significant improvement in cooling performance is achieved when ambient temperatures are high. These types of ACC exchangers provide a maximum surface area on the footprint for greater heat dissipation while providing efficient dry cooling capabilities.
The disadvantage compared to water-cooled towers is that the electricity consumption for driving the fan motors is higher than in the case of water cooling.

The recovering of the cost difference due to electricity consumption in the case of air cooling could be ensured by the no costs for water consumption, sewage, chemical treatments, pump station, equipment and installation costs (ACCs are modular and have fewer components, thus a cost advantage in installation).
The main concern related to this air-cooling condenser regards the operating conditions during Romanian hot summer, but EVAPCO stated that the system was tested in power plants in warmer condition, as in Mexico. The main advantage of the ACC is that the cooling power plant water demand is greatly reduced, namely by an order of magnitude.

I.2. Nuclear Safety & Severe Accidents

I.2.1. New Solutions for Improving of Mixed Catalytic Packing for Heavy Water Detritiation

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The isotopic exchange between hydrogen and liquid water (LPCE) is one of the most efficient process for tritium removal from nuclear effluents, produced in fission and fusion reactors. In order to increase the efficiency of isotopic exchange process, the chemical exchange is usually associated with conventional distillation process so that, the tritium is transferred from tritiated water to the hydrogen gas, through the following consecutive steps:

$$HTO(L) + H2O(V) \rightarrow HTO(V) + H2O(L) \quad (1)$$

$$HTO(V) + H2(G) \rightarrow HT(G) + H2O(V) \quad (2)$$

$$HTO(L) + H2(G) \rightarrow HT(G) + H2O(L) \quad (3)$$

The chemical exchange (reaction 2) becomes effective only in presence of hydrophobic catalyst which repels the liquid water but allows to water vapour and hydrogen gas to reach the active catalytic centre and to accelerate the isotopic transfer process. On the other hand, the process (1) is a conventional water distillation process and need an efficient contact element (hydrophilic packing) in order to increase the contact area between water vapour and liquid water. The mixture consisting of catalyst and hydrophilic packing it's called "catalytic mixed packing" and play the key role in LPCE process.

This paper presents the most significant aspects concerning the manufacture and improvements of various types of mixed catalytic packing, manufactured at tested by authors. The research efforts have been focused on the improvement of the main characteristics of selected Pt/C/ PTFE catalyst (surface area; pores volume; metallic area; platinum particle size etc.) and on the increasing of the wettability of hydrophilic packing. Two procedures, based on chemical and thermal treatments has been applied and tested at normal condition and at increased temperature. The hold-up; wire size, roughness of wire and SEM image have been evaluated and discussed as main parameters of treated packing in deep correlation with the wettability of packing. The results showed as the treatment based on solution B, at increased temperature it is a promising idea and can be applied for the new improved mixed catalytic packing. This type of mixed catalytic packing has been proposed to equip the isotopic exchange columns of the future Tritium Removal Facility from Cernavoda Nuclear Power Plant. Such type of mixed catalytic packing, based on hydrophobic catalyst, makes feasible and more efficient many chemical processes in which the liquid water or humidity are present as reactant or as reaction product.

I.2.2. Evaluation of the Unavailability of the Primary Circuit of TRIGA SSR Reactor, Importance Factors, Risk Criteria Regarding its Components

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PSA - The probabilistic safety assessment analysis is a tool for improving a nuclear installation and its safety regarding the design, construction, operation and management phase. It can also be used successfully in the phases of improvement of the testing and maintenance activities of the nuclear installation. PSA differs from traditional deterministic analysis by the fact that it provides a methodical way of identifying the sequences that results from a wide range of initiating events concerning an accident and it includes systematic and realistic determination of accident frequencies and their consequences. This paper is an analysis to evaluate the unavailability of the TRIGA SSR reactor primary circuit, by obtaining the value of the system unavailability, the main contributors leading to system failure and the calculation of risk importance factors. Two cases were considered, first one does not consider CCF (Common cause failures) in the developed fault tree, and the second case considers CCF on pumps and heat exchangers lines. The purpose of those cases was to see how they influence the unavailability of the primary system. In order to identify the components of the system with a strong impact on the risk (probability of system failure) two importance factors were considered, namely: FV (Fussell-Vesely) and RAW (Risk Achievement Worth). Those factors consider the high impact of failure components on system unavailability and they are used to improve maintenance (preventive and inspection) activities as well as testing. The code used for this analysis is the EDFT code, which is part of the PSAMAN code package, developed in Institute for Nuclear Research Pitesti.

I.2.3. Enhancement of Nuclear Safety and Security in Romania – Improvement of Disaster Resilience and Preparedness for Radiological and Nuclear Events

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The project "Enhancement of Nuclear Safety and Security in Romania – Improvement of Disaster Resilience and Preparedness for Radiological and Nuclear Events" has started in 2019, being carried out in partnership with the Norwegian Radiation and Nuclear Safety Authority and the International Atomic Energy Agency, National Commission for Nuclear Activities Control, CNCAN, as Project Promoter, together with the national partners in Romania, General Inspectorate of the Romanian Gendarmerie and the Ministry of Internal Affairs, as Project Operator.
The project is designed to address many of the challenges faced presently by CNCAN, such as helping the organization to review, update and develop a significant number of regulatory documents, provide the necessary training to CNCAN staff based on the train-the-trainer and other approaches and acquire the equipment, software, and hardware needed.
The project will contribute to strengthening the bilateral cooperation with the donor state by consolidating the exchange of experience between the Romanian and the Norwegian competent authorities, including through the sharing of good practices and the joint participation in training events and exercises.
The project is focused on safe nuclear power operations and safe use of radiation sources with the fundamental aim to avoid any incidents but also on emergency preparedness and response, should an incident occur. The main obtained results will be presented in paper.
The main contributions to be achieved in the project are the improvement of the regulatory framework, processes, practices and training of staff in the field of nuclear safety, security and emergency preparedness and response, as well as protection against ionizing radiation.

I.2.4. Romanian Regulatory Developments Relevant to Nuclear Safety

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The paper presents the most recent regulatory developments in Romania, in the area of nuclear safety. New revisions have been issued by the National Commission for Nuclear Activities Control, CNCAN, in the period 2021 – 2022, such as nuclear safety requirements on ageing management for nuclear installations (NSN-17 rev.1), on event reporting and analysis and on the use of operating experience feedback for nuclear installations (NSN-18 rev.1), on the training, qualification and authorization of nuclear installations personnel with nuclear safety related jobs (NSN-23 rev.1) and on the use of standards for the continuous evaluation and improvement of nuclear safety for power plants (NSN-27 rev.1). In 2021, CNCAN has also issued the revisions of the regulation regarding the protection of nuclear installations against cyber threats (NSC-01 rev.1) and the regulations on physical protection in the nuclear field (NPF-01 rev.1). As well, a new regulation was developed regarding the use of standards for the design, implementation, evaluation and continuous improvement of physical protection systems for nuclear installations, nuclear materials and other related ionizing radiation sources (NPF-04).

In accordance with the provisions of the Law no. 111/1996, CNCAN has the responsibility for reviewing the regulations whenever it is necessary for these to be consistent with international standards and with relevant international legislation in the domain, and for establishing the measures for the application thereof.

Various sources of information relevant for updating the system of regulations and guides are used, including the development of international legislation and safety standards, international cooperation, feedback from the industry and feedback from CNCAN inspectors based on their experience with the enforcement of the regulations, the results of research and development activities.

The paper highlights the most important aspects introduced by the revised and new regulations, taking account of the regulatory and operational experience and the development of international nuclear safety standards.

I.3. Nuclear reactors and nuclear fuels

I.3.1. Evaluation of Spent Fuel Inventory and Radioactivity for CANDU type Fuel Bundles with Increased Number of Elements and U-based Fuels

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Present work aimed to evaluate spent fuel inventory (fissile material, and isotopes of interest for proliferation resistance) and radioactivity (total value in fuel and radioactivity on components - actinides and fission products radioactivity) for some fuel bundle projects with increased number of elements comparatively with the 37-element CANDU standard fuel bundle, C37, with natural uranium fuel, considered as reference. Four CANDU type fuel bundle projects developed by the Nuclear Fuel Performances Group at RATEN ICN have been studied, namely: C43 and CN43 fuel bundles (43 elements), C52 fuel bundle (52 elements), and C61 fuel bundle (61 elements), respectively. The analyses were performed considering natural uranium fuel, and several slightly enriched uranium (SEU) and reprocessed uranium (RU) fuels compositions. From the nuclear safety point of view, a particular interest was in registering the effect of increasing U^{235} enrichment of SEU or RU fuels on spent fuel inventory and radioactivity, envisaging the possibility of developing accident tolerant fuels. Fuel irradiation was simulated by means of ORIGEN-S code included in SCALE6 programme package, assuming identical fuel composition in all fuel bundle elements. Fuel burnup at the end of irradiation (EOI) was of 8 MWd/kgHE (heavy element) for natural uranium fuel, and up to 14 MWd/kgHE for SEU or RU fuels, respectively, with respect to the U^{235} enrichment of the fuel. The parameters of interest were comparatively analyzed against the CANDU standard fuel bundle ones (C37 with natural uranium fuel). Comparison of the same fuel bundle project with different fuel compositions and of different fuel bundle projects with the same fuel composition was also performed. Fissile isotopes inventory (U^{235} şi Pu^{239}) for C37 fuel bundle was greater than the ones corresponding to fuel bundle projects with increased number of elements. At EOI, the inventory of isotopes interesting for proliferation resistance (Pu^{238}, Pu^{240}, Pu^{242}) produced in fuel bundles with increased number of elements was greater than the one characterizing the reference fuel bundle. Regardless the fuel composition (natural uranium, SEU or RU fuel) CANDU standard fuel bundle total radioactivity was greater than the ones correponding to the fuel bundle projects with increased number of elements.

I.3.2. Reactor Physics Study for Advanced Fuel Cycle Options to be used in CANDU Reactors

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In this paper a Reactor Physics study is performed in order to investigate the possibility of using some advanced fuel cycle options in actual CANDU reactors. The fuel cycle options taken into consideration are: Slightly Enriched Uranium (SEU1.1%), Recycled Uranium (RU0.9%), Mixed Oxide Fuel (MOX)-a variant of 3% Plutonium with 1% Gadolinium as Burnable Absorber (BA) in a Silicon Carbide matrix, and three Thorium-based options, with and without BA, in both uniform and mixed CANDU core approaches. Additionally, a sort of Accident Tolerant Fuel (ATF) based on Uranium nitrides was also evaluated with regard to the burnup and general safety performances. Lattice cell calculations were firstly performed in order to generate the cross section tables with respect to the burnup. The next step consisted in performing of core calculations in the Time-Average (TA) approximation using 3D CANDU core models for the DIREN computer code. Different refuelling schemes were used (2, 4 and 2&4 bundle shift), according to the initial bundle fissile content and the maximum bundle powers supplied by every fuel configuration. As the lattice

input parameters (fuel, coolant and moderator temperatures) were used at their average values over the reactor core, the TA approximation is further considered to be fair for such a prospective study envisaging core parameters of interest like: the average discharge burnup, the maximum channel and bundle power, the reference power shape and the maximum linear powers derived using Power Peaking Factors (PPFs) from lattice calculations. The before mentioned parameters values were commented and comparatively presented in tables and figures. Concluding, the advanced fuel cycle options can double the average discharge burnup of the Natural Uranium (NU) fuel while preserving or just improving safety features like the Coolant Void Reactivity (CVR), also illustrated throughout the burnup intervals.

I.3.3. Study of a Loss-of- Cooling Accident at the Spent Fuel Pool of a CANDU NPP containing Natural Uranium or SEU Spent Fuel

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The paper addresses an important aspect of nuclear safety related to the spent fuel management after it was discharged from the reactor core and sent for intermediate temporary wet storage inside the spent fuel pool. A Loss-of Cooling Accident scenario for a spent fuel pool (SFP) belonging to a CANDU NPP was considered for analysis. The cooling of the spent fuel pool is assured, under normal circumstances, by using demineralised water that is circulated thorough a dedicated cooling system. In case of a loss-of-cooling accident (which could occur mainly due to a long-term black-out), the temperature of the demineralised water in the pool cannot be kept anymore in a limited interval. The temperature rising will primarily lead to the boiling of the water in spent fuel pool (after a time interval $t_{boiling}$ with respect to the very first moment of the accident considered as t=0). If the black-out continues, due to water evaporation, after the time $t_{dewatering}$ (considered also with respect to the moment t=0 when the loss-of-cooling accident started) the most likely to be exposed spent fuel assemblies/ elements are the ones positioned at the highest level, on the vertical axis, in SFP. These spent fuel assemblies will be exposed to the atmosphere above the level of the boiling water. The goal of this work was to evaluate, using a dedicated Visual Basic 6.0 code, the parameters $t_{boiling}$ and $t_{dewatering}$ characterizing the loss-of-cooling accident that occurs at the SFP filled with 37-element CANDU spent fuel bundles, for various types of fuel (either natural uranium fuel or SEU, slightly enriched uranium, fuel). The considered U^{235} enrichments of the SEU fuel ranged between 0.90% and 1.50%. The spent fuel residual thermal powers were estimated by using the ORIGEN-S module, included in the software package SCALE4. The considered burn-up for the spent fuel before the moment of discharge was 8000 MWd/tU, which is typical for the ”standard” operation of a CANDU6 reactor (as U1 and U2 from Cernavoda CANDU NPP).

