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PLENARY SESSION – May 27

The Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) Working Group on RD&D Dissemination

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One of the aims of the Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) SecIGD2 project is to support the IGD-TP through its Secretariat to address the networking and structuring of RD&D programmes and competences in countries with less advanced geological disposal programmes including those in the new European Union Member States. This has included setting up a working group specifically to investigate the needs of new member states and to organise two international conferences (2014, 2015) for disseminating the scientific and technical information and results derived from the IGD-TP's Joint Activities as outlined in its Strategic Research Agenda (SRA) and from other RD&D efforts in the field of geological disposal. The working group has produced a questionnaire and collated responses to identify needs from less advanced programmes. The outcomes of the questionnaire showed that there were three categories of needs, specifically: RD&D needs that align with topics in the SRA (e.g. costing, safety case methodology); RD&D needs that can be met through other collaboration initiatives such as Newlancer, ERDO, and IAEA (e.g. development of national policy, establishing regulatory controls); Programme infrastructure, information and processes required to implement geological disposal and fulfil EC Directive 2011/70/EURATOM. The working group will produce a 'mini-roadmap' that is aimed at helping a first response to the RD&D related aspects of the EC Directive. The roadmap will set out the key steps in developing a geological disposal facility programme and strategy based on advanced waste management organizations experience. The roadmap will signpost open documentation and guidelines for specific technical areas.

ALFRED and i-CRADLE: A Distributed Lead Technology Research Infrastructure

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ALFRED, the Advanced Lead Fast Reactor European Demonstrator, has emerged from a European research effort, in the frame of Lead cooled next generation nuclear power plants. The main goal behind ALFRED and Lead technology development is to maintain the nuclear energy source as an important contributor to the development of a secure and low carbon European energy system. ALFRED, as one of the projects supported by the European Sustainable Nuclear Industrial Initiative (ESNII), brings together industry and research partners in the development of so-called Generation IV Fast Neutron Reactor technology, as part of the EU's Strategic Energy Technology Plan (SET-Plan). The paper presents briefly the design status of ALFRED as part of a future technology park to be built in Romania, with open access to European scientists and students for extensive research on operating conditions in a realistic environment. Also included is an overview of the main aspects of the design concerning: system design, safety features and core design. The main design parameters are briefly summarized. In addition to the above described technical overview of ALFRED design, the paper reports about the activities carried out by the FALCON (Fostering ALfred CONstruction) Consortium established in Bucharest on December 18th 2013 by ANSALDO, ENEA, RATEN-ICN, and joined by CVR in December 2014. The main results of the FALCON activities are presented and the present status of the consortium, technical and management activities are summarized. One of the main results of FALCON in this first period of activities is highlighted by the development of the technological roadmap, divided in two main steps: the Lead Technology Development and the ALFRED construction. The first, preparatory step is deemed necessary in order to reach the technology readiness level that is required to build ALFRED. Such development relies on the realization of a number of experimental facilities dealing with the main aspects of lead technology such as: lead corrosion and erosion mechanisms, fuel handling, lead-water and lead-fuel interactions and so on. All these facilities themselves represent the basis of a Distributed Research Infrastructure having its focus on ALFRED, so that an extensive reflection on the challenges and opportunities offered by this approach for LFR development and demonstration is outlined as a conclusion.

PLENARY SESSION – May 28

Advancing the CANDU Technology Option for Europe

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The CANDU® reactor is a unique design characterized by its use of a fuel channel core, heavy water moderator and coolant, and on-power refueling. The design was originally developed in Canada and holds a strong pedigree and performance record. The unique features of the CANDU reactor allow it to utilize various fuel types, including Natural Uranium, Recycled Uranium, Mixed Oxide (MOX), and Thorium. These reactor developments and future evolutions of the CANDU design are supported by Canada's world-class nuclear laboratory.
The Enhanced CANDU Reactor® (EC6®) is a 700-MWe class Generation III design based on the successful CANDU 6® model that has been exported worldwide. Several variants of the EC6 have also been developed, each with its own niche capabilities.
This presentation will focus on the most recent evolutions of the CANDU design for various fuel cycles and the near-term deployment of these projects. This includes the conversion of the fuel in the Qinshan CANDU 6 reactors in China to Natural Uranium Equivalent (NUE), the development of the EC6 CANMOX™ solution in the United Kingdom, and the development of the Advanced Fuel CANDU Reactor (AFCR®) for deployment in China and global markets.In addition to CANMOX™ some of the other fuel options have direct potential application in Europe.

The Implementing Geological Disposal of Radioactive Waste Technology Platform. Main Achievement in 2015

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After decades of bilateral and multilateral cooperation, several European waste management organizations decided, under the auspices of the European Commission (EC), to join their forces to tackle the remaining research, development and demonstration (RD&D) challenges associated with the implementation of their respective geological disposal programs. The main objectives of the Implementing geological disposal of radioactive waste technology platform (IGD-TP) are to initiate and carry out collaborative actions in Europe to tackle the remaining research, development and demonstration (RD&D) challenges with a view to advancing the implementation of geological disposal programmes for high-level and long-lived waste in Europe. This paper presents the organisation of the work and the main Joint activities and projects to date, initiated by the IGD-TP members and supported for some of them by the European Commission under the FP7 framework programme and in the near future under the Horizon 2020 programme. Overpressure rupture of the pressure boundary (2 hours) and first releases of radioactivity into the containment (4 hours), the fluid depletion from all water systems surrounding the fuel in cannot be precluded but can be avoided or significantly delayed by properly considered additional design measures. The consequences of a severe core damage in a typical CANDU reactor currently pose risks that are unacceptable especially after Fukushima and the hype that surrounded the ineffective utility ‘Stress tests’ and regulatory ‘Action Items’ that followed.

ARCADIA PROJECT - Support for ALFRED Demonstrator Implementation

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ARCADIA project (Assessment of Regional Capabilities for new reactors Development through an Integrated Approach) investigates the existing capabilities for the construction of ALFRED in Romania in order to identify the gaps and to select the appropriate methods and tools for competence building. The paper presents some elements developed in the first period of the project: analysis of the regulatory framework, investigation of risks for implementation, benefits and challenges for society, competence building approach, and key elements for feasibility study. All these aspects are analysed starting with the current capabilities of Romanian nuclear sector and continuing with the potential to develop Generation IV based on national resources, FALCON consortium resources, and by capturing external expertise for some peculiar elements. The main outcomes consist of the identification of needs both for competences and infrastructure, the identification of the gaps in competences and infrastructures, and also the approaches to fill these gaps in order to perform preparatory activities, construction and operation of ALFRED demonstrator. The main stakeholders for the process are also identified the approach for participation in decision process is discussed. At the same time some factors and their influence on the implementing process are analysed on the basis of the evolutions in FALCON consortium and at the level of the decision of the national authorities.

I. NUCLEAR ENERGY

I.1. Nuclear Safety and Severe Accidents

I.1.1. Reaction products In the U-Bi-O and U-Pb-O systems: synthesis and characterization

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Characterisation of lead-bismuth eutectic (LBE) - fuel interaction products is of paramount concern in order to qualify the fuel for the MYRRHA accelerator driven system. In the frame of the "Safe exploitation related chemistry for HLM reactors (SEARCH)" FP7 Collaborative Project, an extended study on the chemistry of the ternary U-Bi-O and U-Pb-O was performed. This study resulted in the following U(VI) compounds: bismuth(III) uranate (Bi_2UO_6), lead(II) uranate ($PbUO_4$), and trilead(II) uranate (Pb_3UO_6). The synthesis conditions and the stability domains of these uranates were established. Their thermal expansion and (low- and high-temperature) heat capacity are reported for the first time. A little deviation from the Bi:U:O= 2:1:6 stoichiometry was observed in bismuth(III) uranate, consistent with the $Bi_2UO_{5.94}$ formula already proposed in the literature. Bi_2UO_6 presents two crystalline modifications. The transition temperature of α-Bi_2UO_6 (C 1 2 1 space group) to β-Bi_2UO_6 (P -3 space group) occurs between 873 K and 1073 K. This phase transition was not observed during drop calorimetric measurements or by DTA/TG. The expansion coefficients of the α-Bi_2UO_6 were calculated in the (298 to 873) K temperature interval and the obtained values are discussed with respect of accident scenario in LBE cooled fast reactors.

I.1.2. The In-Vessel Melt Retention Strategy Using the RELAP5/SCDAP Code for VVER-1000 reactor

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After Fukushima accident there is a worldwide strategy to enhance defence in depth (DiD) concept through mitigation of the severe accidents. In-vessel melt retention is a severe accident management strategy which, based on external flooding of the reactor pressure vessel, is aimed at arresting the downward progression of an assumed (unmitigated) core melt accident. The new generation 3+ PWR designs, exemplified by EPR, VVER-1000, APWR-1400 and AP-1000, which employ In-Vessel or Ex-Vessel cooling and retention of the core melt/debris bed that would be produced in the postulated severe accident, are reaching the end state of development for public safety for PWRs. This strategy is a key

for the severe accident management. During the severe accident event, the core materials are melted and slumped to the lower head. These melted core materials are called corium. The corium behaviour in the lower plenum of the reactor vessel during a severe accident is very important, as this affects a failure mechanism of the lower head vessel and a thermal load to the outer reactor vessel under the in-vessel corium retention through external reactor vessel cooling condition. The main objective of this paper is to simulate the VVER-1000 reactor lower head during a severe accident. The RELAP5/SCDAP mod 3.4 applied in a slumping model, is used to study the in-vessel cooling and retention issue using the couple models with user-defined slumping inside the 2D lower head couple mesh. The results illustrate the lower head failure time due to creep rupture, related to the cavity pool water level. The cavity remains intact if it is full of the coolant before the beginning of slumping to the lower plenum. In this case, the relocated core materials are retained within the vessel. In the performed analysis the critical water level inside the cavity for the mitigation of the in-vessel phase of the sever accident, is estimated.

I.1.3. Regulatory Approach to Design Extension Conditions

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In accordance with the definition in the IAEA Safety Standards, design extension conditions represent accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions. This paper provides an overview of the treatment of design extension conditions in the international safety standards and in the regulatory frameworks of other countries. A review of the current national regulatory framework is performed, in order to identify needs for improvement in either mandatory requirements on in guidance with regard to design and safety assessment for nuclear power reactors. The paper presents also a review of the event sequences that can be considered as design extension conditions, based on deterministic and probabilistic considerations, together with the criteria applied to their treatments in safety analysis. The differences between the safety analysis requirements and criteria for design basis and design extension conditions are also presented. The paper also examines the differences in approaches to design basis and design extension analyses for PWRs (Pressurized Water Reactors) and PHWRs (Pressurized Heavy Water Reactors). The original contribution consists of establishing a list of design extension conditions that will be explicitly be included in the revision of the national regulations on the design of nuclear power plants, taking into account the operational and safety assessment experience and the development of international standards and practices.

I.1.4. Progress with the Implementation of Lessons Learned from the Fukushima Daiichi Accident

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The paper presents the review of the main documents issued at international level on lessons learned from the Fukushima Daiichi accident from March 2011 and the corresponding recommendations. The main recommendations have been considered in the framework of the “stress tests” performed for the nuclear power plants (NPPs) after this accident, taken during peer-reviews and constitute the basis for corrective actions and improvements proposed for implementation for each NPP. One of the objectives of this paper is to present the actions resulted from the European "stress tests" together with the status of the implementation of the Romanian National Action Plan (NAP). The paper presents also how the Romanian NAP has been developed for bringing together the actions identified from regulatory reviews, self-assessments, peer-reviews and generic recommendations at international level. The Romanian NAP has been elaborated by CNCAN taking account of the guidance provided by the European Nuclear Regulators Group (ENSREG) and its implementation is permanently monitored by CNCAN. Periodic reports on the implementation progress are published and made available for review to the ENSREG. The Romanian NAP was issued for the first time in December 2012 and it has been reviewed and revised in December 2014. The national action plan is subject to peer-review within ENSREG in the period January - April 2015. Currently, the peer-review of the revised NAP is in progress and conclusions about the results can be drawn. The paper includes also the results of this peer review. The paper highlights the significant improvements to the safety of an operating NPP that can be achieved through a wide range of upgrades to design, operation, monitoring, management and emergency preparedness and response. The improvements related to the regulatory oversight of NPPs are also presented, as well as potential improvements to the safety culture in both regulatory and nuclear operating organizations. All these improvements reflect the cumulative experience and expertise provided by both internal and external specialists and organizations. The original contribution of this paper is represented by the analysis of this integrated contribution to the improvement of NPP safety at international, European and national level, in particular with consideration of the case of the Romanian NAP and its implementation, based on the latest information available on the lessons learned from the Fukushima Daiichi accident.

I.1.5. Cernavoda Unit2 Recirculated Cooling Water Systems Transient Analysis

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The paper is an approach to calculate the response of Cernavoda NPP unit 2 Recirculated Cooling Water (RCW) System to transient regimes during normal and abnormal conditions. In order to develop the RCW model, one started with nominal regime of the system, in summer conditions, with 3 RCW pumps and 3 RCW heat exchangers in operation. This regime was modelled in order to calibrate the model so that all consumers

to be supplied with the required nominal flowrates as per data sheets and design manuals. Then one started to analyse the system response to reactor trip on availability of class III and class IV electrical power, Loss of Cooling Accident (LOCA) with class IV available, LOCA on availability of class III power, Loss of Instrumental Air (LOIA) on class IV available, and LOIA with class IV unavailable. Moreover, one analysed the system transient due to requirement of changeover of a RCW operating pump, planned and unplanned changeover. This is the first transient approach to this system that took in consideration all buildings of the system, obtaining a very large system model, with over 900 pipes, 4 pumps, 50 consumers, and 21 control valves. Previous analyses on the system considered were focused on Balance of Plant (BOP) and pumps station, and for the rest of the system considered as one big consumer for each building, obtaining a simplified/reduced model. The present analysis is the largest transient analysis model for Cernavoda NPP unit 2. The changeover procedure was required to be analysed in order to change the nominal operating mode for unit 2, from current 2 pumps in operation to 3 pumps, operating during summer conditions.
The analysis was performed using PIPENET transient version 1.7, a computer code made mainly for hydraulically computation, for both stationary and transient regimes.
At Cernavoda NPP unit 1, the RCW system normally operates with 3 pumps. The unit 2 of Cernavoda NPP had changed this mode and uses 2 pumps to operate, due to recirculation line vibration problems. After the redesign of recirculation line, the system did not return to operate with 3 pumps, due to concern of the overflow and problems to the new recirculation line. From this analyse one concluded the normal operation mode can be returned to operate with 3 RCW pumps, as the RCW system was designed.

I.1.6. Porous Media Approach In Thermal-hydraulic Core Analysis of Pressurized Water Reactors

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In nuclear-related calculations, especially in the case of reactor core analysis during normal and/or abnormal transients, it has always been interesting to have the most accurate analysis of both the neutronic behaviour and the thermal-hydraulic behaviour in a nuclear reactor core. Any changes in the reactor coolant properties (pressure and/or enthalpy) followed by a transient, will cause a deviation in the coolant density, which in turn leads to feedback on the power generation in the reactor core and to changes on the coolant density again. Thus, a precise thermal- hydraulic analysis of the coolant is needed to accurately simulate these feedbacks during steady and transient conditions. The commonly used methods for simplifying and analysing the thermal-hydraulic behaviour of the coolant flow in the reactor core are the porous media approach and the sub-channel approach. The sub-channel method is mainly a simplified condition of the porous media approach and has proven to be quite reliable in its application to various rod geometries and operating conditions. However, because of its inherent assumption of the existence of a dominantly axial flow, it is doubtful that the sub-channel method can be applied in strong cross flow cases such as a flow blockage or a degraded core geometry, which is encountered in a liquid metal fast breeder reactor (LMFBR) safety analysis. A new approach called the porous body analysis method was developed to provide an alternative for the rod bundle analysis and for application to the general flow cases. The porous body approach is formulated based on the porosities of the control volume for which the conservation equations are written. This allows an arbitrary geometric configuration for the control volume because the geometric effect is taken into account through the surface and volume porosities. In addition, this approach rigorously solves the transverse momentum equation as well as the axial momentum equation. In this study, a thermal-hydraulic analysis of the reactor core is performed using a porous media approach. Based on this approach, each fuel assembly was modelled and was divided into a network of lumped regions, each of which was characterized by a volume average parameter. The conservation equations of mass, linear momentum and energy are derived and discretized using the finite volume method. The pressure, velocity and temperature fields are achieved using a numerical analysis of the above mentioned coupled equations. To validate the applied approach, the conservation equations of mass, linear momentum and energy are derived and discretized by finite volume method in hexagonal coordinate system.

I.1.7. The influence of River Water Temperature Annual Variation on the Moderator Heat Exchangers Heat Flux

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The Main Moderator heat exchangers are the most important consumers supplied by Recirculated Cooling Water (RCW) System. In order to determine an appropriate operating configuration of the RCW system it is needed to determine the flowrate required by the Main Moderator consumers, in real time. From operating experience, the required RCW flowrate necessary to be supplied to the main moderator heat exchangers is much lower than design flowrate. In installation, there are no flow elements that could measure the flow. However, there are two control valves which regulate the flow to the main moderator heaters; they control the outlet temperature of the moderator to 69°C. That leads to the requirement of calculating the flowrate function of the outside temperature for all possible temperatures during a calendar year. Usually, the heat transfer computation for a heat exchanger implies solving a transcendent system of equations with seven equations and seven unknowns. This kind of system can't be solved easy by the system's operating engineers. For this purpose one proposed a mathematical model solved by using MATHCAD 13, in order to immediately compute the required flowrate function of Raw Water System (RSW) temperature. One considered all possible temperatures during an operating year, and more, going beyond design point, up to 36°C, temperature that can occur during quick transients after the fourth RCW pump starting. The calculation was made to verify the capacity of heat exchanger to remove the designed 100 MWt in the new condition of reducing moderator temperature outlet from 77 to 69°C. The change in temperature was required in order to have bigger sub cooling margin in case of LOCA and loss of class III electrical power. The model obtained was validated using field temperature and flow measurements and the conclusion was the model can accurately predict how the RCW system operates in all year operation conditions.

I.1.8. A Technical Approach for Automated Testing of Safety Systems at Cernavoda Nuclear Power Plant

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This paper analyzes the possibilities of upgrading safety systems at Cernavoda NPP, by means of automation of testing process. In CANDU type NPPs there is only one example of automated testing system, at Darlington Plant. Automated testing is provided for the future enhanced CANDU 6 reactor design. We have analyzed all of the four safety systems: Shutdown Systems no. 1 and no. 2, Emergency Core Cooling System and Containment System. In this paper alternatives to replace manual activities with automated actions are shown, given the limitations of the current design of the Cernavoda NPP. Our goal for the proposed changes was to improve the operation at Cernavoda NPP without obstructing manual testing activities.
The proposed Automated Testing System (ATS) consists of three independent programmable logic controllers, having a sufficient number of inputs/outputs for all of the four safety systems. Interfaces with current testing systems are made with relays. ATS allows performing all tests using the current test procedures and determining of the response times for trip channels. ATS will be implemented separately for each Cernavoda NPP Unit.
The implementation of such a system will help to improve the operation at Cernavoda NPP. At Unit 1, ATS will allow implementing of the improvements made at Unit 2, in the Main Control Room system area at the Emergency Core Cooling System.
The conclusion of the analysis is that for all of the four systems it is possible to achieve an ATS for initiating variables, conditioning variables and for systems actuation logic, manual testing should be kept for the actuating elements (shutoff rods and valves).

I.1.9. Evaluation of Thermalhydraulic Parametres for some Important Accident Transients of CANDU Reactor using RELAP5 Code

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The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. It is also suitable to be used for CANDU type reactor because it has the design modeling capabilities such the horizontal fuel bundle geometry and the heavy water library package. Firstly, a simple RELAP model was developed. It describes the thermalhydraulic circuit between reactor inlet header and reactor outlet header coresponding to B11 CANDU fuel channel. Starting from this point, a preliminary model for primary heat transport system of CANDU 600 reactor has been developed, which has two loops and four passes. To check the the validity of implemented model some important accident transients of CANDU reactor has been simulated. The aim of this paper is to identify the applicability of RELAP5 code to CANDU reactors and to present the results obtained for simulation of 80% break at reactor outlet header and 15% break of reactor inlet header. For each type of transient an evaluation of thermalhydraulic parameters has been done. The main results of the paper consist in: the inlet/outlet header

pressure, the break flow rate, the inlet/outlet header flow rate, the cladding and fuel temperatures. The RELAP5 results are in a fair agreement with similar results supplied by CATHENA code, which is the customary code for CANDU reactors. The differences between results are discussed in the paper.

