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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.

The Use of Dynamic Simulation for the Design, Validation and Training for New and Existing Nuclear Reactors

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The resurgence of the nuclear industry has produced many new nuclear designs, resulting in more intelligent control rooms and more complex digital control and safety systems. GSE's real-time, dynamic simulator platform supports the design of new nuclear "First-of-a-Kind (FOAK) plant designs in a way that was not available for the previous generation of plant designs. Nowadays it is possible to integrate and couple advanced engineering analysis codes (like RELAP, SIMULATE, MAAP, MELCOR etc) in the simulator platform and build very accurate predictive dynamic real-time models based on plant design data. The simulator then becomes an important tool to: 1. Reduce risk during the design process, 2. Provide a platform to demonstrate the new design and control system strategies to the regulators, 3. Train operators prior to plant commissioning. FOAK projects are by their nature high risk, ill-defined and require a collaborative approach. GSE developed a phased, agile approach for developing FOAK engineering simulators that integrates with the overall plant design process. In addition to plant design and control system validation, the same simulator becomes an operator training tool and a tool to verify and validate operation procedures for normal, abnormal and emergency plant conditions. The simulator will be a vital tool during the whole plant life period. The presentation will explain the simulator technology of today, the methodology and discuss results from plant designs where GSE's simulators were used.

PLENARY SESSION – May 29

Unfinished Upgrades to CANDU Reactors that Can Reduce Risk from Severe Accidents

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Deterministic analyses to evaluate system response and accident progression after a station blackout scenario have long unveiled a number of CANDU design vulnerabilities that increase risk un-necessarily. Many of these can be fixed with a bit of engineering ingenuity and at no great cost; others require serious retrofits. A number of design improvements have already been incorporated at a number of plants. These include addition of some PARS, containment venting systems, emergency water addition to the reactor vault and acquisition of additional emergency diesel generators. But this has been done without a proper evaluation or consideration of severe accident source terms and a number of fundamental design improvement opportunities remain ignored. Neither the utilities that own and operate the reactors, nor the designers, and still not the unconditionally supportive regulators acknowledge the need to go the extra steps to comprehensively reduce the risk, especially with enhanced safety expectations arising after Fukushima. The utilities and designers have to reconsider test data of critical mitigating equipment and incorporate further improvements to reduce risk. It is undeniable that operating CANDU reactors were never designed with severe accidents within their design basis; yet utilities, designers and the regulators find it compelling to instinctively claim superior ability to meet severe accident challenges even as comprehensive design improvements and supporting accident analyses are incomplete. This paper summarizes the analysis and design evaluations that demonstrate the remaining severe accident related vulnerabilities of a CANDU reactor and shows how the design still inherently causes uncontrolled early pressure boundary ruptures; premature expulsion of critically needed coolant from the main heat transport system and the moderator heat sink; direct exposure of core debris and fission product releases to the containment; thermo-mechanical failure of the much hyped core catcher – the thin shell Calandria vessel at its welds - unable to long term hold hot debris in spite of a relatively cold envelope of water around it; accelerated production of copious amount of combustible gases and containment boundary failures instigated by the relatively low pressure bearing capacity of its penetrations as well as triggered by hydrogen explosions with a grossly inadequate number of hydrogen removal PARS units deployed to actually avoid explosions. The nuclear power regulatory bodies have regrettably failed to provide technical leadership and have not recognized the need for certain critically needed design upgrades to better mitigate severe accidents and to rectify the design errors and omissions. The CNSC in Canada has failed to enforce even the feeble requirements it put in place after a much hyped publication of its Fukushima Action Plan and continues to effectively withdraw under pressure from the utilities many of the requests it did make for design enhancement evaluations. Engineered design measures to avoid some the above undesirable accident progression paths have been put forward and summarized in the paper. While long periods of time are available from the onset of the station blackout to the first uncontrolled

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.

The ALFRED Project: Opportunities for Romania

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Representing one of the few, most advanced systems that will be built in Europe in the next future, ALFRED (the Advanced Lead-cooled Fast Reactor European Demonstrator) will offer a technology platform for the advancement in very high value-adding activities through strengthened research, technological development and innovation. Its realization, along with all the satellite initiatives implied for its achievement, offers an unique opportunity for addressing a number of Romanian interdisciplinary challenges. The wide support required for its realization will create a number of high level job opportunities – aligned with the highest international (nuclear) standards –, thus increasing the sustainability and quality of the employment at national level. Apart from the demonstration role, ALFRED will also represent the prototype of a commercial fleet of LFRs. Based on the fast spectrum technology, the LFR has the great potential for preserving and protecting the environment through the reduction of the volume and the radiological inventory of the waste to be disposed of, thanks to the use as fuel of plutonium. At the same time, the higher exploitation of uranium would allow the promotion of resource efficiency. Accordingly, ALFRED and its successors might not only represent an actual opportunity for Romania to implement an advanced strategy for the management of the spent fuel, but also a more sustainable option for the further exploitation of nuclear energy in the future. Thanks to the intrinsic features of lead, the safety level ensured by ALFRED in preventing and mitigating the risk of severe accident conditions due to any potential initiator – including extreme external hazards – might results a keyword for the public acceptance of such an innovative nuclear energy system, representing an optimal example for Europe to promote its excellence in the nuclear field. The Romanian support to the ALFRED Project will provide an opportunity to define a strategy for stimulating the education system in restoring cohesion, excellence and efficient use of R&D resources and academic environments, further enhancing the capability for high value-adding activities on one side, and the attractiveness of education on the other. Among the most relevant initiatives, the Project will encourage the establishment in Romania of a Center of Excellence (CoE) in support to the advanced technology of Heavy Liquid Metals for reactor applications, notably: competence in the operation of large HLM facilities; experimental capabilities to perform thermal hydraulic simulations of reactor relevant conditions; professional capabilities to perform measurements, develop sensors, calibrate and qualify data acquisition chains; experimental capabilities in environmental compatibility tests of structural and functional materials and components; materials testing capability, accompanied by synergic use of the hot cells associated with the TRIGA Reactor of Institute for Nuclear Research Pitesti. The international support already in place ensures reliability to the project. Opportunities will be taken from the Horizon 2020 Framework Programme to the largest extent, ranging from research programs for the advancement of the LFR technology to supporting actions for strengthening the FALCON Consortium (both in the frame of the EURATOM call), to initiatives aimed at initiating the creation of a CoE in Romania (through dedicated Teaming and Twinning programs).

The FALCON Consortium: Continuing the Successful Italian Romanian International Collaboration

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The history of the international collaboration between Italy and Romania dates back to the early 2000, moving in parallel between the completion of construction and the commissioning of Cernavoda NPP unit 2, and the engagement in the extensive research for lead-cooled subcritical and critical systems. Concerning lead technology, since the ‘90s ENEA and Ansaldo started the development of Accelerator-Driven Subcritical (ADS) systems for the transmutation of the legacies from the spent fuel of LWRs, pursuing the strategic interest of the national industry in the Lead-Fast Reactor (LFR) development. The Romanian Institute for Nuclear Research (INR) Pitesti, joined the activities through the participation to several international research projects supported by the 6th and 7th EURATOM Framework Programmes. Among the many, the most relevant are: ADRIANA, aimed at the investigation of the existing and needed research infrastructures for supporting the development of Generation IV reactors; LEADER (Lead European Advanced DEmonstartion Reactor), with the twofold aim of further developing the design of an industrial LFR (starting from the results of another relevant EU FP6 project, ELSY) and to develop its representative demonstrator, ALFRED (the Advanced Lead-cooled Fast Reactor European Demonstrator); and finally ARCADIA, on the identification of the primary needs for the feasibility of the ALFRED project and the existing National and Regional supporting structures with the aim to draw a map of competences potentially eligible to satisfy the identified needs. The commitment of INR in support to the ALFRED project is considerable, and mostly driven by the intention to construct the reactor on the Mioveni site, close to the present location of the research centre. Currently INR is strongly involved in the development of ALFRED both at technical and strategic levels. To coordinate these activities and take profit of the most intimate synergy, planning and specific agreements have been formalized initially with a Memorandum of Understanding (MoU) signed between ANSALDO, ENEA and INR in November 2011. Recently the aim at optimizing the cooperation through strategic, management, governance, financial and technical work towards the realization of ALFRED in Romania led to the signature of the FALCON

(Fostering ALfred CONstruction) Consortium between the same members in December 2013. The main objectives of FALCON, in its first phase, will be: the joint review of all the technical information and data available as a result of the engineering and experimental activities previously performed; the sharing of information on topics related to the development and progresses of LFRs, also considering the exchange of technical experts whenever beneficial for an efficient progress of the common activities; the joint review, confirmation and constant update of the cost estimates for the design and construction of ALFRED and the identification of a potential industrial supply chain to possibly attract manufacturing companies as potential investors; the identification of the private/public funding mechanisms as potential source to implement the Project, and in consideration of the possible revenues expected from electricity selling when ALFRED will be put in operation; the coordinated promotion of appropriate actions to seek for funding from the member states and the European Community, interfacing with decision makers, in order to allow the implementation of the Project; the promotion of the necessary actions to enlarge the participation of interested and qualified industrial/research organizations to the Consortium itself; the joint definition of a roadmap aiming at setting up the milestones of the Project and at identifying any constraint/risk (technical, financial or geo-political) to be overcome in order to proceed in the implementation of the Project. In this paper, the scope of the FALCON Consortium will be presented along with the present status of the activities and the planning for the successful achievement of its goals.

ARCADIA Project in Support to ALFRED Demonstrator

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Generation IV nuclear systems represent a new and innovative class of reactors that will replace the today's fleet for a more efficient use of the fuel and a better use of uranium resources, for an improved safety, and for the significant minimisation of the radioactive waste.
European Union is deeply involved in the research and technologic development of first three from the six most promising advanced reactor systems and related fuel cycles: SFR, LFR, GFR, MSR, VHTR, and SCWR. SFR was selected as reference technology while LFR or GFR are considered alternative solutions.
Romania as well is engaged together with Italy in development of the LFR systems, whose technical confirmation must be proved by ALFRED demonstrator proposed to be constructed in Romania. To build ALFRED reactor, there are few key requirements to be fulfilled, each of them equally important in the successful implementation.
Scientific and technical competences should be available to solve, in short term, the special technical aspects raised by the innovative character of the ALFRED project. They should be complemented by specific competence in management of an international project, in legal and financial aspects.
Clear steps and requirements for the ALRED site and construction licensing must be established, complying the national and international norms applicable to the Generation IV systems. Public involvement from the very early stage of the project should be also considered. As any investment, ALFRED needs a feasibility study to get all necessary approvals for its financing and construction. The key chapters of the feasibility study must be carefully prepared to emphasize the competitive advantage and core values of the project, and the socio-economic benefits.

Appropriate financial schemes must be identified and proposed in order to ensure the funds necessary for construction and operation. A detailed roadmap, pointing critical milestones and risks, and risk mitigation solutions are also essential elements in the implementation of the ALFRED demonstrator, taking into account, with realism, the national context.
Equally important for the success of the ALFRED project is the national support at political, governmental and also local level. European governance based on the principles of transparency, participation, early involvement of stakeholders could be the best approach to implement an innovative system.
All the activities request a collaborative effort not only of the three partners who signed the FALCON consortium (ICN, ANSALDO and ENEA), but also the contribution of other research and education organisations in the region. All these efforts have been coagulated in the ARCADIA project, financed by the European Commission to support the assessment of ALFRED feasibility in Romania, addressing the elements mentioned above. The paper presents the objectives and activities proposed by the project, as well as the results expected from the 26 partners of the consortium.

MYRRHA Fuel Transient Tests Project at the TRIGA-ACPR Reactor

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Transient testing of MYRRHA fuel will be performed at the TRIGA-ACPR Reactor operated by the Institute for Nuclear Research, Pitesti. The goal of the project is to establish the fuel failure threshold (expressed in deposited energy) next to the development of new testing methodologies and experimental devices for the specific conditions of this Liquid Metal Fast Reactor fuel. The test will be performed within the framework of the European MAXSIMA project. The preliminary assessment shows that the energy deposition level required for possible cladding failure due to the Pellet Cladding Mechanical Interaction (PCMI) can be achieved during the pulse test. Based on the reactor physics assessment and the test fuel segment preliminary design, the conceptual design of a new irradiation rig has been completed. Additional neutron moderator materials are used to modify the neutron spectrum in order to achieve the required fission power density in the test segments. The design of the test rig and the TRIGA-ACPR pulse characteristics have been engineered in such a way that the power and temperature conditions simulate MYRRHA specific transients. Specific attention has been given to conditioning of the Lead-Bismuth Eutectic (LBE) used in the capsule. Results of preliminary safety assessment show that during the test reactor core safety is maintained, there are no effects on the population and environment and there are no unacceptable risks for workers.