I.3.4. Nuclear Safety Aspects associated with Increasing of U235 Enrichment of CANDU Standard Bundles with SEU and RU Fuel

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The paper presents several results of a comparative neutronic analysis on main parameters of interest at cell level calculations, for CANDU fuel bundles with 37 elements and U-based fuel, considering the increasing in U235 enrichment. Two modified fuel bundle projects (C37M1 and C37M2) have been analized in comparison with the standard CANDU fuel bundle design, C37. The compositions considered in this paper were natural uranium (UNAT), slightly enriched uranium (SEU) and recovered uranium (RU), respectively. Elementary cell calculations were performed by means of WIMS and DRAGON codes, using the IAEA's nuclear data library on 172 energy groups. C37M1 and C37M2 fuel bundle projects geometry differs from the one of C37 standard fuel bundle design by a reduced diameter of the central element with 16% and 20%, respectively. Fourteen fuel compositions were selected for analysis, namely: natural uranium - standard composition (0.71% enriched in U235), 4 SEU compositions (enrichment in U235 ranged between 0.9 % and 1.2%), and 9 RU compositions (enrichment in U235 ranged from 0.9% to 1.3%). The paper aimed to evaluate the neutronic parameters of interest for economic considerations (discharge maximum burnup, uniform power distribution in the bundle, U235 consumption to produce a unit of energy), and for nuclear safety ensuring (void reactivity coefficient, Doppler effect, resistance to proliferation based on estimation of minor actinides amount resulting from the fuel burnup). According to both codes, RU and SEU fuel compositions led to higher values for the maximum burnup at discharge by up to 53% comparatively with the value coresponding to C37_UNAT. The highest value was obtained for C37_RU fuel bundles, with 1.3% enrichment in U235. WIMS code estimations were systematically 5% higher than the DRAGON code ones. Regardless the fuel composition, the fuel bundles with modified geometry compared to the standard one does not bring a significant improvement in terms of the burnup uniform distribution. To improve this parameter, fuel compositions with differentiated U235 enrichment on the fuel bundle rings should be used. The lowest U235 consumption was obtained for CANDU standard fuel bundle. SEU and RU fuel compositions led to increasing of U235 consumption by up to 8.8% (SEU fuels) and 11.2% (RU fuels), respectively. Increasing of U235 enrichment in SEU and RU fuels has led to higher burnup comparatively with the reference value, decreasing of plutonium amont (produced by fuel burnup) and reduction of the void coefficient. From the nuclear safety point of view, these aspects support the possibility of using the studied fuels on the development of accident tolerant fuel projects.

I.4. Nuclear Technology and Materials

I.4.1. The Microstructural Investigations of the Liquid Metal Embrittlement Phenomenon on 316L Specimens Tested in the Liquid Lead Environment

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The “liquid metal embrittlement” phenomenon (LME) that can occur in some Generation IV reactor configurations is a complex phenomenon that strongly depends on the specific solid-liquid contact. This phenomenon can lead to the embrittlement of some metals, which are ductile in the air and can become brittle in long contact with certain liquid metals. The paper presents the results of the investigations performed on an atlas of metallographic and SEM analyses which was developed on the samples resulting from the experimental tests, to highlight the fracture characteristics induced by the lead embrittlement on 316L steel.
Fracture in engineering alloys can occur by a trans-granular (through the grains) or an intergranular (along the grain boundaries) fracture path. However, regardless of the fracture path, there are essentially only four principal fracture modes: dimple rupture, cleavage, fatigue, and de-cohesive rupture (ASM Handbook Volume 12, Fractography). Each of these modes has a characteristic fracture surface appearance and a mechanism or mechanisms by which the fracture propagates. The investigations were conducted on the specimens tested in the air and the liquid lead, through metallographic analyzes by optical microscopy and SEM analyses of fracture surfaces.
The metallographic aspects and SEM analyses confirm the propagation character of the cracking front. In principle, this can confirm the ductile, brittle or complex nature of the cross-section fracture of the samples. In this way, a complete and detailed picture of the fracture modes for samples tested in air and lead has been obtained.
Based on the investigation performed, it could be concluded that embrittlement effect on the 316L steel tested in the liquid lead for the temperature range 350°C – 400°C cannot be neglected, especially at low deformation rates.

I.4.2. ACTINIS: Shielded SIMS for Analysis of Highly Radioactive Samples

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Offering extreme sensitivity, high spatial resolution together with high throughput, dynamic SIMS (Secondary Ion Mass Spectrometry) proves extremely useful for a wide range of nuclear science applications. Derived from the field-proven CAMECA IMS 7f-Geo system, ACTINIS is designed to perform high precision elemental and isotopic analyses of highly radioactive samples in a safe environment. It is integrated in a set of biological protections and includes specific adaptations to minimize manual operations such as a full-security sample introduction system. ACTINIS benefits from the close collaboration between CAMECA and CEA Cadarache (France), and from improvements of technical design based on the experience from the first generation shielded instrument (IMS 6fR installed at CEA). ACTINIS offers depth profiling with excellent detection limits (ppb to ppm) and high depth resolution; elemental & isotopic information ranging from low mass (H) to high mass species (Pu and beyond); as well as unique sub-micrometer resolution 2D and 3D imaging capabilities. Studies performed with SIMS on irradiated nuclear fuel focus on three main axes:

1) the nuclear reactions occurring during in core irradiation which are characterized with isotopic ratio measurements,

2) the physical and chemical behavior of fission products which is evidenced by isotopic mapping,
3) the characterization of fission gases which is carried out through depth profiling measurements. Different applications covered by ACTINIS for irradiated fuel analysis will be presented.

I.4.3. Multilayer Feedforward Neural Network Modeling of the Fracture Mechanics Parameters for the Zr-2.5%Nb Pressure Tube

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CANDU reactor uses Zr-2.5Nb alloy pressure tubes as the primary coolant containment. Fracture toughness properties of the pressure tubes are required for evaluation of fracture initiation and leak-before-break. This paper presents Multilayer Feedforward Neural Network (MFNN) modelling of the fracture mechanics parameters for the CANDU pressure tube Zr-2.5%Nb alloy. The database for the MFNN model used an experimental study on the effects of hydride morphology and test temperature on axial fracture toughness of a cold-worked, unirradiated Zr-2.5Nb pressure tube. The specimens were prepared from one tube section which contained as-received hydrogen concentration and another section with 70 ppm hydrogen. The hydride morphologies were characterized with a parameter referred to as the hydride continuity coefficient (HCC), which provided a measure of the extent to which the hydrides were reoriented concerning the applied stress direction. Partially reoriented hydrides with HCC between 0.3-0.4 was formed under the stress and temperature cycles used to precipitate the hydrides. J-R curves characterized the fracture behaviour of the specimens tested at five different temperatures: 25ºC (room temperature), 100ºC, 150ºC, 200ºC and 250ºC. The Multilayer Feedforward Neural Network model was optimisation and implemented in the MATLAB environment. For two fracture mechanics parameters, namely stress intensity factor at maximum load Kml, and J-integral at maximum load, Jml, were obtained the standalone equations as a function of the following parameters: testing temperature, HHC factor, yield stress, ultimate tensile strength. The predicted results employing the MFNN model were compared with the experimental ones, and a good agreement is obtained. The paper was prepared in the framework of research activities carried out in the Fuel Channel Research programme, focused on the structural integrity assessments of the CANDU fuel channels.

I.4.4. Fatigue Behaviour Of Zy-4 Cladding Under Cyclic Loads Using Finite Element Modelling

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Choosing the proper structural materials plays an important role in the evaluation of fuel bundle integrity. Zircaloy alloys have been selected as nuclear materials due to their excellent physicochemical properties such as: low thermal neutron absorption cross section, reasonable mechanical properties at high temperature, high thermal conductibility and excellent corrosion resistance under operating conditions. During reactor normal operation power fluctuations emerge and the fuel element sheath is subjected to cyclic loads. These power variations are associated with power ramps that occur during refuelling, reactor shutdowns, reactivity inserts, and so on. A series of studies available in literature emphasize the appearance of material fatigue phenomenon in such situations. The aim of this paper is to analyse the fatigue behaviour of Zy-4 samples worked from CANDU C43 fuel rods "as received" metallurgical state, subjected to cyclic loads using finite element modelling. The fatigue analyses of C43 rods were performed by means of

reliable and highly used computer programme in nuclear industry, ANSYS 2020R2. In this respect, fatigue behaviour simulations of Zy-4 C43 samples, in which the number of finite elements was varied with the purpose of highlighting the quality and precision of computations, were performed. Low cycle fatigue simulations have been carried out under total strain amplitudes 1%, 2% and 3% at room temperature, while in order to determine the lifetime of Zy-4 the Strain-Life approach was used. A comparative analysis between the simulations results and the experimental data was performed and a good agreement for the simulations results with refined mesh comparatively with the experimental data was obtained.

I.4.5. Experimental Investigation on Hydrogen Absorption Properties of as-cast Zirconium rich U-Zr Alloy, for Use in Hydrogen Storage Applications

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The development of advanced hydrogen storage technologies is a subject of great actuality, interest and also, a necessity for the transition to a sustainable low carbon society. In search of alternative energy sources, hydrogen appears as a promising, renewable and environmental friendly option for energy transport and storage, but also for many other everyday applications.
The research for new hydrogen storage materials is essentially to promote the development of future hydrogen energy applications. Metal hydrides are among the most common materials used for hydrogen storage and represent nowadays an extremely important research field. Even if a large number of materials were investigated in order to use them for hydrogen storage, intensive research efforts are still needed for the development of functional storage systems, in order to find materials fulfilling all the requirements imposed for the efficient storage of hydrogen. The high absorption/desorption temperatures, slow kinetics and degradation upon successive absorption/desorption cycles are the main problems which needs to be solved for hydrogen storage applications.
In this context, the present paper aimed to investigate the hydrogenation properties of an uranium alloy, searching for new materials for hydrogen storage. The material used for this study was an U - 70 wt. % Zr alloy, prepared in the Institute for Nuclear Research Pitesti laboratories, using the powder metallurgy route; its composition, structure, morphology and thermal properties were investigated using different techniques: DRX and SEM analyses, optical microscopy, DSC.
The hydrogen absorption properties of the alloy were investigated using pressure-composition-isotherm method, at two temperatures: 260°C (the hydrogenation temperature of U) and 750°C (the hydrogenation temperature of Zr). The quantity of absorbed hydrogen and the disintegration degree were also evaluated.

I.4.6. The Usefulness of International Collaboration Focused on the Systematic Ageing Management for Nuclear Power Plants

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Effective ageing management throughout the service life of Systems, Structures and Components (SSCs) requires the use of a systematic approach to managing ageing that provides a framework for coordinating all programs and activities relating to the understanding, detection, monitoring, control and mitigation of ageing effects of the plant components or structures, including maintenance, in-service inspection, testing and surveillance, as well as operations, technical support programs (including the analysis of any ageing effects and degradation mechanisms) and external programs such as research and development.

The current NPPs have been designed for 30 to 40 years of operation. There is evidence that ageing in some cases has been underestimated during the original design, construction and commissioning or has not been accurately taken into account during operation. It has also been recognized that the ageing of the plants must be assessed, and an effective management strategy developed in a timely manner, to ensure the necessary technical basis for maintaining safety margins throughout the plant operation.
Many IAEA Member States have already taken actions to address the topic of ageing in their nuclear power plants. In 2009, the IAEA conducted a Technical Meeting where Member States recommended establishing an international platform for discussion between regulators and utilities regarding implementation of acceptable Ageing Management Programs (AMPs). The recommendations were to:
- Develop and maintain a document which can serve as a practical guide for implementing, maintaining and improving AMPs, made up of best practices and universal knowledge on proven AMPs for safety related SSCs;
- Establish a common basis for discussion between regulators and utilities with regard to implementation of acceptable AMPs.

In response, the IAEA commenced the International Ageing Lessons Learned (IGALL) program.
Romania, together with 25 other IAEA member states, has contributed to the achievement of the IGALL objectives since 2016. Specialists from Cernavoda Nuclear Power Plant and Institute for Nuclear Research Pitesti were involved in the IGALL Working Groups: WG1 Mechanical Components, WG2 Electrical and Instrumentation and Control Components (I&C) and WG3 Civil Structures.
The paper describes the structure of IAEA IGALL Program and the contribution of the participants in the development of a common internationally agreed basis on what constitutes acceptable AMPs, as well as on ageing management knowledge base for design of new plants, design reviews, safety reviews (such as periodic safety review), etc., and serves as a roadmap to available information on ageing management.
The IAEA IGALL program assures that information contained in the IGALL safety report will be kept updated and will create an international network for continuous discussion and development of AMPs and Time Limited Ageing Analyses (TLAAs) as recommended tools to manage ageing.

I.4.7. Study of Mechanical Fatigue on the Zr-2.5%Nb Pressure Tube Specimens

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In-service inspections of CANDU Zr-2.5Nb pressure tubes may reveal blunt flaws that include debris fretting flaws (DFF), fuel bundle bearing pad fretting flaws (BPFF), mechanical damage flaws and crevice corrosion marks. Although the flaws are not an immediate threat to the tube integrity, a fatigue crack may initiate flaws due to cyclic loads. Fatigue crack initiation must be accounted for in design and during service when blunt flaws are dispositioned. For this reason, it is necessary to evaluate the flaw for fatigue crack initiation. The CSA Standard N 285.8 contain acceptance criteria to evaluate DFF and BPFF type of flaws

The paper presents a methodology of the mechanical fatigue tests performed in RATEN ICN on the Zr-2.5%Nb pressure tube specimens containing complex flaws with a given depth. The specimens are the dog-bone type with complex flaws and hydrogen concentrations exceeding terminal solid solubility. The paper describes the experimental procedure with the following main steps: specimens preparing (formation of notch-tip hydrides), mechanical fatigue tests and data processing. Hydride formation involves the hydrogen migration under thermal cycling and stress gradient, hydrides precipitation and reorientation in the region of the stress spots. The mechanical fatigue test for this study consists of cycling loading with a given mechanical load amplitude, up to the fatigue crack initiation. The lifetime is determined at the mechanical fatigue stress by a Basquin type relation, represented graphically as fatigue crack initiation curve in the number of cycles & mechanical stress coordinates. The Basquin equation was obtained by fitting the experimental fatigue points, having the coordinates (number of cycles & amplitude of the equivalent stress Von Mises).