I.1.10. Safety and Reliability of the Nuclear Power Plants that use Different Thermal Cycles in order to Increase their Global Efficiency

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The Nuclear Power Plants (NPPs), looking to their global efficiency, have large thermal power losses. Use of this thermal power for desalinisation or for ambient heating increases the NPP electrical power, raising their global efficiency. During power operation, a large quantity of heat is evacuated to the environment, heat that changes the environment state parameters. The solution to prevent heat loses is not only to stop these loses but also to convert the thermal power in electrical power. This conversion is possible in many ways, with specific advantages and disadvantages.
The transmission of the heat by means of the steam generator is performed with a high efficiency; also the steam turbines have a good conversion efficiency.
In the thermal power plants based on Rankine cycle most of the heat is lost in the condenser. To convert this heat in electrical power it is necessary to have other cooling agents with appropriate thermal and chemical properties.
Use of Kalyna thermal cycle means using of two thermal agents (a liquid one and a gaseous one) in the secondary circuit of a NPP. That means to have two types of turbines (a steam turbine type and a gas turbine type). This solution increases the global efficiency of the nuclear power unit.
The paper presents the main elements of a nuclear power plant unit that is based on Rankine and Kalina thermal cycles, these cycles being described also.
A PSA study case is also presented in the paper in order to estimate the variation of the accident risk level in the presented NPP and also the modification of the global reliability.

I.1.11. Considerations on assessment of different time depending models adequacy

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Most of nuclear facilities were commissioned in the 70s – 80s, the extension of operating licenses over 40 years for power reactors being a viable option for operators of nuclear power plants to ensure the adequacy of future capacity of power generation, in terms of economic benefits.

The operating period of nuclear power plants can be prolonged if it can be shown that their safety has remained on a high level, and for this, it is necessary to estimate how the aged systems, structures and components (SSCs) influence the NPP reliability and safety.
A very important issue of the ageing analysis and assessments is to estimate the values of the time-dependent reliability parameters. In estimation of time-dependent parameters the frequentist, or classical method can be used, but ussually the graphical approach is preferred due to facile and expedite estimating parameters approach.
To emphasize the ageing aspects the case study presented in this paper will assess different time depending models for rate of occurrence of failures with the goal to obtain the best fitting model. A sensitivity analysis for the impact of burn-in failures will be performed to improve the result of the goodness of fit test. Based on the analysis results, a conclusion about the existence or the absence of an ageing trend could be developed. A sensitivity analysis regarding the reliability parameters will be performed, and the results will be used to observe the impact of using the time-dependent rate of occurrence of failures over the global results.

I.1.12. Requirements to Amend the Main Influence Factors of the Safety Culture after Fukushima Accident

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The safety culture continues to be of great interest to the international nuclear community and constitutes a major challenge with regard to the integration of the lessons learned and improvements into a better approach of the safety.
This paper presents a general model that provides a framework for the safety culture assessment, creating the possibility to identify factors that can significantly influence the safety culture. The main safety culture influence factors (SCIF) used by model are the following: regulatory environment, organizational environment, worker characteristics, socio-political environment, national culture, organization history, business and technological characteristics.
After the analysis of the deficiencies and weaknesses of SCIF in evolution of the Fukushima accident, some issues that may become necessities and requirements to change and improve both the safety culture and safety of the nuclear installations were highlighted. For each influence factor were identified requirements to amend. The results will emphasize the necessity of the human – technology - organization system assessment. Hence it was demonstrated that the safety culture results from the interaction of individuals with technology and with the organization.
The main conclusion of this paper refers to the complexity of the safety culture field. It is demonstrated that safety culture is a multidisciplinary field with accent on the data and principles of the behaviour, social and cultural sciences, engineering and organizational culture. The development, maintenance and improvement of a strong nuclear safety culture represent an ongoing process that requires sustainable commitment.

I.2. Nuclear Reactors and Gen IV

I.2.1. Fractional Neutron Transport in Finite Clumpy Reactors with Higher-order Scattering

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The problem of neutrons transport in nuclear reactors has received a formidable interest. The neutrons transport in clumpy reactors needs a rigorous mathematical model to emphasize such fractional structure. Therefore, the space-fractional neutron transport equation is used in this work to describe the neutrons transport in clumpy reactors. These problems arise when the environmental properties of the background material of the reactors, with which the neutron interact, are fractional in their space or position. As general, the diffusion equation describes only asymptotical solution of the Boltzmann transport equation. For complete description, the passage to fractional derivatives should be performed not from the diffusion equation but from the Boltzmann equation. Boltzmann equation describing the anomalous transport of neutrons within processes of scattering, fission, and absorption, was generalized to fractional form and its solution was considered under more realistic conditions in earlier work. The nuclear reactor dynamics are studied using fractional neutron point kinetics equations. Throughout this work, the space fractional linear Boltzmann equation is approximated using the Pomraning-Eddington technique into two space-fractional differential equations in terms of neutron density and net neutron flux. The resultant two equations are coupled into a space-fractional diffusion-like equation for the neutron density. Laplace transformation method is employed to solve the obtained space-fractional diffusion-like equation. The solution is represented in terms of the Mittag-Leffler function and its different orders. The scattering is considered as high-order scattering to characterize the lightly scattering cases as neutron-nucleus scattering. Numerical calculations are presented graphically to investigate the effect of the fractional order in addition to the effect radiative-transfer properties on the interesting physical parameters (reflectivity, transmissivity, neutron energy, and net neutron flux).

I.2.2. Preliminary Results in Even-Odd Effects in Prompt Emission for ^{234}U(n,f) at 14 Incident Energies

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Even-odd effects in prompt emission for an even-odd fissioning nucleus ^{234}U(n,f), at 14 incident energies (from 0.2 to 5 MeV) are investigated. Using the available experimental Y(A,TKE) data and the Zp model prescriptions the even-odd effects in fission fragment distributions were also studied. The obtained charge polarizations ΔZ and root mean square of the charge distribution exhibit oscillations with a periodicity of about 5 mass

units. The amplitude of these oscillations decreases with increasing incident neutron energy (En). The global proton (Z) even-odd effect in Y(Z) distributions is decreasing almost linearly with increasing incident neutron energy. The neutron (N) even-odd effect in average neutron separation energy <Sn> was also studied and it decreases with increasing En. The even-odd effect in prompt emission, which is due to the Z even-odd effect in fragment distributions and due to the N even-odd effect in <Sn>, is most visible in prompt neutron multiplicity, especially in ν(Z). The increasing multiplicity with En mainly for the heavy fragments group, observed for ν(A), is also observed for ν(Z). The global Z even-odd effect in prompt neutron multiplicity is slightly decreasing from about 9% at En=0.2 MeV to 8 % at En=5 MeV, respectively. These values are at the same level of magnitude as the global Z even-odd effect in prompt neutron multiplicity for the even-even neighbours $^{233,235}U(n_{th},f)$.

I.2.3. Preliminary Results of Total Kinetic Energy Modelling For Neutron-Induced Fission

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The total kinetic energy as a function of fission fragments mass *TKE(A)* is an important quantity entering in prompt emission calculations. The experimentally distributions of *TKE(A)* are referring to a limited number of fission systems and incident energies. In the present paper, a preliminary model for *TKE* calculation in neutron induced fissioning system is presented. The range of fission fragments is chosen as in the Point by Point treatment. The model needs as input only mass excesses and deformation parameters taken from available nuclear databases being based on the following approximations: total excitation energy of fully accelerated fission fragments *TXE* is calculated from energy balance of neutron-induced fissioning systems as sum of the total excitation energy at scission E^*_{sciss} and deformation energy E_{def}. The deformation energy at scission is given by minimizing the potential energy at the scission configuration. At the scission point, the fissioning system is described by two spheroidal fragments nearly touching by a pre-scission distance or neck caused by the nuclear forces between fragments. Therefore, the Columbian repulsion depending on neck and, consequently, on the fragments deformation at scission, is essentially in *TKE* determination. An approximation is made based on the fission modes. For the very symmetric fission, the dominant super long channel is characterized by long distance between fragments leading to low *TKE* values. Due to magic and double-magic shells closure, the dominant S1 fission mode for pairs with heavy fragment mass A_H around 130-134 is characterized by spherical heavy fragment shape and easily deformed light fragment. The nearly spherical shape of the complementary fragments are characterized by minimum distance, and consequently to maximum *TKE* values. The results obtained for *TKE(A)* are in good agreement with existing experimental data for many neutron induced fissioning systems, e.g. $^{233,235,238}U(n, f)$, $^{237}Np(n,f)$ at different incident energies with only one parameter adjustment.

I.2.3. Potential Implications Of Temperature Gradient Of The Spent Fuel Bay On Corrosion Rate Of The Spent Fuel Cladding Elements

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This paper aims to develop a thermal-hydraulic analysis of the Spent Fuel Bay for a CANDU 6 nuclear power plant.
The Spent Fuel Bay Cooling and Purification System provide cooling and purification of the water within the spent fuel storage bay and three auxiliary (discharge, reception and defective fuel storage) bays.
In order to control the corrosion of metal surfaces of both fuel elements sheaths and underwater bay components is performed the chemical control of the Spent Fuel Bay System. However, corrosion processes can’t be arrested and is dependent on several factors, such as temperature.
In the paper will be developed an analyze of the potential implications of temperature values variations of the spent fuel bay on the corrosion behavior of fuel elements claddings.
Based on Flowmaster calculation code there will be modeled temperature evolution of the spent fuel bay in order to quantify the effects of temperature value modification on the corrosion rate of fuel elements claddings.
The Flowmaster calculation code will be used to develop models and calculation assumptions, their geometric configuration and, also, to define input data for hydraulic analysis and calculation assumptions or input data for thermal calculation verification operation of heat exchangers that are part of this system.
The output data from Flowmaster calculation code iterations will be the basis for discussions and conclusion regarding potential implications of temperature and heat flux variations on the amplitude of the corrosion phenomena.

I.2.4. Preliminary Cell Calculations to Support T43 Fuel Design

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Using innovative fuel in CANDU is a long-time challenge addressed by Material Science, as well as Reactor Physics, Thermal-hydraulics and Fuel Performance. Given the flexibility of the new 43 elements CANDU bundle design, several types of innovative fuels were studied, e.g. SEU/RU43 and T43. The latter contains a mixture of Thorium and Slightly Enriched Uranium oxide and also a certain amount of burnable poison in the central element, to reduce the void effect, if possible. It is also recommended for safeguard reasons, since the large amount of highly active minor actinides makes spent T43 fuel less attractive for reprocessing. The first step to move further throughout the large range of variants concerning U235 enrichment, pellet manufacturing and bundle assembling conditions should be the selection of the "best fit" design based upon the fulfillment of several goals such as improved discharge burnup, uniform power distribution in the bundle, smaller Uranium consumption, minor actinide concentration, reduced void reactivity and so forth. The aim of this paper is to investigate the meeting of these

requirements by cell calculations estimating lattice parameters for a CANDU reactor loaded with T43 bundles. The cell calculations were performed using the CANDU-dedicated DRAGON transport code together with the associated WIMS library based upon ENDF/B-VII evaluated nuclear data. The selected fuel bundles are to be tested against more specific requirements regarding reactor operating conditions.

I.2.5. A Neutronic Modeling Approach of LOCAs in a CANDU Core Fed with Thorium-based Fuel Bundles

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As the flexibility of CANDU reactors to use advanced fuel cycles has been demonstrated along the years, new challenges arise from the associated nuclear safety considerations. A largely spread safety analysis regarding the new fuel cycles introduction is the Loss of Coolant Accident (LOCA) analysis. In this paper some LOCA scenarios calculations regarding Rupture Inlet Header (RIH) with magnitudes up to 35% have been performed using the DIREN code- a 3D diffusion computer program developed by Institute for Nuclear Research (INR) Pitesti. The transients considered in this study were modelled using the cuasistatic approximation of time-dependent diffusion equation solved in DIREN program on time steps. Inside of the advanced fuel bundle SEU-43 developed in INR Pitesti, several combinations of Thorium based fuels were placed, in fact pure thorium dioxide in the first 8 innermost fuel elements along with Slightly Enriched Uranium (SEU1.8%) dioxide in the 35 outermost ones. The neutronic amounts of interest supplied by DIREN code were the neutronic flux amplitude along with bundle and channel power peaks during the simulated transients. These physics amounts are illustrated in the paper comparatively to those supplied at the use of Natural Uranium (NU) as fuel in the CANDU standard 37-rods bundle. In spite of larger values obtained for the Thorium based fuel amplitude and power peaks comparatively with the case of NU, a simply estimation of the fuel heat accumulated until transient termination by Shut Down System no 1 showed that the fuel integrity is kept.

I.2.6. A Analysis of CANDU Fuel Element Behaviour in LOCA Test

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After a brief introduction of general LOCA (Loss Of Coolant Accident) conditions applied to CANDU fuel, the paper presents the LOCA test considered for analysis, based on available data and information included in the OECD/NEA-IFPE Database (IFPE/CANDU-FIO-131 NEA-1783). The EXP-FIO-131 experiment, performed at the Chalk River Laboratories in 1983, was a part of an experimental program on fuel performance under high temperature transient conditions such as those associated with the onset of a loss of coolant accident (LOCA). This LOCA test is a complex instrumented experiment that provided quantitative verification data, offering in this way a support for the development of fuel element modeling by comparing the results. The LOCA test simulation has been

performed by using the TRANSURANUS fuel performance code, developed by Institute for Transuranium Elements (ITU), Karlsruhe, Germany, for the thermal and mechanical analysis of fuel rods in nuclear reactors. TRANSURANUS computer code was applied at Institute for Nuclear Research (RATEN ICN), Pitesti, Romania, to study CANDU PHWR fuel elements behavior, in the framework of some licensing agreements between the two nuclear research institutes. CANDU PHWR fuel corresponding fabrication data and in-reactor specific operating conditions are introduced as input data in the code. Also, some models options also specific to CANDU fuel are appropriately selected, in order to obtain a good simulation with TRANSURANUS code. The results presented in this work are illustrating the evolution both of inner fuel element pressure and fuel temperature, during transient conditions. A good agreement between measured maximum centreline temperature and the calculated by TRANSURANUS has been found. Also, good qualitative results were been obtained for the cladding outer surface temperature, fuel surface temperature, outer cladding diameter and thickness of the oxide layer outside cladding. The fuel element survived the LOCA transient without experiencing cladding rupture and the TRANSURANUS code registered also no cladding failure message.

I.2.7. Shielding design proposal for mobile firefighting robot used in fire prevention and explosion situations

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Worldwide direct property damage produced by fire and also fire causalities have gone up significantly with the development of industrialization and urbanization. In many critical situations manual fire fighting forces that are called upon to deal with a fire or explosion scenario have limited action range. A firefighting mobile robot represents an interesting and practical remote controlled machine that successfully replaces fire fighters to perform dangerous firefighting task. The use of a mobile robot firefighter unit enhances the safety of the public and firefighters from fire and related hazards, providing useful data about enclosed buildings environment.
This paper presents a shielding design proposal for a firefighting mobile robot that is used in fire and explosion prevention in enclosed buildings, in very hostile environment, with temperature above 300°C. To protect robotic mechanisms and electronics from intense heat, different shapes of the frontal shield and also materials available on the market have been tested and a design solution was proposed for the frontal shielding of the mobile firefighting robot.

I.2.8. Simulation of the nitrogen liquefaction installation behavior and the yield mapping turbodetentor from tritium separation plant

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Paper aim is to set up contributions to the development and improvement of the optimization and simulation system for tritium separation plant.

Optimization and simulation of industrial processes, in this case the nitrogen liquefaction process from cryogenic distillation module (named installation 300) necessarily must be completed with the acquisition of experimental data using the model or experimental prototype.

Optimization, logic process by excellence, can be achieved both through analytical way and experimental method and frequently the two methods are used as complementary methods.

The main objective pursued by the paper is to achieve optimization, targeting mainly turbodetentor maximization from cryogenic distillation module on the nitrogen liquefaction cycle.

This optimization is useful for achieving a low-temperature nitrogen as necessary to obtain low temperature hydrogen (24 K) on the top of cryogenic distillation column.

Primarily, it was followed to develop the software programs to simulate the evolution process of the liquefaction nitrogen installation in the cryogenic distillation module from experimental pilot tritium separation, according to the important parameters purchased.

Optimization methods, tracked for maximizing the turbodetentor are mainly:

- Determining the optimal values of the input and output pressures from turbodetentor;
- Determining the optimal values of the entry respectively exits of the temperatures from the turbodetentor.

I.3. Nuclear Technology and Materials

I.3.1. Fuel safety investigations at the Institute of Transuranium Elements

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The Institute for Transuranium Elements (JRC-ITU) is one of seven research institutes of the European Commission's Joint Research Centre (JRC). JRC-ITU was founded in 1963 on the site of the German nuclear research centre Karlsruhe. Most of its scientific activities today are dedicated to safety and security of the nuclear fuel cycle, basic actinide research as well as education and training.
New sample preparation methods and safety performance of plutonium and minor actinide containing fuels have been investigated. Samples were prepared at JRC-ITU, have been irradiated in research reactors and returned to ITU for post-irradiation examinations. The tested fuel samples were based on metal alloys (e.g. MASURCA, METAPHIX), mixed oxides (e.g. SUPERFACT, TRABANT, OMICO, SPHERE, MARINE), nitrides and carbides (e.g. NIMPHE, NILOC, GOCAR, POMPEI), but also included Inert Matrix Fuels (IMF) for transmutation targets (e.g. EFTTRA-T4, ECRIX, CAMIX, COCHIX, HELIOS, FUTURIX). These irradiation campaigns have provided significant knowledge on safety performance and properties of solid fuels for various reactor systems.
Current activities concentrate on safety research on fuels for existing power reactors and fuels for transmutation. Ongoing irradiation campaigns, SPHERE and MARINE, focus on safety aspects of mixed oxide transmutation fuels with significant americium content for homogeneous and heterogeneous fast reactor minor actinide

I.3.2. Current Status of the Chemistry and Conditioning Programme for MYRRHA

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In 2010 the Chemistry and Conditioning Programme (CCP) has been established to provide R&D support for the engineering and licensing of the MYRRHA nuclear system. MYRRHA is an accelerator-driven subcritical nuclear reactor, using liquid lead-bismuth eutectic (LBE) as spallation target material and coolant. The CCP team studies various chemistry-related aspects of LBE which are important for safety, operation and decomissioning of MYRRHA.
A large portion of our research is devoted to the measurement and control of dissolved oxygen in LBE. Our achievements in this domain include the development of a family of new oxygen sensors that perform reliably down to 200 °C, several advanced designs of lead-oxide based solid mass exchangers for oxygen supply to LBE and a unique electrochemical oxygen pumping system to precisely regulate dissolved oxygen. We have also constructed HELIOS 3, an installation optimized for reduction of oxygen in LBE by gas-liquid interaction through finely dispersed bubbles of hydrogen-containing gas. Another installation is dedicated to the testing of

various filter concepts for removal of suspended particles from LBE. Our experimental work on oxygen conditioning is supported by a variety of theoretical calculations. Thermochemical calculations have been very successful in predicting the influence of temperature and impurities such as corrosion products on measured concentrations of oxygen in small-scale setups with simple geometries. Detailed CFD calculations coupled with chemical reactions are used to assess oxygen distribution and transport in complex components of the primary system of MYRRHA. An integrated experimental study of oxygen control, sensing and filter/cold-trapping is carried out in our recently commissioned large chemistry loop termed MEXICO. CCP is also responsible for the exploitation and chemistry control of the CRAFT corrosion loop.
A second priority of the CCP team is the experimental study of evaporation of several safety-critical radionuclides from LBE. These radionuclides are formed in the LBE by activation of the coolant (Po), by spallation (Hg, …) or may be released into the LBE through leaking fuel pins (fission products such as I). For the experimental study of polonium release, considered to be one of the most important safety issues of LBE-cooled reactors, a dedicated Po lab has been set up. The evaporation of other elements is typically studied using stable isotopes in a newly built heavy metals chemistry lab. In close collaboration with colleagues from especially the Swiss Paul Scherrer institute, we have discovered several physicochemical mechanisms by which polonium can be released from LBE and we have performed exploratory studies on methods to capture particularly volatile gaseous Po molecules.

I.3.3. Probabilistic fracture mechanics applied for LBB case study: International Benchmark

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The paper is focused on the application of probabilistic fracture mechanics to evaluate the structural integrity of a case study chosen from experimental Mock-ups of STYLE project. Existing Engineering Assessment methods often handle residual stresses in an over-pessimistic fashion, which is necessary when no reliable information on the real residual stress distribution present in a component under investigation can be obtained. The experimental and numerical residual stress estimations is going to allow the methods in defects assessment procedures (R6,etc.) to be benchmarked and improved. A reliability model for probabilistic structural integrity focused on the assessment of TWC in the pipe weld under complex loading (bending moment and residual stress) has been setup. The basic model is the model of fracture for through-wall cracked pipe under elastic-plastic conditions. The corresponding structural reliability approach is developed in MATLAB environment with the probabilities of failure associated with maximum load for crack initiation, net-section collapse but also the evaluation the instability loads. To verify the predictions accuracy of the reliability model, a numerical benchmark is provided by using a case study from literature. The probabilities of failure for a through-wall crack in a pipe subject to pure bending are evaluated by using crude Monte Carlo simulations. The results of the numerical benchmark are consistent with those obtained from the literature for the mentioned case. The intention is to provide detailed steps of using the probabilistic approach for the structural integrity assessment in the context of ageing and lifetime management of pressure boundary/pressure circuit component.