I. NUCLEAR ENERGY

I.1. Nuclear Safety and Severe Accidents

I.1.1. Requirements and Challenges in Development of a New Integrated Severe Accident Simulations Code ROSHNI for PHWR Level 2 PSA Applications.

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Severe accident transients are typically evaluated using an integrated computer code that models thermal hydraulic and thermo-mechanical-chemical responses of the reactor core, its cooling systems and the interfacing safety and process systems including containment. Implicit is the code's ability to intelligently model all dominant severe core damage related phenomena and processes in an integrated manner with proper consideration of feedback between processes and systems and a detail that is consistent with the application of the code. The complex processes of modelling overall core degradation and disassembly, debris heatup and relocation, fission product releases and hydrogen production requires special models, intricate logic and computational acumen. Historically, the overall core representation has been typically undertaken with certain simplifications not necessary in the dedicated licensing safety analysis codes. Severe accident progression analyses for CANDU PHWRs have so far been undertaken using three main methodologies. Two MAAP based computer codes MAAP-CANDU in Canada and ISAAC in Korea were developed over 20 years ago with differing details of CANDU core specific models to interface with the MAAP code. In their application to the single unit CANDU 6 reactor they have consistently produced results that differ from each other considerably and an examination of the systems and phenomena they model reveals substantial weaknesses and un-necessary simplifications and assumptions contradictory to common wisdom. They have also failed to make even simple requisite adjustments to the MAAP code such as to use of heavy water properties instead of light water that MAAP codes were designed to model. Use of ISAAC has been limited to Korea while MAAP-CANDU has been used extensively as an Industry Standard Toolset in Canada and has also seen application in other countries. The third methodology has involved adaptation of codes like RELAP, MELCOR and non integrated Indian Code packages and has found limited and time consuming applications with results that capture even less CANDU specific phenomena. Of the three methodologies MAAAP-CANDU computer code has managed to capture more of CANDU specific severe accident related phenomenological issues. With consideration of the present arrangements of access to MAAP-CANDU, its ageing capabilities, arrested development and use of data not relevant to PHWRs, it is proposed that certain advanced users can benefit from a new code with a modelling approach that is more detailed than in the existing MAAP-CANDU code which has apparently not seen any fundamental improvements in CANDU specific modelling for over 20 years. A new computer code ROSHNI is under development drawing from the experience of earlier codes. Challenges in development of updated severe accident methodologies include recognition of importance of phenomena and accident progression pathways that the present methodologies miss but also inclusion of phenomena that have so far been poorly understood. A more detailed modelling of the horizontal PHWR reactor channels, fuel

bundles, end fittings and feeders with an advanced, more CANDU specific consideration of solid debris behaviour in the Calandria vessel is proposed in a new computer code destined also for an interface with a severe accident simulator useful for training the operators, plant personnel, regulators etc. It includes accident progression pathways and severe accident phenomena previously not considered. The paper details the challenges, modelling approach, describes the phenomenology and presents some results to illustrate code capabilities. It is expected that the source terms can then be evaluated with lesser uncertainty and with greater degree of control available to the analysts. Users are able to perform parametric and uncertainty analyses with greater ease and development of a CANDU severe accident simulator reaches a first important milestone.

I.1.2. Methodology for Time Window Analysis to Resolve HRA Issues

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One of the main tasks of every probabilistic safety assessment (PSA) is human reliability analysis (HRA). In order to be performed, the time available for the operator to accomplish a manual action must be calculated by quantitative evidences. In PSA terminology, the process is called time window analysis. Based on the complexity of an accident and availability of safety systems, minimum time window for the operator could be totally different for particular manual actions. Therefore for every sequence of the PSA event trees which requires human intervention, a separate deterministic calculation is needed. This is normally achieved as part of success criteria analysis that is thermo-hydraulics calculations in support of PSA. This paper explains a procedure for conducting time window calculations in compliance with ASME PRA standard. For the sake of demonstration, the methodology will be implemented on the process of feed and bleed through primary loop of a typical PWR. The feed and bleed process is manually actuated or initiated by the operator in the case of failure to remove decay heat through the secondary loop of a PWR. Here the available time window for the operator to initiate feed and bleed process will be illustrated. The results are summarized in graphs showing that the operator is able to mitigate the accident if his action is accomplished earlier than 1 hour from the accident onset. It will be also discussed how sensitivity analysis is carried out on the deterministic results to find out the operator's time window for the required sequences or sequences.

I.1.3. Regulatory Review of Emergency Operating Procedures and Severe Accident Management Guidelines

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The paper presents lessons learned from past review activities conducted by regulatory organizations in other countries to assess the implementation of emergency operating

procedures and severe accident management guidelines and gives an overview of the regulatory expectations in Romania in this field. A new regulation has been issued in 2014 - "Nuclear Safety Requirements on the preparedness of the response to transients, accidents and emergency situations for nuclear power plants". The new regulation provides requirements on: objectives, principles and factors to be taken into account for the response to transients, accidents and emergency situations on-site; transient and accident scenarios to be addressed in / covered by the EOPs; severe accident scenarios to be covered by the SAMGs; emergency situations to be covered by the on-site emergency response plan and emergency response procedures; establishment of the minimum number of staff with necessary qualifications to manage all scenarios required by the regulation; facilities and equipment to be available for accident management and on-site emergency response, including in situations caused by extreme external events; development and validation of procedures; documentation of the technical basis for the procedures; configuration management in relation to the procedures and systems credited for accident management and emergency response; training programmes and exercises; use of operational experience for the improvement of accident management and emergency response; records from exercises and from real events. Systematic regulatory reviews and inspections for assessing compliance with the new regulation are scheduled for 2014, supplementing the reviews performed in 2011 in the framework of the European "stress tests" post-Fukushima. The regulatory activities will include review of procedures, inspection of control rooms, secondary control areas and emergency control centre, participation full-scope simulator exams and to emergency response exercises, review of records from past exercises and monitoring of the implementation of the post-Fukushima action plan.

I.1.4. Thermo-Chemical Behavior of Disassembled CANDU Fuel Channels and Debris during a Severe Accident.

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Progression of a severe core damage accident in a CANDU reactor is expected to be significantly more complex than for a PWR. Disassembly of segments of horizontal fuel channels upon a sustained loss of cooling and depletion of heavy water coolant requires an addition loss of moderator that envelops the horizontal channels. Integrity of fuel channels in presence of an enveloping liquid moderator has been demonstrated as part of safety submissions in support of their licensing. An accelerated heatup of uncovered fuel channels upon additional degradation of external cooling in the moderator space is expected to not only lead to gross fuel bundle deformations but also gradual dissociation of individual fuel bundle lengths (along with associated pressure and Calandria tube segments) from the channel. This solid debris is expected to be initially formed at discreet locations and build up as more segments of underlying and overlying channels disassemble. These segments are termed suspended debris and their heatup, ensuing thermo-chemical deformations and fission product releases that are directed quickly into the containment are affected by a number of factors such as their location, composition, geometry, decay power levels, ingress of steam for metal water reaction, cooling by steam flow fields inside the vessel and melting of Zircaloy and its relocation, including flow of melt into the underlying water by the still submerged and intact channels. The motion and geometry of the debris is also affected as well by the proximity to and interactions with the large number of in-core devices. The evaluation of accident progression and generation of source terms that challenge containment integrity require a best effort modelling approach, including that at various locations and times for the suspended debris whose contribution to the source terms of deuterium and fission products can be significant. Debris interaction with underlying water by melt relocation and by energetic quenching upon their drop into the water below can determine pressure loads whose evaluation for calandria vessel integrity assessments is an important analytical step. Treatment of thermo-chemical and mechanical behaviour including motion and potential contribution to pull out of some underlying channels is an important CANDU severe accident progression phase that includes a number of distinctly different and often competing phenomena. The purpose of this research study is to identify the important phenomena, discuss their treatment in integrated codes used for severe accidents analysis and relevance of available experiments. A number of analytical treatments for the process of debris formation and the thermo- chemical and mechanical deformations are also discussed. The computational simulations of severe accidents need to become more realistic and need to be accelerated in wake of the Fukushima accident because one of the main lessons learned after this accident is related to the need of strengthening the safety culture that surrounds the field of evaluation of severe accident progression simulations. This is necessary in interest not only of a more realistic evaluation of accident consequences but also in support of recovery actions that are contemplated in Severe Accident Management guidelines for timely and well informed interventions that must be undertaken to reduce the risk that severe core damage accidents potentially pose. The work performed in this paper is in line with this lesson learned because it will allow pointing out some gaps in the analytical treatment of the phenomena highlighted in the study and create prerequisites for improving the CANDU severe accident simulation.

I.1.5. Numerical Investigation of Natural Circulation Flow in AP 1000 Passive Heat Removal System Using ACSLXtreme

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In the past few years a lot of interest was shown towards the implementation of passive systems in nuclear power plants (NPPs). These systems present a series of advantages over the active systems because they rely on natural forces and they don’t require an active source for initiation. Many advanced reactors use passive systems as a mean of cooling the reactor either for normal operation or in case of accidental conditions. AP1000 is an III+ generation reactor and employs the natural circulation concept, e.g. gravity, to enhance the NPP’s safety. Following a Station Black Out (SBO) the control rods are inserted in the core shutting down the reactor. After 2 minutes the steam generator water level decreases and activates the Passive Core Cooling System. Natural circulation occurs when the colder, denser water in the Passive Heat Exchanger flows by way of gravity down to the reactor vessel and removes the decay heat from the reactor core. In this present study, a mathematical model that investigates the thermal hydraulic characteristics of the natural circulation loop was developed using ASCLXtreme. The results have shown that the fuel temperature and mass flow rate are dependent on the vertical position of the heat sink relative to reactor pressure vessel.

I.1.6. Technical and Nuclear Safety Management during Cernavoda Units 3&4 Pre-Project Activities

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The decision making process for continuation of the construction works at Units 3 and 4 of the Cernavoda Nuclear Power Plant (Cernavoda NPP Units 3&4) has been and continues to be supported by the tasks of the technical and nuclear safety management. From these various tasks, some of them have been highly important and they were implemented with very good results in the pre-project phase of the Cernavoda NPP Units 3&4. Some good practices and lessons learned were retained following the implementation and completion of those tasks related to: the issuance of the European Commission viewpoint on the project in accordance with Article 43 of the Euratom Treaty (Romania being an European Union Member State), the acquisition of the technical and nuclear safety documentation in support of the development of the project, the issuance of a Letter of Comfort by the Romanian Regulatory Body - National Commission for Nuclear Activities Control based on an extensive and comprehensive package describing significant aspects for safety of the project, and the issuance of the Environmental Agreement based on an extensive environmental impact assessment process including national and transboundary (with neighbour countries) consultations. Several aspects on the good practices and lessons learned mentioned above are analysed and emphasized in this paper since they may worth to be considered in nuclear projects further developed in Romania.

I.1.7. Nuclear Safety Culture - Lessons Learned from Past Accidents

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The paper presents an overview of the lessons learned from past nuclear reactor accidents and their relevance for the development of nuclear safety culture. Since the introduction of the concept by the International Nuclear Safety Advisory Group (INSAG), many definitions proposed for safety culture over the years have been. The definition of safety culture used in this paper is that proposed by the International Atomic Energy Agency (IAEA): “the assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance”. Although the term "safety culture" emerged after the Chernobyl accident, the factors that contributed to earlier accidents, of which the most notable was the accident of Three Mile Island Unit 2, are also relevant for nuclear safety culture. As regards the Fukushima Daichii accident from 2011, safety culture was once again brought into discussion. As part of its safety guides on management systems for nuclear facilities and activities, the IAEA has provided a sound framework for the assessment of safety culture, based on a set of 37 attributes, grouped into 5 areas corresponding to safety culture characteristics. In this paper we will analyze the lessons learned from past nuclear accidents by making use of the 37 attributes of a strong safety culture, promoted by the IAEA.