The results can be used for the structural integrity analyses of pressure tubes in CANDU fuel channels subject to periodic inspections.

I.4.8. Fuelling Machine Head Testing Loop - ADAM Control And Data Acquisition System

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The Fuelling Machine Head is a mechanical robot controlled by a hydraulic driving which performs operations required in a complete refuelling cycle, but also different fuel channel interventions. Prior to the integration in the CANDU Fuel Handling System, the Fuelling Machine Head must be the subject of several validation tests, to determine concordance with working conditions in the CANDU Nuclear Reactor. The Institute for Nuclear Research Fuelling Machine Head Testing Loop located in the Out of Pile Testing Department allows the simulation of hydraulic parameters like pressure, flow and temperature similar to those existing in a normal operation fuelling/refuelling cycle. The Testing Rig was completed in 1989 and was used between 2000-2005 to evaluate two of the Cernavoda Nuclear Power Plant Fuelling Machine Heads.
Throughout 2018-2021 were made several upgrades based on the programmable logic controller Advantech ADAM in the Fuelling Machine Testing Loop Control Room in order to increase operating safety and achieving a better parameter monitoring. This system created allows real-time process monitoring using the touchscreen display and different types of data visualisation like bar graph, real-time trend graphs or numerical display on the Testing Rig user-interface, but also signalling critical thresholds and test limits. Another purpose for the so-called ADAM System is real-time recording of process parameters and alarms via Microsoft Excel. Additionally, ADAM System controls part of SCC Heat Exchanger subsystem parameters, using the on-screen operated PID regulators.
ADAM System was used for data monitoring and recording during the 2021 operating campaigns of the Fuelling Machine Testing Rig, all the data from the testing period was recorded and written in ".xls" file continuously at every 10 seconds. Operating the Fuelling Machine Testing Rig using the ADAM System offers an easier overall monitoring of the process parameters on the HMI interface, the alarms display allowing quicker means of locating sensors or other equipment malfunctions. Additionally, for flexibility, the system was created to allow its extension if required.

I.4.9. Fuelling Machine Head Testing Loop - Control Room Safety Increase Through Equipment Upgrading

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The Fuelling Machine Head Testing Rig consists of several technological installations: the Fuelling Machine Head Testing Loop which supplies 4 fuel channels with pressurized water, the Hydraulic Water and Oil Supply System used to control the Fuelling Machine Head actuators, the Carriage for positioning, centering and locking the Fuelling Machine Head onto the fuel channel, Control Room and Process Computer and an individual RAM Assembly Testing Rig.
The Control Room allows obtaining and maintaining required test parameters such as pressure, flow and temperature in the test Pressure Channels. Following the evaluations made on the existing installation, it was decided to create a modern control and signaling system of the Fuelling Machine Head Testing Loop by replacing the physically worn and morally obsolete components, with forthwith benefits on safety in operation.
A series of electric and electronic components were studied to comply with the existing installation assuring in the same time that the characteristics will be maintained according with the default

project. To eliminate existing faults, new sketches of control panels and electronic components were designed using Autodesk Inventor software.
Each item and electrical component used to control the automation installation of Fuelling Machine Head Testing Loop has been tested to ensure a normal operation mode even under a constant working flow.
The contemporary process will be accomplished by the effective achievement of the automation subsystems that will be used in the new control and signaling system. The new system will be tested with installation in stand-by mode and also during Hot Loop/Cold Loop campaigns, in order to rectify any possible inaccuracy in operation, if necessary.

I.4.10. Spectral Analysis of Acoustic Signals Occurring at a Cracked Pipe Crossed by a Pressure Fluid

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The basic idea from which the work started is that when a crack appears on a pipe crossed by a pressure coolant, acoustic vibrations (sound signals) are generated in the pipe environment as well as mechanical vibrations in the body of the pipe.
Acoustic signals are generally analog signals that appear randomly. Spectral analysis of this type of complex signal brings information that can be used in certain applications of interest such as monitoring, detection, location of a acoustic signal.
The authors used a mechanical device to simulate the appearance of a crack in a pipe that allows the passage of the high pressure and high temperature water flow and containing a hole that may have a variable diameter. Passing high pressure hot water through the device will cause the appearance of the sound signal that simulates the cracking of the pipe. Using a unidirectional microphone, the acoustic signal was acquired and processed by the Bruel & Kjaer spectral measure and analysis chain. Following spectral processing and analysis, the acoustic signal in the time domain is converted into acoustic signal in the frequency domain. Spectral analyses of type FFT (Fast Fourier Transformer) and CPB (Constant Percentage Bandwidth) was carried out by applying the Fourier Transform.
The work presents the FFT and CPB frequency spectra obtained during experimental tests to simulate the appearance of a circular crack with a diameter of 1.03 mm on a hot water pipe under pressure.
From the measurement of the amplitudes of the frequency spectra before and after simulating the occurrence of a crack, the authors concluded that the detection of the acoustic signal is possible by monitoring the sound pressure level values obtained by the microphone on the entire area of measurement. By using several microphones spread in the area to be monitored, the location of the crack can also be achieved by phase / frequencies correlations of the acoustic signals acquired from each microphone and the distance at which they are located from each other.

I.4.11. Induction Heating Process Modeling of Nuclear Fuel Rod

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The paper presents a numerical study of high-temperature induction heating in vacuum beyond the melting point of the U-$ZrH_{1.6}$ nuclear fuel rod containing 45% weight of uranium, with enrichment up to 20%. The fuel rod and crucible are placed inside a tubular vacuum chamber and the inductor coil is placed outside of the vacuum chamber. The induction heating computer modelling is useful for process design and optimization, including coil design. The induction device modelling and numerical simulation of induction heating have been performed with FLUX2D software which uses the finite element method for electromagnetic and thermal calculation.

The information obtained from the induction heating simulation provides input design data for the experimental process of inductive heating up to melting the fuel rod, as inductor shape, number of turns, positioning of inductor in relation to fuel rod to be heated by induction, as well as the estimation of the electrical parameters (voltage, current density of the inductor) required to reach temperature values around 1760 °C. Metallic materials used in the numerical study are isotropic materials without phase transformation at Currie point and characterized by a magnetization law which is independent on application direction of magnetic field.

Numerical simulations used the electrical parameters of an induction heating installation existent at Institute for Nuclear Research (INR) Pitesti namely a solid-state frequency converter with output of 8 kHz and a rated power of 20 kW.

The simulation procedure is verified and calibrated against several induction heating experiments on single-material parts with simple geometry, with steady electric parameters at the induction generator.

UZrH melting study is intended to yield feasible options for enriched uranium recovery from waste resulted during fuel rods fabrication in INR and also for sintered fuel densification, as parts of the project for future refuelling of the Romanian TRIGA SSR-14MW.

I.4.12. Embedded Web Server for Industrial Automation Using Programmable Logic Controller

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In the present era of automated industrial process, there is a great need of fully automated industrial system. Modern automation technology increasingly integrates IT technologies together with an integrated Ethernet-based communication, providing a direct access to the automation system via local network (intranet). This automation systems to be fully integrated it needs to have an embedded web server to allow them to be self-dependent of the platform operating system, by using web browser as an interface between user and automated industrial system. A web server is software and hardware that uses HTTP (Hypertext Transfer Protocol) and other protocols to respond to client requests made over a network. The function of a web server is to display web page content to users through storing and processing. The aim of the project presented in this paper is to integrate devices capable of displaying web page (PC, Smartphones, tablet), to access a Programmable Logic Controller (PLC) within the local network. These devices are connected via network router using the TCP IP in a wired connection or using a wireless connection on the local network. PLC has a webserver technology implemented that stores useful information and is accessed via a html page designed to control and monitor the process. This paper presents a project that is used to monitor and control the status of industrial and nuclear equipment's connected to PLC. For implementation of this project the HyperText Markup Language (HTML) programming language is used due to wide compatibility with embedded systems. The user can monitor and control the status of the process via local network from any location of the plant.

I.4.13. Design of a Manufacturing Execution System (MES) Used for LEU Type Nuclear Fuel Manufacturing Process

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The manufacturing process of the LEU fuel used in TRIGA reactors is well known and fully technically developed. Due to the aging of industrial equipment used in the manufacturing process, there is a need to upgrade and replace them with newer models that have intelligent functions and which can be used to develop and implement a Manufacturing Execution System (MES). This MES control system manages the Distributed Control System (DCS) or Supervisory Control And Data Acquisition (SCADA) applications to achieve good planning, assembly, and continuity of production regarding LEU-type nuclear fuel elements. This article proposes the development of an MES system for the manufacturing process of LEU-type fuel elements. As the fuel manufacturing chain implemented at the Nuclear Research Institute will undergo the modernization process, based on the information presented in this article, such a system may be developed and implemented in the near future. The paper will present in detail the architecture of the proposed MES system and a partially developed structure in this regard. Siemens PLC systems from the S7-1200 family will be used to present the proposed concept.

I.4.14. Location of an Acoustic Signal Source in the Two-Dimensional Plane by the Estimated Delay Time Method

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The location of a fixed source of acoustic signal of unknown position, based on spectral analysis of the signal received from several microphones distributed within an area, generates particular practical applications. One of these could be the detection and localisation of (micro) cracks occurring in the pipes from technology installation or distribution / cooling system that moving pressurised fluids on the basis of the specific noise caused by the escape of air or liquid from the cracked pipe.
The work proposes to experiment with the location in a two-dimensional plane of a fixed sound signal source whose coordinates are unknown, based on the estimated delay time method (TDE), using Bruel & Kjaer's acquisition and spectral analysis system and three unidirectional microphones.
The location method is based on the determination of the delay time of a signal emitted by a fixed sound source, which will be received by two microphones placed at a known distance from each other. The source emitting the audible signal has an unknown position which will have to be determined. The signal will first be received by the microphone nearest to the signal source and then by the second microphone. The delay time resulting from the receipt of the signal by the two microphones is constant and determined solely by the speed of the sound sign in the air and by the distance between the signal source and each microphone. It is important to say that the two microphones form a pair that will generate a delay time specific to that pair.
Experimental tests involved generating an audible signal by a signal source and acquiring it from the three microphones that are connected to Bruel & Kjaer's acquisition and spectral analysis system. Pulse LabShop software records and makes an FFT spectral analysis (Fast Fourier Transformer) on signals acquiring from microphones. FFT spectral analysis of sound signals makes it possible to calculate the cross-corelation function between the signals of two microphones forming a pair. The cross-corelation function measures the degree of similarity between two functions: a (t) and b (t), the latter one having a shift in time with the value of δ. The maximum cross-corelation function is the delay time between the two microphones.
The results obtained can determine the coordinates of the signal source with fairly good accuracy. A monitoring system may be designed to detect and locate cracks in fluid-flow piping systems. A practical application could be to locate and detect the occurrence of a crack for a reactor's primary coolant system, where strong radiation conditions do not allow human presence for a long time and where it is vital to locate a crack from a distance.

I.4.15. Embedded System Using Microcontroller for Gamma Radiation Detection

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This paper presents a modern solution for the realization of a microcontroller computing system that works together with a detection assembly for measuring radioisotope activity.
The development of digital signal processing has allowed increasing the number of simultaneous acquisitions by realizing complex measurement systems, supervised by microcontrollers or digital signal processors.
The use of microcontroller-based systems can be a viable solution both in terms of competitiveness and in terms of coupling to new technologies. In this work, it was decided to use an Atmega 328-PU microcontroller, in order to work together with the signal processing module within the detection assembly for measuring the activity of radionuclides.
The signals provided by the detection assembly are in the form of a voltage pulse train, the main function of the microcontroller being to measure the pulse frequency. Also, in addition to this function, the microcontroller must also perform data processing, display the result and the unit of measurement.
The measurements made with these instruments are performed automatically, and the user has the possibility to add new functions through the program or to modify the way of presenting the results.
The fields of measurement for the activity of gamma radiation emitting sources are within the requirements imposed for this type of equipment, by international standards.
The microcontroller computing system for measuring gamma radiation is designed to offer the possibility of measuring, monitoring and acquiring data without depending on a computer, which makes the equipment stand-alone and also very easy to use. In order to measure the range of radioactive activity, it will be necessary to use only the portable electronic instrument and one of the two ionization chamber type radiation detection assemblies with well (AD-A-01 and AD-A-02).

I.4.16. CFD Model For The Double Ice Plugging Process Of A Dn 200 Pipe

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The ice plugging (pipe freezing) technique is primarily implemented for maintenance or repairing activities on the water cooling circuits of nuclear power plants, but can also be useful in other industrial installations, residential pipelines, etc.); its main advantage is that the maintenance or repair activities can be carried out without shutting down the entire plant or draining the entire installation. Although the operation usually requires forming a single ice plug, in some cases (large pipe diameter or high water temperature) multiple ice plugs are required.
Perfecting the operation of a system or a technological process often involves the analysis of several physical phenomena as well as the interactions between them. Optimizing the technique of pipe freezing using multiple ice plugs requires understanding the flow characteristics and temperature distributions during the process. Computational fluid dynamics (CFD) has been used for the present study; a 3D double ice plugging process simulation has been conducted using ANSYS Fluent. The results are useful for further research and for better understanding heat transfer analysis of the pipe freezing process.
This paper proposes a computational model for the heat transfer during the operation of forming two controlled ice plugs in a horizontal pipeline in order to isolate it from the rest of the circuit. The simulation results can be used to calculate the total cost of the operation regarding the total downtime of the installation, and the liquid nitrogen consumption. In some cases, the solution of double pipe freezing may be more advantageous in order to reduce the installation downtime and can thus provide important cost savings. The CFD model should also be used for further research regarding the design of the freezing devices and also streamlining the multi-ice plugging process

by determining the optimal installation distances for the devices on the outer wall of the pipes considering their nominal diameter and also the initial water temperature.