I.3.4. Experimental Research Concerning the Degradations of CANDU Steam Generator

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Nearly all of the steam generator fouling processes and the related degradations were attributed to secondary side water chemistry conditions and excursions, many of which resulted from condenser cooling water ingress.
The investigation of the structural materials corrosion in correlation with the water chemistry, as well as the impurities and corrosion products concentration and deposition and their removing from the CANDU steam generators is a very active field and both the experimental works and the understanding of the mechanisms involved are submitted to some rapid changes and permanently open to the research. To provide information about the corrosion behaviour of the structural materials from CANDU steam generators under normal and abnormal conditions of operation and to identify the failure types produced by corrosion we performed a lot of corrosion experiments. These experiments consisted in chemical accelerated tests, static autoclaving and electrochemical investigations.
The purpose of this experimental research consists in the assessment of corrosion behaviour of the tubesheet material, carbon steel SA 508 cl.2, at normal secondary circuit parameters (temperature - 260^{0}C, pressure - 5.1MPa). The testing environment was the demineralised water without impurities, at different pH values regulated with morpholine and cyclohexylamine (all volatile treatment – AVT).
The results are presented like micrographics and graphics representing weight loss of metal due to corrosion, corrosion rate, total corrosion products formed, the adherent corrosion products, released corrosion products, release rate of corrosion products and release rate of the metal.
The gravimetric method, optical metallographic microscopy as well as electrochemical measurements have been used to evaluate the corrosion behavior of the tubesheet material.

I.3.5. Characterisation of Oxides Developed in Supercritical Water on 18Cr-20Mn ODS Steel

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The international programs for fusion have developed and tested ODS steels at elevated temperature. Fuel cladding and structural applications of materials in next generation of nuclear reactors, generation IV for fission, requires low activation steels and few alloying element like Ni, as austenite stabiliser, are undesired. In SCW reactors, supercritical water is an aggressive media and the compatibility of materials with this coolant has to be study.
In this paper, new austenitic steel, 18Cr-20Mn reinforced with yttrium and titanium oxides dispersion, was tested in supercritical water in a static autoclave at 550^{o}C and 600^{o}C temperature values and 25MPa pressure up to 1600 exposure hours. The oxidation kinetics obtained from weigh gain measurements has proven a weight loss signifying non-adherent oxides and that are lost in the media.
The surface of oxidised coupons was investigated by scanning electron microscopy using secondary electrons (SE). Oxide layers formed on 18Cr-20Mn ODS alloy during exposure

at 600°C were cross-sectional examined using backscattered electron (BSE) signals. And the stratification of oxides in depth has been evaluated using Energy Dispersive X-Ray Spectrometry (EDS). With these different techniques has been estimated the protective character of oxide films formed on surface.

I.3.6. Cool-Down Oxidation in Steam and Air of Preoxidized ZY-4 Cladding

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The purpose of this paper is to present the results obtained in cool-down oxidation test in steam and air of the pre-oxidized Zy-4 cladding at different cooling rate. In a SETARAM termo-balance, 15 samples were first pre-oxidized in steam at oxide scale of 20 to 80 μm thickness. The range of the oxide thickness for 10 air pre-oxidized samples was between10 and 80 μm. All the pre-oxidized samples were then submitted to a cool –down test between 1700K and the room temperature at cooling rates between 5 and 90K/min. The weight gain and the temperature data were registered and plotted. From the experimental curves using nonlinear kinetic equation the order of the reaction, the kinetic coefficient dependence of temperature and the activation energy of the reaction were obtained. The fitted curves for same cooling rate and different levels of pre-oxidation are compared. The results obtained for a given level of pre-oxidation, at different cooling rates, are also compared and discussed. For all samples the reaction follows a parabolic law and practically stops at 900K. The weight gain descries with the increasing cooling rate. The activation energy in each case, have different values depending on temperature ranges: between 1500 and 1800K, between 1100 and 1400K and bellow 1100K.

I.3.7. Analysis of Fission Gases Released in the Void Volume of Irradiated Fuel Rods

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The gaseous fission products tend, due to their nature and their insolubility, to produce changes in the fuel pellet volume and increase pressure in the void volume of irradiated fuel elements. Considering the impact that these changes can have in terms of nuclear safety, it is necessary to study the behavior of fission gas during irradiation. In order to study the production and release of fission gas during irradiation and post-irradiation experiments, various experiments can be designed. This paper presents the installation for cladding puncture and analysis of fission gases released into the void volume of irradiated fuel elements and recent experimental results obtained on CANDU type fuel elements.
The installation for cladding puncture and fission gas analysis was designed and manufactured at RATEN ICN Pitesti. It is used for:

- Measurement of the pressure and volume of gases in the void volume of the fuel rod;
- Measurement of the fuel rod internal void volume;
- Determination of the chemical composition of fission gases, including isotopic composition of the fission gases where applicable.

The paper contains also a description of the method used for the analysis of fission gases. A special attention is paid to the calibration method used for gas analysis by quadrupol mass spectrometry. A dedicated device was designed in order to mix pure gases in different concentrations for the calibration of the mass spectrometer.

I.3.8. Investigation on mechanical alloying process for V-Cr-Ti Alloys

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This work is part of a global trend of using new processes for obtaining and processing advanced materials for nuclear industry. Mechanical alloying (MA) is a powder metallurgy technique efficient for fabricating advanced materials, and has been used for strengthening structural materials including vanadium alloys.

In this study a high-energy ball mill is used to obtain a V-Cr-Ti alloy. The process parameters used are: 72 hours milling time, a rotation speed of 400rot /min in cycles of 10min milling and 5min pause and argon protective atmosphere. It is known that in the initial milling stages ductile particles get flattened and weld together by ball-powder-ball collisions and this leads to an increase in particle size, and with continued milling the particles are refined and the grain size decreases. But brittle particles get fragmented and tend to become occluded by the ductile constituents and trapped into them.

After 72 hours grinding, the element distribution of V-Cr-Ti milled powder is investigated by scanning electron microscopy (SEM) and energy dispersive X-ray (EDX) analysis. It is observed that the particles size increase up to hundreds of microns and the V, Cr and Ti are uniformly distributed in the MA powder.

Changes in the material composition are analysed, also, by X-ray diffraction (DRX). It seems that part of the Ti remains non-dissolved in the V matrix. After milling there is a change of crystal lattice by increasing network defects and network deformation. Crystallite size distribution and strain microtensions II distribution are analysed by Warren-Averbach method. There is a decrease in the crystallite size to 12.7nm, compared with the one of the original (40-50nm). Deformation induced by microtensions II increases during processing, from approximately 0.4% for the initial powders to 0.6% after milling.

The MA powder is consolidated by using the not too common but rapidly developing technique of spark-plasma sintering (SPS).

KEYWORDS: vanadium alloys, mechanical alloying, spark plasma sintering.

I.3.9. Research for the Evaluation of the Chemical Decontamination Processes Effect on Metallic Surfaces Using ASME and API Standardized Technologies

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In nuclear facilities, radioactive material from both internal and external surfaces of components and systems is removed by chemical, mechanical, electrochemical decontamination procedures. Decontamination is necessary for reducing radiation field prior to inspection, maintenance or repair activities of reactor systems or their components. Prior the use of any decontamination procedure it is necessary to understand if there exist any impact on material degradation and component reliability. Some of the decontamination procedures might affect or damage the surface of the item treated to the point where the item cannot be safely used. The present paper aim is to focus on the potential effects on functional characteristics and reliability of certain well known chemical decontamination procedures used with nuclear components and systems. Chemical decontamination is based on the dissolution of the oxide layer of the contaminated surfaces by use different chemical reagents. A review of relevant literature and experimental data revealed that the significance of electrochemical aspects has not been taken into account in the evaluation of chemical decontamination procedures. It is attempted an approach to establish if standardized ASME and API methods can be used to evaluate the impact of chemical decontamination processes on metallic surfaces together with a better understanding of the governing mechanisms.

I.3.10. Some Calculation Formulae for the CANDU 6 Hypoid Gearing.A Physical Point of View

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The CANDU 6 reactor’s Adjuster Unit consists of a tubular, stainless steel, neutron absorbing element which is raised and lowered within a guide tube. The Adjuster Unit is suspended from a cable which is wound around the sheave of a drive mechanism. To provide a very fine “up-down” displacement of the absorbing element, the adjuster drive mechanism incorporates a hypoid-gearing containing a conical spiroid pinion (N_1 =1) and a conical spiroid gear (N_2 = 150), having their axes at rights angle. Here N_i represents the wheels’ teeth number. By its conical spiroid pinion, its conical spiroid gear with double-involute teeth, and its reduction (transmission) ratio $i = N_2 / N_1 = 150$, this hypoid-gearing seems to be a special type mechanism, not encountered (by the authors, at least) in the classical engineering textbooks, where the most analyzed examples only are for $N_1 \geq 5$ and i = 50÷100. Elaborated from a physical point of view, the paper demonstrates and proposes a number of calculation formulae for the main geometrical elements of a same CANDU 6 hypoid-gearing. Finally, the paper presents a very good (exact, practically) superposition between the values obtained using the proposed calculation formulae, and the known CANDU 6 hypoid-gearing’s geometrical values. Al these general calculation formulae are helpful for well understanding the dynamics and kinematics of the adjuster

drive mechanism, and represent the minimum needed knowledge for some possible applications, in other mechanical systems, of a hypoid-gearing with a so large reduction ratio (i ≥ 150).

I.3.11. "Analysis of Incoloy 800HT alloy tested in Thermal Transient Conditions"

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This paper investigated Incoloy 800 HT (UNS N08811) alloy after some thermal transient tests. The study continues prior tests realized in ICN Piteşti concerning utilization of some nickel-based alloys in the steam generators construction. The experiment consisted in thermal transients using the following conditions/scenarios: fast heating rates (50° and 90°C/minute) up to 1,000°C, maintaining this temperature level (0 and 60 minutes), and slowly or fast cooling. This alloy is one of the candidate materials for construction of the steam generators of the future NPP reactors which must operate in severe conditions (high temperature, thermo-mechanical stress, and aggressive media). The analysis consisted in metallographic examination (microstructure, micro-hardness) and traction tests (strength resistance). After thermal transient tests, the samples were prepared by metallographic methods (cutting, mounting, grinding, polishing, and etching) and then investigated using the Olympus GX 71 optical microscope, the OPL Microduromètre with automatic cycle and WALTER BAI traction device. The average grain size was determined by linear interception method (Heyn). Average grain size ASTM (G) is in accordance with the structural requirement of ASME SB-409 standard (G=5.0 or coarser). The micro hardness was calculated by the relationship from the device technical book. Also, on the traction diagrams the following mechanical characteristics: strength resistance (R*m*), elongation at rupture (*A*) and elastic modulus (E) were obtained. Strength resistance and elongation is conform to the material requirements of ASME SB-409 standard (Table 3: R*m*=min. 450 MPa; *A*=min. 30%). The tested alloy was compared with the „as received" material, and the results showed a good behaviour of this alloy in the presented conditions/scenarios. The results can serve to create a database with candidate materials for Generation IV reactors heat exchangers construction, tested in different conditions.

I.3.12. THE assessment of temperature influence on UO_2 fuel pellets fragmentation in air containing atmospheres

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The aim of this work was the assessment of temperature influence on UO_2 fuel pellets fragmentation by oxidation in air containing atmospheres. In order to achieve this, unirradiated fuel pellets were manufactured from UO_2 powder by pressing and sintering. The oxidation tests were performed using the FIPRED equipment. The main parts of this equipment are: a horizontally ceramic tube furnace mounted on a travelling track, a stationary stainless steel tube, measurement and control equipment and a nacelle. The

samples were heated in nitrogen atmosphere until the testing temperature (673- 1273K) was reached. Then, a mixture of air-nitrogen (20, 40, 60 and 80%) was introduced to oxidize the samples. The nonfragmented part of samples was weight after every hour of testing in order to evaluate the fragmentation rate.
The techniques used to evaluate the morphological aspects and the particles size distributions of the powders resulted by fragmentation were scanning electron microscopy and sieving.
The graphs presented in this paper show the dependence of samples fragmentation rate and particles size distributions with temperature. The SEM images presented in the paper reveal the morphological aspects changes with the testing temperature.
Also, the kinetic curves of samples fragmentation obtained for a content of 80% air in nitrogen were fitted with parabolic function to determine the kinetic parameters.
The fragmentation rate was the highest for oxidation at 773K.
The sizes of particles obtained after fragmentation increases with temperature increase, for values of air content of 20, 40, 60 and 80% in nitrogen.

I.3.13. Development of the Qualitative Techniques for Monitoring of the Power Cables Condition Using Infrared Analysis

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The results are useful to identify, model and manage the power cable material ageing phenomenon in the NPP. There is an issue to provide the non-aged samples of many years back manufactured cables for laboratory tests.
For assessment of the insulation material ageing degree for cable, by infrared (IR) analysis used the equipment Perkin Spectrum 100. The specimens analysed have been sampled from CYY 3 x 25 mm type cable sections. These sections were 3.5 mm long and have been accelerated thermal aged (by Joule Lenz effect), the thermal ageing time equivalent for NPP operation being of 10, 20, 30, 40 and 50 years.
This technique is using the fact that, the polymers are degrading, the structure changes taking place leads on development of some new cross-linking having different absorption characteristics (wave length) than the initial un-aged material linking.
A characteristic absorption band for the same functional bunch (inside of molecule) is recovering at approximately the same band number values in the IR spectrum of any molecule (the bunch characteristic vibration). Their presence assumes replacing of one or several Hydrogen atoms from a hydrocarbon molecule with atoms or groups of atoms of the organogenic elements.
The dominant oxidation mechanisms for the aged polymers in air are producing carbonyl species that absorb infrared light at characteristic wave numbers around 1720 cm^{-1}.
The absorption intensity gives the possibility to determine the oxidation levels. For each sample established, using the specific equipment software, the absorption intensity absolute values of IR radiation due to C=O carbonyl species vibration. The characteristic absorbance of this polymer spectrum is founding out to the wave number value of 2000 cm^{-1}.The absorbance trends, at 1720 cm^{-1}, to grow related to increasing of degradation.
The modern spectrometers FTIR (IR spectrometers using the Fourier transform) works analysing the infrared light reflected by a sample surface and can generate spectral data

as identical shape with those produced by films analysing. This means that an intrusive technique requiring sampling converts strictly in a condition monitoring technique available currently, indeed non-intrusive. This technique isn't applying to polymers containing heavy absorbent materials such as carbon black (i.e. GSPE and PCP).

I.3.14. Metallographic Examination of (UTH) O_2 and UO_2 Fuel Tested in Power Ramp Conditions in TRIGA Reactor

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The purpose of this paper is to determine, the behaviour of two fuel experimental elements (EC1 and EC2), by destructive post-irradiation examination. The fuel elements were mounted inside a pattern port, one in extension of the other and irradiated in power ramp conditions in order to check their behaviour. Fuel element 1 (EC1) contains $(UTh)O_2$ pellet, and other one (EC2) UO_2 pellet. The results of destructive post-irradiation examination are evidenced by metallographic and ceramographic analyses. The data obtained from the post-irradiation examinations are used, first to confirm the security, reliability and nuclear fuel performance, and second, for the development of CANDU fuel. The EC1 fuel pellet presents radial fissures on the whole section and another one circular. There are not visible effects of fuel – sheath mechanical and chemical interaction. Fuel microstructures are consisted of a homogeneous phases mixture, thorium and uranium. Because the compounds of thorium are very stable is difficult to obtain the grains microstructure. It is noticed hydrates plates uniformly distributed inside the sheath. It presents on the outer side a zirconium oxide layer, continuously and uniform. The EC2 fuel pellet presents radial fissures on the whole section, a circular fissure and a central hole. There are not visible effects of fuel – sheath mechanical and chemical interaction. Fuel microstructures are consisted by columnar grains, equiaxial grains and unaffected grains. It is noticed hydrates plates parallel distributed with the sheath. It presents on the outer side a zirconium oxide layer, continuously and uniformly. The results obtained by destructive examinations regarding the integrity, sheath hydrating and oxidation as well as the structural modifications are typical for fuel elements tested in power ramp conditions.

I.3.15. Corrosion of dissimilar welds between martensitic stainless steel and carbon steel from secondary circuit of CANDU NPP

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Corrosion damages of welds occur in spite of the fact that the proper base metal and filler metal have been correctly selected, industry codes and standards have been followed and welds have been realized with full weld penetration and have proper shape and contour. It is not unusual to find that, although the base metal or alloy is resistant to corrosion in a particular environment, the welded counterpart is not resistant. However, there are also

many cases in which the weld exhibits corrosion resistance superior to that of the base metal.
In secondary circuit of a Nuclear Power Station there are some components which have dissimilar welds.
Our experiments were performed in chloride environmental on two types of samples: non-welded (420 martensitic steel and 52.2k carbon steel) and dissimilar welds (dissimilar metal welds: joints beetween 420 martensitic steel and 52.2k carbon steel). To evaluate corrosion susceptibility of dissimilar welds was used electrochemical method (potentiodynamic method) and metallography microscopy (microstructural analysis).
The present paper follows the localized corrosion behaviour of dissimilar welds between austenitic stainless steel and carbon steel in solutions containing chloride ions.
We have been evaluated the corrosion rates of samples (welded and non-welded) by electrochemically.

I.3.16. General Corrosion Properties of Austenitic Alloys In Supercritical Water

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Based on safety and efficiency, the Generation IV forum has selected six innovative concepts for nuclear reactors. The supercritical water reactor (SCWR) is a promising design with many advantages (a single phase coolant with high enthalpy, the elimination of components and a higher efficiency 45% vs. 33% in current Light Water Reactors). Operating above the thermodynamic critical point of water (374 ^{0}C/22.1 MPa), supercritical water (SCW) is expected to be more corrosive to conventional structure materials commonly used in actual nuclear power plants. The performance of materials used as thin-walled components in this environment continues to be a challenge so, the design for the SCWR calls for advanced cladding materials with high irradiation and corrosion resistance, as well as high temperature strength properties .
This paper presents a preliminary study regarding general corrosion behaviour of some austenitic stainless steels (304L, 310S, 316L and 321) immersed in supercritical water at 550 - 600ºC up to 1680 hours. After exposure to supercritical water were evaluated the weight changes using gravimetry, oxide film thicknesses using scanning electron microscopy (SEM) and chemical distribution of the elements in oxides using energy dispersive X-ray spectroscopy (EDS). The thickness of the oxides was evaluated by optical microscopy. Gravimetric analysis showed the lowest weight gainin case of 310S alloy. After surface analysis it was determined that 310S alloy had the thinnest and adherent oxide layer, followed by 316L and 321 alloys. Further investigation is needed to confirm other material properties of the austenitic alloys under supercritical water conditions.

I.3.17. Neutron Scattering, X Ray Diffraction and Electron Spectroscopy Characterization of Microstructures Developed on 316L AND 304-L Steels by Plasma Electrolysis Processing

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Alloys 304-L and 316-L are austenitic steels that have been used extensively as structural nuclear materials. The candidate materials for Generation IV nuclear power concepts include modified stainless steels, to improve corrosion resistance in extreme conditions (liquid metals, temperature higher the upper limit of conventional austenitic stainless steels). It has been demonstrated that, especially in the high temperature range, the corrosion resistance of structural materials can be enhanced by FeAl alloy coating .Design of stainless steel AISI 316L and AISI 304L steels used in reactor technique is closely related to the development of physico-mechanical models to forecast materials' serviceability and endurance in the most severe operating conditions The surface modification of austenitic stainless steels as AISI 304L and AISI 316L was performed by complex surface treatments, including surface nano-structuring of steels by Plasma Electrolytic processing .
Was investigated the new structure created by specific surface treatment for above mentioned steels using the SANS instrument existing at BNC. The application of magnetic field (up to 2 T in the samples) helped to magnify the contrast of different grains of steels. The measurement was done at wavelength λ=0.6 nm in order to avoid double Bragg scattering on grains. Consequently the momentum transfer range was $q=(4\pi/\lambda)\sin(\theta/2)=0.1\text{-}2.7\ nm^{-1}$ which covers the scales R ~1/q ~ 0.3-30 nm of nano-defects. The result of SANS experiment elucidated the details of grain size distribution, their changes under heat and chemical effect.
Were also performed diffraction measurements at LNF-DUBNA, FSD diffractometer to put in evidence the phase transition under the treated surface, for the same samples measured before at the SANS instrument.
Stacks of several samples were used in transmission geometry to increase the volume affected. The results of measurements by neutron scattering techniques will be correlated with results of measurements by complementary techniques (Electron spectroscopy, XRD) which were also performed for characterization of microstructures developed on 316L and 304-L steels by Plasma Electrolysis Processing.