I.1.8. Optimizing Maintenance Crediting Maintenance Rule: A Case Study on Safety Injection System

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Nuclear power plants (NPPs) routinely apply on-line maintenance (OLM) to improve plant reliability, safety and economic performance. According to Maintenance Rule (i.e. 10 CFR 50.65) before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the NPPs shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety. The key point is to assess and manage risk prior to taking risk-significant equipment out-of-service. Four principles govern optimization of planned maintenance with respect to nuclear risk: ensuring nuclear safety by understanding the impact of equipment unavailability; managing risk by limiting the duration of the unavailability of equipment; maximizing the efficiency and effectiveness of the plant staff and other resources by integrating risk-insights into the work management schedule, and identifying the impact of work by effectively communicating to the plant staff. In this study a framework for online maintenance optimization in nuclear power plants systems is proposed crediting the Maintenance Rule using the above principles. The Safety Injection System (SIS) of a third generation PWR is optimized as a case study. The importance of the Safety Injection System under study is significant in the total risk (Core Damage Frequency) of the plant, because SIS comprises both high pressure and low pressure injection capabilities by using only one pump in each train. SIS is modelled with Reliability Block Diagram (RBD) and simulation is completed by considering reliability and maintenance profiles of the system. The optimum availability of the system is estimated by studying different maintenance policies.

I.1.9. A Commentary on Success Criteria Analysis in Support of PSA

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Success criteria in the context of Probabilistic Safety Assessment (PSA) level 1 is defined as minimum requirements of the system for mitigation of an accident. The methodologies for success criteria analysis (SCA) are constructed on deterministic evaluation (based on PSA assumptions) of the system response to a postulated accident and probabilistic interpretation of the results for PSA needs. The PSA supportive thermo-hydraulics (TH) calculations play a crucial role in PSA of nuclear power plants and could affect the quantitative outcomes of PSA such as core damage frequency (CDF). Therefore SCA is an area where deterministic and probabilistic approaches meet. Success criterion is defined at system level as well as sequence level; hence we distinguish these two terms: “top event success criteria” and “accident sequence success criteria”. Although the methodologies are known for industrial and technical communities, no reference systematically explains the procedure for success criteria analysis. This paper is aimed to cover the topic for filling the gap in this area. Moreover, SCA methodologies of already performed PSAs will be critically reviewed and the technical challenges will be categorized. It is discussed in details how the quality of PSA could be affected, by taking into account the following four approaches that are believed to be misleading or technically erroneous and it is recommend to avoid them: Approach 1 - Use of expert judgments; Approach 2 - Use of design information of the systems; Approach 3 - Use of thermo-hydraulics calculation for Design Basis Accidents (DBA); Approach 4 - Neglecting the containment role in level-1 PSA. A framework for performing the analysis in compliance with ASME PRA standard is proposed. The main goal of this article is to critically review the available approaches; furthermore the methodology will be implemented on a typical PWR for the sake of demonstration.

I.1.10. Relationship between Security Culture and Safety Culture

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The operation of nuclear power plants (NPPs) requires careful attention to safety, security and safeguards. Safety is aimed at preventing accidents; security is aimed at preventing intentional acts that might harm the NPPs or result in the theft of nuclear materials; and safeguards are aimed at preventing the diversion of nuclear materials for nuclear weapons purposes. Although these activities have different goals, actions that are taken to further one activity can have implications for the others. While both nuclear safety and nuclear security consider the risk of inadvertent human error, nuclear security places additional emphasis on deliberate acts that are intended to cause harm. Because security deals with deliberate acts, security culture requires different attitudes and behaviour, such as confidentiality of information and efforts to deter from malicious acts, as compared with safety culture. Therefore, the main shared objective of security culture and safety culture is to limit the risk resulting from radioactive material and associated facilities. In this context, the paper presents the relationship between security culture and safety culture. There will be presented how safety and security cultures coexist and the need to reinforce each other because they share the common objective of limiting risk. A fully discussion about the need to strengthening nuclear safety worldwide (in light of lessons learned so far from the Fukushima Daiichi accident) and about the basic concepts and elements of nuclear security culture is the main purpose of this paper. A special attention will be given to the common objective of safety and security: the protection of the public and the environment with a typically reflection of a common philosophy of defense in depth.

I.1.11. Simulation of Molten Corium Concrete Interaction for the Reactor Case in Initial Wet Conditions

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In case of a hypothetical severe accident in a nuclear power plant, the corium consisting of the molten reactor core and internal structures may flow onto the concrete floor of containment building. This would cause an interaction between the molten corium and the concrete (Molten Corium Concrete Interaction - MCCI), in which the heat transfer from the hot melt to the concrete would cause the decomposition and the ablation of the concrete. The potential hazard of this interaction is the loss of integrity of the containment building and the release of fission products into the environment due to the possibility of a concrete foundation melt-through or containment over-pressurization by the gases produced from the decomposition of the concrete or by the inflammation of combustible gases. In the safety assessment of nuclear reactors, it is important to know the consequences of such a phenomenon. Considerable uncertainties on the MCCI knowledge still exist and concern the distribution of the heat flux at the pool interfaces, the transfer of heat and mass between the metal and oxide layers in a stratified configuration, and changes to the configuration of the pool. The purpose of this paper is to perform a set of sensitivity calculations using the MEDICIS-CPA modules of ASTEC code on a generic CANDU 6 reactor case. The calculations are carried out with different pool configuration models and different values of the parameters of the heat transfer model at the corium pool interfaces. As there is water in the reactor vault at the vessel failure time, the corium quenching is taken into account. The results show the significant impact on reactor predictions of the solidification temperature and of the pool configuration assumptions. Also, the heat transfer coefficient at the metal/oxide interface has an important impact.

I.1.12. A Parametric Study of Candu Fuel Channel Using RELAP5 Code

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The RELAP5 code has been extensively applied at the Pressurized Water Reactors (PWR) safety analysis, being one of the most used codes worldwide, including at the regulary analyses. The RELAP5 code is a flexible tool for computerized simulation as its approach allows to models as much as needed of a particular thermalhydraulic system, with use both for anticipated transients of nuclear power plants or research reactors, and also for small scale test facilities. Although the RELAP5 code was not extensively assessed for CANDU, the code's generalization, development and its capability to model any complex thermo hydraulic system enable its application to heavy water reactors. Supplementary, the diffusion of the code, the quality of the produced results and its flexibility in developing nodalization and continuous development and improvment makes RELAP5 suitable for this purpose. A parametric study of CANDU fuel channel is performed with RELAP5 code. The aim of this paper is to identify the applicability of RELAP5 code at CANDU reactors and to present the results obtained for simulation of a power transient

inside a CANDU fuel channel. The RELAP5 results agreed very well with the CATHENA results, which is the customary code for the CANDU system.
In general, the results presented on this paper, fall within the foreseeable expectations with reasonable trends.

I.1.13. INR Preliminary Activities for ALFRED Demonstrator PSA Level 1

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Probabilistic Safety Assessment (PSA) has been widely applied in the nuclear power field since the late 1970's and has become a mature tool to identify potential accident sequences, quantitatively estimate their frequencies of occurrence, and also probabilistically estimate the consequences associated with these postulated accidents. Together with deterministic methods, the PSA methodology has been recognized to demonstrate the safety of nuclear power plants (NPP) worldwide. Applied until recently after the finalization of NPP design or even after the plant was built and operated, PSA was mainly used as a way to measure the level of risk associated with plant's operation. Along with the development of evolutionary NPP (Generation III), the importance of using PSA in the design phase was recognized. In order to have an integrated framework for assessing risk and safety issues for use throughout the Generation IV (Gen IV) technology development cycle, an Integrated Safety Assessment Methodology (ISAM) for Gen IV nuclear systems has been elaborated by Gen IV International Forum's Risk and Safety Working Group (RSWG). For Gen IV nuclear systems, the ISAM is intended to support achievement of safety that is “built-in” rather than “added on” by influencing the direction of the concept and design development from its earliest stages. The ISAM consists of five distinct analytical tools (Qualitative Safety Features Review, Phenomena Identification and Ranking Table, Objective Provision Tree, Deterministic and Phenomenological Analyses and Probabilistic Safety Analysis) which are structured around the last one, PSA. Having in mind that PSA will play an important role in the three major phases of Gen IV technology development: design, licensing, and the operational stage, in Institute for Nuclear Research (INR) Pitesti, efforts have been made (and are still on-going) in order to develop a PSA level 1 study for Advanced Lead Fast Reactor European Demonstrator (ALFRED). The paper will focus on some preliminary activities for reaching this objective: the initiating events (IEs) taken into consideration, the development of “Total loss of flow – IE” event tree (ET) and also the development of Decay Heat Removal system fault trees, taking into account the ET's requirements. It is necessary to mention that the results of these activities embed large uncertainties especially due to lack of project's details, lack of reliability and experimental data, lack of deterministic analyses results, etc.

I.2. Nuclear Reactors and Gen IV

I.2.1. Reactor MARIA in Poland, its History and Future

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Multipurpose high flux research reactor MARIA is situated at Otwock near Warsaw and operated by National Centre for Nuclear Research (NCBJ). MARIA has been designed and constructed by Polish industry. It is a water and beryllium moderated, water cooled reactor of pool type with pressurized fuel channels containing concentric multi -tube assemblies of fuel elements. This reactor has been designed with a high degree of flexibility to provide possibilities of radioisotopes production, of physical and irradiation experiments. Operational power is 30 MW and thermal neutron flux in core is $4x10^{18}$ n/m^2s. First criticality of the MARIA reactor was reached in December 1974. After a few months it was gradually brought to full power. The reactor was in operation since 1975 until 1985. In July 1985 it was shut down for modernization. MARIA reactor was put again into operation in 1993. As the research reactor MARIA is a multipurpose reactor and remains only one high flux research reactor in Poland it has been designed to operate with several experimental devices in parallel. The most important of them are: vertical irradiation channels for radioisotope production; reactor test rigs for structural materials and reactor fuel studies under stationary conditions; horizontal experimental channels for neutron beam studies. The main reactor possibilities are as follows: radioisotopes production; neutron physics; neutron activation analysis; neutron radiography doping; boron neutron capture therapy. The present exploitation experience and future projects will be presented.

I.2.2. Stochastic Neutron Transport in Fluctuating Reactors with Pure-Triplet Scattering using RVT Technique

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The problems of neutrons transport in stochastic media have been intensively studied being considered of very high interest. These problems arise when the environmental properties of the background material of the reactors, with which the neutrons interact, are random functions of position and time. This randomness is in scales of, or longer than, the mean free paths of the transported neutrons. For instance, in Boiling Water Reactors, coolant density and temperature fluctuations can be considered responsible for introducing randomness in the flux by causing the fluctuations of the neutron mean free path. An application that motivated the present work is the transport of neutral atoms in tokamak plasmas. Such plasmas are known to be turbulent, and to a good approximation, the plasma density variations are well represented by a Gaussian stochastic process. In this work, a new algorithm is presented in order to solve a problem related to the continuous stochastic transport. It is mainly based on the Random Variable Transformation (RVT) technique, which is a powerful technique used to get the complete solution, represented by

the probability-density function (PDF) of the solution process, for the stochastic differential equation (SDE). The exact PDF of the solution process can be obtained using RVT technique if the exact deterministic solution of the SDE is calculated. This enables us to calculate any statistical quantity related to the solution. On the other hand, RVT together with some numerical deterministic methods can be used to solve some SDE’s to get a good approximate PDF of the solution process. Throughout the present paper, RVT technique is presented in details. In this algorithm, a simple integral transformation to the input stochastic process (the extinction function of the medium) is applied. This linear transformation enables us to rewrite the stochastic transport equations in terms of the optical random variable *x*. Then the differential equation is solved deterministically using the separation method to get a closed form for the solution as a function of *x*. This solution is used to obtain the PDF of the solution process applying the RVT technique among the input random variable (*x*) and the output process (the solution). Some numerical results for pure-triplet anisotropic scattering considering various distributions of the optical random variable, *x*, are finally presented.