I.4.17. Corrosion Susceptibility Assessment of Chromium Nitride Thin Layers Applied on Stainless Steels

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Austenitic stainless steels are considered a class of materials suitable for use in supercritical water-cooled reactors (SCWRs) due to their high resistance to oxidation, creep and good radiation performance. Although these types of steels have a high oxidation resistance, they are susceptible to intergranular corrosion and appearing of secondary phases (e.g sigma phase) under aggressive conditions like supercritical water (550°C, 250 atm). Therefore, one of the ways to improve the performances of these alloys is the application on the surface of thin ceramic layers by various deposition techniques.
The aim of this paper is to present results obtained after characterization of the chromium nitride thin layers applied on the surface of 310 H stainless steel (SS) using the thermionic vacuum arc (TVA) method. The surfaces of coated samples have been analysed using techniques as scanning electron microscopy (SEM), energy dispersive spectroscopy (EDX) and X-ray diffraction (XRD). In parallel, using electrochemical methods (linear potentiodynamic polarization and electrochemical impedance spectroscopy) have been assessed the general corrosion susceptibility of coated samples, respectively the protective character of the ceramic layers.
According to the corrosion parameters from the potentiodynamic polarization tests, the chromium nitride coated 310 H SS sample showed smaller corrosion susceptibility than the uncoated sample. In the same time, the EIS analysis with Bode and Nyquist plots, confirms that the chromium nitride coated 310 H SS substrates are more protective, being in concordance with the polarization tests results. Because the electrochemical tests were performed at room temperatures, in the next steps the samples will be tested in supercritical conditions to confirm the protective character of chromium nitride layers.

I.4.18. Oxygen control system using gas phase in molten lead corrosion test facility

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This paper describes an acquisition and control method for the liquid lead corrosion test facility using the gas phase control method. A graphical interface has been developed in the LabView environment for the acquisition and processing of process parameters, which are taken over by the Compact DaQ equipment from National Instruments. The interface with the operator offers the possibility of graphical monitoring of the values coming from the sensors located in the corrosion testing installation. The process parameters are acquired from the oxygen and temperature sensors of the test facility. The signal coming from the oxygen sensor is measured in volts, in the range of 0-5V, being acquired with a precision on 12 bits / sample. The application controls the oxygen concentration using gases which could be a combination of Argon + 4% Hydrogen or Argon, respectively. The user interface displays the values provided by the sensors (oxygen,

temperature) and the useful process parameters (errors, system status, completion of certain actions). Also, the process values from the sensors are acquired, processed and stored for offline analysis. The application also allows, in addition to the control of the oxygen concentration and the communication with the temperature controller through the ModBus RTU protocol, for the heating of the lead vessel. Finally, it is desired to control the oxygen concentration (wt%) and to maintain it at a desired value in order to avoid the oxidation of the materials to be tested.

I.4.19. Plasma Electrolytic Oxidation of Titanium and its Surface Chemistry Investigation

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The research in the field of structural materials in order to improve their properties is of real interest and continuously evolving. Titanium belongs to an advanced clad materials category having the technology readiness level (TRL) 1. Thus, a significant level of further development of this material is required before commercial use is possible. In this context, the development of environmentally friendly surface engineering techniques is still a great challenge. 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 present study focuses on the development of titanium oxide/phosphate films on commercially pure titanium substrate by plasma electrolytic oxidation and assessment of their surface chemistry at the metal-electrolyte interface. Post-deposition X-ray photoelectron spectroscopy analysis of the samples highlighted the successful formation of a surface coating consisting of a mixed oxide, containing both titanium oxide and titanium phosphate, as a result of an applied potentiostatic regime in NaH_2PO_4 electrolyte during the PEO process. These results, obtained in the first layers of the surface, are of great interest for understanding the chemical behavior of titanium-based materials in oxidizing environments, important for generation IV nuclear energy systems.

I.4.20. The Influence of Electrolyte Composition and Current Regime on the Micro-Discharges of Plasma Electrolytic Oxidation on Zr-2.5Nb and Titanium Alloys

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The research in the field of structural materials in order to improve their properties is of real interest and constantly evolving, so the development of the environmentally friendly surface engineering technology is still a major challenge. One of these new surface techniques is Plasma Electrolytic Oxidation (*PEO*). Over the last years many *PEO* studies have been performed on zirconium and titanium alloys, so in this paper we study the *PEO* process in parallel on the two substrates, in order to develop the skills to use this technique.
The plasma electrolysis tests were performed on titanium grade 2 and on *Zr-2.5Nb* samples with a programmable *DC* power supply, on the pulsed galvanostatic mode. The applied current density and the t_{on} value were kept constant, while t_{off} values between pulses were varied. Two types of aqueous electrolytes were used, namely $NaAlO_2$:Na_3PO_4:KOH and Na_2SiO_3:KOH:$NaAlO_2$, in different compositions. The *Voltage-time* anodization curves were recorded. The duration and rate of the four characteristic stages of the *PEO* process were correlated with the visual micro-discharges, like intensity, spatial density and color characteristics. One can observe that the presence of *KOH* under the tested conditions accelerates the *PEO* process on titanium, while on Zr-2,5Nb it slows down the *PEO* process. The presence of aluminate does not favor the *PEO* process on titanium; in its absence the process takes place successfully. On the other side, aluminate improves the evolution of the *PEO* process on *Zr-2,5Nb*.

It has been determined that the precence of the KOH, as well as aluminate have different influence depending on the substrate. The work contibutes to the development of new materials, with special properties, suitable for use in extreme conditions.

I.4.21. The Pulsed Current Mode and Current Density Influences on Plasma Electrolytic Oxidation of Zr-2.5Nb and Ti Alloys

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This paper is part of the efforts performed in acquiring the knowledge and the skills necessary to use the Plasma Electrolysis Oxidation (PEO) technique in order to improve the wear and corrosion resistances of Zr and Ti alloys used in extreme conditions. The aim of this work is to identify the influence of pulse parameters and current density on the evolution of the discharge events taking place during the PEO treatment of Zr-2.5Nb and Titanium grade 2 alloys. PEO tests were performed in an aqueous solution based on $NaAlO_2$, Na_3PO_4 and KOH, in galvanostatic mode, using a programmable DC power supply. By analyzing the Voltage-time anodization curves the influence of pulse parameters and current density on the PEO processes was evaluated. The PEO process is accelerated on Ti alloy by increasing the pulse duration (t_{on}) or the current density. The increase of the current density has the same effect on Zr-2.5Nb alloy. The increase of the time between pulses (t_{off}) slows down the PEO process on both alloys. The pulsed current mode influences the duration of the process and the intensity of the micro-discharges. On Zr-2.5Nb alloy the stages of the PEO process are much shorter and more intense, the time required to obtain the coating being much shorter than on grade 2 Ti alloy.

II. ENVIRONMENTAL PROTECTION

II.1. Radioactive Waste Management

II.1.1. The Management of Intermediate Level Radioactive Waste Generated during Cernavoda Unit 1 Retubing Project

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The design life of a CANDU reactor is about 30 years at 80% average capacity factor. By replacing the major reactor components including fuel channels, briefly by retubing, the life of a CANDU reactor could be extended with 25 – 30 years of operation. Cernavoda NPP Unit 1 retubing will generate significant quantities of radioactive waste including the replaced metallic component of the reactor. Partly this waste is classified as Intermediate Level Waste – Long Life (ILW-LL), and require adequate methods for collecting, packaging and interim storage before the disposal in an underground facility. This paper presents a synthesis of the main aspects related to the management of Intermediate Level Waste Long Life, ILW-LL, produced during Cernavoda NPP Unit 1 refurbishment and the proposed conceptual solutions to fulfil the legal provisions for nuclear and radiological safety of personnel, population and environment.

II.1.2. Radiological Safety Evaluation of Recycling Facility for the Dismantled Concrete Waste

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The spatial dose distribution and exposure dose of workers was analysed for the radiological safety assessment of the recycling facility. When decommissioning a nuclear power plant, a large amount of radioactive waste will be generated. It must be disposed of in a safe and efficient way. About 20% of radioactive waste is concrete waste. Among them, the bio-shield is expected to be highly activated compared to others due to continuous radiation by neutron flux. Thus, the spatial dose distribution of the recycling facility and external exposure of workers must be performed when recycling or disposing of bio-shield concrete. This study is a preliminary evaluation for the construction of a recycling facility for the dismantled concrete structures. In this study, the radiation dose was simulated using VISIPLAN software, the radiation dose evaluation code, in order to simulate the spatial dose distribution and the external exposure of radiation workers.
Through the VISIPALN code, the spatial dose distribution of the facility and exposure dose of workers was calculated to determine whether the dose limit of workers is satisfied. The cases were divided into two categories according to the specific radioactivity of the concrete waste. Case 1 was set when the part closest to the nuclear reactor was cut vertically, and case 2 was set when the concrete waste exceeded the allowable concentration for self-disposal. The maximum spatial dose of case 1 was 1.4E-02 mSv/hr and that of case 2 was 8.2E-03 mSv/hr.
According to the Enforcement Regulations on Radiation Safety Management, workers must not exceed 100 mSv for 5 years within the range not exceeding 50 mSv per year. It was assumed that one worker for each equipment worked for 8 hours a day, 5 days a week for 1 year. The annual maximum dose limit was 17.16 mSv/yr for case 1 and 9.82 mSv/yr for case 2. In both cases, it was confirmed that the dose limit was not exceeded. The dose evaluation on the work process of the concrete recycling facility is expected to be applied for the actual management of worker's radiation safety.

II.1.3. Uncertainty Quantification Applied to the Radiological Characterization of Radioactive Waste

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The characterization of radioactive waste supposes the establishment of the list of radionuclides together with their activities. The estimated activity levels are compared with the limits given by the national authority in the field. Therefore, quantifying the uncertainty affecting the radionuclide concentration is essential to estimate the acceptability of waste for the final disposal, but also to control the segregation and radiological characterization phases.
This study presents methodologies used to estimate the uncertainties of hard to measure and easy to measure radionuclides in radioactive waste packages.
The scaling factor method is an alternative method widely used to evaluate the activity of hard to measure radionuclides. The method is based on the correlation between easy to measure radionuclides called key radionuclides and hard to measure radionuclides. The activity of hard to measure radionuclides evaluated using the scaling factors method has an uncertainty that depend on the physical parameters of the radioactive waste package and on the gamma spectrometric measurements performed to evaluate the activity of key radionuclides.
Based on the experience and lessons learned from decommissioning of the VVR-V nuclear research reactor from Magurele, this study will led to the improvement of methodologies used to estimate the activity of hard to measure radionuclides and establish the strategy for quantifying the associated uncertainties. The results have a major impact in the radiological characterization of the waste generated in the decommissioning of nuclear installations, because they cover many important aspects in assessing the uncertainties of radionuclides encountered in radioactive waste. These methodologies can be used in radioactive waste management within the Department of Radioactive Waste Management (DMDR), IFIN-HH.

II.1.4. Modelling of Cement Hydration using PHREEQC Code

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Modelling the hydration process of cement-based materials is a very useful method to predict, understand and describe the behaviour of this very complex process which is crucial for undertaking the long-term performance and service life predictions of cement-based materials.
Based on PHREEQC geochemical software capacity to perform thermodynamic equilibrium calculations by solving the law of mass action (LMA) and using an appropriate thermodynamic database, hydration products formation and precipitation were thermodynamically predicted for a commercial cement type using its oxide composition in powder form.
To build the model which simulates the phase hydration kinetic and phase assemblage over time, the method developed by Parrott and Killoh was used and integrated in PHREEQC. This method is a mathematical representation of the three main processes involved in clinker dissolution and hydration products formation: nucleation and growth, diffusion and formation of a hydration shell.
Furthermore, for a comparative study PHREEQC and GEMS softwares were used and the results are in good agreement.
The results show that PHREEQC can reasonably accurately predict the mass of fully hydrated products, clinker phases kinetic hydration and phase accumulation, proving that PHREEQC has the potential to be a very powerful tool that with a proper programming may be used to predict the behaviour and durability of cement based materials in various environments, a very important aspect for demonstrating the safety function associated with cementitious engineered barriers in a radioactive waste repository.

II.1.5. Effect of Organics and Cement Degradation on Ni-63 Solubility in Cement Pore Waters

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Cementitious materials are widely used in deep underground repositories for radioactive wastes as construction materials in the engineering barrier system and also as waste conditioning matrices. Radionuclide migration depends, among other, on the interaction of dissolved radionuclides with backfill materials and organic matter. In order to predict the radionuclide release into the far field, their concentrations in cement porewaters are of critical importance.
In cement based repositories for radioactive waste, nickel isotopes are considered to have a safety relevant role in performance assessment. ^{63}Ni, a pure beta emitter, is an activation product generated as a result of neutron activation of stable ^{62}Ni present in structural components of reactor vessels. The presence of organic materials in cement pore waters has the potential to influence radionuclide solubility and also influence the performance of the disposal system.
In this study, the solubility of nickel in cement pore waters has been investigated in an experiment under conditions corresponding to the initial stage of cement degradation, state III of cement degradation and for both degradation states, the influence of organic molecules has been assessed. The organics considered in this study is formic acid, released in the disposal environment by spent ion exchange resins degradation.
The paper presents the results of a solubility test using nickel solutions labelled with ^{63}Ni. The test has been carried out to assess the solubility limit of nickel in cement pore waters and the influence of formic acid. Cement pore waters studied in this work have different composition corresponding to different degradation states of cement: state I – the alkali content and pH (~ 13.30) have higher values and state III – governed by calcium depletion of the CSH gel and presenting a lower pH (~ 11.30). The formic acid concentration selected in the system has been 10^{-3} M and the total nickel concentration in the experiment ranged from 10^{-8} M – 10^{-5} M. The results obtained show that the measured solubility of nickel is lower than predicted by most thermodynamic data.