I.3.18. Hydrogen Concentration Determination In Pressure Tube Samples Using Differential Scanning Calorimetry (DSC)

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Zirconium alloys are widely used as a structural material in nuclear reactors.

It is known that zirconium based cladding alloys absorb hydrogen as a result of service in a pressurized water reactor.
Hydrogen absorbed (during operation of the reactor) in the zirconium alloy, out of which the pressure tube is made, is one of the major factors determining the life time of the pressure tube. For monitoring the hydrides, samples of the pressure tube are periodically taken and analyzed.
At normal reactor operating temperature, hydrogen has limited solubility in the zirconium lattice and precipitates out of solid solution as zirconium hydride when the solid solubility is exceeded.
As a consequences material characterization of Zr-2.5Nb CANDU pressure tubes is required after manufacturing but also during the operation to assess its structural integrity and to predict its behavior until the next in-service inspection.
Hydrogen and deuterium concentration determination is one of the most important parameters to be evaluated during the experimental tests.
Hydrogen present in zirconium alloys has a strong effect of weakening.
Following the zirconium-hydrogen reaction, the resulting zirconium hydride precipitates in the mass of material. Weakening of the material, due to the presence of 10 ppm of precipitated hydrogen significantly affects some of its properties.
The concentration of hydrogen in a sample can be determined by several methods, one of them being the differential scanning calorimetry (DSC). The principle of the method consists in measuring the difference between the amount of heat required to raise the temperature of a sample and a reference to a certain value. The experiments were made using a TA Instruments DSC Q2000 calorimeter.
This paper contains experimental work for hydrogen concentration determination by Differential Scanning Calorimetry (DSC) method. Also, the reproducibility and accuracy of the method used at INR Pitesti are presented.

I.3.19. Dimensional Measurements and Eddy Currents Control of the Sheath Integrity for a Set of Irradiated CANDU Fuel Elements

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During irradiation in the nuclear reactor, fuel elements undergo dimensional and structural changes, and changes of sheath surface condition as well, which can lead to damages and even loss of integrity. This paper presents the results of dimensional measurements and of examination technique with eddy currents for three fuel elements of an irradiated CANDU fuel bundle. One of the fuel elements (FE), which is studied in detail, presented a crack about 40 mm long.
Dimensional changes mainly consist in increased diameter, bending, cambering and stretching of fuel element and are the result of swelling of the sheath and of the fuel and also of the fuel-sheath interaction induced by nuclear radiations. The measurement system is a machine equipped with step by step engines, for vertical movement and rotation of the fuel element, a console for diameter measurement with two displacement transducers, diametrically opposite, mounted on the machine and a control command console. Diameter measurement was performed along the fuel element with a 1 mm step, on three longitudinal directions: 0°, 120° and 240°. For the studied FE, the most pronounced dimensional change was found to be an increase in diameter of 250 µm on a length of 60 mm in the crack area; the maximum diameter is 2.03% larger than the reference diameter provided by the manufacturer.

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). The control with flaw detector was performed along the FE, on 80 equidistant generators. After longitudinal scans, circular scans were also made, in the detected points of interest. The defects differentiation was done taking into account the phase of the signal, indicating the nature of the defect (external unpierced, internal unpierced, pierced) and the amplitude of the signal which indicates the size of the defect. As a result of control with eddy currents, areas with unbored pores, unbored external crack, mechanical dents, scratches were detected. These defects were also highlighted by plain visual examination.

I.3.20. Study of CANDU Fuel Elements Irradiated in a Nuclear Power Plant

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The object of this work is the behaviour of CANDU fuel elements after service in nuclear power plant. The results are analysed and compared with previous result obtained on unirradiated samples and with the results obtained on samples irradiated in the TRIGA reactor of INR Pitesti. Zircaloy-4 is the material used for CANDU fuel sheath. The importance of studying its behaviour results from the fact that the mechanical properties of the CANDU fuel sheath suffer modifications during normal and abnormal operation. In the nuclear reactor, the fuel elements endure dimensional and structural changes as well as cladding oxidation, hydriding and corrosion. These changes can lead to defects and even to the loss of integrity of the cladding.
This paper presents the results of examinations performed in the Post Irradiation Examination Laboratory (PIEL) of INR Pitesti on samples from fuel elements after they were removed out of the nuclear power plant:
- dimensional and macrostructural characterization;
- microstructural characterization by metallographic analyses;
- determination of mechanical properties;
- fracture surface analysis by scanning electron microscopy (SEM).
A full set of non-destructive and destructive examinations concerning the integrity, dimensional changes, oxidation, hydriding and mechanical properties of the cladding was performed. The obtained results are typical for CANDU 6-type fuel.
The obtained data could be used to evaluate the security, reliability and nuclear fuel performance, and for the improvement of the CANDU fuel.

I.3.21. Study of Archaeological Objects by Neutron Imaging, XRD and XRF

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The activity was performed in connection with the research contract with IAEA entitled "The neutron and gamma imaging method combined with neutron-based analytical methods for cultural heritage research", in the frame of a current CRP, "Application of Two and Three Dimensional Neutron Imaging with Focus on Cultural Heritage Research", that helps curators to reveal the internal structure and composition of the objects. Archaeological objects were borrowed from Arges County Museum and investigated at Institute for Nuclear Research by neutron imaging, X-ray fluorescence and X-ray diffraction. Were investigated metallic objects made in iron, copper and its alloys, silver and zinc representing arrowheads, a knife, head pins, coins, a brooch, bracelets, pin belts, buckles, brackets, crest fragments and round disks discovered in southern part of Romania mostly Dacian and Roman origin. For neutron imaging was used the neutron imaging facility from tangential channel of the TRIGA ACPR to put in evidence the internal structure of the objects. To put in evidence the elemental and chemical composition of the objects were performed investigations by X-ray fluorescence (semi-quantitative analysis) with ARL ADVANT 2500 spectrometer and X-ray diffraction with X'Pert PRO MPD diffractometer. After neutron imaging the elemental composition and concentration levels in samples were determined by X-ray fluorescence and information was completed by chemical composition by X-ray diffraction to offer valuable information in archeological research about composition, structure of the bulk, presence of alteration, inclusions, typology of the location of material extraction, manufacturing techniques etc. This work is an example of application of neutron imaging and other penetrating radiation-based analytical methods for cultural heritage research that had the aim to involve some of the non destructive investigation methods available at the Institute for Nuclear Research.

I.3.22. Hydrogen Absorption Properties of U-Zr Alloy

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The development of hydrogen storage materials with high capacity is one of the key issues for wide practical applications in hydrogen energy systems. The uranium-zirconium alloy is a promising material for hydrogen storage. In order to develop new hydrogen storage materials, hydrogen absorption properties of zirconium-rich alloy of 55.8wt.%Zr (U-55Zr) were examined at temperatures from 350°C to 600°C and the hydrogen pressure slightly lower than one atmosphere. Hydrogen-absorption isotherms were measured at 350, 400, 500 and 600°C. The hydrogen absorption measurements have been made on four alloy specimens in SETARAM SETSYS EVOLUTION 24 thermo balance, by thermo-gravimetric (TG) analysis. The U-55Zr alloy was prepared from the constituent elements by powder metallurgy using depleted uranium and sponge zirconium and studied in as cast condition as hydrogen storage material. Microstructure observation was conducted for the starting alloy specimens and hydrogenated specimens using a scanning electron microscopy (SEM) coupled with energy dispersive spectroscopy (EDS). Crystal structures of the specimens were also analyzed with X-ray diffraction technique. The hydrogen capacity of U-55Zr alloy was determined by weighing the samples prior to and after the hydrogenation process and the kinetic of hydrogen absorption related to the hydride composition was analysed and discussed. Initially, at a hydrogen pressure of one atmosphere at 600°C, U-55Zr absorbed an amount of hydrogen. Then by decreasing the temperature stepwise by 100°C steps, the hydrogen concentrate increased and the specimen not disintegrated into fine powder. The alloy consisted of two crystallographic phases before hydrogenation and the hydrogenated specimens mainly consisted of the ternary hydride (UZr2Hx), but contained also zirconium hydride (ZrH2, ZrH1.6) and other phases. This compound slowly reacted with hydrogen to form ternary hydride and it showed high resistance to powdering. Hence, the U-55Zr alloy has a potentiality to be a suitable material for non-powdering hydrogen storage or for others.).

I.3.23. Microbiologically Influenced Corrosion of SA106gr.B Carbon Steel in Raw Water

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Many of the secondary circuit pipes of a CANDU nuclear power plant are made from carbon steel SA 106gr.B. This material is susceptible to microbiological corrosion, which has major implications in the degradation of the carbon steel components.
Microbiologically influenced corrosion (MIC) is the corrosion resulting from the presence and activity of microorganisms, including bacteria and fungi.
This paper presents the evaluation of microbiological corrosion susceptibility of carbon steel SA106gr.B in raw water.
The experiment consisted of a series of electrochemical accelerated tests which evaluated the pitting corrosion susceptibility and determined corrosion rates before and after the immersion.
The microbiological analysis of the water determined the types of bacteria and bacterial concentration present in water and in biofilms.
Microbiological analysis of the water sample emphasized the existence, in small numbers ($10\text{-}10^{1}$ ml^{-1}), of heterotrophic aerobic bacteria, sulphate-reducing bacteria and iron-oxidizing microorganisms. Along with sulphate-reducing bacteria, the heterotrophic aerobic bacteria and the iron-oxidizing microorganisms are categorized as having an important role in the corrosion of metals, including steel.
The surfaces of the tested samples were analysed using the optical and electronic microscope, and emphasized the role of bacteria in the development of biofilms under which appeared characteristics of corrosion attack.
Potentiodynamic polarization tests showed an increase in the corrosion current (from 26.05 µA to 61.04 µA) and the corrosion rate (from 0.076 mm / year to 0.179 mm / year) after immersion of the samples in water.

I.3.24. Evolution of the Irradiation Parameters of Compact-Tension (CT) Pressure Tube in C5 Capsule

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The irradiation devices provide technical support for irradiation tests on structural materials and nuclear fuel for CANDU type nuclear power plants.
Capsule C5 is an irradiation device of TRIGA SSR, which is designed for irradiation of structural materials in an inert environment for mechanical behaviour characterization and the material microstructure evolution during irradiation.
Capsule C5 was designed and manufactured entirely at INR in 1984. Initially it was designed to test stainless steel samples.
The test section of C5 capsule includes the capsule pressure tube, probe port and test section support. The pressure tube provides a tight inert helium atmosphere around irradiated probes. The probe port provides the samples support in maximum flux reactor core area. There are six probes supports which are located in the lower part of the fuel port, 350 mm in length. They are designed of aluminium to enhance the heat transfer. Every probe support forms a level of irradiation probes. Each piece of the probes level is provided with radial channels to fit the probes.
C5 capsule is provided with the following instrumentation:

- temperature;
- flux;
- pressure.

Temperature monitoring is provided by 12 Cromel-Alumel T/C.
Neutron flux is measured in the proximity of the probes by 2 Ag colectrons, with compensation, type Neutrocoax with a 200 mm active part.
C5 capsule has been used in 2010÷2015 to perform an irradiation test of compact-tension (CT) pressure tube which is useful for the hydring cracks (DHC) phenomena analysis.
The paper presents evolution of the irradiation parameters in 1370 hours from 04.12.2014 to 05.03.2015. It is expected that the results presented in this paper will be useful when designing the instrumented capsules for an irradiation test of ODS.

I.3.25. Demineralized Water Flow Cancelling Experiments with Ice Plug into High Diameter Horizontal Tube (300 Nominal Diameter)

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The isolation with ice plug of a high diameter horizontal pipeline section is a specific technique for repairs activities/ replacements of components owning to thermo-hydraulic installations working with liquid agents. The application of such technique don't assumes stopping of the entire system. The ice plugging inside of the pipeline assumes using of a special device and of an own specific technology for application. The device, assembled on the horizontal pipeline which have to isolate in the system, it forms an annular space with outer tube surface, the space drowned continuously by the liquid nitrogen. The heat transfer from the working fluid to the liquid nitrogen generates its boiling in the annular space, the vapours blowing off at the upper part of the device through one or two holes. Into the pipeline, in the plugging device influence area the ice deposits in subsequent layers. The shape and dimension of the deposits depends on the fluid agent temperature, its flow rate, the injection way of the liquid nitrogen, the pipeline diameter and on the component parts of the plugging device. The paper contains a brief description of the experimental technological facilities used, followed by setting off the main moments in evolution of two experimental tests carried out on the test section with 300 nominal diameter for demineralized water. The results analysis and some conclusions outlined at the end is pointing out to the fact that the shape of the ice deposits depends on tube diameter size, the liquid nitrogen injection method, the demineralized water temperature evolution, and especially, by the flow rate. In our case, the tube diameter value and the liquid nitrogen amount available for a test influenced major the sizing of the device. The paper is dedicated to the specialists working in the research and technological engineering.

I.3.26. The Influence of the Preliminary Clamping Force in Garter Spring Simulator Behaviour after First Six Years of Operation

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The garter spring spacer is a torsional spring of special constructive appropriate for fitting-out the CANDU 6 fuel channel. The pressure tube ageing lead to decrease of gap made with calandria tube. Continuous decrease in time of the gap affects directly the behaviour of garter spring spacers during fuel channel assembly operation. Inside of garter spring spacer – pressure tube – calandria coupling, the coils being in direct touch are turning round resulting in compression of the free coils. The sag developed in free coils of the garter spring spacer lead to increase of its pre-clamping force outside of pressure tube. The influence of the ageing phenomenon of the elastic/resilient coupling between pressure tube – garter spring spacer – calandria is researched to understand the garter spring spacer behaviour. The pre-clamping force value of the garter spring spacer at assembling is all-important for its behaviour in time. Instead of the original garter spring spacers, the tests carried-out with hand-made garter springs using spring wire, taking as reference element the geometrical characteristics of the original spacer. The increase of the preliminary-clamping assembling force has performed by reduction of the coils number. The paper contains a brief description of the experimental technological facilities used, followed by outlining of few significant moments in development of some experimental tests carried out in laboratory conditions, avoiding pressure boiled water and radiation working condition. The results analysis and some conclusions outlined at the end is pointing out to the fact that an increase of the garter spring spacer preliminary clamping force value lead to an increase of its stiffness, increasing the wear of touching component parts and, at certain moment, to loss of the torsion coils integrity. The paper is dedicated to the specialists working in the research and technological engineering.

I.3.27. Characterization of Corrosion Deposits on Components of CANDU Steam Generators.

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The identification of corrosion compounds that can form in corrosive process, function of chemistry of circuit environment, represents an important step in establishing and modelling of corrosion mechanisms.
X-rays diffraction method is often used to identify the compounds formed at the surface of the corroded metal as well as the crystallization forms of phases produced. As technique of XRD method, the grazing incidence brings supplementary information in case of very thin layer or about the sandwich oxide layers structure by the emphasizing of oxide forms distribution in the layer.
In our work, it was investigated oxidised samples obtained from Incoloy 800, steel 516 and Inconel 600, materials used to manufacture different components of CANDU steam generators. The oxidised process consisted in the exposure of samples for 157 days in demineralised water with LiOH (pH=10.5), at 310^{0}C and 5MPa (the conditions corresponding to the chemistry of the CANDU primary side).
After exposure in corrosive environment, samples were tested by GIXRD (Grazing Incidence X Ray Diffraction) and SEM (Scanning Electron Microscopy). In the case of 516 steel it was observed that the magnetite is formed at interface metal-oxide, while the external layer is formed preponderant from hematite particles. The external layer present on the surface of Inconel 600 is formed by nickel hydroxide, while the spinelic compounds such as $(Cr, Fe)_2O_4$ are distributed in the inner layer. Because the oxide formed on the surface of Incoloy 800 sample is very thin, there is not major difference regarding the composition in the depth profile. The X-rays results were sustained by the SEM results.

I.3.28. Preliminary parametric analysis of the primary pump of the lead-cooled nuclear reactor ALFRED

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One of the key issues in the development of the Lead Fast Reactor (LFR) technology is the design of Primary Pumps evolving liquid lead and having both the appropriate performance for the required circulation of lead in the primary circuit and the high reliability required for the use in a nuclear power plant. In the present paper the Primary Pump for the LFR demonstrator (ALFRED) is analysed. The Primary Pump of ALFRED has to meet a number of constraints: evolving the required mass flow rate of the coolant, creating the required head in the circuit, maintaining the speed of the coolant in the pump as low as possible to prevent erosion of the components. This work proposes and investigates the performance of several types of pumps (volumetric and radial) in different geometric configurations. A number of locations of the primary pumps inside the pool are analysed from the point of view of the resulting flow field. The methodology for optimising the

configuration is also presented. The volumetric pumps (Archimedean's screw pumps) with straight shaft or with bulbed shaft, with or without variable blade step are analysed in the present investigation. The results shown in this paper summarise a significant effort in term of geometrical modelling as well as CFD numerical simulation. As an example, a few hundreds CFD simulations for tens of variants of screw-pumps were required in order to select the best performing variant. This analysis shows that the best performance for the given initial and boundary conditions are obtained with variable step for which the angular difference between the inlet and outlet angle of attack of the blades is around 33 degree.

I.3.29. Magnetic Field and Liquid Metal Interaction, MHD Modeling

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Today the nuclear field reliability standards demands are very high. To pump pressurized liquid metal trough pipes is quite challenging. Is quite known that the pump reliability rely on a crucial component which is the sealing. One way to get rider of the old problem of sealing is to use electromagnetic pumps (EMP). The most efficient EMP are those with permanent magnets. The design of an EMP is simple but to estimate the flow mass and pressure drop is not so simple, but using computer codes for fluid dynamics combined with electromagnetic codes, named magneto hydro dynamics codes (MHD), can offer a good choice. The present paper is focusing on the results for EMP MHD computation. The main goal is to compute the impact between uniform magnetic field and melted liquid. This is possible by using MHD module from FLUENT computer code. The obtain results are presented as tables for different inlet velocity, the values were set between 0.1 m/s up to 5 m/s, and different magnetic field intensity, starting at 0.1 T up to 1.5 T. For better view graphs were used to view the internal local velocity, local pressure drop and current density. The interaction between magnetic field and liquid metal was as expected, the current density was in the expected place and having expected shape. Also the central pressure drop shape was very good. The future goal will be to use an external magnetic field.

I.3.30. Preliminary Experimental Tests in the Liquid Lead Environment

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Research projects on the use of lead-bismuth alloy (LBE – Lead-Bismuth Eutectic) as coolant in nuclear reactors were initiated in the early 1950s and operational since 1963 in Russia for military submarine propulsion.
Coolant liquids as lead or Pb-Bi eutectic have good thermo-physical and technological properties regarding low working pressure in cooling circuit that facilitate better operating conditions in the primary circuit. High lead boiling temperatureremoves the possibility of an explosion in system sections where power density is increased.

Generation IV reactors are in progress with the design phase and therefore an important aspect is the selection of structural materials to operate properly in the environment of liquid lead. The following are some conditions that must be fulfilled these materials:

- Dimensional stability under thermal creep, stress relaxation and irradiation;
- Resistance to fatigue cracking, embrittelement due to hydrogen in helium;
- Compatibility of the physical and chemical properties between coolant agent and other material from the system.

RATEN ICN is involved in several European projects aimed to Generation IV research activities. In a first stage a laboratory material tensile testing installation has been set up that have the advantage to perform the experimental tests in the liquid lead environment.
In the FP 7 MatISSE RATEN ICN undertakes tensile tests on samples in liquid lead, to evaluate the influence of various coatings on mechanical properties in this environment.
The paper describes the preliminary tests to establish the work methodology, the functionality of the installation and the corresponding results are evaluated in the scope of future experimental activities.

I.3.31. Evolution of the Irradiation Parameters of Compact-Tension (CT) Pressure Tube Samples in C5 Capsule

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C5 Capsule is an irradiation device of TRIGA SSR, which is designed for irradiation of structural materials in an inert environment for mechanical behaviour characterization and the material microstructure evolution during irradiation.
C5 Capsule was designed and manufactured entirely at INR in 1984. Initially it was designed to test stainless steel samples.
The test section of C5 capsule includes the capsule pressure tube, sample holder and test section support. The pressure tube provides a tight inert Helium atmosphere around irradiated samples. The sample holder provides the samples support in maximum flux reactor core area. There are six sample supports which are located in the lower part of the sample hoder, 350 mm in length. They are designed of Aluminium to enhance the heat transfer. Each sample support forms a level of irradiation samples, and each piece of the sample support is provided with radial channels to fit the samples
C5 capsule is provided with instrumentation to measure the following parameters :

- temperature ;
- flux;
- pressure.