I.2.3. Even-Odd Effects in Prompt Emission in Fission

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Compared to proton even-odd effects in fission fragments distributions, that were extensively studied, the even-odd effects in prompt emission received less attention. In this context the present paper is focusing on the even-odd effects (in both Z and N) appearing in prompt neutron and gamma ray emission. The even-odd effects in prompt emission were studied in the case of four even-even fissioning systems, namely: $^{233}U(n_{th},f)$, $^{235}U(n_{th},f)$, $^{239}Pu(n_{th},f)$ and $^{252}Cf(SF)$, respectively. The results of this study show that the even-odd effects are mainly due to the Z even-odd effects in fragment distributions and charge deviations and N even-odd effects in average neutron separation energies. The most pronounced even-odd effect is visible in prompt neutron multiplicity. Average multiplicity, as a function of Z, ν(Z), has a saw tooth shape and exhibits a visible staggering for asymmetric fragmentations. The size of the global even-odd effect in prompt neutron multiplicity shows a decrease when increasing the mass of fissioning nucleus. The N even-odd effect in average neutron separation energy often influences the global even-odd effects in prompt neutron multiplicity in an opposite sense comparatively with the Z even-odd effect. The average prompt neutron multiplicity as a function of TKE shows an increase of the Z even-odd effect when increasing TKE. The Z even-odd effects in the average prompt neutron energy and the average γ-ray energy have comparable sizes and are lower than those in prompt neutron multiplicity. Different average prompt emission quantities (as a function of A, of Z, of TKE and total averaged ones) - that are fragment distribution dependent- are affected by the even-odd effects in Y(A,Z,TKE). The primary multi-parametric matrices of prompt emission quantities (e.g. ν(A,Z,TKE))-that are not fragment distribution dependent- are affected by the N even-odd effect in neutron separation energy and by the pronounced Z even-odd effect in the charge deviation ΔZ(A).

I.2.4. Preliminary Neutronic Analysis of some Alloys if Used in Lead Cooled Fast Reactors

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The new generation of innovative nuclear systems requires structural materials able to withstand a harsh environment due to the intensity and the energy of the radiation, to the heavy liquid corrosion/erosion at high temperatures, etc. One of the most important factors in assessing potential materials for nuclear power applications is the material performance under neutron irradiation. During time a lot of stainless steels and alloys have been considered and investigated as candidates for GEN IV nuclear reactors and a large number of studies have been performed and reported regarding their behavior under different neutron spectra and irradiation temperatures. The objective of the present work is to provide a preliminary analysis of the impact of using Molybdenum (TZM, Mo-Re5% and Mo-Re47.2%) and Vanadium (V-4Cr-4Ti, V-4Cr-4Ti-0.15Y and V-4Cr-4Ti-0.5Y) based alloys as fuel cladding materials on some neutronic parameters of a lead cooled fast reactor type demonstrator (ALFRED). The quantities envisaged are the following: multiplication factors, neutron fluxes, heat deposition as well as the isotopic inventory of these alloys at various irradiation and cooling steps. The evaluations are performed using the Monte Carlo code MCNPX 2.6.0 (and the JEFF 3.1 neutron cross section libraries) for the actual, detailed three-dimensional geometry of the ALFRED reactor. It should be mentioned that the reference active core configuration was developed at ENEA-Bologna in the frame of the FP7-LEADER Project (WP2 Core Design). The FISPACT code (EASY-2005 code system) is also used in order to provide information about the isotopic inventory and the hazard factors. It is well recognized that the simulation of the isotopic composition with the irradiation provide useful information about the possible crystallographic changes that could lead to modifications of the material properties. The isotope inventories, evaluated at various irradiation and cooling time steps, could be also used as input for further analyses regarding the radioprotection or waste characterization and disposal

I.2.5. Estimation of Spent Fuel Parameters Evolution during Thorium-Based Fuels Irradiation in CANDU Reactors

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The paper presents some preliminary analyses on Thorium/Uranium (Th, U)O_2, and Thorium/Plutonium, (Th, Pu)O_2, mixed oxides fuels behaviour to irradiation in CANDU reactors by means of the estimated spent fuel characteristic parameters (radioactivity, thermal power and gamma energy, respectively) evolution with fuel burn-up. Above mentioned analyses have been performed considering the CANDU type fuel bundle with 43 fuel elements, SEU43 (developed in Institute for Nuclear Research Pitesti) with the following fuel compositions: Th_SEU – ThO_2 in the thick fuel elements (central element + 7 inner elements) and Slight Enriched Uranium, SEU (1,8 wt% enrichment in ^{235}U) in the thin fuel elements (14 intermediate elements + 21 outer elements); Th_Pu – ThO_2 + PuO_2, with

3%, 4% and 5%, respectively, PuO_2 in the fuel bundle (all the 43 fuel elements of the bundle contain the same fuel composition); Th_Pu_Mix – ThO_2 + 7% Dy in the central element and the 7 inner fuel elements, ThO_2 + 4% PuO_2 in the 14 intermediate fuel elements and ThO_2 + 2% PuO_2 in the 21 outer fuel elements, respectively. The considered Th-based fuels were irradiated in CANDU reactor specific conditions in order to reach 20 MWd/kgHE (Heavy Element) fuel burn-up, in steps of 2 MWd/kgHE, the specific powers for the working cases being 32 kW/kgHE (Th_SEU case) and 37.7 kW/kgHE (Th_Pu and Th_Pu_Mix cases), respectively. The fuel burn-up was simulated using ORIGEN-S code, included in SCALE6 programs package, developed by Oak Ridge National Laboratory. The spectral neutron cross-sections weighting factors (used as input data for fuel burn-up simulation in CANDU reactor) were given by DRAGON3.05E code lattice cell calculations. For the considered Th-based fuels compositions spent fuel characteristic parameters values were estimated and their evolution with the fuel burn-up have been presented comparatively, emphasizing on the actinides and fission products contribution to the total value of the interest parameter.

I.2.6. The Relationship between Experimental Data and Fuel Modeling in the Fuel Behaviour Analysis

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The paper highlights the importance of a good knowledge of the fabrication data and operating conditions on the fuel behaviour simulation during irradiation experiments with the Fuel Performance codes. In this aim, TRANSURANUS fuel performance code was used to simulate the CANDU fuel elements behaviour during irradiation experiments FIO118 and FIO119. TRANSURANUS is a very complex code, developed by the Institute for Transuranium Elements (ITU), Germany, and was adapted at the Institute for Nuclear Research (INR), Nuclear Fuel Performances Group, to simulate the CANDU nuclear fuel behaviour. The experiments FIO118 and FIO119 are described in details in the technical documentation from the International Fuel Performance Experiments (IFPE) database and were performed in order to investigate the effect of various design and operating parameters on the dimensional response of CANDU fuel elements during irradiation. In the present work, the parameters of the fission gas models and those of the densification model along with some of the irradiation conditions (fast neutron flux and coolant temperature and pressure) were varied in order to investigate their influence on the code results. Simultaneously, the code results were analysed in correlation with the experimental measurements, as follows: dimensional, during the test; for microstructure and fission gas volume in the post irradiation examinations. The paper results point out that a good simulation with a fuel performance code implies a good knowledge of the code abilities in order to adjust appropriately the models options for each case analysed correspondingly to the microstructural particularities of the fuel along with the specific irradiation conditions

I.2.7. The Influence of CANDU Fuel Bundle Central Element Size on Lattice and Core Reactor Physics Parameters

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The development of advanced fuel designs for existing and future CANDU reactors is a permanent research task in the Reactor Physics, Nuclear Safety and Nuclear Fuel Performances Department of INR Pitesti. Nowadays, a new CANDU fuel design has been proposed by Nuclear Fuel Performances Group, namely the 37M fuel bundle design. It is based on the actual 37-rods CANDU standard bundle design, the main difference being the central fuel element which is thinner by about 12% than the CANDU standard one. To find out the behaviour of 37M bundle design placed in a CANDU power reactor, lattice and core calculations have been performed using the WIMS and DIREN computer codes, respectively. The main neutron physics parameters of interest pursued during calculations were: k-infinity, power peaking factors (PPFs), coolant void reactivity (CVR), channel and bundle power distributions. The lattice results showed a slightly better flattening of ring power distribution for 37M bundle design face to the 37-rods bundle one, while the time average core calculations led to extremely close values of integral parameters for both fuel bundle designs involved in analysis. Peculiar behaviour has been discovered after core-follow simulations, when 37M fuel bundle design led to a bit smaller Uranium consumption in the same 710 days reference burnup period. The paper concludes that in some specific conditions, less Uranium per fuel bundle can supply a better average fuel burnup if an adequate fuel management strategy is pursued.

I.2.8. Trainings in support of the development of Generation IV nuclear reactors

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In order to work towards a safe and efficient nuclear power industry as well as to support the development of new nuclear concepts, many variables has to be taken into account. In particular, decision makers need to consider the actual state of the nuclear society and make long-term commitments for the growth of new generations.
The establishment and confirmation of good scientific and technological infrastructures is of paramount importance. Moreover, training and education programs covering all the phases of reactors building, operation, dismantling have a key role for the formation of professional management, financial experts and operators.
In this aim, the Centrum výzkumu Rez, CVR, is working towards the establishment of infrastructures and training programs in support of the development of a safe Gen IV culture.
The centre has two research nuclear reactors, and is building new hot cells and experimental loops covering applications in a wide range of technologies. In particular, the research reactor LR-0 (zero-power light-water reactor) gives the opportunity to train young scientists. Currently, in the reactor, several types of courses have been developed in several areas: reactor physics, gamma spectroscopy, neutron spectroscopy, nuclear safety and heavy liquid metals technology.

In general, the courses are structured in order to give to the trainee a theoretical background and also a possibility for practical interaction with the facilities.
Moreover, in the frame of the project SUSEN (Sustainable Energy), there are currently being built several technological loops for research applied to GenIV nuclear reactors and fusion. This unique infrastructure will provide tools for new and innovative research and will enable also to design new topics for training of young engineers for the future of the safety culture.

I.2.9. Application of Computer Code KENO to the Assessment of Nuclear Fuels Criticality

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The assessment of fresh nuclear fuel criticality is required either to establish the fuel capacity to achieve the necessary criticality for the nuclear reactor operation, or for analyzes that should demonstrate the ensuring of nuclear safety in fresh fuel transport, handling and storage stages before loading it in the reactor core. Currently, there is a global growing interest for Small and Medium Sized Nuclear Reactors, developed based on classical typologies (PWR, BWR, HWR, GCR, MRL). At the same time, there is a constant interest for getting the alternative nuclear fuel based on thorium for the reactor types that are already in operation. CITON Safety Department has to solve some questions about criticality safety assessment arisen in the practical case of the manufacturing of CANDU fuel for Cernavoda NPP. This paper aims to analyze the achievement of criticality using the computer code KENO-V.a for a few different types of fresh nuclear fuel. For a first step, the criticality of fresh fuel configurations, PWR type, for Modular Advanced Reactor with Integrated Primary System (SMART) and CERMET type for a fixed bed nuclear reactor (FBNR) has been assessed. Then, the criticality calculations were performed for two fresh fuel types based on Thorium, proposed as feasible alternatives to nuclear fuel CANDU type with 37 elements (Th with up to 5% U-233 enrichment) and to the changed nuclear fuel type LWR or PWR (based on Th, with addition of Pu-239 and Pu-241). Finally, the nuclear material based on natural Uranium, in a few possible configurations during the fabrication process, was analyzed in terms of criticality. The obtained results showed that the analyzed nuclear fuels have the capability to achieve criticality when they are inserted in the reactor core. In the case of the fresh fuel based on natural Uranium, one has been demonstrated the safely handling in the precursory stages to its loading the reactor. As a conclusion, the results obtained in the described cases shown that the computer code KENO is an effective tool for calculating nuclear fuel criticality configurations.