II.1.6. Assessment of Gamma Dose Rate for Waste Packages with Radioactive Concentrate using MicroShield Software

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MicroShield is a useful computer program with a wide range of applications including designing shields for photon and gamma rays, dose assessment, and minimizing exposure of the workers. The basic concept of MicroShield program is Point Kernel deterministic method which involves dividing the volume of the source into a large number of small point sources and each contribution of point sources are summed up.
In nuclear activities the applicability of the MicroShield program is very useful in assessment of dose rate as well as in radiological characterization of different radioactive sealed source and conditioned radioactive waste packages for their final disposal.
The purpose of this paper is to verify the applicability of MicroShield program in the assessment of gamma dose rate for drums with radioactive concentrate resulted by evaporation of β-γ liquid radioactive waste generated by TRIGA reactor.
Inputs of data for each radioactive concentrate drums were created to assess the dose rate using MicroShield. These data inputs contain information about source dimensions, coordinates of the points for dose assessment, density of shielding materials, and radiological characteristics of the

drums to be analysed. The activity concentration of gamma radionuclides in the drums with radioactive concentrate was determined by gamma measurements of representative samples using an HPGe detector. The dose rate at contact and one meter distance from the analysed drums was measured using a Berthold radiometer.
The discrepancy between the dose rate values obtained with MicroShield and experimental measurements are ranging between 1% and 35%, and the modeling and experimental values have similar linear trends. The results obtained by MicroShield simulations depend of the accuracy of the input data and therefore the data characteristics of the source and the configuration geometry are extremely important in the mathematical modelling. On the other hand, the experimental measurements are influenced by counting statistics. Considered the uncertainties associated both with modeling and experimental measurements, the discrepancies between experimental and modeling data are considered acceptable.
The results obtained show that MicroShield program is a rapid and useful program for dose rate assessment, limiting the exposure time of the working personnel. Based on nuclide vector, the dose rates estimated by modeling can be used to evaluate the radioactivity of the waste drums.

II.1.7. Site selection criteria for the deep geological repository in Romania

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Romania's deep geological repository (DGR) is planned to be operational around 2055 and it is intended to dispose of the spent nuclear fuel, high-level radioactive waste (HLW) and long – lived low and intermediate level radioactive waste (LILW-LL). A geological disposal system confines the radioactive waste disposed of and provides a unique and long-lasting level of protection for human and the environment. The radiological safety of this disposal facility is ensured by using a multi-barrier system which consists of the form and properties of the radioactive waste, the engineering barriers (such as the container in which the waste is placed, the buffer, the backfill, etc.) and the natural barrier provided by the host rock formation.
In order to be considered in the selection process, a site must meet a number of criteria, both technical and social. In an initial examination, based on the available information, the candidate sites must meet a minimum set of criteria such as having the dimensions necessary to accommodate surface and underground facilities, be outside of protected areas, not contain economically exploitable natural resources, not located in a seismic zone, etc. Sites that do not meet these initial criteria will be excluded from the more detailed evaluation process, and those that meet the criteria will be investigated through a series of scientific and technical studies.
In this paper, a documentary analysis of geological formations at national scale was performed. Using the experience of countries with advanced geological disposal programs, several categories of criteria have been proposed to be included in the site selection strategy to be applied in Romania to select the host rock and site for DGR. The selected criteria reflect the best knowledge and experience of the advanced programs and are comparable to the criteria used by them, as well as to the recommendations of international agencies such as IAEA (International Atomic Energy Agency) and NEA (Nuclear Energy Agency).

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

II.2.1. Individual Dosimetry Program at CNE Cernavoda NPP – Good practices in the Dosimetry Lab

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Cernavoda NPP has two CANDU 600 reactors in commercial operation, first since December 1996 and the second one since November 2007. For a CANDU reactor the major contributor (90%) to the external dose is gamma radiation. The major contributor to the internal dose of professionally exposed workers is the tritiated heavy water (DTO) – up to 30% of the total effective dose. The main purpose of Individual Dosimetry Program is to implement a "Monitoring, Evaluation and Recording of Individual Doses Program" and ensure that all the exposures are kept ALARA. Dosimetry Lab's mission is to ensure measurement, assign and record all the significant radiation doses (Hp(10), Hp(0.07)) received by an individual during activities performed at Cernavoda NPP. For all the persons entering radiological controlled areas (Cernavoda NPP employees, short-term atomic radiation workers, contractors and visitors) Dosimetry Lab provides individual (external and internal) dosimetric surveillance. At Cernavoda NPP, Individual dose monitoring is provided by Dosimetry Lab which is a licensed dosimetry service, approved by the Romanian regulatory body, National Commission for Nuclear Activities Control (CNCAN). Dosimetry Lab provides dosimetry services for measurement, evaluation and recording of all significant ionizing radiation doses, using good equipment and good practices, by having 25 years of experience in this field. Cernavoda NPP's Dose Monitoring, Evaluation and Recording program is based on the latest ICRP recommendations and also, on the EU requirements, national laws and regulations.

II.2.2. Exposure Evaluation within Um-Safi Mine (Egypt) due to Radionuclides determined by High-Resolution Gamma-ray Spectrometry

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The natural radioactivity of bedrocks of Um Safi mine, Egypt, is investigated. The mine is a part of the Arabian-Nubian Shield, being situated on the territory of Egypt. Following the IAEA protocol, TECDOC-1415, 20 samples were collected from the pre-determined locations which are represented by serpentinite talc carbonate rocks, basic to intermediate metavolcanics, and related volcanoclastics, acidic metavolcanics, and related volcanoclastics, metagabbro–diorite complex, and older granitoid. The radiometric measurements were done using high-resolution HPGe spectrometer at RATEN-ICN Romania. The gamma-emitting nuclides library has been created to assess the activity concentration of radionuclides belonging to the natural series of uranium, and thorium, including also potassium-40. The true coincidence summing corrections have been applied by using the Genie-2000 software, and LabSOCS calculated total efficiencies. The descriptive statistics have been performed, and the obtained concentrations reflect the high radioactivity of Uranium and Thorium in the investigated area according to the UNSCEAR, ICRP, and IAEA recommendations threshold. The radiological hazards have been calculated by using EPA modelling coefficients. The obtained data are considered as baseline data for characterizing the Um-Safi mine in terms of the environmental radioactivity and protection. Furthermore, the resultant measurements help in detecting any change in the radioactive background level due to the geological processes, and to achieve radiation protection strategies for occupationally exposed persons.

II.2.3 Assessment of Radiation Dosimetry and Excess Cancer Risk due to Terrestrial Radionuclides for Al-Zubair Petroleum Station (Iraq)

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The radioactivity distribution pattern has been estimated for naturally occurring radioactive materials due to exploration and deep drilling in the Al-Zubair petroleum station from Iraq. The investigated samples were collected from Al-Zubair station and their surrounding areas concerning extraction locations of petroleum ores and storage places of NORM residues. The radionuclides library has been designed to involve the natural uranium and thorium decay series. Based on the measured activity concentrations, the radiation doses were evaluated by applying the conversion dose coefficients recommended from IAEA, ICRP, and UNSCEAR. During the drilling process, the samples collected from the surface until the depth 2.5 km approximately. For the surface samples till the samples at the depth of 1 km, the resulting values are significantly normal compared to the standard references given by the Environmental Protection Agency and IAEA in this regard. The overall mean value of the measured activity concentrations of ^{226}Ra, ^{232}Th, and ^{40}K are: 3627.66, 192.92, and 630.89 Bq/kg, respectively. Going deeply (> 1 km), the soil sludge from the drilling accumulated in the workspace of the station. It is close to the petroleum layer and it contains an abundance of NORM. The corresponding calculated dose indicated a non-permissible level, so strong safety and security precautions must be applied to protect the personnel from the radiation hazards. To furnish the data of radiation protection and human safety, integral organ doses and the excess lifetime cancer risk have been calculated in order to make recommendations for the adequate safety principles to occupational radiation protection programs and perspective on the assessment of environmental NORM risk. This study is a part of the master thesis of the first author who is working in Iraqi Ministry of Health.

II.2.4. On Radioactivity in Timis

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Although there are no mining works in Timis County, we found areas with high radioactivity. Among the radioactive sources, an important presence is radon, a noble gas that comes from the Earth's crust and is identified in: soil and rocks, surface and deep water, natural gas, but also inside buildings. Radiation from the direct decay of radon and its offspring greatly affects the health of the population. We place the existence of these radiations on the presence of some anomalies in the perimeter of the tectonic faults in the investigated area, the relative displacement of which causes seismic movements. These, in turn, accelerate the emission of radioactive gases from the Earth's crust. The relationship between earthquakes and radon emissions has been studied in many countries (Italy, Japan, USA). We made measurements in several locations / areas (Sacosu Mare, Darova, Gataia) about which we had information that there were such emanations, on which occasion we found radiometric values higher than the values specified in the CNCAN regulations†. The results of the measurements led us to draw some conclusions: (i) it is necessary to make a correct map of radon emissions; (ii) radiometric measurements must be supplemented with heavy metal measurements; (iii) it is necessary to apply specific regulations in constructions (ventilation paths, construction authorizations in areas with high radioactive activity) and for construction materials; (iv) continuous assessment of the radon level in risk areas and medical institutions

located in such areas; (v) increase the awareness of the population who live in areas with high radioactivity and identify measures to reduce its harmful effects on health.

II.2.5. Radiation Protection and Climate Change: How Do the Dose Response Models Influence the Choices of Solutions to Mitigate Climate Problems

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It has become more apparent, even for the most sceptical, that the climate change is real and does present a serious challenge for our infrastructure and our ways of life. Science has also reached a rather broad consensus that one way to mitigate the climate change means to reduce the CO2 footprint of our society. As nobody is really willing to return to a pre-industrial era (and this would not help much, anyway) and our need for energy production increases, some very serious decisions have to be taken regarding on the ways in which we generate energy. Fossil fuels are by far the most polluting (and not only with CO2) so this basically leaves us with three main choices: nuclear, hydro and renewables.
Some countries show an extremely strong opposition to nuclear, bordering the irrational, due to the low understanding about how nuclear industry works, what are the safety standards and, most of all, of the real effects of radiation on the human health.
The model used right now for all risk estimations and thus for communicating with the public is the LNT model. Besides the fact that, by now, there is an important body of evidence that suggest that this model is highly inaccurate, the message conveyed by this model is quite threatening for the general public: Every dose carries a risk. The presentation explores the way in which the perception of risk generated by the LNT gives shapes the public response to nuclear energy production. A short study across various media seems to indicate a wide overestimation of the dangers posed by the nuclear industry, a low knowledge of the real environmental impact of the renewables and the underestimation of the risk that a nuclear phase out will pose for the environment The work also presents some ideas about lines of communication with the public in such a way that people will have a better understanding of the real dangers of the nuclear industry (rather that over- or underestimate them) and how does the CO2 footprint of a nuclear power plant compares to the CO2 footprint of wind turbines and the solar panels.

II.2.6. Improving analytical performance through proficiency testing activities

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One of the most important parts of environmental monitoring program worldwide is the measurement of radionuclides activity concentrations in order to be able to assess the human and biota exposure due to radiation. An important tool for external quality control that provides opportunity to improve analytical skills and to validate the techniques in the field of radioactivity measurements is the participation in proficiency tests (PTs), being at the same time an essential requirement of ISO 17025 standard.
The aim of this paper is to present the obtained results of Radiation Protection, Environmental Protection and Civil Protection Laboratory (LRPMPC) from the latest five environmental radioactivity PTs, organized by IAEA ALMERA. ALMERA is a network of Analytical Laboratories for the Measurement of Environmental Radioactivity coordinated by IAEA. The network aims to enhance the capability and performance of participating laboratories with a view to providing timely and reliable measurement results in emergency situations.

Yearly, ALMERA provides to each member laboratory a set of PT samples and the member laboratories have to conduct the radioactivity measurements and report the results. All analysis results are collected and interpreted. Each result should pass accuracy and precision test to be awarded the status “Accepted”, otherwise it lies on the status “Warning” or “Not Accepted”. Thereby, the PTs exercises allow the laboratory to improve the adopted analytical methods, through the corrective actions implemented for unacceptable results.
The results of the presented proficiency test exercises organized by IAEA ALMERA network between 2017 and 2021 are satisfactory, more than 80% of the reported results being accepted.

II.2.7. Management System Regarding the Release of Gaseous Effluents in Nuclear Installations

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The use of computerized systems for acquisition, processing and storage offers essential advantages such as: high-capacity databases, flexibility, high reliability, acquisition speed, high processing power. In addition to these performances, computer systems offer the possibility of communication, which allows connection in local computer networks or over large areas. Integration capability is an essential advantage of modern systems. Databases are an essential component of a modern radiation protection system. Their main functions are: storage, protection and security, sorting and presentation of data.
The paper presents the stages regarding the development and implementation of a data management system related to gaseous effluents from nuclear installations. The management system is based on data collection, manipulation and centralization in a protected database structure. The Oracle APEX 11G development environment was used to ensure high performance, robustness and data integrity. The paper presents the architecture of the implemented database and various functions used for the interaction and manipulation of stored data. Information management is done through a modern interface using high-level web page objects.