Temperature monitoring is provided by 12 K-type thermocuples (two thermocouples on each of samples levels for redundancy), and 1 K-type thermocouple for radial temperature gradient determination. Thermal flux is measured by two SPND-s with Silver emitter with compensation
C5 capsule has been used in 2010÷2015 to perform an irradiation test of compact-tension (CT) pressure tube samples which is useful for the delayed hydring cracks (DHC) phenomena analysis.

The paper presents evolution of the irradiation parameters in 1370 hours from Dec. 04, 2014 to March. 05, 2015. It is expected that the results presented in this paper will be useful when designing the instrumented capsules for an irradiation test of ODS.

II. ENVIRONMENTAL PROTECTION

II.1. Radioprotection & Air, Water and Soil Protection

II.1.1. Investigation of Optical Properties of $Zn_xCd_{1-x}O$ Semiconductor Oxides Fired at Different Temperature Ranges

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The main objective of the present work is to investigate the optical properties of $Zn_xCd_{1-x}O$ thin film semiconductor oxides samples for possible usage in radiation dosimetry. In-depth understanding of the changes in the optical properties of these oxides thin, when they are exposed to ionizing radiations, is necessary in order to design and develop thin film based gamma radiation dosimeters. The samples of semiconductors $Zn_xCd_{1-x}O$ have been prepared using solid solution reaction, where x is the concentration of Zn. The powder of ZnO and CdO was mixed by using different x compositions and then pressed at 70 KN. The specimen discs were fired at the temperatures of 1273 K and 1373 K, respectively, for two periods (2 hours and 5 hours). The colour of fired discs visually varied from brownish for x = 0.00 to whitish-yellowish for x = 1.00, with poor optical transmittance. The optical properties for samples with x values in the range from 0.95 to 1.00, fired at high temperature for 5 hours exhibited weak transparency and had an average transmittance less than 70% in the visible region. Meanwhile, the samples with concentration x < 0.95 had their transmittance close to zero both in the visible and UV regions (i.e. opaque). The band gap energies of specimens fired at 1273 K and 1373 K for 2 hours, and 1373 K for 5 hours, estimated from the absorption spectra, were 2.22 eV, 2.55 eV, and 2.58 eV, respectively. These thins films of $Zn_xCd_{1-x}O$ oxides with versatile band gap energies make these semiconductor materials to be one of the possible promising materials for radiation dosimetry.

II.1.2. Considerations on the Dose – Response Relationships and Their Implications on the Radiological Protection Regulations

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The entire philosophy of the radiological protection regulations is based today on the linear no threshold (LNT) theory, which considers that any dose, no matter, how small, is detrimental to health and therefore exposure to radiation should be kept, both for the general public and the professionally exposed, at the lowest reasonable possible level. This dose – response theory is mainly based on the extrapolation of the data obtained

from the survivors of the atomic bomb. Lately, however, other studies, like the health impact of CT examinations and the studies performed by Anders Pappe Moeller on the effects radiation on wildlife in the Cernobyl contaminated areas seem to validate this theory with data using small doses, and not just by extrapolation from high doses.
On the other hand, an increasing number of studies, like those regarding the high background areas and the studies on the contaminated buildings in Taiwan plus many in-vitro studies seem to point out that the LNT theory is incorrect and that there is a threshold dose below which not only there is no detrimental effect, but the radiation has even a stimulating effect on the immune system. These studies seem to indicate that the current dose limits imposed by the radiological protection regulations are far too low and they impact negatively on the efficiency and costs of the nuclear industry.
The most important part, however, seems to be left out: how is it possible that different studies lead to such radically different conclusion? Why, for instance, papers like those of Ian Fairlie in J. Env. Rad. 133(2014) and the one of J. D. Mathews et al. in BMJ 346 (2013) seem to point out that even low doses can have detrimental effects while papers like the one of S.M.J. Mortazavi et al. in Dose Response 12 (2014) lead to such different results?
In our opinion, the one element that has not been taken into account is the dose rate. The present paper will present a proposal for a new approach on the dose – response relationship, one which is based on the dose rate rather than on the dose. This approach basically unifies the LNT theory and the adaptive theories, since each one applies to a different range of dose rates. Thus, based on the Ramsar studies it seems like dose rate up to a few tenths of microGy/h are quite safe, whilst accepting, in the same time, that a CT examination can pose a health risk because the doses in the range of 10 mGy are delivered within seconds or, sometimes, even miliseconds

II.1.3. Rapid Methods for Radiological Monitoring in Emergency Situations

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Institute for Nuclear Research Pitesti (ICN in Romanian) operates nuclear and radiological facilities that require planning and preparation activities for response to nuclear accident and radiological emergency situations. These activities are being performed following Romanian regulations and IAEA Safety Standards being validated by participation in national and international exercises. They were carried out over time by the Radiation Protection Laboratory which is also coordinating a research and development programme addressing the emergency planning and preparedness technical and scientific support. Within this framework, ICN developed a series of practical tools such as computer codes for the assessment of the radiological consequences in case of nuclear accidents, testing, validation and implementation of procedures for rapid determination of environmental radioactive contamination, in situ gamma spectrometry, dose rate mapping, search and location of lost sources and hot spots. The ICN's Radiation Protection Laboratory is an active member of the IAEA's networks RANET and ALMERA which are organizing regularly training workshops, inter-comparison exercises and proficiency tests on rapid methods for determination of radioactivity in the environment.

The paper presents the techniques and methods for field monitoring used for radiological characterization of contaminated areas in case of a nuclear accident. Described methods have been applied by the authors for radiological characterisation of contaminated sites due to previous activities involving NORM or artificial radioactivity and were validated in practical workshops and field exercises organized by RANET and ALMERA. Some of the results obtained by the ICN's Radiation Protection Laboratory during the above mentioned activities organized at international level are presented in the paper.

II.1.4. Methodology for Radiological Characterization of Large Items Contaminated with Gamma Emitting Radionuclides

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Radiological characterization of components of a nuclear installation is an important activity throughout its lifecycle giving the possibility for optimization of the exposure of personnel both during operation and during interventions or decommissioning activities.
In situ gamma spectrometry is the only non-destructive method which can provide information related to the contamination status of a nuclear installation.
Due to the high variability of geometrical configurations of contaminated components, the efficiency calibration for a spectrometric system, used for characterization activities, represents the main issue concerning application of in situ methods.
The efficiency transfer method is one of the options for establishing of the response function of a spectrometric system in case of extended geometries.
The experimental tests have shown that enough precision could be achieved when calculate the efficiency for extended detection geometry by applying efficiency transfer corrections to measured efficiencies determined for simple reference geometries. The level of accuracy achievable depends on the complexity of the model used for the extended geometry and, usually is high enough as to meet data quality objectives for radiological characterization activities.
An analysis has been performed concerning the sensitivity of the model related to the input parameters, thus being established the critical parameters of the method. One has determined that the density of the source and absorbent media in between source and detector is of a great importance as concerns the efficiency transfer factor, its underestimation being especially amplified in magnitude by the model.
Starting from a set of data quality objectives, common for most of the radiological characterization strategies, one has developed a methodology for determination, by in situ gamma spectrometry, of the radioactive content of large items with complex geometries. This methodology is suitable for both determination

II.1.5. Measurement of ^{90}Sr-^{90}Y in aqueous samples using exclusive Cherenkov counting and combining Cherenkov counting with LSC

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^{90}Sr is one of the most important anthropogenic radionuclides from the radiation protection point of view due to rather long residence time in the biosphere, bio-availability for certain dietary products (especially milk) and high dose contribution after ingestion due to high-

energy beta radiation and preferential fixation into the skeleton (the chemical analogy of strontium and calcium leads to the long-term storage of ^{90}Sr in bones). For this reason, is important that ^{90}Sr to be monitored regularly in many types of environmental samples related to the human food chain. Also, even that the radiostrontium is released in insignificant amounts during normal operations of nuclear reactors, its presence in effluents should be continuously monitored to verify the compliance with the limits required in the operation license.
^{90}Sr decays with a half-life of 28.8 yr by emitting beta radiation with a maximum energy of 546 keV into ^{90}Y, which decays with a half-life of 64.1 hr into stable ^{90}Zr, emitting beta radiation with a maximum energy of 2.28 MeV. The ^{90}Sr activity of a sample can be determined either by analyzing ^{90}Sr itself or its daughter nuclide, ^{90}Y. Since both nuclides are pure beta emitters, both types of analysis require radiochemical processing of the sample and prior to measurement they must be separated from the sample. Some separation methods exist for strontium extraction from aqueous samples, based on either precipitation/coprecipitation, or solvent extraction or even ion exchange chromatography. Gas flow proportional counting, liquid scintillation counting (LSC) and Cerenkov counting are the 3 principle detection methods used in low-level analysis of ^{90}Sr-^{90}Y.
In this work two procedures for ^{90}Sr-^{90}Y measuring in aqueous samples are presented, one using exclusive Cherenkov counting and the second combining Cherenkov counting with LSC. In both methods the strontium separation is based on the use of strontium specific extraction chromatographic resin, SrResin, from Eichrom Inc. The methods were tested on aqueous samples spiked with known quantities of a standard solution of ^{90}Sr-^{90}Y in equilibrium. ^{90}Sr and ^{90}Y activities determined experimentally were compared and correlated with the activities of the two radionuclides, calculated on the basis of their radioactive decay. Based on these correlations, the Cherenkov and LSC efficiencies of the two radionuclides were determined. The values of 0.606 and 0.017 were found for Cherenkov efficiencies of ^{90}Sr and ^{90}Y respectively, these being in very good agreement with similar values found in the literature.

II.1.6. Environmental radioactivity – ICIT Valcea

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Romanian national nuclear regulations require that for every new nuclear installation a preoperational program for the monitoring of the environmental radioactivity around the installation should start at least two years before the commissioning of the installation.
The goal of this paper is to present the results recorded in the last year on levels of the beta/gamma radioactivity around the Pilot Plant for Tritium and Deuterium Separation at ICIT Ramnicu Valcea.
Hundreds of samples of soil, sediment, water and vegetation, over a circle of 5 km radius around ICIT, have been collected and analyzed according to the frequencies established within the preoperational program, and all results are presented here. Also, we will describe some methods of sample preparation or measuring.
For gamma measurements we used a high resolution gamma-spectrometric system consisting of a multichannel analyser with HPGe detector, and for global beta activity we used a gas-flow proportional counting, model MPC9300.

II.2. Radioactive Waste Management

II.2.1. Carbon-14 Source Term in Geological Disposal:The EC Project CAST

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The EC CAST Project (CArbon-14 Source Term) aims to develop understanding of the potential release mechanisms of carbon-14 from radioactive waste materials under conditions relevant to waste packaging and their disposal to underground geological disposal facilities. The CAST consortium brings together 33 partners with a range of skills and competencies in the management of radioactive waste containing carbon-14, geological disposal research, safety case development and experimental work on gas generation. The project began in October 2013 and lasts for 54 months. The waste materials being examined are: irradiated steels, Zircaloys and graphite; and spent ion-exchange resins. Reviews of the current understanding of the degradation and release of carbon-14 from these materials have been undertaken. CAST includes a work package to integrate the results from the experimental studies in the context of national programmes and safety cases and a work package to disseminate CAST activities and results more widely. This paper will discuss progress on CAST and the key conclusions from the current status reviews. More information, including project publications are on the CAST website: http://www.projectcast.eu/. The project has received funding from the European Union's European Atomic Energy Community's (Euratom) Seventh Framework Programme FP/2007-2013 under grant agreement no. 604779, the CAST project.

II.2.2. Future prospects for the Management of Radioactive Waste in Greece

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In Greece, like in many other countries having only research reactors or other small nuclear installations, there isn't yet any decision for construction of a disposal facility. Since the predisposal management of radioactive waste should be aligned with the disposal solutions, the determination of the disposal options is essential for the selection of the technology needed for treatment and conditioning of the wastes. The scope of the present study is the investigation of the disposal options for Greece. Firstly, the study deals with the preliminary inventory as well as the classification of the existing radioactive waste and the prediction of the expected waste from decommissioning of the open pool type at 5 MW Greek Research Reactor (GRR-1). Furthermore, it concerns the investigation of those waste streams which are appropriate for long-term interim storage in order to decay and meet the clearance criterion within a reasonable time span. The existing radioactive waste includes the institutional waste from the operation of GRR-1 and associated facilities as well as orphan sources and other radioactive items collected in the frame of emergency by the Greek Atomic Energy Commission and kept at the interim

storage of the National Centre for Scientific Research "Demokritos". Based on the present inventory of RW, the establishment of a small scale and cost affordable LILW geological repository seems to be the appropriate and most acceptable by the public disposal solution.

II.2.3 Use of Experience from Previous National Nuclear Projects in Support of Planning a Sustainable Geological Disposal R&D Program

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In a previous systematic study which was based on expert judgment and reliable methods and tools the authors identified that, the current national context would raise several risks when the Romanian geological disposal program would be developed. The study of the national context risks made in different scenarios has identified the need for a phase for preparation of the national geological repository (NGR) program and allowed defining an optimum solution for integrating risk responses through several processes which contribute to planning and implementing a sustainable program and might support this integration.
Several significant national context risks should be treated before the NGR program would start, since the analyses have shown that they could affect the technical activities of the program and have impact on cost, schedule and/or acceptance of the program. Among them there are risks which individually or combined with other risks might affect R&D activities of the NGR program, i.e. absence of a national coordinated R&D capacity, difficult and costly access to data and information on the national territory and resources and existence of many types of potential host rocks for NGR.
Experience and lessons learned from previous national nuclear projects have offered and could further offer valuable information allowing judgment on the way the national context risks might act and they should be treated.
The paper will present in brief the study of the national context risks that was made by using such relevant experience and a solution recommended for preventing that actions of such risks to affect R&D activities when planning a sustainable NGR program.

II.2.4. THE Management of Financial resources intended for radioactive waste and decommissioning of the nuclear facilities in the European Union

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The European Commission has developed policies and made recommendations on how financial resources should be established and managed by European Union Member States for the purpose of radioactive waste management and decommissioning. The manner in which these recommendations have been accepted, and are applied, varies between European countries. Also the current practice in the European Union related to decommissioning of nuclear installations and management of radioactive waste. To some

extent, this variation reflects the maturity of the nuclear programs in each country and whether or not nuclear facilities are largely state or privately owned and operated.
This paper reviews the European Commission’s policy on financial resourcing for radioactive waste management and decommissioning and evaluates how financial resources are practically established and managed by European Union Member States.
The findings from the review are then used to benchmark the situation in Romania. In special the review of the Romanian regulations on management of financial resources for radioactive waste management and decommissioning against the European Commission's recommendation. As well highlight some recommendations for the national Agency regarding the management of the financial resources necessary for the safe management of the radioactive waste and the decommissioning of the nuclear and radiological facilities in Romania.

II.2.5. Estimate of Cesium Transport From Aqueous Radioactive Waste Using Emulsions as Carrier

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In view of establishing the technology for treatment and conditioning of β-γ liquid radioactive waste generated from Romanian TRIGA research reactor operation different treatment methods including organic emulsions were tested. The aim of the present work was to evaluate the global efficiency of caesium extraction and consists in the estimation of cesium transport phenomena from a phase to another one through a liquid organic membrane used as carrier. In these experiments a mixture of dichloromethane, carbon tetrachloride and p-tertbutilcalixarena as carrier and distilled water as receiving phase were used.
A β - γ radioactive waste from TRIGA reactor was used in the tests performed in an installation comprising of: a vessel (in which the carrier is placed and the receiving phase is poured), a magnetic stirrer and a tube for the source phase introducing.
Several experiments were conducted in which the contact time, the ratio between the carrier and receiving phase, and the type of calixarene were varied.
It was determined that the transport flux for Cs-134 and Cs-137, at a carrier/ receiving phase ratio of 2.35, for 6 hours, is about 10^{-3} moles/s$\times$m^2 and 10^{-11} moles/s$\times$m^2, respectively.
For the experiments where the carrier /receiving phase ratio was 0.9, the extracted Cs-137 increased with the contact time between the two phases. In this case, the transport flux for caesium ranged from 10^{-15} to 10^{-16} moles/s$\times$m^2. It was also concluded that p-tert-butilcalix[8]arene is a more efficient carrier than p-tert-butilcalix[6]arene.

II.2.6. EAGLE – Public Perception on Ionizing Radiation Communication in EuropeVisions about Future – Forecasting the Evolution of Long Term Radioactive Waste Disposal Systems Based on Archaeological Inventories

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The topic of the paper is focused on safety demonstration of the long term disposal systems intended for radioactive waste generated during the operation, refurbishment and decommissioning of nuclear installations.
It is well known that, worldwide, researchers had concentrated, in the past decades, on critical evaluation of existing data for several archaeological artefacts in order to assist the demonstration process for long term disposal of radioactive waste and to give a higher confidence degree for the extrapolations done for certain experimental data from research activities dedicated to RW disposal.
The authors will perform a short review of the state-of-the-art of the actions related to artefacts studies to assess the capability to derive valuable conclusion to be used in the management of radioactive waste. The study will be focused on the ability of manmade items to maintain, for extremely long periods, theirs geometrical appearance, function and material characteristics in burial condition.
A discussion will be made on the existent knowledge for the estimation of corrosion rates, corrosion mechanisms, corrosion layers formation / damage kinetics, occurrence of any potential detrimental and insidious localized corrosion phenomena (pitting / crevice corrosion, stress corrosion cracking, microbiological corrosion, selective de-alloying etc.) for several preserved artefacts.
The current context in Romania will be shortly analysed and the authors will discuss the potential benefits for the geological (but not only) disposal program, benefits to be achieved from conclusion related with material demonstrated behaviour in archaeological sites. The analysis will be developed for iron, copper and bronze artefacts.
Some recommendations on possible integration of archaeological data with repository design data will be issued to ease final decisions and extrapolations for long term disposal of radioactive waste (RW).

II.2.7. Sequential Separation of Cs, Ca and Ba for ^{90}Sr Assessment

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The ^{90}Sr radioactive isotope is not a naturally occurring radionuclide and it exists because of human activities. The behavior of ^{90}Sr in various media such as atmosphere, soil, water, plants and human body, transforms it a potential long-term hazard to the environment and people if it is not properly disposed.
The ^{90}Sr radioisotope is a pure β emitter (Emax = 546 keV) with a half-life of about 29 years. Measurement of ^{90}Sr radioactivity is based on detection of its beta particles emission by liquid scintillation cocktails technique (LSC) which requires a chemical separation and purification from other interfering radionuclides before determining its activity concentrations.

The radio-chemical separation and purification technique of ^{90}Sr is based on development and testing of the analytical methods applicable to natural isotopes in the form of Sr^{2+} species.
A two-steps chemical treatment technique for strontium separation from aqueous samples is described. The method was applied to simulated samples containing stable elements of Sr, Ni, Cs, Ca, Ba, Mn, Fe, Co and Eu. The transition elements (Ni, Mn, Fe, Co, Eu) were precipitated as hydroxides, followed by the use of sodium carbonate solution to precipitate the strontium, calcium and barium carbonates and to separate them from cesium. The Sr Resin from Triskem International was used for Sr separation from Ca and Ba on the base an extraction system that presents selectivity for strontium. The strontium affinity for the resin increases with the nitric acid concentration, reaching a maximum bonding strength for a concentration of 8 M nitric acid.
The fractions collected in the different stages of the method were analyzed by the inductively coupled plasma-optical emission spectrometric (ICP-OES) technique in order to assess the mass concentration of the elements.
The combination of successive precipitations with extraction chromatography for removing of the other interferences from Sr matrix leads to a recovery of about 91 % for Cs, Ca and Ba, and of about 93 % for Sr. The decontamination factor values of the presented method were 10^2 to 10^5 for all interfering elements.

II.2.8. ^{14}C Determination in Irradiated Graphite from the Thermal Column of the VVR-S Reactor

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The release and migration of carbon-14 (^{14}C) has been identified as a key issue for geological disposal radioactive wastes due to ^{14}C long half-life. A significant fraction of ^{14}C in the Romania inventory is derived from irradiated graphite.
Determination of radiological characteristics of irradiated graphite structures after the operational period of nuclear reactors which use graphite as a moderator or reflector is a high important issue from the point of view of their decommissioning
The composition of radioisotopes in reactor graphite depends on impurities and irradiation variables, namely the spatial flux of neutrons, activation and cooling time.
This paper presents the methodology and the experimental results of the variation of ^{14}C concentration with the distance in irradiated graphite of the thermal column of the VVR-S Research Reactor (under decommissioning). Small samples have been sampled from the graphite rods of the first graphite disc located near reactor vessel. Some pieces were cut from rod's edge near the reactor vessel and the others pieces from the opposite part of the rod.
An estimation of the radionuclide inventory of the graphite samples collected from thermal column of the VVR-S reactor block and their specific activity was performed using a gamma spectrometer with HPGe detector type GMX ORTEC.
The irradiated graphite samples were ground to a fine powder and the sample material was weighed on a microgram balance.
Based on our results the overwhelming amount of ^{14}C is located in the first disc of the thermal column of the reactor and have demonstrated that the ^{14}C activity decreases along the thermal column (due to the thermal neutron flux attenuation) from $3.4x10^4$Bq/g, next to the reactor core

to $2.6x10^2$Bq/g for the first graphite disc. The results obtained will be used to establish the ^{14}C inventory in the decommissioning phase of the VVR-S nuclear reactor.