I.2.10. FireWater Pumping Station From NPP Cernavoda - Transient Analysis

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In this paper we analysed the operating of the Firefighting Water Pump Station (SPAI), both on transient and stationary regimes. The station was tested to variable flow rate

required for various fire overlapping scenarios in the Nuclear Power Plant (NPP). In the past, there have been performed analyses for this system only considering the stationary regimes. The present study is the first transient analysis performed for the Cernavoda NPP SPAI. Moreover, in this paper we proposed some changes to be made in the SPAI configuration in order to get a more reliable pump station. For the SPAI one had considered three fire zones, namely: inside fire (10 l/s on a period of 10 minutes); outside fire (25 l/s on a period of 3 hours); cable/trafo station fire (120 l/s on a period of 40 minutes). Supplementary, a scenario assuming the maximum overlapping of all three types of fire was considered, having as result a maximum required flow rate of 155 l/s as maximum load for SPAI. In the paper, 7 transient regimes and 4 stationary operating regimes have been analyzed, as follows:

• *Transient regimes* - Regime 1) Outside fire; Regime 2) Outside and inside fire; Regime 2') Outside and inside fire (Pump P504 in stand-by); Regime 3) Outside, inside and trafo fire; Regime 3') Outside, inside and trafo fire (Pumps P504 and P508 in stand-by); Regime 4) Outside, inside and trafo fire with return to the initial state; Regime 4') Outside, inside and trafo fire with return to initial state (Pumps P504 and P508 in stand-by); • *Stationary regimes* – Regime 5) Outside fire (10 l/s) - changed SPAI; Regime 6) Inside fire (25 l/s) - changed SPAI; Regime 7) Trafo fire (120 l/s) - changed SPAI ; Regime 8) Inside, outside and trafo fire (10 l/s, 25 l/s and 120 l/s, respectively) - changed SPAI. The conclusions accomplished at the end of this analysis sustain that the SPAI should be changed in order to get more reliable pumps in the system and to simplify the operation of the station by reducing the number of pumps from 9 old type pumps to 2 modern, variable speed pumps capable to cover all design regimes.

I.3. Nuclear Technology and Materials

I.3.1. Heavy Liquid Metals: experience and lessons learnt

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Heavy Liquid Metals (HLM) such as Pb band PbBi eutectic are considered as reactor core coolant for fast reactors and spallation targets in Generation IV reactors. However, the use of HLM poses a big challenge in terms of selection of structural materials. In fact, beside the high irradiation doses expected in the Gen-IV fast reactors, the interaction between steels and HLMs is a critical issue in terms of aggressiveness of the environment towards the structural materials. Issues such as corrosion and degradation of mechanical properties have been studied for several years and different materials have been proposed.
The work carried out in the CVR laboratories, covered a wide range of materials and their study from the corrosion and mechanical properties point of view. Moreover, over the years, the technology for oxygen monitoring has been developed and implemented in all the facilities. This is an important tool to ascertain that in the LM there is a certain amount of oxygen dissolved and, thus, structural materials are able to develop protective oxides.
By testing materials both in Pb and PbBi many similarities and differences were observed. Overall, although from the technical point of view, working with PbBi is much easier, due to the lower melting temperature, Pb was observed to be far less aggressive towards the materials. In addition, dosing and monitoring of the oxygen content was also observed to be more stable in Pb, suggesting that the chemistry of the 2 LM is also different.
This work reports the main results obtained in both environments, highlighting the main similarities and differences

I.3.2. Current Understanding of Polonium Evaporation from Irradiated Lead-Bismuth Eutectic.

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Understanding polonium evaporation from lead-bismuth eutectic (LBE) is required for the design of nuclear installations that use liquid LBE as coolant or spallation target. In LBE, substantial amounts of polonium are formed by neutron capture on bismuth. Also in nuclear systems cooled with lead, polonium will be formed, by activation of bismuth impurities. In lead-cooled systems the amount of polonium will be much smaller than in LBE, but quantification of its release remains of importance, especially because lead-cooled systems are foreseen to operate at higher temperature. In the present report we give an overview of the research on polonium release from LBE that was conducted at

SCK-CEN during the last two years in the frame of the MYRRHA accelerator driven system project. Our work has demonstrated that polonium release by evaporation has a much richer phenomenology than initially anticipated. Roughly, one can distinguish between two regimes. At elevated temperatures, typically above 600 °C, the equilibrium between dissolved and evaporated polonium is well-described by Henry's law and a temperature correlation for Henry's law constant which has been independently established by several research groups including our own. The combined results suggest that in this high temperature regime polonium evaporation is rather insensitive to hydrogen and water vapor with concentrations up to a few percents in argon cover gas. At temperatures below *ca.* 600 °C, polonium release is more complex. Experiments at the Paul Scherrer Institute, which were confirmed by us, revealed that below 600 °C polonium evaporates much more than expected on the basis of the high-temperature behavior. Recently we conducted time-dependent polonium release experiments which provided insight in the increased polonium volatility below 600 °C. The results indicated that volatile polonium is formed at the free surface of the LBE and that the extent of its formation strongly depends on the concentration of hydrogen and water vapor in the cover gas. These observations of increased polonium volatility under conditions at which heavy liquid metal based facilities are foreseen to operate, are of particular interest for the design of safety systems and the licensing of such installations.

I.3.3. Wetting Parameter Determination for Ultrasonic Sensors in Heavy Liquid Metals

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The thermo-physical properties of Heavy Liquid Metals (HLM), like lead or its alloys (Lead Bismuth Eutectic), makes them attractive as coolant candidates in advanced nuclear systems. The opaqueness, that is common to all liquid metals, disables all optical methods. Therefore visual inspection is done using ultrasonic waves. The transducers used to send and receive the ultrasonic waves are specially designed for use in harsh conditions (high temperature 200°C - 500°C, corrosive nature of lead or LBE and strong gamma radiation). The piezoelectric elements are bonded to a thin protection layer or a wave guide made of stainless steel, what enables to obtain good sensitivity and bandwidth in liquids. Experiments show that HLM wetting of stainless steel surfaces is worse than in water. Thus, acoustic contact between transducer and liquid metal is not perfectly and the transmission and reflection of ultrasonic waves are not good. So, liquid metal wetting of stainless steels surfaces play a major role in the ultrasonic waves pass through interface. Material and surface conditions of “protector” in front of piezo are crucial to get a good acoustic contact between transducer and liquid metal. In order to solve the issue of initial wetting and have a good acoustic contact between ultrasonic transducer and liquid metal is needed to remove the oxides on the surface of the transducer. However, the long term wetting behaviour in an operational system is still unclear. Thus, it is necessary to introduce a correction factor, defined as "wetting parameter", for acoustic parameters calculation. This report shows an ultrasonic measurement technique for wetting parameter determination on the surface of the transducer, and the experimental results got in lead bismuth eutectic at 200^{o}C temperature.

I.3.4. Therapy Proton Beam Generation by RPA Mechanism

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Among all the types of corpuscular and photon radiations, hadrons (protons, antiprotons, ions, mesons etc.) show the most attractive distribution of the in-depth absorbed dose for to be used in radiotherapy. Starting from that advantage, at present there are medical charged particle accelerators, such as cyclotrons, synchrotrons and linacs, employed for the treatment of malign tumours. The accomplishment of the 10 PW APOLLON laser systems on Magurele Platform has induced the idea of its application for the elaboration of some proton radio-therapeutic methods, as an alternative to the conventional proton accelerators based on radio-frequency. The present paper presents a way to generate the proton beams with energies ranging between 50 MeV – 250 MeV and intensity of 10^{11}protons by means of the acceleration system based on the laser radiation pressure mechanism (RPA). According to such a mechanism, the energy of the laser beam photons is transformed into proton kinetic energy using the laser radiation pressure, P [Gbar] = $0.67 \times 10^{-18} \cdot I_0$ [W/cm^2], where I_0 is the laser radiation intensity. By the use of such a mechanism in the "Hole Boring" (HB) regime, the paper presents the results of an evaluation of the main physical parameters of the proton beams, laser beams and for a proton target

I.3.5. Assessment of Oxidation in Supercritical Water of Alloys with Different Chromium Content

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The Supercritical Water-Cooled Reactor (SCWR) was selected as one of six candidate reactor types by the Generation IV International Forum (GIF). This type of reactor operates in severe environments with a temperature range from 290 ^{o}C to higher than 600˚C and high doses of neutron irradiation. Corrosion is one of the most critical issues for designing internal components for this type of reactor. Owing to the good corrosion resistance in conventional nuclear reactors, ferritic–martensitic steels, austenitic stainless steels, Ni-base alloys and oxide dispersion strengthened (ODS) alloys are considered as promising candidate materials for SCWRs and must be investigated to evaluate their applicability in this aggressive environment. As in other high temperature environments, corrosion resistance in supercritical water is given by quality of the oxide layers developed on the materials surface. Oxidation, the most encountered form of high-temperature water corrosion is not always detrimental, since most heat resistant alloys develop a protective oxide film which provides resistance to corrosion. The aim of high-temperature alloys is to avoid excessive metal loss by scale formation (external oxidation) and to minimise the penetration of oxidation products into the alloy (internal oxidation). This paper presents a preliminary study regarding corrosion behaviour of two austenitic stainless steels (304L

and 310S) and two Ni based alloys (Incoloy 800HT and Inconel 718) immersed in supercritical water at 550 - 600ºC up to 1200h. After exposure to supercritical water were evaluated the weight changes using gravimetry, oxide film thicknesses and chemical distribution of the elements in oxides using scanning electron microscopy/energy dispersive X-ray spectroscopy (SEM/EDS). The weight change decreased with Cr content, at both temperatures, for all investigated alloys. After testing, it was determined that some alloys undergo internal oxidation, other show pitting corrosion, while others develop secondary phases (sigma phase).

I.3.6. Mechanical Properties of Two Austenitic and Ferritic/Martensitic Steels Strenghthened with Dispersed Oxides

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ODS Steels development for structural applications in nuclear reactors for fusion and generation IV for fission requires an advanced characterization, mechanical behaviour assessment at high temperature and high dose conditions, extended operation period, coolant compatibility and qualification for safety assessment. In order to achieve these objectives it is necessary to develop and qualify new materials for key structural components. This study prove differences between tensile properties of two ODS steels, austenitic and F/M types, strengthened with yttrium and titanium oxides dispersion. The unirradiated alloys have been austenitic 304ODS steel, ferritic/martensitic 16Cr-ODS steel and SA 270 grade 304L austenitic stainless steel. Both ODS materials have been manufactured by same processes, and the austenitic ODS steel experienced in addition an annealing process. At temperatures from RT until 700^{o}C, tensile tests have been performed. The results shown obvious strength advantage of ODS steels at higher temperatures. Strain to fracture has been recorded at 55% value at RT and tends to 40% at 600^{o}C in the case of 304L, while a decrease with temperature from 40% to 20% is displayed in the case 304ODS. For 16Cr-ODS steel the strain to fracture behaviour is opposite; an increasing with temperature was recorded from a minimum of 6% at RT up to 20% at 700^{o}C. The reduced ductility was noted for ferritic/martensitic ODS steel. The paper presents the fracture surface features examined by scanning electron microscopy.

I.3.7. Detritiation of Heavy Water by Catalysed Isotopic Exchange in Liquid Phase

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During the operation of CANDU reactors the radioactivity of heavy water used as moderator and cooling agent increase due to neutron activation and the tritium removal it’s strongly requested by safety and environmental reasons. Due to its high separation factor, the catalytic hydrogen isotopic exchange process (LPCE) between hydrogen gas and liquid water is one of the most desirable processes of tritium separation from gaseous

and liquid effluents produced in nuclear reactors. The author's extensive hands-on experience and the thorough literature review provide the foundation for this paper which is an updated overview of the R&D work related to the LPCE development and its application to deuterium and tritium separation technologies. During LPCE process, tritium from tritiated water is transferred to hydrogen gas by two successive processes. The first is in fact a conventional water distillation process that requires the presence of an efficient contact element (hydrophilic packing). The second , isotopic exchange possible only in the presence of a hydrophobic catalyst that repel liquid water, but allows both gases (H_2) and vapours to reach the active catalytic centres and / thus speeding up the isotopic transfer process. This hydrophilic packing-catalyst mixture labelled as "mixed catalytic packing" is usually used with in LPCE columns under various structures, geometries and ratios. Based on the LPCE process several countries (i.e. Canada, Korea and Romania) developed their own tritium removal facilities while other countries are in the process of developing and implementing similar technologies. The paper presents current status of activities developed on the world and at ICIT for developing a suitable mixed catalytic packing to promote LPCE process. The main goal are: to setup and develop a database to design the best suited mixed catalytic packing; to evaluate and propose new ways to improve the LPCE process efficiency. Current status of the activities concerning the implementing the LPCE process at Cernavoda NPP it is also presented. The authors propose several conclusions and general recommendations aimed at the selection of the best suited mixed catalytic packing and operating parameters to improve of the overall efficiency of the LPCE process.