III. SUSTAINABLE DEVELOPMENT

III.1. Education, Training and Knowledge Transfer

III.1.1. Augmented Cooperation in Education and Training in Nuclear and Radiochemistry – The A-CINCH Project

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European Commission, identified a loss of the younger generation's interest for specialized nuclear knowledge and a risk that the current workforce, progressively retiring, couldn't be replaced as one of the current main concerns in nuclear sector. As a consequence, under the EURATMOM call it was decided that the topic addressed by the A-CINCH project could tackle this issue. Highly educated personnel with very specific knowledge, skills and competences on nuclear radio-chemistry will be still required in the future regardless of the development of nuclear power sector in the EU, as either new builds, development of innovative and advanced reactors, long-term operations, shut-down, decommissioning, waste management and radiation protection all necessitate qualified staff. This attitude can also be seen at other industrial and medical applications making use of radionuclides and/or ionising radiation. This situation persists already since the turn of millennia when the OECD/Nuclear Energy Agency's report, "Nuclear Education and Training: Cause for Concern?" (2000), demonstrated that many nations are training too few scientists to meet the needs of their current and future nuclear industries and authorities and that the European educational skill base has become fragmented to a point where universities in many countries lack sufficient staff and equipment to provide education in all, but a few, nuclear areas.
A-CINCH aims to tackle these challenges by bringing nuclear-and-radiochemistry teaching to a new level: state-of the art innovative teaching methods will be developed, including 3D virtual reality courses and electronic distance learning. At the same time, hands-on-training courses in radiochemistry laboratories across Europe will be made available on a continuous base. A-CINCH will comprehensively include all the tools and the material produced by all the former CINCH-project series in a homogeneous virtual reality-based setup and will make all courses available conveniently by the e-Shop.
A-CINCH project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945301.

III.1.2. TOWARDS OPTIMIZED USE OF RESEARCH REACTORS IN EUROPE – THE TOURR PROJECT

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Europe has a broad and very diverse landscape of RRs, many of them since long time in operation, well maintained and regularly upgraded.

Nuclear research reactors (RR) have been constructed in countries implementing nuclear power plants and used in experiments necessary to develop commercial reactors and training programmes. Neutron irradiations techniques have found new applications in the adaption and production of existing and new materials, as well as medical radioisotopes. The latter enabled development of new diagnosis and treatment techniques, for the benefit of millions of people.
Yet a combination of declining interest and the absence of a sound financial model, led to closure of many RRs and a few others will close soon.
This negative scenario calls for a coordinated European action to assess the impact of the decreasing number of RRs, identify future needs (including new neutron sources), draw a roadmap for upgrade of the existing RR fleet, and a model for harmonized resource management.
TOURR project is a response to this challenge. Its primary objective is to develop a strategy for RR in Europe and prepare the ground for its implementation. This strategic goal can be divided into specific objectives: assessment of the current status of European RR fleet, including plans for upgrade; evaluation of urgent EU needs; developing tools for optimal use of RR fleet; rising awareness among decision makers on the (future) role of RRs. The ambition of TOURR project is to secure access and availability of RRs as a vital part of the European Research Area and to support stable supply of medical radioisotopes.
The TOURR project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945 269.

III.1.3. Training in COVID -19 pandemic context – challenges and opportunities

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The purpose of the paper is to present the challenges and the opportunities that Cernavoda NPP has encountered in the training activity from the beginning of the COVID-19 pandemic. The challenges were varying from changing the traditional classroom training to adapting the courses content for distance learning (e-learning and virtual training). With changed working conditions (working from home or limited number of employees physically present on site) and all the sanitary restriction imposed by the government and the organization to maintain the safe operation of the power plant, Cernavoda NPP had to adapt and develop new ways to meet the training requirements of the organization. The paper also presents the new approaches implemented to deliver training to our personnel and also to external personnel in order to maintain their qualifications and competencies for a safe operation of the plant. To adapt to the ‘new normal’, Cernavoda NPP caught the new training opportunities that aroused and transformed them in an added value for our organization. The training mix used by Cernavoda NPP from the beginning of the pandemic context proved to be beneficial to a large influx of new employees created by the generations change and the new projects that the company is currently involved in. The young generation employees responded favorable to the new mix of training methods and their integration was speeded up. In the conclusion section, the paper presents the key take-away for the lessons learned.

III.1.4. Nondestructive Examination Techniques on CANDU Fuel Elements

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During irradiation in nuclear reactor, fuel elements undergo dimensional and structural changes, and changes of surface conditions sheath as well, which can lead to damages and even loss of integrity.

One of the nondestructive examination techniques for CANDU fuel elements is represented by visual examination and photography in hot cells with the help of a periscope. Hence, macroscopic changes of the fuel element sheath (oxidation, cruel deposit, deposits of corrosion products, corrosion effects, deformities, defects, colour etc.) due to the irradiation conditions and manufacturing conditions as well are observed. Dimensional changes mainly consist in increased diameter, wrinkling, stretching, bending and ovalization of fuel element and are the result of swelling of the sheath and fuel-to-fuel sheath interaction induced by nuclear radiations. Unirradiated Zircaloy-4 tubes were used for calibration purposes, whereas irradiated Zircaloy-4 tubes were actually subjected to visual inspection and dimensional measurements. Another non-destructive technique, the control technique with eddy currents obtains information about the irradiated nuclear fuel sheath integrity or about the existence of defects produced by irradiation (cracks, holes, external and internal notches, changes of sheath wall thickness, inclusions, etc). The control equipment consists of a flaw detector with eddy currents, operable in the frequency range 10 Hz ÷ 10 MHz, and a differential probe. The calibration of the flaw detector is done using artificial defects (longitudinal, transversal, external and internal notches, bored and unbored holes) obtained on Zircaloy-4 tubes identical to those out of which the sheath of the CANDU fuel element is manufactured (having a diameter of 13.08 mm and a wall thickness of 0.4 mm). Measuring the oxide layer thickness, by eddy currents, formed on elements sheath is useful for characterizing the behaviour of fuel elements at high fuel burn-up or during a long-term storage. Eddy current technique has the advantage of allowing rapid measurement of Zirconium oxide layer thickness along the length of the fuel element, while the metallography on the sheath allows only local measurements. To measure the thickness of the oxide layer, a high frequency electromagnetic field is generated, that induces eddy currents in the substrate conductive material. The amplitude of these eddy currents depends on the distance between the probe coil and the substrate material. The measurement signal is produced by the variation of the impedance reflected in the probe coil by the eddy currents generated in the substrate material. The changes of the coil impedance depend on the distance between the coil of the control probe and the conductive substrate material and its dimensions.We present results of measurements done by eddy current techniques on Zircaloy- 4 tubes, unirradiated, but oxidized in an autoclave prior to examinations. These tubes were longitudinally scanned along three generators at 120˚ with respect to each other and also radially, on the circumference, with a step of 9˚. The measured thickness of the oxide layer varies between 1.83 µm and 2.96 µm, with an average of 2.29 µm. These results are in very good agreement with the oxide layer thickness values obtained by optical microscopy.

III.1.5. Determination of the Functional Parameters of the Hot Thermomechanical Loop During the Injection of Water into the Fuel Channel

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The hot loop is a technological installation component of the Re-fueling Machine Stand (RFM Stand), used for the preparation and hot water supply of a PHWR-600 type fuel channel called test section in order to achieve the parameters (flow, pressure, temperature) necessary for acceptance and pre-acceptance hot tests.
The work aims to perform hot tests of the RFM Stand, to verify its functional capacity. The hot loop in the RFM Test Stand supplies water to a PHWR-660 type fuel channel called the test section to simulate the conditions in the CANDU nuclear reactor after the onset of nuclear reactions during operation.
The hot loop is designed to supply hot water to the test sections of the working parameters required by the test specifications (flow rate Q = 23.49 kg / sec, working pressure P = 110 bar, working temperature T = 312˚C).
Within the technological installation, tests will be performed to verify its functional capacity at the working parameters imposed by the test specifications, and also the test of injection of water into the fuel channel (ST3 - test section 3) using the mechanical simulator (SM).

In order to verify the operational status of the technological installations, the functional tests with the hot loop of the RFM Stand, will follow the normal operating parameters and their evolutions.
To verify the operational status of the hot loop as well as the involved technological installations, the following tests shall be performed:
- checking the functional capacity of the hot loop in the RFM test stand;
- test of injection of water into the fuel channel (ST3) using the mechanical simulator (SM).

During the "Hot Loop" tests, the operating loop of the operating parameters specified in the test requirements shall be monitored. As a result the functional capacity of its RFM hot loop will be checked and its commissioning, bringing it into normal operating parameters and monitoring its operation over time will be monitored. In conclusion, functional problems were performed according to the PS-TH-70 procedure. The working parameters will be monitored to see if they adhere to the values specified in the test requirements. In addition, for a more accurate visualization of the pressures between the mechanical simulator and the terminal fitting of the fuel channel, a glycerin manometer (measuring range 0-250 bar) will be fitted at the ST3 outlet.

III.1.6. Evaluation Technique for Power Cable Ageing

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The electric cables are sensitive elements are susceptible to degradation caused by ageing and have to be included in the ageing management programme (AMP). The development of the cables condition monitoring activities (CM) is a significant aspect of the power cables AMP. The purpose of the CM activities is to verify that the qualification status has not been exceeded. Therefore, the value of the condition indicator (CI) at the end of the qualified lifetime (QLD) and the CI variation over time have to be determined.
Due to different laboratory conditions, these parameters have different values from one laboratory to another. Therefore, each power plant has to elaborate a baseline condition indicator database. This paper describes the operation by analysing a PVC cable.
Five sections of a 3.5-meter-long PVC cable were sampled (CTT 3x25 mm type). The samples were subjected to thermal accelerated ageing (by Joule Lenz effect). The equivalent NPP operating time is 0, 10, 20, 30 and 40 years.
The paper describes the conditions under which the cable samples were artificially aged.
The elongation at break (A%) is an internationally accepted condition indicator for assessing the degradation of cable insulation materials. The tests and analysis to determine elongation at break, performed at INR, have shown that it is a sensitive parameter for material degradation. The tests are destructive for the cable integrity and are carried out in laboratory conditions on cables similar to those installed in the plant.
In order to practically apply the results, non-destructive monitoring methods are required for the condition of the PVC cable jacket inside the NPP, and the correlation of the results.
The indenter modulus (IM) is a 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. The results show that the indenter modulus is a sensitive indicator for evaluating the ageing degradation of PVC insulated the power cables.
There is a good correlation between the mechanical (A%) and physical properties (IM) of the studied power cable insulation material.

The results are useful for identifying, modelling and managing the power cable material ageing phenomenon in the NPP.

III.1.7. Characterization of the Ramberg-Osgood Constitutive Equation for 316L Stainless Steel in Liquid Lead at 400°C

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In the research field for the Generation IV nuclear reactors, managing 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). This phenomenon can lead to the embrittlement of some metals, which are ductile in the air and can become brittle in long contact with certain liquid metals. Many theoretical and experimental studies in the literature are dedicated to Liquid Metal Embrittlement (LME) to understand its basic mechanisms. To date, there are still gaps in the understanding of LME and there is still no agreement on the assessment methods and design, which are essentially based on experimental correlations. The paper presents the results of the experimental tensile tests performed on 316L steel samples, in the liquid lead environment, to highlight the LME mechanism determined by the molten lead. The stress-strain experimental curves were processed in the Matlab environment to obtain the parameters of the equation constituting mechanical stress - deformation, as Ramberg - Osgood type, to be used in the structural integrity analyses. Also, the paper presents the metallographic and SEM analyzes of the samples resulting from the experimental tests to highlight the fracture characteristics induced by the LME embrittlement of lead on 316L steel. The work belongs to the integration effort of the methodology developed in the framework of the CANDU Fuel Channel Program, and is aimed to obtain the material properties for the Generation IV innovative materials with a focus on those planned to be used for ALFRED components.

III.1.8. Study of Pressure Tube Behavior in case of Flow Blockage in a CANDU Fuel Channel

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The current paper aims to present the evolution of parameters in the case of flow blockage in a channel loaded with typical CANDU fuel. Reducing the flow in a fuel channel can occur due to events affecting a single channel, such as blocking the flow in a channel.The consequences of this event depend on the extent of the blockage. At this stage, a conservative study has been carried out based on the current main geometrical and physical features of the CANDU fuel channel with a standard 37-element fuel bundle. Results indicate that decreasing the coolant flow will increase its enthalpy and also increase the coolant temperature along the channel. Reducing the flow rate to 60% of the nominal value, will increase the void coefficient, but will not change the average temperature of the coolant, which remains at saturation. When the flow is reduced below 60% of the nominal value, both the cladding and fuel temperature will rapidly increase due to the appearance of the Critical Heat Flux (CHF). At about 20% of the nominal value of the flow, the boiling point will move upstream on the channel, and the amount of steam produced will gradually increase. Coolant temperature will rise rapidly and leads to a rapid decrease of the efficiency of heat transfer. At about 11.5% of the nominal value, the heat flow produced in this reaction becomes significant, the pressure tube (PT) rupture should be expected due to non-uniform strain (circumferential temperature gradient).