II.2.9. The CITON Contribution on the Improvement of Technologies to Radioactive Waste Facilities, a Responsible and Ethic Activity

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According to IAEA classification, Romania is considered to be a country with an average nuclear activity. In Romania there was an especial interest in the management of radioactive wastes generated by the use of nuclear technology in industry and research in order to ensure the human and environment protection. Using the most advanced technologies, Romania successfully accomplished to solve all management issues related to radioactive wastes being addressed all safety concerns. Every major nuclear installation was accompanied by the suitable waste management facilities. As examples, the Russian Research Reactor from Magurele was provided with an English Treatment Plant at top level in the 70's and the American Triga Research Reactor from Pitesti was accompanied by the French Treatment Plant being at the highest level in the 80'.

This paper presents the main characteristics of radioactive waste management systems from Romania, a country with one of the highest carefulness on responsible and ethic concept for the safety radioactive waste isolation at the top level recommended by IAEA. All existing and future Waste Management Facilities are the main subject of the continuous improvements of technologies and long-term safety.

The main objective of this work is to present information about the effort of CITON specialists, involved in all important parts of radioactive waste management in the country, in order to reduce the radioactive impact on humans and environment by improvement of some critical aspects of radioactive waste management strategy.

The existing National Repository for Radioactive Wastes at Baita-Bihor (DNDR) is designed only for the disposal of institutional waste. Following a recent safety analysis performed in the frame of the PHARE project, the safety of this repository will be improved according to the last recommendations. This activity is in development by CITON in the frame of a National funded project related to disposal galleries filling improvement and repository closure according to best practices.

The National Nuclear and Radioactive Waste Agency's (ANDR's) strategy on low level waste (LLW) and intermediate level waste (ILW) disposal from Fuel Cycle waste has as objective the putting in operation of the Final Repository for LILW (DFDSMA) Saligny, in 2020. This facility is in the responsibility of the ANDR. However, wastes arising from the Cernavoda NPP must be treated, in order to achieve the Waste Acceptance Criteria (WAC) of DFDSMA Saligny. The related Radioactive Waste Treatment Facility - RWTF (STDR), which does not exist yet, is the responsibility of Cernavoda NPP. The main requirement for the RWTF is the necessity to achieve the treatment and conditioning of radioactive wastes that arise both from NPP operation and from future decommissioning activities, as to ensure the compliance with the WAC of the DFDSMA Saligny.

The processing and final disposal of wastes from decommissioned nuclear facilities will use an important amount of the allocated budget for a nuclear facility and in this case an important effort is required to diminish this cost; this activity constitutes a key objective for the management of wall radioactive waste

The Public is informed periodic by CITON on the progresses encountered in radioactive waste management in Romania in order to obtain the acceptance of Nuclear Power development. The present Conference contributes to the general strategy for Public information.

As a general conclusion, it may consider that all proposed or already performed improvements by CITON in the Radioactive Waste Management Domain, is an example for human and environmental protection.

All CITON's activities are conducted in a responsible and ethic manner which will be demonstrated in the paper.

II.2.10 Laboratory Experiments to Measure Cs-137 Transport Parameters in Porous Media

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Laboratory transport studies generate important information on the fate and behavior of contaminants and their distribution in natural and engineered barriers of a radioactive waste disposal facility. For the contaminant transport simulation in porous media, most computer software use the distribution coefficient, Kd, to describe the partitioning of the contaminant between the solid and aqueous phases and diffusion coefficient to describe the diffusional transport. This paper presents an experimental study of the Cs-137 sorption and diffusion in two porous media typically used in the engineered barrier system: sand and concrete. Flow through method (in Plexiglas columns) permits to evaluate sorption behavior and through-diffusion experiments (using Plexiglas diffusion cells) are performing to measure the Cs diffusion coefficient. Since both methods turned to be time consuming and experiments are still ongoing, preliminary sorption data were achieved by batch method both for sand and for concrete. To evaluate cesium sorption characteristics on the investigated matrix, the contaminant concentrations in solution, at equilibrium, were plotted versus the amount of contaminant sorbed on the solid phase, and sorption isotherms were obtained for the concentration range of interest. To obtain these isotherms, and to evaluate the distribution coefficients (Kd) on a large concentration range, cesium concentration was varied while other parameters were held as constant as possible. For sand samples, the experimental data were best fitted by linear sorption isotherm with a Kd of 29.2 L/kg, in agreement with literature data for cesium sorption on sand. The experimental data obtained for concrete sample seems to be best fitted by an anti-Langmuir isotherm, although it is well recognized that cesium sorption on concrete is best described by a linear isotherm. The observed sorption behavior could be a result of an unexpected foaming process occurring in the centrifuge tubes during the sorption test, process initiated most probably by a water-limestone reaction.

II.2.11. Comparative Study of Solid Waste Combustion and Microwave Digestion Methods for C-14 Measurement by LSC

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In this paper two methods for simulated solid radwaste dissolution were tested to recommend the most appropriate one to be implemented in ^{14}C measurement at Cernavoda NPP.
During the combustion process, all carbon isotopes, including ^{14}C are oxidized to gaseous carbon dioxide that is subsequently trapped in form of carbonate in a column filled with a carbon dioxide absorbent (Carbo-Sorb E). The carbonate is flushed into the counting vial together with the liquid scintillation cocktail (PermaFluorE+) and the resulted solution is ready to be measured by liquid scintillation counting (LSC).
In the microwave digestion method the solid sample is transformed totally or partially in liquid phase depending on the sample matrix using adequate digestion reagents. The digestion solutions are directly heated through the absorption of the microwave radiation by the polar digestion reagent. After the digestion procedure is completed the resulted liquid is properly prepared for ^{14}C measurement by LSC
Tests performed on simulated solid radwaste (paper, textile) showed a ^{14}C recovery of 90% by combustion and higher then 75% by microwave digestion method.

II.2.12. A Destructive Sample Preparation Method for Radioactive Waste Characterization

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The accurate knowledge of the radionuclide inventory present in the radioactive waste is an essential step in selection of the adequate methods for their further management. The radioactive waste characterization is carried out through “non-destructive methods” when the concerning radioisotopes are gamma emitters and through “destructive methods” when they are pure beta or alpha emitters.
The destructive characterization methods often suppose the conversion of the radioactive waste matrices in liquid aqueous fractions, before the radiochemical analysis to be carried out. The chemical processes through which the total dissolving occurred rely on the oxidation-reduction reactions, such as alkaline fusion, acid leaching and acid digestion. Nowadays the acid digestion in microwave field represents an efficient method for solid sample preparation, often used in analytical chemistry.
Acid digestion, using the microwave power, was applied for “dissolution” of different materials corresponding to the radioactive waste matrices resulted from a nuclear power plant operation, including exchange resin (cationic and mixed), concrete, paper, textile and oil. A small aliquot of solid sample (0.1-0.5g) was mixed with a known volume of digestion reagents (HNO_3 67%– H_2O_2 30% or HNO_3 67% - HCl 37%, with HF addition if the SiO_2 was present in matrices) in a 100 ml PTFE vessel and it was mineralized using a Berghof digestion system, Speedwave 4. Starting from the manufacturer procedures, the technical parameters (temperature and mineralization time), the types and quantities of digestion reagents were optimized. After the mineralization process, the samples were transferred in a centrifuge tubes, separated at 3500 rot/min and visual analyzed. The obtained solutions were clear, without suspended or deposed materials and separated phases, ready for future separation processes of the “difficult to measure” radioisotopes.

II.2.13 Efficiency evaluation of nuclear grade resins selected for improvement of the aqueous radioactive waste treatment technology

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INR Pitesti applies the treatment of aqueous radioactive waste by ion exchange, for waste generated in decontamination processes. The technology used was licensed in year 2000, being designed for treatment of radioactive liquids from decontamination centre of Cernavoda NPP. Nowadays, this technology, based on the pair of ion exchangers (cationite and anionite): C100H Purolite/A600 Purolite, is implemented in a mobile, compact treatment plant, for processing small volumes of waste with variable composition, from Cernavoda NPP. To develop a new technology, with improved performances, a couple of nuclear grade resins from Purolite Co. were tested. In the present paper, the treatment efficiency obtained with the pair of resins NRW1600/NRW5050 is presented as compared to the same parameter obtained for the pair C100H/A600. It was observed that the breakthrough point for the column with NRW1600 cationite is around 600 BV, while for the column with C100H cationite is around 500 BV, for the same waste. This leads to the conclusion that by using nuclear grade resins the volume reduction factor can be increased with over 20%. The analysis of the economic efficiency shown that by changing the ion exchangers with nuclear grade types, despite of the increased price of the last one, an overall reduction of costs might be achieved due to the reduction of the secondary waste production which will lead to a decrease of the disposal costs.

II.2.14. Performances of Solidification Agents used for Treatment of Tritiated Organic Liquid Waste

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The paper presents the author's research on solidification of organic radioactive waste using the pair of absorbent polymers (WasteLock® 770 and OrganoLock™ N-65). The research studies aimed to determine the performances of the selected solidification agents applying the following bench-scale tests:

- Absorption of the organic and aqueous waste in order to meet Paint Filter Test Requirements (EPA 9095);
- Evaluation of the incinerability of the solidification product;
- Determination of the heat release during solidification process;
- Determination of the tritium exposure hazard due to solidification product.

The absorbent polymers are used in treatment of a large variety of organic liquid waste (waste water, oil, solvent, liquid scintillation cocktail). Reasonable results were obtained with the investigated absorbent polymers.

The results of Paint Filter Test (EPA 9095) show that the selected solidification agents are suitable for the proposed goal. The polymers have an absorbent capacity over 2 (w/w) with a minimal volume increasing.
The temperature recorded during the solidification process of the organic liquid waste with the selected absorbent polymers increases with 1÷2,5^{o}C in the first 15÷30 minutes, depending on the waste type.
For the all types of organic waste tested the solidified product can be incinerated resulting in a small ash percentage (less than 0,2%).
The tritium exposure hazard associated with the solidification products of the pair of absorbent polymers complies with the requirements of the national regulations concerning radioactive waste management activities.

II.3. Air, Water and Soil Protection

II.3.1. Solid Recovered Fuel as a possible fuel

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Solid Recovered Fuel (SRF) is a solid fuel prepared from non hazardous waste to be utilised for energy recovery in incineration and coincineration plants. Energy intensive industries are looking for alternative fuels in order to save primary fuels, and by doing so, enforce the development of sustainable development. The waste management sector industry has developed, since many years in western countries, ways to produce secondary fuels e.g. SRF's with reliable qualities, which are used successfully regarding economic and environmental aspects. SRF was successfully introduced by Lagan as a fuel in 2009, but in Romania SRF's are unknown there potential. It is a renewable, high quality, clean burning fuel that is produced from the non-recyclable part of normal household and office waste. This waste is processed to remove recyclable or compostable materials.
A sustainable, renewable fuel. More than 60% of the energy from SRF is derived from carbon neutral biomass and so provides a potent source of renewable energy.
SRF has many important environmental/economical benefits: replaces imported fossil fuels with locally produced renewable 'greener' fuels derived from the waste sector; support domestic UE waste management requirements and possible Romanian jobs; reduces requirement for landfilling, in line with EU environmental requirements; generates no waste 'incinerator-type' ash, that would require further disposal; avoids methane production from landfilling, methane which have almost the same importance as CO_2 in climate change.
Typical composition of SRF's: paper; plastic; wood; textile; miscellaneous.
In our paper, we had proposed, with our skills and equipment's from ICSI Solid, Liquid and Gaseous Fuels Characterisation Laboratory, to show fuel potential of SRF's. Five homogenised samples of SRF prepared special in our laboratory with a composition like: paper 40-58%; plastic 10-20%; wood 1-5%; textile 7-15%; miscellaneous 0-2% was analysed as a fuel.
Results of SRF characterisation:
1. It was determined the initial moisture W_i^i content in a special oven at T=70^{0}C for 8h until constant mass. Then the samples was put again in the oven at T=105^{0}C for 1 and a half hour until constant mass to determine the hygroscopic humidity W_h^a.
2. After determination of initial moisture, the samples was cut in small pieces, collected in a container, mixed prepared at 0,2mm, then put in a very small tin capsule to could determine the C-H-N-S/O with the help of elemental analyser Flash2000 by combustion and pyrolysis methods. We use in our cause BBOT calibration standard; an thermal conductivity detector TCD, special columns fitted with molecular sieve 5A and porapaq Q+R at T=65^{0}C, to could separate gases which results from combustion and pyrolysis processes, like CO_2 for Carbon determination, SO_2 for Sulphur determination, NO_2 for nitrogen determination, H_2O for Hydrogen determination and CO for Oxygen determination; 2 furnaces with 2 special reactors fitted with cooper chemical, oxide cooper used at T=950^{0}C for combustion and T=1060^{0}C for pyrolysis method.
3.The caloric potential of this samples it was established with a calorimeter. The calorimeter IKA5000 uses combustion method to determine NCV and HCV with the help of a oxygen bomb

which is feed with pure oxygen to burn the sample to generate calories/heat. Necessary mass of the sample for this experiment was approx. 1 grame.

4. Ash content was determined at T=815^{0}C for 2h in a calcination oven.

II.3.2. Underground water dating and age correction using radiocarbon

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Isotopic techniques provide tools for research regarding ways of carbon exchange between atmosphere, biological systems and geological systems. Main objective of this paper is to correlate environmental isotopes (^{14}C, ^{3}H, ^{18}O and ^{2}H) in order to estimate recharge conditions of studied aquifer located in the city of Băbeni, Vâlcea County. For this purpose were collected water samples from eight drillings, three domestic wells located further along the path of flow of aquifer and from two rivers that influence the aquifer, river Olt and river Bistrița.

Groundwater ages were estimated by radiocarbon dating and subsequently were corrected using alkalinity correction (ALK model), chemical mass balance correction (CMB-Chem model and CMB-Alk), δ^{13}C and exchange matrix model (Fontes-Garnie model). Obtained data conducted to a mean transit velocity of studied aquifer of 0.55 m/year, value which is specific for this type of aquifer. This value was determined taking into account the difference in average age of drillings with old underground water and the distance between them.

Although radiocarbon measurement type proposed in this paper is a classical one, optimizations to the preparation technique allow reduction of time and financial costs and can be successfully applied to a large number of samples, common in hydrological studies.

II.3.3. Determination Polychlorinated Biphenyls of Soil by Chromatography

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PCBs, synthetic oils are from persistent organic pollutants group banned under the Stockholm Convention since 2001. They have toxic effects as dioxins, so can be called dioxin. On the other hand, can be identified as form of emissions during incineration of municipal waste but also in the industry, these effects are carefully controlled during operations. Soil is the most prone contamination environment being an important reservoir for storing of PCBs in the terrestrial environment. In this context, a high priority is monitoring the level of contaminants deposited in sensitive areas such passions.

The purpose of this work consists of identification and quantification polychlorinated biphenyls, samples from different soils, sampled taken from regions of Romania. PCBs concentration from soil samples were analyzed using GC-ECD equipment according to ISO standard ISO 10382/2007. After analyzing samples using GC-ECD, was identified and

quantified PCBs types with different concentration levels. The next phase of this work consists of a comparative analysis of concentrations levels between PCBs results and other samples of PCBs.
After analyzing, the concentrations of PCBs quantified from samples taken soil have values between 0.030 µg / kg and 0.5 ng / g. On soil samples were analyzed 6 indicator PCBs and 12 dioxins PCBs.

III. SUSTAINABLE DEVELOPMENT

III.1. Polices and Strategies in Nuclear Research

III.1.1. Progress in the Implementation of the National Strategy for Nuclear Safety and Security

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In July 2014, the National Strategy for Nuclear Safety and Security was officially approved by the Romanian Government and by the Supreme Council of National Defence, has been published and has come into force. The work on a national strategy for nuclear safety and security started from a recommendation received from the 2011 IRRS mission. Although most of the elements required by such a strategy were considered to be already in place, it was recognized that a national strategy may bring better coordination and coherence in addressing all the aspects and measures that have an impact on nuclear safety and security. Therefore this work has been considered of added value and all the national authorities with roles and responsibilities related to the nuclear field, as well as the organizations owning and operating the major nuclear facilities, have participated in the development of the strategy, which was coordinated by CNCAN. The development of the national strategy started in the beginning of 2013. Based on the current regulatory framework and on the trends observed at international level with regard to the improvement of the synergy between safety and security, it was decided that a national strategy addressing both nuclear safety and security is justified, taking into account also the provisions in the IAEA Safety Standards. The strategy includes a policy statement with nuclear safety and security principles, including the ten fundamental safety principles outlined in the IAEA SF-1 Publication, and takes account of the relevant provisions of the IAEA GSR Part 1 Publication. The strategy will be reviewed and revised as necessary, at least every 5 years. A process will be established to monitor the implementation of the strategy and of its corresponding action plan, and the results would be presented annually to the Government.

III.1.2. Nuclear Energy in the context of the EU Energy Policy

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The most well known definition of sustainable development states that it represents the *development that meets the needs of the current generation without compromising the* ability of *future generations to meet their own needs*. In the field of energy we can associate this concept with the availability of energy, the impact upon environment due to air emissions and climate change and energetic resources use in relation with their availability for the future generations. The European Union is aiming through its energy policy in orientating the energy sector to a sustainable path. In practice it wants to meet its energy needs at an affordable price and in the same time to reduce GHG emissions. Through its effort the UE wants to become a model in the area of energy policy. Currently the EU represents 12.7% of the global energy consumption, behind China and the USA, with a declining trend. However due to its modest energetic resources and declining production in contrast with its consumption levels, the energetic dependency rate is on an ascending trend. This is especially challenging knowing that about 70% of the gas imports originate from only three suppliers and coal use tends to be reduced due to environmental considerations. In this context the European Commission is aiming in redeveloping the EU energetic sector by promoting the use of renewable resources and the improvement of energy efficiency. Due to its attributes, nuclear energy can contribute to the sustainable development of the European energy sector. The present paper aims to highlight the potential of nuclear energy from the perspective its costs, ability to reduce the GHG levels and to contribute to the energetic security of the Union. In area of competitiveness, the research aims to present the contrast between Germany, that will phase out its nuclear reactors until 2022 and sustains the deployment of renewables, and France, where the nuclear energy represents the main component of the energy mix. The analysis is performed with the help of the INPRO methodology by comparing the levelized cost of electricity and net present value of different energy sources from the two countries. Nuclear energy proves itself to be a reliable source of energy, with an attractive and stable cost especially for low discount rates and reduced period of plant construction. In contrast with gas it is not vulnerable to fuel price escalations. Form from the perspective of climate change it presents lower rates of GHG emissions as its fossil counterparts. In the area of energetic security uranium resources present a high energetic content and are available in sufficient quantities to meet demand in the long term. We conclude that nuclear energy proves itself to be a reliable, competitive and sustainable source of energy, capable in contributing to the EU long term energy vision.

III.1.3. Evaluation of Nuclear Power Development Scenarios in Romania Envisaging the Long-term National Energy Sustainability

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The paper summarizes the results obtained by the RATEN ICN Pitesti experts in the IAEA's Collaborative Project INPRO-SYNERGIES. In the framework of SYNERGIES project, the Romanian team performed a case study included in the project final report under Task 1 ″Evaluation of Synergistic Collaborative Scenarios of Fuel Cycle Infrastructure Development″. Romanian study proposes to evaluate and analyze the development of the nuclear capacity and increasing of its share in the national energy sector, envisaging the long term national and regional energy sustainability by keeping options open for the future while bringing solutions to short/medium-term challenges. The study focused on the modelling of national NES (Nuclear Energy System) development on short and medium-term (2050 taken as time horizon), considering the existing NFC (Nuclear Fuel Cycle) infrastructure, provisions of strategic documents in force and including also the possibility of regional collaboration related to U/fresh fuel supply and SF (Spent Fuel) storage, as services provided at international market prices. The energy system modelling was realized by using the IAEA's MESSAGE program. For electricity demand evolution, a scenario developed in the framework of ″Nuclear Energy System Assessment in Romania using INPRO methodology″ national project has been considered. The analyses were performed for three distinct NES development scenarios, as follows: a) <u>reference case</u>: 4 PHWR, CANDU type (existing CANDU U1&U2 reactors, 700 MWe each, in operation, and new CANDU U3&U4 reactors, 720 MWe each, with projected in-service after 2020); b) <u>low case</u>: 2 PHWR, CANDU type (existing CANDU U1&U2, in operation); c) <u>high case</u>: 4 PHWR, CANDU type (as in reference case) and other NPP solutions, advanced PWR (1000 MWe each) or advanced HWR (720 MWe each), with projected in-service after 2035. Romania adopted an open fuel cycle, OTFC (Once-Through Fuel Cycle), characteristic for CANDU reactors. For considered time horizon, no changes in NFC were assumed, including the enrichment and/or reprocessing activities not supported by national legislation in force. The study results may offer a clear image and also the possible answer to several key questions regarding: the potential of nuclear energy to participate with an important share in national energy mix, in conditions of cost competitiveness, safety and security of supply; impact on national energy mix portfolio of capacities and electricity production; impact on Uranium domestic resources; economic projection/investments needed for new nuclear capacities addition; fresh fuel requirements for nuclear capacities; SF annually discharged from the reactors and transferred to interim wet storage for cooling; SF volume in interim dry storage, etc.