I.3.8. Measurements of Fatigue Initiation of Carbon Steel in High Temperature Water Using Blunt Notch Compact Tension Specimens

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The US Nuclear Regulatory Commission (USNRC) has recently issued guidelines (NUREG-CR/6909) for the prediction of fatigue endurance of steel in high temperature water under conditions typical of those in Pressurised Heavy and Light Water Reactors, and Boiling Water Reactors. The work reported in this paper compares the guidelines provided in that document with data from experiments on blunt notch compact tension specimens fabricated from moderate sulfur carbon steel. The specimens were subjected to cyclical loads at a load ratio of 0.05 in oxygenated lithiated water at 300ºC or in air. The initiation, coalescence and growth of the resultant fatigue cracks was monitored in real-time using the direct current potential drop technique. Steady state long crack growth rate measurements were in good agreement with literature values. In addition, it was concluded that the fatigue endurance (25% load drop condition) referenced in the USNRC document is equivalent to a crack advance from a plane surface of 300-500µm. In general the trends observed for the number of cycles to achieve this length of crack were consistent with the predictions of NUREG-CR/6909 for triangular load-forms. The study was further extended to consider the effect of triangular loading with hold periods. When the hold was imposed at the point of minimum load, there was found to be no effect on the number of cycles to initiate a 300µm crack. However, when the hold was imposed at the point of maximum load, the number of cycles to achieve crack initiation was significantly

increased. This finding could have a profound effect on the calculation of fatigue usage factors which are used in Safety Justifications for nuclear plant.

I.3.9. Application of Neural Network Model for in-reactor deformation of CANDU Zr-2.5Nb pressure tubes

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The parametric models have been successfully applied for creep phenomena for CANDU Zr-2.5%Nb pressure tubes. However, they not yet succeeded to deal with large number of variables. The neural network (NN) method is a more modern technique which is in a full progress and complexity. The NN method has now been demonstrated to be superior in many cases in the extrapolation and representative experimental data, also for the creep data. The Neural Networks methodology could be applicable in virtually every situation in which a relationship between the predictor variables (independents-inputs) and predicted variables (dependents-outputs) is requested, even when that relationship is very complex.
In the function approximation two useful neural networks are the Multilayer Feedforward Layer (MFL) and Radial Basis Function (RBF) networks. The MFL networks is perhaps the most popular network architecture in use today. In the MFL the units each perform a biased weighted sum of their inputs and pass this activation level through a transfer function. As result it gets their output, and the units are set in a layered feedforward topology. The layered feedforward topology is a specific connection structure of a neural network, where neurons of one neuron layer may only have connections to neurons of other layers. The network thus has a simple interpretation as a form of input-output model, with the weights and thresholds (biases) the free parameters of the model. Such networks can model functions of various complexities, with the number of layers, and the number of units in each layer.
The Multilayered Feedforward Neural Networks (MFNNs) are an important class of the neural networks. This type of networks consist of a set source node that constitute the input, one or more hidden layers of computation nodes and an output layer of computation nodes. The inputs are processed through the network in a forward direction, on a layer-by-layer basis. This architecture of the Multilayer Feedforward Network will be used for creep assessment problem, for CANDU Zr 2.5%Nb pressure tube.
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I.3.10. Non-Isothermal Decomposition of Titanium Hydride Powder in Air

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Titanium hydride has attracted wide attention not only because it is a potential hydrogen storage material, but also it has many applications in other fields: source of pure hydrogen, to synthesize other hydrides, the manufacture of metal foams etc. All these applications are dependent on the decomposition of TiH_2 or the release of hydrogen. The thermal decomposition behaviour of titanium hydride powder and hydrogen release during the heating process was analysed in air atmosphere, by thermo-gravimetric analysis (TGA). Using this method, a fairly accurate indication of when hydrogen starts to evolve can be made. Heating was performed in the temperature range of 25-1200°C in air at different heating rates. The TGA curves were obtained at heating rates of 5, 10 and 20°C/min. The titanium hydride powder used in this study was obtained by means of hydriding of titanium powder (99.4%). In accordance with the results the TGA curves for titanium hydride powder show, in all cases, a similar trend. An increase in mass loss with increasing temperature, after which a steady increase in mass is observed. As a result, the hydrogen release mechanism does not change with increase in heating rates. The kinetic of hydrogen release related to the hydride composition is analysed and discussed.

I.3.11. Average Diffusion Coefficient Measurement of the Oxygen in Zy-4 Cladding, at Temperatures between 873 and 1673 K

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The main purpose of this paper is to present the results obtained for diffusion coefficient measurement of the oxygen in Zy-4 cladding material, from the initial gradient of the sorption curves. Because the oxidation kinetic of the zirconium alloys are mainly governed by the diffusion, the curves of absorption can be directly related to the diffusion coefficient. If the initial gradient of the sorption curves give the average value of the diffusion coefficient, in the later stage the influence of the tensile stresses at oxide – metal interface, the evolution of the surface of this and the porosity development in the oxide layer will strongly affect the diffusion. Zy-4 cladding samples were oxidized in dynamic oxygen atmosphere at 16ml/min flow rate and 1bar pressure, under isothermal conditions, at temperatures between 873K and 1673 K. With the average values of the diffusion coefficients obtained from the initial gradient of the sorption curves plotted against the square root of time , the temperature dependence of the diffusion coefficient were obtained , both from alfa and beta structures of the zirconium. The results obtained, related also from the structural changes, the oxide layer thickness, the contributions of the tensile stresses and the surface of diffusion are presented and discussed.

I.3.12. Influence of the Samples Preparation on Hydrogen Concentration Determination Using the Differential Scanning Calorimetry Method

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Hydrogen present in zirconium alloys has a strong effect of weakening. Zirconium is known for its affinity for hydrogen. 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. 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. 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. In this paper, it is analyzed how the sample preparation method influences the determination of the concentration of hydrogen. The paper presents the results obtained on samples of pressure tube with masses between 5 and 50 mg. There were analyzed samples with different forms such as 0.1 mm thin foils of and rectangular pieces. It was also watched how the results are influenced for a sample consisting of several stacked thin foils.

I.3.13. Fabrication of Metallic Mo Targets for Technetium Generator

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Technetium-99 isotope is an important metastable element used in nuclear medicine for fast diagnosis. The ^{99m}Tc isotope generation takes place by the decay of ^{99} Mo from irradiated targets. The purpose of this work was to fabricate sintered metallic molybdenum targets for technetium generator and to characterize them. The sintered metallic Mo targets will be irradiated in ICN TRIGA reactor to obtain technetium-99 radionuclide. Pure molybdenum wastes were dissolved in nitric acid to obtain molybdenum trioxide (MoO_3) for the fabrication of metallic molybdenum targets. The molybdenum trioxide was conditioned by washing and calcination. The reduction of molybdenum trioxide to metallic molybdenum is done stepwise in a reducing hydrogen atmosphere in a controlled atmosphere furnace. The reduction of MoO_3 to MoO_2 was done in hydrogen atmosphere at a temperature of 500ºC for 2h. The metallic Mo powder was obtained by reduction of MoO_2 in H_2 atmosphere at a temperature of 900ºC for 2h. The characteristics of the powders are very important in the sintering process, so after each reduction process the obtained powders were analysed by size distribution of particles, X-ray diffraction (XRD) and scanning electron microscopy (SEM). The Mo targets were obtained by pressing and then sintering of the green compacts in a gas controlled furnace at 1650ºC for 2h in dynamic atmosphere of hydrogen. The sintered Mo targets were characterized by density and dimensional measurements, X-ray diffraction and scanning electron microscopy.

I.3.14. Production of ^{99}Mo using LEU targets at INR Pitesti

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Technetium-99m (6.02 h – half-life) is and promises to remain the most widely used radioisotope in nuclear medicine, being used in more than 80% of all diagnostic nuclear medicine procedures. Technetinm-99m is almost exclusively produced from the decay of its (66-h – half-life) parent ^{99}Mo which is produced at present mainly by nuclear fission of ^{235}U. In the last years Institute for Nuclear research (INR) Pitesti was involved in various international collaborations on this topic, including the IAEA Coordinated Research Project (CRP) named "Developing techniques for small scale indigenous Molybdenum 99 production using Low Enriched Uranium fission or neutron activation" (2007-2012). In this context an experimental process for the production of sodium molybdate, based on a modified version of the Cintichem process, was carried out in the fume hood of the Post-Irradiation Examination Laboratory of INR Pitesti. The work we present here shows recent experimental results on testing and improving the modified Cintichem process. The most sensitive aspects of the Cintichem process are ^{99}Mo recovery and purification by its precipitation with α-benzoin oxide. The final sodium molybdate solution was found to contain no other gamma emitter than ^{99}Mo. Based on the activities of ^{99}Mo measured for each step of the process the yields were calculated for each of the steps of the process and for the whole process.

I.3.15. Experimental Study on the Sintering Behavior of Powder Mixtures of Triuranium Octoxide in Hyperstoichiometric Uranium Dioxide

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The purpose of this study was to obtain from powder mixtures sintered pellets that meet the following requirements: to have a stoichiometric ratio O / U (Oxygen/Uranium) closer to uranium dioxide (UO_2), the height diameter ratio (H / D) greater than 1, a sintered density more than 95% of the theoretical density (TD) uranium dioxide (10.96g/cm^3). The uranium dioxides was obtained from ADU (ammonium diuranate) by calcination in air and reduction in hydrogen atmosphere. By using waste powders of this type of $UO_{2,}$ mixed with higher uranium oxides obtained by calcination at 665^{0}C temperature these requirements have been met. The powder mixtures were 5%, 10%, 20%, 30%, 40%, 50% of the higher oxides of uranium in hiperstoichiometric uranium dioxide. The started density of green pellets had been more than 45% theoretical density. The proposed sintering diagram leads to fulfillment of these requirements. Metallographic analysis shows that sintered pellets obtained from 20%, 30%, 40%, 50% mixed powders has uniform dimensional distribution and homogeneous shapes.

I.3.16. Cernavoda Nuclear Power Plant Assessment Techniques for Pressure Tubes Degradation Mechanisms

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CANDU which stands for CANada Deuterium Uranium is a nuclear reactor designed by the Candu Energy Inc. the former AECL (Atomic Energy of Canada Limited). It uses natural uranium as fuel and heavy water as both cooling agent and moderator. This type of nuclear reactor is a special one because it doesn`t have a pressure vessel (like the majority of reactors) instead it has 380 fuel channels also known as CANDU Pressure Tubes (PT`s). The PT`s are critical components of this type of nuclear reactor and we have a special inspection program for our degradation mechanisms called fitness-for-service inspections. There are a series of possible degradation mechanisms: delayed hydride cracking (DHC), irradiation enhanced deformation (creep), corrosion, deuterium ingress and changes in material properties: reduction in ductility and fracture toughness. In order to discover these mechanisms the inspections are visual, dimensional, ultrasonic and radiographic. Pressure tube deformation has been managed such that it alone could not cause a safety or structural integrity concern for Cernavoda Nuclear Power Plant (NPP) fuel channels. Assessment methodology and prediction capacity enables Cernavoda NPP safe operation with certain conservatism. This paper presents the experimental assessment techniques for CANDU Pressure Tubes degradation mechanisms for the Cernavoda Nuclear Power Plant.

I.3.17. Development of Practical Techniques to Estimate the Power Cable Ageing by Thermal Methods

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At most polymers, used to power cables insulation and jackets structure, the dominant processes related to radiation and temperature degradation of the insulating material are under oxidation control. The Oxidation Induction Time (OIT) and Oxidation Induction Temperature (OITP) depend on the remaining level of antioxidant and extent of oxidation (or degradation) and remaining lifetime of the polymer thereby. The research, tests and analysis for installed power cable ageing management have been made to develop the monitoring thermal techniques of the insulating materials condition. CYY 3x25, C2XY 3x25 type sections of 3.5 m length have been accelerated thermal aged for ageing times equivalent to plant thermal field of 10, 20, 30, 40, 50 years. Have taken material samples (about 10 mg) from thermal aged cable insulation to determine OITP and OIT. The SETARAM SETSYSEVOLUTION 2400 installation used for data acquisition regarding sample temperature and heat rate. Analysing data gained noted the following elements specific to heating transformation processes: the baseline appears inclined; the peaks split (one of six samples); the OITP threshold is different from one sample to another; OIT determination is not completed because a 20 hours transformation at 200°C test temperature didn’t start. The elongation at break is a condition indicator accepted internationally to assess the degradation of power cable insulation. The tests and analysis to determine elongation at break, performed in INR, have shown that it is a sensitive parameter for material degradation. The tests are destructive versus the cable integrity and carried out in laboratory conditions on cables similar to those installed in the plant. The OITP values for the cable jacket material installed have to relate to elongation at break values resulted on cable insulation tested in laboratory to develop a non-destructive method to assess the power cable ageing. 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 original manufacturer cables for laboratory tests.