III.1.9. Fluorimetric Method Applied to Determination of Uranium Content in AQUEOUS Samples

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Naturally occurring radionuclides represent a very important class of environmental contaminants. Uranium concentration levels in aqueous samples are of great importance because uranium presents both chemical and radiological hazard to the environment. Therefore, it is necessary to have available accurate, reliable and precise analytical methods to determine concentrations of uranium in aqueous samples. These methods should also be as fast as possible in order to give results in emergency cases to take decision for radioprotection of the environment and human beings.
Uranium content in aqueous samples.can be measured by using radiological and chemical methods.
Both the chemical methods: spectrophotometry and fluorimetry and radiological methods: liquid scintillation (LSC), gamma spectrometry and alpha spectrometry are used for estimating uranium concentration in aqueous samples.
A simple and inexpensive chemical method for analysis of uranium is the fluorimetric technique. Uranium concentration in an aqueous solution is determined by measuring the intensity of retarded fluorescence of uranyl ions (λ=530 nm) after activation by ultraviolet radiation. The luminescence of the solution is increased by addition of sodium polysilicate (pH 8–10).
Alpha spectrometry, using solid-state detectors, is a promising method for determining the individual uranium isotopes.
Determination of uranium in samples by alpha spectrometry requires several radiochemical procedures to separate these radionuclides from the matrix. The following steps must be performed: conversion of uranium associated with the matrix into soluble form, radiochemical separation and preparation of the source for measurement.
Liquid scintillation counting is used for measurement, mainly of α and β emitters, in liquid samples. The sample is directly dissolved in the liquid scintillation solution (scintillation cocktail) and the generated light output measured by photomultiplier tubes.
The aim of this work is to present these methods emphasizing boththeir advantages and disadvantages of each of them and and to evaluate the performance of the fluorimetric method applied to determine the natural uranium content in aqueous samples.

III.1.10. Evaluation of Uranium Utilization for Some Advanced Nuclear Fuels Suitable to be Used in a CANDU Reactor

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This paper aims to assess the degree of Uranium utilization in a generic CANDU nuclear power reactor when some advanced nuclear fuel designs are used. In the view of this paper, the degree of uranium utilization is defined as the ratio between the Average Discharge Burnup (ADB) obtained through Time-Average (TA) core calculations and the specific Conversion Ratio (CR) for every fuel design. The CR coefficient is defined as the amount of Natural Uranium (NU) consumed to obtain the unit mass of that advanced fuel having a higher enrichment in U^{235} comparatively with NU, while the remaining depleted tail is assumed to be 0.25%. Three advanced fuel designs were considered, as follow: a 0.85% Slightly Enriched Uranium (SEU) fuel design, a Recycled Uranium (RU) fuel variant with an initial enrichment of 0.96% in U^{235} and a Thorium based nuclear fuel (a mixture of pure Thorium and SEU-1.8% dioxides). The neutronic properties were firstly generated for every fuel design, according to the burnup, in the form of nuclear cross section tables. Then, core calculations have been performed using a finite differences computer program in order to find out the ADB in a generic CANDU reactor. The paper conclusions reveal the improvements in

Uranium utilization brought by the considered advanced nuclear fuel designs along with some comments on the core neutronic parameters.

III.1.11. Study of the Elastic Anisotropy of CANDU Pressure Tube by Ultrasonic Methods

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The cold-worked Zr-2.5%Nb alloy is used as material for the pressure tubes of CANDU nuclear reactors. Due to the limited slip system, it has developed a strong texture during extrusion process, leading to anisotropic properties. The material properties are strongly dependent of the grains orientation distribution. For this reason, results a directional anisotropy of elastic coefficients.

During the service life in reactor, diffusion of hydrogen and/or deuterium in the pressure tubes wall may occur. Below a certain temperature, when solubility limits are exceeded, a brittle second phase (hydrides) appear.

The hydrides, even in small concentrations, can potentially have a dramatic effect on the structural integrity of zirconium alloys nuclear components.

To characterize the degree of anisotropy and the hydrogen influence it is necessary to determine the anisotropic elastic modulus on the main directions (axial, circumferential and radial) of the tube samples.

In the present paper, the most usual elastic modulus on a given direction (axial, circumferential and radial) of the tub (Zr-2.5%Nb alloy) have been investigated using non-destructive method based on measurements of ultrasonic velocity.

Thus, both longitudinal V_L and transversal V_T phase velocities have been experimentally determined for each direction by pulse-echo method.
The measurements were performed at room temperature on the rectangular samples, in “as received” state and at different hydrogen concentrations, machined from the pressure tube.

III.1.12. Factors that Influence the TRIGA Steady State Reactor Core Reactivity

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Present paper aims to illustrate TRIGA SSR 14 MW core reactivity evolution due to fuel burnup during an operation cycle. TRIGA SSR is a pool-type, light water moderated reactor. It has a nominal power of 14MWt, was commissioned in 1979 and it is used for irradiation of nuclear fuel and materials, radioisotope production or gamma and neutrons applications. The reactor core contains 29 LEU fuel assemblies, 8 control rods, 44 beryllium reflectors, 15 experimental locations and 36 plugs. The original core configuration arrangement was provided with 3 large experimental locations corresponding to 4 locations each, and 3 small experimental locations corresponding to one location each. TRIGA reactor core also allow irradiation devices holder arrangement in 20 beryllium reflectors with central cavity. The most important phenomena during reactor operation are criticality condition evolution and core changes. By taking account the factors that alters the core reactivity (i.e., fuel burnup, temperature negative prompt coefficient of the TRIGA fuel, Xe and Sm production, reactivity introduced by experimental devices) we have to ensure a reactivity excess depending on operational goals (i.e., desired nominal power and period of operation). Reactivity measurements are performed periodically in order to: determine and verify the safety margins for each core configuration; verify the irradiation devices induced reactivity; establish core reactivity excess at the beginning and at the ending of each irradiation campaign. Using a reactivity computer, the reactivity excess at the beginning and at the ending of each irradiation campaign was determined. Measurements showed that due to fuel burnup the reactivity excess has diminished with 68.5 cents for a released energy of 393 MWD.

III.1.13. Vacuum Induction Melting - VIM

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The paper describes a method developed in Institute for Nuclear Research (INR) Pitesti for obtaining high and very high temperatures in order to melt UZrH rods or debris resulted from UZrH fuel fabrication finally leading to recovery of valuable enriched Uranium.
In order to obtain the high and very high temperatures necessary for the process, an equipment was created consisting of the electric system (medium frequency generator, a load impedance adapter, water-cooled copper pipe inductor), the vacuum system (integrated turbomolecular vacuum pump, vacuum chamber, gauges) and the control system (pyrometer, data acquisition system).
The aim was to reach a temperature of 1650 ^{o}C, a temperature high enough to melt UZrH with 45% wt U content.
The material to be melt is placed inside an alumina crucible, surrounded by a tantalum susceptor.
With the given configuration and with proper choice of the impedance adapter parameters, the temperature can be raised at an average rate of 100 ^{o}C / min.
During the experiment, a maximum temperature of 1800 ^{o}C was obtained inside the alumina crucible. This means that the equipment and method can really be applied as part of the INR technology for UZrH LEU fuel elements fabrication.

III.1.14. Recovery of Uranium from Secondary Products obtained in the Manufacture of LEU Experimental Nuclear Fuel

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The purpose of this paper is to present the importance of methods for recovering uranium from by-products resulting from the manufacturing process of low-enriched uranium fuel rods. Testing of several uranium recovery processes has led, with very good results, to the recovery of up to 60% of uranium (in the form of UO_2) using several chemical and thermal treatments. The method we used involved high-temperature calcination of by-products recovered from the production of nuclear fuel to obtain metal oxides. This mixture is treated with concentrated nitric acid to solubilize uranium. Zirconium and erbium oxides are insoluble in concentrated nitric acid and remain in the form of a residue. The impure formed uranyl nitrate solution is subjected to liquid-liquid extraction with an organic mixture of tri-n-butyl-phosphate in kerosene and then re-extraction in aqueous medium to purify it. The chosen organic solvent has a high affinity for uranium. As the pure uranyl nitrate solution obtained is diluted, it is concentrated by evaporation. Uranium is precipitated from the solution in the form of ammonium diuranate using ammonia. The ammonium diuranate obtained is insoluble in water. After washing the ammonium diuranate precipitate and drying it in the oven, it is calcined to form U_3O_8 and then reduced to UO_2 with hydrogen. Applying this method allows the recovery of up to 60% of uranium, which means an increase in the amount of uranium

available to the energy industry. The rising price of electricity and uranium are additional arguments in favor of applying the presented recovery method.

III.1.15. Study on the Measurement of Oxygen Concentration in Molten Lead

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Lead and eutectic Pb-Bi alloy (LBE) have been proposed and investigated as coolants for fast reactors since the 1950s. More recently, based on these cooling agents, the following reactor concepts are currently being developed: Lead Cooled Fast Reactor (LFR), Subcritical Reactor Cooled by Pb-Bi (Accelerator Driven System-ADS) and Pb-Bi Fast Reactors.
Corrosion and erosion of structural materials is the main problem of lead alloy cooled systems. It has been observed that one method to mitigate the corrosion effects is to maintain a specific oxygen concentration inside the liquid metal. In order to have an active control of the oxygen concentration dissolved in the liquid metal, the oxygen concentration has to be measured. In this regard, oxygen sensors for heavy liquid metals are being developed. They
measure an electric potential, and are usually composed of a ceramic solid electrolyte with high oxygen ion conductivity and a reference electrode with well-known oxygen activity located inside the ceramic element.
The solid electrolyte could be MSZ (Magnezia Stabilized Zirconia), YSZ (Yttria Stabilized Zirconia) or CSZ (Calcia Stabilized Zirconia). Among them, YSZ is more used due to its better mechanical proprieties. Given the mentioned solid electrolytes, the following reference electrode materials were tested: Pt-air and metal/metal oxide references.
The sensors were calibrated in oxygen saturated liquid lead, and the measured values have been compared with the theoretical ones, in order to determine the accuracy of the sensors for a specific temperature range.

III.1.16. Methods of Acquiring Oxygen from the the Molten Lead Medium

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This paper describes the method of controlling the oxygen in the liquid lead medium by using potentiometric oxygen sensors. The principles of a potentiometric oxygen sensor is based on the use of a solid electrolyte (an oxygen ion conductor) separating two electronically conducting electrodes, the reference and working electrode. At each electrode-solid electrolyte interface a chemical potential of oxygen (μO2) is present upon the oxygen equilibrium partial pressure. A difference in oxygen chemical potential gives rise to an electric potential difference (E) measurable with a high-impedance voltmeter (V).The electrochemical pumping of oxygen principle is based on the transfer of oxygen ions through solid electrolytes under an external applied electric potential. Like the potentiometric oxygen sensor, the reference electrode (air / electrode) and the working electrode are physically separated by an electrolyte (lead). Moreover, possible poisoning effects oxide formation at the interface between lead and YPSZ could decrease the electrochemical oxygen pump active area. To add oxygen to the liquid metal, the use of electrode materials with higher oxygen reduction reaction kinetics than air/perovskite oxides might decrease the total cell resistance of the electrochemical pump with as consequence an increase of the current density. Moreover, the effect of metallic impurities dissolved in the liquid metal on the oxygen pump performance while adding oxygen to lead need to be further investigated as well.

III.1.17. Using NAA Method to Determine the Concentration of Retained Elements in the Air Filters of the Ventilation Installation from the TRIGA Reactor

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The purpose of this research work was to determine the elements retained in the filters of the ventilation installation inside the TRIGA reactor building. To achieve this goal, 3 samples of mechanical pre-filters and 3 samples of HEPA filters were taken from the caisson of the ventilation boiler – CV 1. CV 1 ventilation unit contains a battery of 8 caissons of three filter cartridges each (pre-filter + HEPA filter). The samples (pre-filter and HEPA filter) were carefully taken from boxes numbered 3, 5 and 8 in accordance with all radiation protection regulations, and were subsequently placed in zip-lock bags to be processed for irradiation in the reactor. A sample of unused mechanical filter and HEPA filter were taken for the control. After drying in the oven or calcining in the electric oven, the pre-filter and filter samples were placed in polypropylene cartridges to be irradiated in the neutron field. The calcination operation was performed for the best possible homogenization of the filter samples. Irradiation took place in the pneumatic mail of the TRIGA SSR reactor operated in steady state at a thermal power ranging from 11.50 MW to 13 MW. A set of four irradiations was performed at location K-11 of the TRIGA SSR reactor. For each irradiation, a polypropylene cartridge with a pre-filter sample, a HEPA filter sample (both from the same caisson) and a flow monitor were introduced - in this case a gold monitor sheet Au^{197} with a concentration of 0.1 %, all inserted in a larger irradiation cartridge also made of polypropylene. By measuring the spectra of the gold monitors, an accurate description of the thermal neutron flux over the entire irradiation period was made. Depending on the contact dose rate value, the samples were allowed to settle in the reactor pool for a predetermined time so that the short-lived isotopes disintegrate completely. A short irradiation (120-seconds) was also performed to determine short-lived isotopes. Samples were measured on a high resolution gamma spectrometry detection system with hyperpur germanium crystal. Following the measurements, the gamma energy quantum spectrum was obtained, and will be further processed with the GENIE IV acquisition / analysis program. The elemental concentrations in the samples were determined applying, the neutron activation analysis method (NAA- k_0) using k_0 standardization. The main purpose of the work was to determine the trace element concentrations retained in the collected air filters after the long-term operation of the TRIGA reactor. As a result of the work carried out, it was concluded that mechanical pre-filters retained most of the elements (^{87}W, ^{139}Ba, ^{140}La, ^{56}Mn, ^{239}Np, ^{46}Sc, ^{59}Fe, ^{51}Cr, ^{60}Co, ^{42}K, ^{65}Zn). HEPA filters retained a percentage of the elements that were not retained on the pre-filters. It should be noted that the values of calculated elemental concentrations of radioisotopes ^{65}Zn, ^{69m}Zn, ^{122}Sb şi ^{124}Sb can be neglected because the embedded frame of filters and pre-filters is made of galvanized sheet, and was already determined in the filter and pre-filter samples used as a control.