III.2. International Partnership for a Sustainable Development

III.2.1. INR Activities Regarding ASAMPSA_E Project

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The FP7 ASAMPSA_E (Advanced Safety Assessment Methodologies: extended PSA) project aims at examining in detail how far the Probabilistic Safety Assessment (PSA) methodology is able to identify any major risk induced by the interaction between a nuclear power plant (NPP) and its environment, and to derive some technical recommendations for PSA developers and users. The project is structured on 3 phases: phase 1 focuses on Topics identifications; phase 2 on Guidance development and phase 3 is necessary for Final review, conclusions, and validation of the results. The paper will present the contributions that Institute for Nuclear Research (INR) PSA team members have brought to the project, with specific reference to the deliverables to which they have contributed. More specifically, it will discuss the work performed in the frame of the first two work packages. The first work package deals with relations with PSA end-users, and a questionnaire has been developed and sent to nuclear stakeholders (including utilities, vendors, safety authorities, research and technical support organizations). Questionnaire topics will be presented, together with some reflections on the preliminary results. The second work package is referring to identification of initiating events (internal and external hazards), single and correlated, that needs to be taken into consideration in an extended PSA, and to the implementation modalities (for both external hazards and their combinations) in level 1 (L1) PSA. The conclusions drawn after analyzing the existing guidance on the implementation of external hazards in extended L1 PSA, specifically for human reliability assessment (HRA) aspect of external hazard analysis will be presented. The proposals for further work will be highlighted.

III.2.2. The Achievements of the "Study Case: the Development and Implementation of the Knowledge Transfer and Preservation Process at Cernavoda NPP" PROJECT

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Between 2010 and 2013, the Nuclear Knowledge Management (NKM) team of Institute for Nuclear Research Pitesti with the valuable collaboration of the experts of the Personnel Training and Authorization Department from Cernavoda Nuclear Power Plant (NPP) performed a national project in NKM area. The title of the project was "Study Case: The Development and Implementation of the Knowledge Transfer and Preservation Process at Cernavoda NPP", and was approved in the frame of the IAEA's Coordinated Research Project (CRP) named "Increasing NPP Performance through Process-oriented Knowledge

Management Approach". The partners in the project were the following Members States: Bulgaria, China, Malaysia, Romania, Russia and United States, each of them with their national project. The objective of our project was to develop and then to implement a coherent and consistent strategy for implementing the knowledge transfer and the preservation process at Cernavoda NPP, employing a process-oriented approach. This paper intends to highlight the main results obtained during the development and the implementation of the project, passing through the all phases (visioning, planning, execution, ongoing). The main conclusion that can be highlighted after we performed this project is that this project was a real help in promoting how important is the way to develop and implement the knowledge management in the knowledge-based organizations. More exactly, we can affirm that the process-oriented approach is the adequate one for the nuclear operational organizations.

III.2.3. Upgrading Capacity to Develop and Implement the Technology for Tritium Removal from Heavy Water at the Cernavoda Nuclear Power Plant

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The main idea of this IAEA project (Technical Cooperation Programme) is the exchange of ideas, problems, and results which are involved in develop and implement the technology for tritium removal from heavy water at the Cernavoda nuclear power plant. The coordinator of the project is carrying on related researches and knows the real technical problems associated with tritium removal from heavy water. The main beneficiaries of the knowledge acquired in the project are stakeholders imply in tritium removal facility (TRF) from Cernavoda, e.g. the state company Nuclearelectrica, ICIT Rm-Valcea, CITON and CNCAN. The project is focused on current status of activities developed on TRF. Each partner has a well-defined role in the project: ICIT Rm-Valcea, as the owner of experimental TRF and supplies of the technology and the design of TRF to be implemented at Cernavoda NPP; CNCAN as the regulatory body which will issue all necessary licenses for operating of Experimental TRF with high tritiated water, construction license for industrial TRF and will check the conformity of specific radioactive waste management and TRF' decommissioning plan; Nuclearelectrica SA as final end-user of the implemented technology. The specific seminars, meetings and workshops focused on open issues in the field were carried out. Alongside the technical and scientific contributions, the project will contribute to the confidence and public acceptance of TRF technology.

III.3. Education, Training and Knowledge Management

III.3.1. Developing Criteria for Mutual Recognition

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Development of Nuclear Education, Training and Knowledge Management (NETKM) at national, regional and European level is based on the principles of Bologna process which provide a common basis for mutual understanding between all interested parties. This is a complex action requiring methods and tools for the common understanding of competences based on learning outcomes as defined by knowledge, skills and attitudes/competences.
The development of any national nuclear energy programme is dependent on the successful development of the workforce, through a government and industry supported nuclear educational and training programme. Following the international trend, in Romania two major projects are undergoing in the nuclear sector. One of them is the development of a Generation IV reactor – ALFRED (newest generation reactor) and Extreme Light Infrastructure – Nuclear Physics (ELI-NP). A vital component of any nuclear workforce is the nuclear engineer.
Based on existing experience, one should consider that the requirements of a nuclear engineering programme are based on: high standards of education and training; an inherent adherence to a strong culture of safety and security and compliance to the national system of education. Only when all three aspects are in place it will be possible for a nuclear engineering programme to provide the required human resources.
Nuclear engineering is only one of the professions requiring nuclear knowledge and competence in the nuclear industry. Similar analysis should be undertaken for all professions to ensure that the same high standards of nuclear knowledge and competence are maintained. Through collaboration between all nuclear professions, information and analysis can be shared for the benefit of all stakeholders and continuing improvement of the worldwide nuclear safety culture.
The nuclear industry needs are reflected in the curricula. Thus the Educational Curricula should contain general courses, considered as prerequisites to the Nuclear Engineering, and core courses. For the nuclear engineer, it is important that these topics are well integrated together to produce a well-prepared graduate who can enter into the training programmes for a specific nuclear power plant and reach the required level of competence to successfully carry out his or her responsibilities for safe, secure and economical operation.
To achieve this, the Nuclear Engineering curriculum should contain general courses of mathematics, general physics, chemistry, engineering and computational courses which support the core courses and give the student the required level of competency to access the core courses.
A competent nuclear engineer can be produced through varying contributions of formal academic programmes and industry training but it is important to recognize that neither education nor training on their own can produce the high level of competence required by the nuclear industry. Therefore, in order to evaluate the capacity of competence building, a series of accepted common benchmarking criteria are needed, regarding: policy, strategy, vision and mission of the educational organisation; capacity to deliver nuclear engineering programmes; educational Curricula; outcomes of the programme; quality assurance and accreditation; human resource policy; national and international dimensions; collaboration with industry.

III.3.2. Determination of Trace Elements Content from Steam Generator Deposits by TOF ICP MS

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The investigation of the chemical composition of deposits from a **N**uclear **P**ower **P**lant (NPP) **S**team **G**enerator (SG) is of high importance in order to understand the degradation mechanism and to propose suitable cleaning procedure of these equipments. The deposits properties are different from one NPP to another but they are mainly composed (over 95%) by iron oxides such as Fe_3O_4 and $\gamma\text{-}Fe_2O_3$. The paper presents the method developed for trace level elements identification in this type of samples using the Optimass 8000 TOF-ICP MS. In order to analyze the samples by mass spectrometry, a microwave digestion procedure using a mixture of nitric and hydrochloric acid was also established. The advantages of this sample preparation procedure consist in the minimization of volatile compounds losses and the possible elimination of sample contamination. After digestion the samples were diluted and analysed. All reagents used were of analytical grade and standards have been prepared by dilution of 1,000mg/l certified solutions. For preparation and dilution of samples ultrapure water of 18.2 MΩ.cm was used. The calibration technique used was external standard. Several parameters, such as: torch position, nebulizer flow, plasma flow, auxiliary flow and generator power were optimized before measurement. The instrument settings, the calibration curves and the spectra obtained by analysing the samples are presented in the paper. The elements quantitatively determined in samples were Li, Fe, Cr, Ni, Mo, Mn, Ca, Mg, Ti, Si, Al, si Cu. The acquisition time was set to 5 seconds and 5 replicates were made for each measurement. To avoid isobaric interferences the isotopes: ^{7}Li, ^{25}Mg, ^{27}Al, ^{28}Si, ^{44}Ca, ^{48}Ti, ^{52}Cr, ^{55}Mn, ^{57}Fe, ^{60}Ni, ^{65}Cu, ^{98}Mo were chosen. Optimass 8000 through its time of flight mass analyser provides high speed and simultaneous acquisition of almost entire mass range from the light ions to the heavy ions. This is a fast, precise and accurate multi-element analytical technique for the determination of trace elements (< 0.1 wt. %) in liquid samples.

III.3.3. Thermophysical Properties of UO_2 at High Temperatures

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The paper presents the results obtained in measuring the thermophysical properties of UO_2 at high temperatures. Using Differential Scanning Calorimetry / Calorimeter the heat capacity and the specific heat capacity were determined at temperatures ranging between 200 and 1700 K. Prior, the apparatus was calibrated for temperature, heating rate and heat flow measurements using certificated standards with melting points covering the whole temperature range. Also for each heating rate, the measurement of the blank was needed. The curves corresponding to the blank was subtracted from the experimental curve. The samples were prepared from UO_2 pellets. The specific heat dependency on temperature was obtained by polynomial regression from the experimental curves, after

the corrections were made. For thermal diffusivity measurements the differential thermal analyses were used, also after an appropriate calibration. The temperature range in this case was between 400 and 2100 K. Another set of tests were performed for measuring the linear expansion coefficient up to 2000 K. The dilatometric measurements allow obtaining the equation of the density change with temperature. Using this three parameters, obtained experimentally, the thermal conductivity was calculated. The experimental results are presented and also the equations obtained for each of these parameters. The results are in good agreement with those published in literature.

III.3.4. Determination of the Tritium Activity Concentration in the Molecular Sieve Resulted from the Normal Operation of CANDU Nuclear Power Plants

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Molecular sieve are used in the D_2O Vapour Recovery System to CANDU nuclear power plants. After several cycles of regeneration, the molecular sieve are replaced and declared as radioactive waste that has to be characterized. At each step of the radioactive waste management the activity of radionuclides from the waste package must be well known. Laboratory tests for measuring the concentration of tritium activity of the molecular sieves are important because the efficiency of the procedures and the obtained results are dependent on the laboratory equipment. The paper presents the samples preparation techniques and parameters to be considered in applying the selected methods for measuring the tritium activity by liquid scintillation spectrometry. The specific activity of tritium in molecular sieve is difficult to analysed and require evaluation of several techniques for sample preparation. The paper proposes the comparative use, of two preparation techniques, namely: acid digestion in microwave field and contact with water of the molecular sieve samples. Speed wave Microware Digestion System BERGHOF was used for microwave pressure digestion experiments of the molecular sieve samples. TriCarb Liquid Scintillation Analyzer model 3110TR (manufactured by Packard Instrument Company) was used to determine the tritium specific activity in the resulted solutions obtained by both preparation techniques. The requirements for counting vials and the liquid scintillation cocktail are also presented in the paper. The experimental study developed on the similar molecular sieve samples demonstrate that the both techniques for sample preparation lead to comparable results and, therefore, both are suitable for achieving the proposed paper’s goal.

III.3.5. Separation and Activity Evaluation of ^3H and ^{14}C in Graphite Samples

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This paper focused on separation and activity evaluation of ^3H and ^{14}C in graphite samples from thermal column of Romanian Training, Research, Isotopes, General Atomics (TRIGA) 14 MW research reactor. ^3H and ^{14}C in various kinds of samples can be detected through liquid scintillation counting (LSC), which is an analytical technique defined by the incorporation of the radioactive sample into a homogenous distribution with the appropriate liquid scintillation cocktail. The graphite samples must be solubilized prior to dissolving them into the liquid scintillation cocktail. The graphite is considered as a difficult-to-dissolution sample; therefore, combustion using a PerkinElmer Sample Oxidizer equipment was applied. The samples are combusted in an oxygen rich atmosphere and the hydrogen present is oxidized to water while any carbon is oxidized to carbon dioxide. For samples containing ^3H and ^{14}C, the combustion products will be ^3H_2O and $^{14}CO_2$. Prior to combustion of the graphite samples, Sample Oxidizer performance specifications ($^3H/^{14}C$ recovery degree, $^3H/^{14}C$ retention degree in the circuits of combustion system, $^3H/^{14}C$ spillover) were evaluated. The combustion method provided accurate results, as follows: the degree of ^3H and ^{14}C recovery was higher than 99 %, respectively 98 %; the ^3H retention degree in the circuits of combustion system was 0.04 % and for ^{14}C was 0.03 %; ^3H and ^{14}C spillover was 0.01 % and 0.02 %, respectively. Five i-graphite samples were collected and analyzed from different locations of a cell removed from thermal column of the reactor and the obtained values of activity ranged between $(6.87…19.76)\cdot10^3$ Bq/g for ^3H and $(0.89…1.55)\cdot10^3$ Bq/g for ^{14}C, respectively.

III.3.6. The Elastic Characterization of Zr-2.5%Nb Alloy 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. It has developed a strong texture due to the limited slip system during extrusion process, leading to anisotropic properties. The material properties are strongly dependent on the orientation distributions of grains. Therefore, it 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. Zirconium alloys have a low solubility for hydrogen. Thus, when solubility limits are exceeded, a brittle second phase (hydrides) appears. 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) was

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.

III.3.7. Characterization of mechanical properties of Zircaloy-4 cladding by burst tests

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This work aims the study of test models and thermo-mechanical analysis for the materials which are part of CANDU fuel elements used to cladding of used fuel in Units 1 and 2 from Cernavoda Nuclear Power Plant. The paper presents the tests modes for Zircaloy-4 tubes by mechanical, static and dynamic tests. The subject of the work is the determination of mechanical parameters by burst tests on samples prepared from Zircaloy-4 tubes used to manufacture standard CANDU fuel bundle with 37 elements. The installation for test, measuring and control equipment related to the installation, test mode, the samples after the test and determination of the Total Circumferential Elongation (TCE) and Ultimate Hoop Stress (UHS) are presented in accordance with the standards and procedures used at RATEN ICN Piteşti for this type of testing. The samples were also examined and analyzed in term of microstructural by optical and Scaning Electron Microscopy (SEM). The results obtained for the determined parameters are consistent with the technical specifications used as reference for the performance of Zircaloy-4 tubes used. This type of test is one of the conclusive tests for analysis of the behaviour of Zircaloy-4 alloy in normal operation and especially in case of accident.

III.3.8. A Study Regarding the CANDU Fuel Bundle Geometry Influence on Lattice Parameters of Interest for Nuclear Fuels with Natural and Slightly Enriched Uranium

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This paper was elaborated following the cooperation agreement between Institute for Nuclear Research Pitesti and University of Pitesti. The goal of our work was to find out the geometry influence of some CANDU fuel bundle designs on lattice parameters of interest when such nuclear fuel carriers are filled with Natural Uranium (NU) and Slightly Enriched Uranium (SEU). The lattice parameters of interest are Power Peaking Factors (PPFs), which represent the ratio between average linear power on a fuel rod ring and the average linear power on the entire fuel bundle. The PPFs have been calculated using the widely spread transport equation solver computer code WIMS (Winfrith Improved Multigroup Scheme), available in INR Pitesti in D5B NEA version. The working configurations correspond to both fresh fuel and irradiated fuel up to the discharge burnup from a generic CANDU reactor. Three Uranium enrichments were used: 0.72 wt% U235 (corresponding

to Natural Uranium), 1wt% U235 and 1.5wt% U235 corresponding to SEU fuel. The PPFs represented with respect to the fuel burnup show closer values as the burnup increases allowing for a better fuel ring power flattening-one of the main goal in fuel bundle design process. The advanced geometry design of the 43-rods bundle design also led to a more efficient power flattening over the fuel bundle.

III.3.9. Implementation of Defence in Depth for Lead Fast Reactors

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The current operating Nuclear Power Plants (NPP) have been designed taking into account the defence in depth (DiD) concept, which represents the most important and practical strategy in order to realize a high level of safety. This concept is universally accepted as being the most basic and most effective safety principle, therefore it must remain essential for the safety of Generation IV (Gen IV) systems. To compensate for potential human errors and mechanical failures, the implementation of DiD is based on levels of protection including succesive barriers preventing the release of radioactive materials to the environment. There are four barriers that must retain the fission products (the fuel matrix, the fuel cladding, the boundary of the reactor coolant system and the containment system) and five levels of protection associated with the five NPP operational states (normal operation including shutdown state, anticipated operational occurrences, complex operational occurrences and design basis accidents, severe accidents, post-severe accident situations). This paper applies the Objective Provision Tree (OPT) as a tool of the Gen IV Nuclear Systems Integrated Safety Assessment (ISAM), in order to evaluate the levels of the DiD for Advanced Lead Fast Reactor European Demonstrator (ALFRED). In order to implement the DiD for ALFRED, the three fundamental safety functions have been considered: control of the reactivity, removal of the heat from the core and confinement of radioactive materials. Taking into account the design characteristics of ALFRED, for levels 1 to 4 of the DiD and for each of the safety functions, those provisions to guarantee the maintaining of safety functions have been identified, by means of OPT. Therefore, starting from the objective of each level of DiD and from the acceptance criteria of each safety function, the hierarchic structure of OPTs identify a set of possible challenges which could have impact on safety functions, the plausible mechanisms which can materialize these challenges and, finally, the provided provisions to prevent, control or mitigate the consequences of the challenges/mechanisms. The considerations for level 5 of the DiD refer to the fact that the complete dismissal of offsite emergency plans it would be inconsistent with DiD concept, even if one of the specific safety goals for Gen IV system is “Gen IV nuclear energy systems will eliminate the need for offsite emergency response”.

III.3.10 Treatment of Liquid Radioactive Waste by Ion Exchange – Organic vs. Inorganic Ion Exchange Media

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The liquid radioactive waste can be generated in various systems of a nuclear power plant, thus the radiochemical composition and, therefore, their proper treatment methods are various, too. Ion exchange methods have extensive applications in nuclear fuel cycle operations and other activities involving radioactive materials. Process water systems from nuclear power plants typically use organic ion exchange resins to control system chemistry. Organic resins have also being used in several chemical decontamination or cleaning processes. The organic ion exchange resins have relative low radiation stability and are inappropriate for treatment of solutions with high concentration of dissolved salts. Most of the ion exchange process limitations can be reduced or eliminated only by using ion specific exchangers, which are usually inorganic materials. The purpose of this paper is to emphasize the advantages and limitations of ion exchange process, by comparing organic ion exchangers with inorganic ones. The data reported in the literature have been reviewed, by analyzing the following: the stability, the exchange capacity, the selectivity, the mechanical strength, the cost, the availability and the immobilization possibility for both ion exchanger types. In order to develop an ion exchange process for a specific application, it is necessary to evaluate the options for conditioning of the spent ion exchange materials, considering the requirements for their disposal. The organic and inorganic ion exchangers have both advantages and disadvantages. The balance between them will lead to the selection of the optimal method for a specific application, but it should be taken into consideration that a modern trend among authorities and industry is to avoid using of organic based materials with limited radiation stability.

III.3.11 Increasing the Performance and the Efficiency of the New Generation of Nuclear Power Plants

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The actual Nuclear Power Plant units design, as regarding to their performance and efficiency, have to be improved and upgraded. Looking to the secondary circuit global efficiency, the upgrading possibility exists. Necessity to increase by design the nuclear power and correspondingly the electrical power is possible but the steam turbine becomes high volume, especially on the low-pressure stage. Discussing about the global NPP unit efficiency, the transmission of the heat from the reactor by means of the steam generator is performed with a high efficiency; also the steam turbines have good conversion efficiency. In the thermal power plant units, based on different thermal cycle, the secondary thermal circuit evacuates the heat through the condenser. The main desire or goal of the power plant unit is to produce electrical power, so to convert the thermal power in electrical power by any means. One of the solutions is to use appropriate cooling agents

that have effective thermal and chemical properties. To use an appropriate thermal cycle with the optimum heat transport agents, in the secondary circuit of a NPP unit, is very important. Using different design on the BOP of NPP unit performs this; some of the effects are on the accident risk and also to the NPP unit reliability. The paper presents different nuclear power plant units that use advanced fuel cycles and efficient thermal cycles on the secondary thermal circuits. Some examples of how is evaluated the accident risk and the plant unit reliability are presented in the paper in order to optimise the design and the investment costs for an accepted accident risk level and a given NPP unit reliability.