I.3.18 The CANDU Outlet Feeder's Wall Thinning by Flow Accelerated Corrosion – A Minimal Predictor Model

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The Flow Accelerated Corrosion (FAC), also named Flow Assisted Corrosion or Erosion-Corrosion, is a degradation process which affects the metallic material, as in the case of the significant pipes' wall thinning observed in CANDU reactors, PWRs, BWRs, and fossil power plants. The FAC can appear in both mono- and diphase flows, and is a fundamental combination between: electro-chemical phenomena in the pipe's material, water's chemistry, and mass transport. The CANDU outlet feeders' wall thinning phenomenon relieves a temporally augmentation of the probability of pipes' fissure. To avoid such sever events, it is necessarily to early replace the alarmingly affected pipes. Cleary, to be also financially efficient, this replacement action must be sustained and planed on periodically measurements of the wall's thickness, theoretical modelling and extrapolation of the results. In the above context, concordantly with the three most used descriptions of the Flow Accelerated Corrosion (FAC) phenomenon, namely Berge, Burrill and Sanchez-Caldera models, which construct a constant rate for the mass degradation dm/dt, the paper presents a minimal predictor model for the pipe's wall thinning by FAC. The model was used with good results to fit and extrapolate the measurements made between 1997-2012 at Cernavoda Nuclear Power Plant (a CANDU type reactor), concerning the outlet feeders' wall thinning by Flow Accelerated Corrosion. More exactly, correlated solely with the bulk of measured data, we analysed a number of 1140 cases for the in-time evolution of FAC phenomena, with an acceptable relevance between theoretical and experimental values.

I.3.19. A Practical Tool for the Common and Not Common Frequencies Detection in Two or More Congruent Spectra

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Sampling a time-varying signal and his spectral analysis are, both, subjected to theoretically compelling, such as Shannon's theorem and the objectively limiting of the frequency's resolution. Usually, the signals' spectra are processed and interpreted by a scientist who, presumably, has sufficient prior information about the monitored signals to conclude, for example, on the significant frequencies. On the other hand, processing and interpretation of signals' spectra are routine tasks that can be automated using suitable software (PC

applications). In the above context, the paper presents an intuitive and practical approach of the (automatic) detection of the common and not common frequencies in two or more congruent spectra. Our paper is not about time-varying signal's sampling and/or his spectral analysis theories, subjects moreover fully analyzed in the very rich special literature. Our paper signals but it is not about some possible algebraic structures and/or other involved mathematical aspects. We only present (and propose) a practical tool which can help the experimenter in a quickly (and automatically) identification of the common and not common frequencies existing in two ore more spectra. A tool which uses, to conclude on the significant frequencies, the intuitive discernment criterion based on the magnitudes of the spectral lines, i.e.: *the more important frequency in signal has the greater magnitude from the spectral lines*. Starting with four unconventional definitions for: *discrete frequencies spectrum*, *spectral line*, *congruent spectra* and *correspondent spectral lines*, the proposed method is mathematically funded and fully illustrated by numerical simulations made using the Microsoft Office Excel computational tools, and can be implemented in any automated signals monitoring task (e.g. PC applications), with an adequately degree of reliability. Indeed, the method was already used in the analysis of the noise existing in the experimental measured signals of two absolute pressure transducers installed at the ends of a pipe through which water circulates, some surprisingly results being presented also in our paper.

I.3.20. The Influence of Hydrazine as Oxigen Scavenger on Incoloy 800 Tubing of Nuclear Steam Generators during the Wet Lay-Up

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During the wet layup of steam generators (SGs) from CANDU NPP, hydrazine is commonly used to scavenge the dissolved oxygen in the secondary heat transport system, thus reducing the corrosion of many SG materials exposed to aqueous environments. Although oxygen scavenging is expected to decrease the corrosion of carbon steel support materials, the presence of hydrazine in large concentrations may compromise the integrity of passive films known to protect alloys used for SG tubes fabrication. The aqueous medium chemistry during shutdown and wet layup must be effective in prevention of contaminants ingress and their transport and in corrosion products formation and their transport to those areas where their deposition will occur. It is very important to determine the optimum hydrazine concentration for the unique combination of SG materials - tubing and tubesheet - from CANDU SGs. Moreover, it is desirable to optimize the hydrazine concentration used during layup periods in order to minimize the release of hazardous chemicals into the environment, while providing adequate corrosion protection SG components. The aim of this paper is to assess the corrosion behaviour of SG tubing material - Incoloy-800 - during SG wet layup period. For this purpose several electrochemical methods were used to characterize the corrosion behaviour of Incoloy 800 during wet layup simulated conditions. The electrochemical tests have been performed in demineralized water containing some hydrazine concentrations (max. 50ppm) having pH~9.5, adjusted with morpholine and cyclohexylamine (All Volatile Treatment – AVT) and the conductivity 60÷95μS. The results presented in this paper referring to the Iy-800 corrosion behaviour under SG simulated layup conditions can be used to determine the optimal hydrazine concentration necessary to ensure an adequate corrosion protection to Incoloy 800 components. In the long run, the goal is to minimize the environmental impact and

costs of the use large volumes of hydrazine to provide an acceptable corrosion protection to SG materials.

I.3.21. Microstructure and properties of thin films formed by Plasma Electrolysis Processing of iron based structural materials

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The objective of this study is the improvement of iron based structural nuclear materials properties by Plasma Electrolysis techniques. Electrochemical Plasma Technology (EPT) is an effective surface engineering tool that combines cleaning and coating of metals. During EPT processing, D.C. voltage is applied to the electrodes in aqueous electrolyte, which produces plasma at the surface of the workpiece. Thermal, chemical, electrical and mechanical effects imparted by EPT to the workpiece create unique surface characteristics. Ability to form plasma on the surface of the work piece in liquid electrolyte gives the capability to carry out various treatments on metal surface (generally cathode). Plasma electrolytic treatments of carburizing, carbo-nitriding, and boro-carbo-nitriding were applied on austenitic stainless steel samples (AISI 316L and AISI 304L). The stainless steel substrates were cathodically polarized. Experimental conditions of EPT treatments are presented. The coatings obtained in various experimental conditions have been investigated by means of optical microscopy, XPS (including "depth profiling" and "ball cratering depth profiling"), and EIS. Microhardness measurements were performed and corrosion behavior of treated samples was investigated by electrochemical techniques. The obtained results demonstrate that we can select the processing parameters for essential improvement of corrosion behavior in some aggressive medium and high values of micro-hardness. An overall description of the processes involved in the surface properties improvement, and some considerations about the further new materials development for various applications are presented

I.3.22. Assessment of Some Nickel-Based Alloys After Thermal Transient Tests

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This paper continued the study "The Thermal Transient Effect on Some Nickel-Based Alloys" presented at the 6th NUCLEAR 2013 Conference. The same nickel-based alloys were tested, Inconel 617 (UNS N06617) and Haynes 230 (UNS N06230). These materials are studied to be used in the 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 new experiments consisted in thermal transient tests using the following scenarios: fast heating rates with 90°C/minute up to 1,000°C, maintaining this temperature level 0 (zero) or 60 minutes, and then slowly or fast cooling. The examination methods used were, as follows: optic microscopy, Vickers micro-hardness and traction tests. 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 microdurometer with automatic cycle and WALTER BAI traction device. The metallographic analysis of samples consisted in microstructure examination, micro-hardness determinations

and traction tests. The average grain size was determined by linear interception method (Heyn-ASTM E 112) and reported as number G (ASTM Grain Size No). Average grain size (G) is in accordance with ASTM SB-168 and ASME SB-435 standards. The micro hardness of materials was calculated by using the relationship from the device technical book (OPL-France). On the traction diagrams, representing unitary strength variation σ [MPa] as a function of elongation ε [%], the following mechanic characteristics were obtained: tensile strength (R*m*), elongation at rupture (*A*) and elastic modulus (E). The tested alloys were compared with the received materials. The results can serve to create a database with candidate materials for Generation IV reactors heat exchangers construction, tested in different conditions.

I.3.23. Mechanical Tests on CANDU Pressure Tubes Performed at Institute for Nuclear Research Pitesti

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This paper fits the investigative technique of CANDU pressure tubes out of service. There are two types of tests for Pressure Tubes (PT) properties characterization and to define the mechanisms responsible for their degradation: tensile tests and fracture mechanics tests. These tests for analysis of pressure tubes are to be implemented at Post-Irradiation Examination Laboratory (PIEL) from Institute for Nuclear Research *(*INR) Pitesti. For curved compact tension specimen (CT type) tests with Instron tensile Machine, model 5569 from PIEL of INR Pitesti was designed and manufactured a gripping view of all the observations made in this paper and in previous tests. The high-strength material, which rupture is preceded by low levels of plastic deformation associated brittle fracture type, accepts the hypothesis linear elastic behavior in the region of the crack tip is primed and develops the break. Material feature is *the stress intensity factor "K"* size which is a measure of the stress increase in the presence of cracks from tensions in its absence. The tests performed on CT samples are made to determine the stress intensity factor for pressure tubes replaced from Cernavoda NPP. This paper has applicability by supporting research topics and development on the methodology for the analysis and evaluation of critical components of CANDU fuel channels proposed by Cernavoda NPP within the Collaboration Protocol with INR.

I.3.24. Localized Corrosion of Dissimilar Welds between Austenitic Stainless Steel and Carbon Steel Components from Secondary Circuit of Nuclear Power Plant

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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 (304 L austenitic steel and 52.2k carbon steel) and dissimilar welds (dissimilar metal welds: joints between 304 L austenitic steel and 52.2k carbon steel). To evaluate corrosion susceptibility of dissimilar welds was used electrochemical method (potentiodynamic method) and optical 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. The corrosion rates of samples (welded and non-welded) by electrochemical methods have been evaluated

I.3.25. Impurities Determination on Nuclear Fuel Pellets Using

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The manufacturing of nuclear fuel pellets used for research purposes at Institute for Nuclear Research Pitesti requires a series of chemical and metallurgical processes. For monitoring contaminations during manufacturing, the determination of impurities should be performed on the obtained pellets. Several impurities such as: Cd, Gd, Dy, B, Na, Mg, Al, Si, Ca, Cr, Mn, Fe, Ni, Cu and Mo in uranium dioxide fuel pellets were quantitatively determined by mass spectrometry using an ICP-oa TOF mass spectrometer. The analysed samples were calcinated and then dissolved in high-purity nitric acid. The uranium was selectively extracted with tributylphosphate from the uranyl nitrate solution to avoid matrix interferences. The samples were then diluted and analysed. All reagents used were of analytical grade and standards and were prepared by dilution of 1,000mg/l certified solutions. For this application, all solutions and samples were prepared in 1% ultra pure nitric acid. Ultrapure water of 18.2 MΩ.cm was used for preparation and dilution of samples. The calibration technique used in this case was external standard addition. To optimize the analytical conditions of the technique, several parameters, such as: torch position, nebulizer flow, plasma flow, auxiliary flow and generator power were varied. The used instrument settings are presented in the paper and also the spectra obtained analysing the aqueous solution of impurities. The best mass resolution obtained in this study is ~2300 for Dysprosium.