III.1.18. Probabilistic Safety Assessment - an Important Tool for the Maintenance Activity Optimization

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The safety related maintenance has been recognized as essential to plant safety. The goal of this work was to propose an approach to maintenance optimization of the nuclear power plant components, which can contribute to increase plant reliability, availability and safety.
In order to evaluate the impact of the maintenance on the quantitative basis the Probabilistic Safety Assessment (PSA) methodology and the software code EdFT (**Ed**it **F**ault **T**ree) as part of the computer programs package PSAMAN (**PSA Man**ager) developed in RATEN INR Pitesti by the

PSA team were used; the reliability data used for evaluation of the developed fault tree (FT) is the generic database of the EdFT code.
The paper includes an example, Raw Service Water (RSW) System from CANDU6 NPP that was modelled with the methods and methodology specific to the PSA. An essential role in the evaluation of the system logic model is the available information: failure rate and mode, average repair time, maintenance frequency and duration, test interval and duration. The study focused primarily on the preventive maintenance activities of the tested components (pumps, valves, motors, etc.) considering different preventive maintenance frequencies and durations. The optimum test interval was estimated with the optimization method based on risk information. The calculated optimum test interval of the selected components is being used to observe the impact on the system unavailability and a comparison of data results is made. For some components as valves the recommendation is to have larger test intervals and for pumps and their related subcomponents the recommendation is to have a frequently test interval.
The results of the study demonstrate that the probabilistic safety analysis can assist in the development and planning of maintenance activities.
The paper has an educational purpose and was elaborated following the cooperation agreement between The Institute for Nuclear Research Pitesti and The University of Pitesti.

III.1.19. Short Term Irradiated Ghraphite Leaching Test

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Understanding the radionuclide behaviour, especially of long-lived ones such as ^{14}C (T1/2=5730 years) in the near and far-fields of a radioactive waste repository is essential when performing safety assessment of long-term storage and disposal of irradiated graphite. Factors like graphite type, ^{14}C inventory and speciation in the irradiated graphite, release rate, groundwater composition, pH-and Eh- conditions, etc influence the release behaviour of ^{14}C in the geochemical environment.
The paper presents the result and the methodology of a short term (90 days) leaching test conducted to evaluate the ^{14}C release from TRIGA irradiated graphite samples, under disposal relevant conditions. The leaching approach is based on a short term semi-dynamic test, with a precise time scale sampling/replacement of the leachate. The leaching test was performed at room temperature, in aerobic conditions.
The irradiated cylindrical graphite samples were leached in 0.1 M NaOH solution, with high pH (~13) that is relevant for disposal in cementitious environment and avoid ^{14}C precipitation. The ratio of solution to exposed surface area of the sample was kept constant and not-exceed 0.1m. After each sampling steps, the ^{14}C concentration activity in leachate solutions were measured by liquid scintillation spectrometry.
Leaching of ^{14}C from TRIGA irradiated graphite was quantified both as cumulative leaching fraction of ^{14}C and by incremental leaching rate. The values obtained were corrected with the dilution factor and reported to the initial ^{14}C content. The results obtained from the leaching test show a very low ^{14}C release rate. Higher leach rates are found in the first days of testing, indicating an initial rapid release rate, followed by a slower release rate.

III.1.20. Obtaining U-Zr-Er Alloys by Applying the Powder Metallurgy Method

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The TRIGA 14MW Steady-State Reactor owned and operated by the RATEN ICN uses fuel assemblies consisting of U-Zr-Er alloy wich mainly contains zirconium hydride (ZrH1.6) and uranium metal (45%) distributed as a dispersion of metal particles. The fuel rods also contain a small amount of erbium, which acts as a "neutron poison", being added to correct the neutron flow diagram of the fuel at the beginning of its irradiation. The production of U-Zr-Er alloy rods has as an intermediate stage. In this paper, technological tests were performed in order to obtain sintered compacts of U-Zr-Er alloy (using depleted uranium), by the method of powder metallurgy. The aim of the paper is to establish the optimal parameters, which lead to obtaining compact U-Zr-Er alloy with density values as high as possible. Vacuum sintering treatments of metal hydride compacts were performed to obtain U-Zr-Er alloy rods at different temperatures. The values of the parameters characterizing the sintering treatments (time, temperature and pressure) are monitored through a program developed especially for this installation. The U-Zr-Er metal compacts were analyzed non-destructively (determination of mass, measurement of length and diameter of each sample) and their density was determined by immersion in demineralized water. The obtained results were analyzed to identify the values of the parameters that lead to the obtaining of sintered U-Zr-Er rods with high density. The high value of the U-Zr-Er alloy density is important, because it influences the burning degree of compacts.

III.1.22. Determination of Peak Efficiency for an HPGe Detector Used in Gamma Spectrometry of Environmental Samples

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The environmental radioactivity monitoring in normal circumstances is performed to ensure compliance with the basic safety standards and comprises two complementary parts: global monitoring of the areas outside the zones where significant nuclear activity is carried out and close monitoring around sites where an activity liable to have a radiological impact on the environment is carried out.

The high-resolution gamma-ray spectrometry is one of the most frequently used methods for environmental radioactivity monitoring. The quantitative determination of radionuclides from environmental samples by gamma spectrometry requires the knowledge of the full energy peak efficiency. The efficiency depends not only on the detection system but also on the sample shape and the sample matrix with different characteristics (dimensions, composition or density) and the detection geometry (the relative position of source and detector and the presence of materials between these). The current paper presents the experimental determination of the efficiency of the high-purity germanium coaxial detector used in Radiation Protection Laboratory.

Aiming at the purpose of the paper there were used three different standard sources containing different radionuclides with known activities and gamma emissions covering the energy range of interest, in the usually used geometries for environmental samples: water, soil and vegetation. The standards can be purchased from authorised supplier or can be made using "in house procedures". In our case the standard sources for water and vegetation were made by homogeneously incorporating of certified solutions of radionuclide into inactive matrices with the same composition and density as the sample to be assayed. Because of the low radioactivity level of the environmental sample, the sources packed in cylindrical polypropylene containers were placed as close to the detector crystal as possible (on contact with the detector).

The software used to obtain the efficiency calibration curve was conventional Genie 2000 package software from Canberra, the same software used for analyse gamma spectra for environmental samples.

III.1.23. Calibration in Energy and Efficiency of the Gaseous Effluent Monitoring System

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The paper presents the principle of operation of a gaseous effluent monitoring system and the activities performed for its calibration in energy and efficiency.
Environmental radioactivity monitoring covers the arrangements and activities necessary to determine the levels of radioactivity in the environment, to assess the radiological impact on public health and the environment due to effluent discharges.
Monitoring of the environmental radioactivity allows prevention of risk of irradiation or contamination of the population, as a result of the release of radioactive substances in the environment, in accordance with the specific requirements of CNCAN.
The main function of the gaseous effluent monitoring system (SMEG) is the continuous monitoring of the exhausted air contamination, to ensure that the permitted release limits for noble gases, aerosols and radioactive iodine are not exceeded.
Calibration is the establishment of a mathematical correlation between the values recorded by the device and the radiological properties of the measured calibration sources (standards, reference materials).
For energy calibration of a gamma spectrometer one or more standard sources are used, which emit gamma radiation with exactly known energies and which are recommended to cover the whole range of energies to be analysed. The energy calibration helps to identify radionuclides in the sample by analysing their specific energy spectra.
The efficiency calibration is performed by achieving the desired measurement geometry, acquiring and processing the gamma spectrum of the calibration source and determining the parameters of the efficiency calibration curve.

III.1.24. Geopolymers Matrices for Radioactiv Waste Conditioning

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Inorganic polymers, more commonly referred as geopolymers, are aluminosilicate materials with excellent physical and chemical properties and a diverse range of potential applications, such as: concrete pavements and products, containment and immobilisation of toxic, hazardous and radioactive waste, advanced structural tooling and refractory ceramics, and fire-resistant composites used in buildings, aeroplanes, shipbuilding, racing cars, and the nuclear power industry.

This paper presents formulation of an original geopolymer composition based on local raw materials: granulated blast furnace slag from Liberty Steel Galați (which is a by-product from iron industry) and volcanic tuff from Bârsana.

Alkali-activated binders, also called geopolymers (GP), are produced through the reaction of an aluminosilicate precursor with an alkaline activator. Aluminosilicate precursors, used in powder form, include industrial by products such as blast furnace slag or fly ash and natural materials such as clays. Alkaline activator consists usually of a concentrated aqueous solution of alkali hydroxide, silicate, carbonate or sulphate.

Based on oxide composition, the materials selected for GP formulation tested in this study are blast furnace slag from Liberty Steel Galați, with a basic character (frequently used as calcium precursor) that was dried, grounded and sieved (<75μm) – volcanic tuff from Bârsana that was calcined at 80˚C and sieved (<75μm), and standardized sand usually used to obtain cement mortars. For alkaline activator a mixture of Na_2SiO_3 solution and 8 M NaOH solution was used.

Mechanical tests carried out to test the performances of the GP matrices were performed according to SR EN 196-3. Setting time and soundness were estimated using a VICAT equipment, while the compression strength was measured using a MATEST equipment.

The results obtained for the geopolymer matrices were comparable with the characteristic of cement-based matrix indicating the potential of the geopolymers tested to be used for different categories of radioactive waste conditioning.

III.1.25. Analysis of Natural Uranium in Liquid Radioactive Waste

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Uranium is a long-life alpha-emitter radionuclide, which it is naturally present, in low concentration, in the environment (soil, rocks and groundwater). Three main radioisotopes are present in natural uranium (U-nat) composition: U-234, U-235 and U-238 in different mass ratios (U-238 – 99.275%; U-235 – 0.719% and respectively U-234 – 0.0057%).

Knowledge of the uranium content in the environment is important mainly because of its high toxicity.

The concentration of natural uranium released in environment from nuclear activities is regulated by the current legislation that limits the maxim discharge concentration to 1 mg/l. As the liquid radioactive waste generated by the CANDU Nuclear Fuel Factory (FCN Pitesti) exceed the maximum discharge limit, these waste are treated in the RATEN ICN Pitesti for U-nat recovery and the U-nat recovered is sent back to FCN Pitesti.

The natural uranium concentration in liquid radioactive waste is usually measured by chemical/radiochemical methods.

In this paper is presented a comparative study regarding determination of natural uranium concentration in liquid radioactive samples using three different methods: fluorimetry, liquid scintillation spectrometry (LSC) and gamma spectrometry. The proposed methods require a minimal sample preparation: for gamma measurement no sample preparation is necessary, the measurements are performed directly on 100 ml plastic bottles with liquid radioactive waste; for the fluorimetry and LSC measurements the liquid sample is mixed with 47% nitric acid (HNO_3), evaporated to dry salts and the residue is dissolved in small volume of 0.1M HNO_3.

In present study, the natural uranium was determined by gamma spectrometry converting the activity of U-235 measured at 185.7 keV (with 57.1 emission probability).

For determination of U-nat by LSC, a quench curve (CPM/ U-nat concentration vs tSIE) was developed and applied first on the liquid sample with known U-nat concentration for confirmation. After that, the method was implemented for the measurement of U-nat from the radioactive liquid samples with unknown U-nat concentration.

The results obtained by gamma spectrometry and LSC were compared with the concentration of U-nat measured by fluorimetric method, as this last method is commonly used in the RATEN ICN laboratory for determination of U-nat concentration in the liquid radioactive samples. The results obtained are comparable with the fluorimetric measurements, but the liquid samples with low content of U-nat must be concentrated (2 or 3 times) to obtain reliable results.

III.1.26. Determination of HPGe Detector Efficiency Calibration Curve for Liquid Radioactive Waste Measurements

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Gamma ray spectrometry is a very powerful analytical tool for gamma emitters measurement using a high resolution HPGe detector. A gamma ray spectrometer is an instrument for measuring the distribution of the gamma radiation intensity versus the energy of each photon. Gamma-ray spectrometry provides two kinds of information: qualitative and quantitative.

The most important step in performing gamma ray analysis of radioactive samples is the energy and efficiency calibration of the detector used for different energies, source-detector geometries, and composition of the samples or sources. The most difficult task in term of calibration is the efficiency calibration, due to the multitude of factors that influence the detection efficiency.

Based on this aspect, the aim of this paper is to establish the energy and efficiency calibration curves for a liquid source in a cylindrical geometry (250ml plastic vial) at contact with the detector.

The energy calibration consists in the experimental determination of a function, usually a first- or second-degree polynomial curve, describing the energy dependence of the channel number in the spectrum.

The efficiency values were determined using the amplitude spectra acquired for every radioactive source, namely from the net area in the interest region of the energy peak, divided by the activity of the sources, acquisition time and intensity of the energy line.

To perform an accurate and reliable calibration curve a standard radioactive source with certified activity of each radionuclide (Am-241, Eu-152, Cs-137, Co-60) covering a wide spectral range between 50 keV and 2000 keV was used. The characteristic spectrum of the calibration source was acquired with GammaVision software and the calibration curves were performed using Genie2000 software.

The efficiency curve was obtained by interpolating the calculated efficiency values of each full energy peak of the standard source by a polynomial function.

The energy and efficiency curves with good relative errors calculated from theoretical and experimental activity values characteristic of calibrated source were obtained.

The resulted efficiency curve was used in quantification and identification of gamma ray emitters contained in samples of liquid radioactive waste generated from operation of TRIGA reactor.