III.3.12 The Behavior of Steam Generator Tubing of a CANDU NPP on Stress Corrosion Cracking in Caustic Environment

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Incoloy-800 (type UNS N08800) is an austenitic alloy Fe- Cr- Ni used for the manufacturing of the steam generator pipes of CANDU NNP. This alloy is resistant at general corrosion, but, in certain environments and in the presence of a stress state, is susceptible to Stress Corrosion Cracking.
Stress Corrosion Cracking (SCC) in caustic environment may affect the secondary side of the steam generator in CNE-Cernavoda, Romania, because of the accidental formation of an alkaline environment, especially in areas with restricted flow, due to the leaks of cooling water from the condenser (the water from the Danube River which contains dissolved alkaline salts). The stress state can appear in the rolling expansion regions of the steam generator tubes at the joint with the tube plate and in the region of the tube supports.
For the testing of this material at SCC there have been used type C-ring samples. In order to create stress concentrations, on the outer side of the C-ring samples there have been made mechanical cracks of different sizes. The samples have been stressed using a screw that was made also of Incoloy-800. The value of the stress at the top of the mechanical cracks was estimated using the ANSYS calculation code.
SCC tests were performed in 10% NaOH solution (pH=13), at 260^{0}C and 50atm. The total testing period has been 57 days. Initially, the samples were tested for 20 days, but there could only be observed cracks in the areas of maximum stress estimated by the ANSYS code. After another 37 days of tests there could have been seen by optical microscopy the existence of SCC transgranular cracks. The depth of the cracks penetration in the walls of the Incoloy-800 tubes depends on the dimension of the mechanical defect.
The SEM analysis has confirmed the transgranular character of the cracks, the areas of transgranular cracking areas being clearly separated from the areas of mechanical crack. The film formed on the samples tested in these conditions is made out of spinel compound like magnetite (Fe_3O_4), trevorite ($NiFe_2O_4$), chromite ($FeCr_2O_4$), and sodium based compounds.

III.3.13 Failure analysis on CANDU structural materials by DHC mechanism

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Zircaloy-4, widely used as nuclear fuel cladding, has proven very reliable in CANDU reactors and also meets the requirements to operate under specific thermo-mechanical, corrosion and irradiation conditions due to low thermal neutron capture cross-section, good mechanical properties, resistance to water corrosion at high temperature. Despite these advantages, the structural integrity of these reactor components can be affected under certain conditions including stress, temperature and hydrogen concentration exceeding the limit of solubility. When the concentration in solution exceeds the TSS (**T**erminal **L**imit of **S**olubility), the excess hydrogen precipitates as platelets of hydride. Under specific conditions including thermal cycling and stress gradient, the hydrides, oriented after the fabrication schedule in the circumferential direction, tend to reorient. This phenomenon is responsible for a time-dependent failure through the hydride zone. If the process is repeated sequentially, a stepwise crack growth in a stable fashion will take place, the so-called **DHC** (**D**elayed **H**ydride **C**racking) mechanism. This paper presents the experimental method to determine the threshold factors responsible for DHC phenomenon. The tests were performed in temperature conditions (282^{0}C) on hydride PLT (Pin Loading Tension) samples obtained from Zircaloy-4 cladding tube. Some results concerning DHC susceptibility of zirconium based alloys are presented. Metallographic investigation was also performed to observe and measure the crack extension during fatigue pre-cracking and K_{IH} test by using the nine-point or five-point average method.

III.3.14 Laboratory Experiments for Evaluation of Cs Transport Parameters through Geologic Environments for Radioactive Waste Disposal

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For the safety assessments of a radioactive waste repository it is important to know the radionuclide transport parameters through engineered and natural barriers of the disposal system. The main transport mechanisms for radionuclide migration are advection and diffusion while the processes such as mechanical dispersion and sorption act as retardation mechanisms on the radionuclide transport. Batch and column experiments were performed on limestone samples from the Saligny site, selected for the disposal of low and intermediate level short lived (LIL–SL) radioactive waste. Radionuclides of particular interest for this paperwork are Cs–137 and H–3. Tritium was selected as nonreactive contaminant, while cesium is known as a very reactive contaminant especially in the geological formations with high clay content. Experiments were run on crushed limestone with particle size distribution between 75-500 µm (fine particles) and 500-2000 µm (coarse particles). Experimentally values for transport parameters obtained for columns packed with finer limestone particles can approximate the transport parameters for intact healthy limestone while those gotten for coarser particles can approximate the transport parameters through the fractured, weathered limestone. Experiments were

performed under saturated conditions using Saligny simulated water containing the main cations and anions found in the natural groundwater. After ~ 55 days only tritium was detected in the collected effluents, while Cs was completely retained on the limestone particles. The tritium breakthrough curves were fitted using CXTFIT code. This code was developed for transport parameters estimation from laboratory and field tracer test experiment and it is implemented under STANMODE (Studio for Analytical Models for Solving the Convection-Dispersion Equation). A gamma probe was used to scan the columns and estimate the final distribution of Cs-137 along the columns. In batch experiments performed to measure the distribution coefficients of Cesium on the two types of limestone showed that this radionuclide is moderately sorbed on limestone with distribution coefficients of 17.93 ℓ/kg for finer limestone and 10.18 ℓ/kg for coarser limestone, respectively.

III.3.15. The Development of Irradiation Testing Technology for SEU-43 Advanced Nuclear Fuel

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At a worldwide level, the need for energy it is in continuous growth. For this reason, it is necessary to develop new technologies for nuclear power plants (NPPs). One of the aspects on which the scientists are focusing is the development of new types of nuclear fuels, which should increase the efficiency and safety of NPPs. SEU-43 (**S**lightly **E**nriched **U**ranium) nuclear fuel is an advanced nuclear fuel derived from CANDU fuel and designed to be used in CANDU-6 NPP. In order to observe the behaviour of this type of fuel during the residence in the reactor, it was necessary to develop irradiation testing technologies to determine the fuel's characteristics and its evolution during the irradiation in CANDU reactors. The purpose of these experiments is represented by the qualification of SEU-43 nuclear fuel for use in CANDU-type NPPs. The testing of SEU-43 fuel required the design of an experimental infrastructure based on Loop-A 100kW irradiation device, device designed for irradiation testing of nuclear fuels and structural materials in TRIGA SSR-14 MW reactor. Using this loop-type irradiation device, an upgraded version of IRENE irradiation device used in the OSIRIS reactor facility (France), several tests have been realized on nuclear fuels and structural materials which have later been used for CANDU nuclear reactors at the Cernavoda NPP. After the implementation of the experimental infrastructure, it was necessary to test its functionality and the compatibility of the components. The test consisted in bringing of the parameters at nominal levels, with the test section by-passed, and maintaining of the parameters at respective values for a predetermined period of time. After these tests, it was concluded that the Loop-A irradiation device, after the upgrade of its components, is capable to fulfil its functions and the experiment can be realized without any concerns regarding the nuclear safety. These conclusions are supported by the acquired data during testing, data which describes the behaviour of the device and also by the numerous systems and functions with active role in the nuclear safety.

III.3.16 Investigation of Reinforcement Corrosion from the Reactor Concrete Containment

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Containment provides the ultimate barrier between the primary reactor systems and the outside environment and is designed to withstand the loading resulting from the external and internal events
Nuclear safety related concrete structure is composed of several constituents: concrete, conventional steel reinforcement, pre-stressing steel and steel liner plate.
Corrosion of steel in concrete is a well known problem with large consequences in terms of economics. In appropriate conditions (e.g., high pH), the steel from concrete will be protected against corrosion, but the protective film formed on the steel surface in concrete under specific conditions, can be destroyed by lowering the alkalinity (e.g. by carbonation and / or the presence of chloride).
The paper discusses results of experiments performed in conditions simulating steel state in concrete, for both protective (0.02M $Ca(OH)_2$ (pH about 12.5), solutions simulating concrete pore water [0.6M KOH, 0.2M NaOH and 0.001M $Ca(OH)_2$] having a pH of ~13 and deionized water, respective corrosive conditions by adding NaCl.
Electrochemical behavior of steel in alkaline environment is complex and not well understood. In this work has been applied an electrochemical methods to evaluate the corrosion behavior of reinforcing steel in concrete and also the metallographic microscopy examination to emphasize the possible corrosion attack produced by test solutions.

III.3.17 Curriculum Design for VET in Nuclear Domain. EQF, ECVET and other specific methodologies and tools

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In this paper the authors, based on their previous experience in developing a VET Curriculum for different other education and training courses, briefly present the way to build-up a curriculum to be introduced in the portfolio of the organisations delivering HE-VET training to the specialists working in the nuclear domain and related fields. A coherent methodology that starts with the definition of Learning Outcomes (LOs) in terms of knowledge (K), skills (S) and competences (C), the aggregation of KSC's in "Learning Units" (LUs), the calibration of these LUs from the ECVET perspective and suggestions related to the main themes to be approached in the training process is also presented within this paper.
Assessment, evaluation and certification of the educational and training process is also described. In the same time, the paper emphasizes on European Tools for Mobility such as Memorandum of Understanding, Learning Agreement and Transcript of Records and provides suggestions for potential adaptation to the specificities of the nuclear domain.

Last but not least, the authors try to correlate this methodology with current developments acquired in the framework of different projects such ARCADIA, looking for possible spin-off effects and added values to the project outcomes.
A clear relation with the educational demarches accomplished within the University of Pitesti in order to develop a BS diploma in Energetics and Nuclear Technologies is also approached towards the last part of the paper.

III.3.18 Sustainable Development Based on Nuclear Energy – Implications from HR Perspective

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In this paper the authors present the last year tendencies regarding the sustainable development from the perspective of Human Resources (HR) management.
The evolution of HR in nuclear domain is rendered with a focus on the influence of different external factors such as demography, labour dynamics and fluidity of the market.
The paper analyzes the impact of changes in political approach in the last years and the financial pressure on the HR development.
In this context, the role and place of the University of Pitesti as HE provider and as local developer of HR for nuclear domain in Romania, together with its efforts made in the last years in order to preserve the necessary institutional framework and heritage of knowledge in what means HR preparation for the nuclear field is equally described by the authors.
At the end of the paper, the authors explain a clear and coherent methodology that can support the establishment of international partnerships in order to better prepare the HR for nuclear field and this way could share the existing knowledge and infrastructure of the potential partners.

III.3.19 Study of the applicability of CFD codes for thermohydraulic analyses specific for Gen IV Nuclear Reactors

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In this article has been studied the applicability of CFD codes for thermohydraulic analyses specific for Generation IV Nuclear Reactors cooled by molten lead. Studies consisted in calculation the friction pressure losses factor f and convective heat transfer coefficient α, using both correlations found in the literature and numerical simulations using ANSYS CFD. Analyses where made on different configurations of the fuel assembly and at different flow regimes defined by the Peclet criterion. The results obtained using CFD models(with a number of user-defined parameters regarding turbulence models and mesh) are close to those obtained using correlations.In conclusion, these physical models implemented in CFD codes can be trusted for thermohydraulic studies of more complex configurations of the core.

III.3.20. Plasma Electrolysis Processing for the Deposition of a Ceramic-like protecting Aluminium containing Layer on the AISI 304L and 316L Stainless Steels

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By plasma electrolytic treatments, electrodepositions from innovative ionic liquids, duplex treatments, one aims the forming of the ceramic-like films containing aluminium. These films give better corosion resistance in the liquid lead medium used in Fast Lead Reactors. Light metals and those from „valve metals cathegory” have intrinsic resistance at atmospheric corrosion due to the presence of a dense, pasive, protecting layer (in the form of an aluminium oxide or hidroxide on Al substrate, which due to low electrochimic potential, is formed immediately after the metal exposure to the air). For iron or austenitic stainles steels it is necessary the forming of an intermediate dielectric film on the substrate. The properties of the obtained structures by these treatments depend on the initial state of the substrate. Thus, in a preliminary stage, a controlled modification of the substrate was achieved by various treatments, such as by Cyclic Sweep Voltametry (*CSV*) or by autoclaving of the sample. A programable continuous power supplay was used for better control of the experimental parameters. The treatments were performend both in the anodic regime, on the AISI 304L and AISI 316L substrate in the aqueous solution of *0.1 M* $NaAlO_2$ and *0.05 M* NaOH, as well as in the chatodic regime, on the AISI 304L in ionic liquid solution of choline chloride : urea (*1:2* molar ratio) containig *0.5 M* $AlCl_3$. The samples were then characterisated by different techniques: *X* Ray Photoelectron Spectroscopy (*XPS*), *X* Ray Diffraction (*XRD*), Energy Dispersive *X* Ray Spectroscopy (*EDS*), Scanning Electron Microscopy (*SEM)*, metallography, electrochimic techniques. The results show, in the selected experimental conditions, the formation of a surface layer containing aluminium, which has an amorphous and inhomogeneous character, and gives a better corrosion behaviour.

III.3.20. Plasma Electrolysis Processing for the Deposition of a Ceramic-like protecting Aluminium containing Layer on the AISI 304L and 316L Stainless Steels

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This paper aims at analytical methods for the study of long-lived radionuclides at Ultratrace in highly active materials in reactors to reduce contamination of instruments and decreasing the dose received by the operator.
The paper presents the alpha spectrometry (it is a particularly useful method for the quantitative determination of active alpha elements, which under certain conditions can not be determined by other methods) and mass spectrometry with inductively coupled plasma (is an analytical method, highly sensitive for simultaneous determination of concentrations of radionuclides with

long half-life in the trace and ultratrace).
The purpose of the project was to determine the concentration of actinides by mass spectrometry with inductively coupled plasma from spent nuclear fuel.
It presents spectrometer inductively coupled plasma mass ELAN DRC-e, the basic components of an ICP-MS instrument, interference isotope ratio determination, the chemical separation of actinides, actinides using ICP-MS determination site.
Following the results obtained by the two methods of measurement have been found losses of radionuclide sources alfa preparation by up to 20%. Inductively coupled plasma mass spectrometry technique proved to be a useful tool for determining the actinide ppb level.
The conclusions are: using ICP-MS technique were reduced to measure samples at 24 hours it takes to measure alpha sources with an alpha spectrometer multichannel to 2 hours if the equipment is not turned on; unlike alpha spectrometry, inductively coupled mass spectrometry in plasma can follow all stages of separation of actinides and intermediate yields can be determined.

III.4. International Partnership for a Sustainable Development

III.4.1. Public participation in decision making process on nuclear facilities

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Aarhus and Espoo conventions, already implemented in the national legislation, recommend a set of measures for early public participation in decision making process regarding nuclear projects. Aarhus convention is based on three pillars: the access to information, the public participation in decision-making and the access to justice. Espoo convention is based on the understanding of the fact that the environmental threats do not respect national borders. There are many motivations to involve public in decision making process (DMP): sustainable development requirements ask for each and every actor's involvement; a larger participation, generally, determines a better DMP; a larger participation results in a better implementation of the decisions; improvement of the relationship between government and civil society. It should be noted that public participation cannot be made without continuous and adequate information.
From this reason, communication, information or education programmes represent the starting point in a process devoted to train and involve stakeholders and general public in the decision making. Institute for Nuclear Research (INR) Pitesti initiated the implementation of approaches, methods and tools for public participation in Romania in the frame of FP6 project COWAM2. Afterwards in some European projects like CIP (COWAM in Practice), IPPA (Implementing Public Participation Approaches in Radioactive Waste Disposal) and PLATENSO a consolidation of the public debate was targeted. An useful experience for the construction of public participation in decision regarding the implementation of ALFRED demonstrator in Romania was obtained. The demonstrator is proposed to be built on Mioveni nuclear platform based on the FALCON consortium agreement and the Romanian Govern Memorandum declaring the platform as the reference site. A first step is represented by the initiating of the dialogue with the local community, under FP7 ARCADIA project (Assessment of Regional CApabilities for new reactors Development through an Integrated Approach). A Local Group was set-up in order to create an interface between implementer and local community. A bi-directional communication is expected aiming to achieve continuous information to the community on the proposed investment and decision process and to capture the feedback related to public concerns, suggestions, requirements and proposal for improvement. The paper will be focused on the public involvement methods experienced by INR in the above mentioned projects and on the construction of the ALFRED local dialog group, its role, objective and functionality.

III.4.2. Multi-Unit Site and its Challenges for Risk Assessment

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The Fukushima Daiichi accident has clearly proven that there is a significant interaction between the nuclear units located on the same site. In the event of an external hazard occurrence, or even worse, in the event of occurrence of combinations of external hazards, in the assessment of risks, it is important to take into account the sharing of systems and the resources available to undertake severe accident management activities and the capacity to implement the adequate actions, given the sequence of releases from different sources. There is an increasing recognition of the critical need for the evaluation of a site risk in an integrated way, and this approach requires the consideration of the potential accidents that affect multiple installations in the same time frame.

The international nuclear community is taking serious efforts trying to develop and reach consensus on the modalities in which the multi-unit PSA should be conducted and multi-unit safety goals should be defined. The FP7 Advanced Safety Assessment Methodologies: extended PSA (ASAMPSA_E) project is not an exception to these efforts. The project aims at examining in detail how far the Probabilistic Safety Assessment (PSA) methodology is able to identify any major risk induced by the interaction between a nuclear power plant (NPP) and its environment, and to derive some technical recommendations for PSA developers and users.

The paper presents the results obtained in the investigation regarding the use of PSA in the context of a multi-unit site. The work represents an attempt to respond to the question if in case of a site consisting of multiple nuclear installations, the traditional PSA is adequate enough to assess the risk incurred by a severe event affecting all or multiple installations on the site. The latest insights regarding the development and application of a multi-unit PSA, as the issues that require further investigations will be highlighted.

III.4.3. Regional Excellence Project on Regulatory Capacity Building in Nuclear and Radiological Safety, Emergency Preparedness and Response in Romania 2013-2016

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The objective of The Regional Excellence Project on Regulatory Capacity Building in Nuclear and Radiological Safety, Emergency Preparedness and Response in Romania 2013-2016 is to enhance the capabilities of the National Commission for Nuclear Activities Control (CNCAN) in eight specific functional areas of work through exchange of experiences, best practices and capacity building with the Norwegian Radiation Protection Authority (NRPA) and the International Atomic Energy Agency (IAEA). The project has 8 subprojects: CNCAN 1 - Enhancement of CNCAN capabilities for safety analysis; CNCAN 2 - Enhancement of CNCAN capabilities for: integrated management systems and knowledge management; CNCAN 3 - Enhancement of CNCAN capabilities for inspections; CNCAN 4 - Enhancement of CNCAN capabilities for safety and security of transport and transit of radioactive and nuclear materials on the Romanian Territory; CNCAN 5 - Enhancement of CNCAN capabilities for emergency preparedness and response; CNCAN 6 - Enhancement of CNCAN capabilities for ionizing radiation sources control; CNCAN 7 - Enhancement of CNCAN capabilities for radioactive waste, spent nuclear fuel management and decommissioning activities; CNCAN 8 - Enhancement of CNCAN capabilities for safeguards. The project budget is 85% allocated from Norway Grants and 15% provided by the Romanian national co-financing. The main project partner is NRPA

and the partner organization is the IAEA. All project activities are organized under IAEA extra-budgetary programme (EBP). The project represents a platform for sharing experience and best practices between the CNCAN, NRPA and the international nuclear community through the IAEA active involvement. Some of the most important activities of the project are related to the development and updating of the Romanian regulations in line with the EU and IAEA standards and guidelines as well as training of CNCAN staff using effective knowledge transfer methods that will ensure sustainability. These activities will enhance CNCAN's regulatory capabilities in the areas of safety analysis, integrated management systems and knowledge management, inspections, safety and security of transport and transit of radioactive and nuclear materials, emergency preparedness and response, ionizing radiation sources control, radioactive waste and spent nuclear fuel management and nuclear safeguards control.

III.4.4. Perspectives for Nuclear Hydrogen in Romania

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This paper presents recent activities of National Center for Hydrogen and Fuel Cell in programs and major initiatives that can be interconnected with nuclear hydrogen. Romania, through this center is collaborating with Canada on nuclear hydrogen production with the thermochemical Cu-Cl cycle.
The Center currently performs theoretical and experimental research in integrating renewable energy with water electrolysis for hydrogen "green" production, catalytic membrane reformer design, thermo-chemical water splitting. The potential of the Copper-Chlorine (Cu-Cl) thermo-chemical cycle for large-scale hydrogen production using nuclear energy is under investigation to perform integrated lab-scale experiments. The Center is a part of an international project named "Clean Hydrogen Production with Water Splitting Technologies", with Canada as the leader of the project. The Center has participated in projects involving the assessment of the potential and relevant business cases for large scale and seasonal storage of renewable electricity by hydrogen underground storage in Europe – in a HyUnder consortium. The scope of Romanian participation was to identify the most promising site for hydrogen underground storage. For large scale underground storage, hydrogen can be produced by both, renewable and nuclear.