I.3.26. Nondestructive Post-Irradiation Examination of a CANDU Fuel Bundle from Cernavoda Nuclear Power Plant

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This paper presents the results of non-destructive post-irradiation examination of a CANDU fuel bundle, found with defect during the underwater inspection performed at Unit 2 of the Cernavoda NPP. The defect was found after one year burning, and the burn-up of the fuel at discharge was approximately 210 MWh/kg U. The flaw bundle was transferred

to the Post Irradiation Examination Laboratory (PIEL) of INR Pitesti, for examination and determination of the causes that lead to defect. For visual examination, dimensional and defectoscopy detection with eddy currents, three fuel elements (FE) were selected. For visual examination it was used a device which allows the fuel bundle to be placed on it in order to rotate it and move it in front of the periscope for inspecting and photographing its entire outer surface. The periscope provides magnifications up to 11.6x. During the visual examination, there were evidenced many superficial scratches on FE sheaths. For one of the three FE studied, it was confirmed the piercing of the sheath in the skate area near the reference end, as well as the cracking of the end plug-sheath at the non-reference end. This paper contains images of defects and interpretations of the causes of their occurrence. Dimensional examination consists in determination of diameter and arrow axial distributions, of the ovality and of the length for each FE studied. The measurement system is a machine equipped with step by step engines, for vertical movement and fuel element rotation, 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°. From the three FE studied one presented the most pronounced dimensional changes, and it was found an increase by 0.32% of diameter average, compared with the reference diameter provided by the manufacturer. The purpose of the eddy currents control is to obtain information about the sheath integrity of the irradiated nuclear fuel, and about the existence of defects produced by irradiation (cracks, holes, external and internal notches, changes of sheath wall thickness, inclusions, etc). The control with flaw detector was performed along the FE, on 80 equidistant generators. After longitudinal scans, circular scans were 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, no major defects were detected. Just scratches were highlighted, detected on previous visual examination.

I.3.27 Structural characterization of new intermetallic compounds used as targets for iridium and cobalt gamma radiation sources

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The aim of this paperwork is to present the microstructural characterization of some new intermetallic compounds used as targets for iridium and cobalt gamma radiation sources. The envisaged effect is to replace the metallic targets of iridium and cobalt with intermetallic compounds of aluminum in order to improve the quality of the iridium and cobalt based gamma radiation sources, to reduce the irradiation time and to lower the material costs. The samples subjected to microstructural analysis were Al, Co and Ir powders, as well as Co-Al intermetallic compounds sintered in argon and hydrogen atmosphere. The method of analysis used was X-ray Diffraction technique (XRD). XRD patterns were acquired using a Rigaku Ultima IV diffractometer with Cross Beam Optics module (CBO), CuKα X-ray source, 40kV working voltage and 30mA current, for an angular range of 2Θ [150-1550] at an angular step of Δ(2Θ) =0.050 and a time per step of

2s. XRD qualitative analysis revealed the existence of the following crystalline phases: Al (cub. sys., PS cF4/1) and Al2O3 (trig. sys., PS hR/10) for Al powder, Co-alpha (hex. sys., PS hP2/1) and Co-beta, (cub. sys., PS CF4/1) for Co powder, Ir (cub. sys., PS CF4/1) for Ir powder, AlCo (cub. sys., PS CP2/1) and Al2O3 (trig. sys., PS hR/10) for Co-Al intermetallic compounds sintered in argon atmosphere, AlCo (cub. sys., PS CP2/1) and Al2O3 (trig. sys., PS hR/10) for Co-Al intermetallic compounds sintered in hydrogen atmosphere. XRD quantitative analysis revealed the weight ratio (%), average crystallite size (Å) and average microstrain (%) for each phase.
Although XRD analysis has shown information about crystalline phase identification and content and a microstructural characterization has been performed showing crystalites sizes and microstrains, in order to fully demonstrate the efficiency of these intermetalic compounds comparing to the old classic metallic compounds used as targets for iridium and cobalt gamma radiation sources several additional tests are still needed. Irradiation tests on TRIGA reactor are going to provide the specific activity values for these intermetallic compounds in order to be compared with the Co and Ir classic metallic compounds. Electrical resistance tests before and after irradiation will be performed on the selected targets in order to determine the occurrence possibility for structural defects.

I.3.28. Surface modification/ alloying using Plasma Electrolysis Processing as a tool for improving the Corrosion behavior of Austenitic Stainless Steels used as Nuclear Structural Materials

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Alloy 316-L is an austenitic steel that has been used extensively in past sodium cooled fast reactors. From a cost and fabrication experience standpoint this material might be preferred for use for the vessel and other components of the LFR test demonstrator that are maintained at lower temperatures. New surface treatments, including the alloying of steel surface with Al, are currently being investigated looking for better corrosion resistance properties of 316-L SS exposed in liquid Pb and relevant temperatures field for LFR condition. The objective of this study is to improve the corrosion resistance by the development of a modified Al-containing surface layer on 316 – L SS using a complex surface treatment including Electrolysis Plasma Processing – EPP. Ceramic–like surface structures containing Al were developed on 316–L SS, by various surface treatments: EPP by cathodic polarization of 316-L SS in 0.5 M AlCl3 in Deep Eutectic Solvent ChCl-ureea (1:2 molar ratio); EPP by anodizing of 316-L SS surface in aqueous solution of 0.1 M NaAlO2 and 0.05 M NaOH; EPP of 316-L SS with Al layer deposited in Thermionic Vacuum Arc Plasma (TVA method). The coatings obtained in various experimental conditions have been investigated by means of optical microscopy, X-ray Photoelectron Spectroscopy – XPS and Scanning Electron Microscopy – SEM

I.3.29. Study of the Aging Behavior of Materials Important in Nuclear Energy Field - Incoloy 800 HT and 304L Steel - Using Neutron Scattering Techniques

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The Incoloy 800 HT is widely used in power generation for steam generators tubing and high temperature heat exchangers for gas cooled nuclear reactors and as a candidate material for fuel cladding in GEN IV reactors. The 304L steel has a high ductibility. Low yield stress and high tensile strength and is used widely in nuclear power plant. There were manufactured 4 samples of Incoloy 800HT and 4 samples of 304L steel, all of 2x15x25mm dimensions and standing a heat treatment of 60 days at 450, 500, 550 and 600 Celsius degrees respectively. The samples were investigated by neutron diffraction and small angle neutron scattering at the FSD, HRFD and MEREDIT diffractometers, YuMO-SANS and KFKI SANS spectrometers (in function at IBR-2 reactor, Dubna, respectively Budapest). Lattice cell and peak width parameters changes for both sample series (using neutron diffraction measurements) together with information on heat treatment effects concerning precipitate size and volume distribution of alloying elements (using SANS measurements) were found. Structural properties of the above mentioned materials of significant importance in nuclear energy field, revealed through high temperature heat treatment are described.

I.3.30. The Kinetics of Carbon Steel Corrosion in the Presence of Iron Oxidizing Bacteria

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Microorganisms modify the metallic surfaces (biocorrosion) by several mechanisms determined by both the nature of materials, and most of all the environmental conditions. Iron-oxidizing bacteria catalyze reaction of the ferrous with oxygen (dissolved in the water) to form ferric oxide. The Fe_2O_3 excreted is insoluble in water and precipitates, combined with organic matter, to form a shell or tubercle over the source of ferrous ions. This tubercle acts as a deposit and increases under deposit corrosion. The tests were performed on carbon steel samples (SA106gr.B) which were exposed in an inoculated medium with a culture of iron-oxidizing bacteria –Gallionella. Total exposure of the samples in these media was of 100 days. Based on a periodic weighing of the exposed samples in inoculated medium and their descaling, a series of kinetics data specifically of corrosion were calculated: the loss of metal by corrosion; the corrosion rate of the metal; adherent corrosion products; released corrosion products; the released rate of the corrosion products and the released rate of the metal.

I.3.31. Thermal Neutron Tomography at Institute for Nuclear Research Pitesti

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A new thermal neutron radiography facility placed at the tangential channel of the TRIGA-ACPR reactor was commissioned. Facility layout with main components and parameters, experimental set-up, investigated objects and results of the tomography reconstructions are presented. The detector system has been developed based on two interchangeable scintillators, one for neutrons (6LiF-ZnS, 0.3 mm thick) and the other one for gamma radiation (Gd2O2S, LANEX type) and two interchangeable CCD cameras (SXV-H9 from STARLIGHT XPRESS with a XD-4 type image intensifier for static higher resolution and EM-CCD Hamamatsu C9100-02 for static and real time imaging). The imaging investigations involving SXV-H9 camera for a test object and archeological objects have shown the potential for high resolution static imaging based on thermal neutrons and gamma radiations (exposure time is 35 - 45 s/image) for a single or a series of projections for tomography reconstructions. Imaging investigations involving Hamamatsu camera based on gamma radiations for the test object are presented also (maximum exposure time is 10 s and quality of the image is better for gamma imaging). The test object contains aluminum pipes, fuse with high breaking strength, normal fuse, a textolit bar, a brick of lead, borate polyethylene plates, a plastic pipe, iron welding electrode, insulating ceramics and brass in an architecture with a spatial extension that offers the possibility to characterize the spatial resolution and material contrast for neutrons and gamma radiations. Anthropomorphic and zoomorphic clay figurines of prehistoric settlements from Arges County were investigated with promising results to reveal internal structure by tomography method.

I.3.32. Increase of Power Units Efficiency Based on Thermal Energy Conversion

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The actual energy sources are based on fossil or nuclear materials. Prevention of environment change and destruction imposes to use improved technologies for energy production; to don’t exist significant changes in nature and adjacent life. Actually, from Nuclear Power Units and from Thermal Power Units, during power operation, is evacuated to the environment a large quantity of heat, heat that contributes to the environment change. For the nuclear and thermal power plants cases, stopping to throw large quantities of heat in the environment should be very advantageous. In this respect the paper offers some solutions in order to solve such problem. In the thermal cycle, at power production, the residual heat energy is in many designs evacuated to the environment, this decreasing the efficiency of the power unit. The solution that is proposed in the paper, to convert this residual heat into power energy, will increase the efficiency of the power production cycle. To estimate the effect of air

and water warming, in the paper were performed special calculations. The paper gives some solutions in these directions, one being the possibility to use the residual heat and convert it in other forms of energy. A second benefit will be stopping to thermal pollute the environment. Paper shows also specific elements related to the possibilities that exist in order to store high quantities of energy. In the paper were included specific solutions on how to use the residual heat energy, from the thermal cycle, by converting it into other forms of energy.

I.3.33. Improving Electromagnetic Flow Meter Sensibility

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In some parts of industry, measuring different types of parameters is not possible using standard equipment, due to many facts. In our case, this is due to high temperature 500oC and liquid properties. Measuring the flow speed without modifying the pipes will offer a good reliability in time. This activity of building an electromagnetic flow meter started as a necessity for us. Having an experimental LEad COrrosion TEsting LOop (LECOTELO), imposes to optimize the flow trough pipes and different components. One part in which we can improve pressure loss is in using flow meters without intrusion in the pipes. The main goal was to find the best shape and properties for coils used to build an electromagnetic flow meter for melted metals. The first challenge was to determine the sensibility of entire process. The second stage will be to build the electromagnetic flow meter, and the third one will be to determine the relationship between speed and output signal the electromagnetic flow meter, using the data obtained from the first one, which is a Venturi flow meter. The first experiment has been performed to show how the coils are influencing the sensibility of the measuring system. The second experiment using a pipe with the diameter and thickness of the wall as same as the pipe used at LECOTELO did not act as expected, the signal on the receiving coil was too small compared to the noise generated by the whole system. Installing all three coils on the same casing did not allow adjustment of the distance between the coils; therefore null signal could not be obtained in the receiving coil. The flow meter sensitivity can be improved by directing a large part of the magnetic field trough the flowing liquid. This can be done by using pipes with thin walls, decreasing the number of copper layers of the coil, installing the coil closer to the metal pipe. Using shielded wires to collect signal from receiving coil reduces a lot the noise from the main power supply.

I.3.34. Plugging Tests for Stopping of the Industrial / Service Water Flow through Horizontal High Diameter Tube (200 mm Nominal Diameter) with Artificial Ice Plug

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The local isolation of a pipeline section, placed horizontally in an industrial / service water supply system installation, can be made with an ice plug resulted after a controlled process inside of pipeline. The technique is applied in order to execute repairs or replacements of components, without stopping and draining the system. This isolation represents a technological process developed continuous especially by the area-line firms, according to each application's requirements. On principle, the technique is simple and assumes applying of a special built sleeve/ collar for each case. Assembled outside on the tube in the area chosen for flow plugging, it form an annular space with tube surface or in its right closeness. This space is allocated, in our case, to vaporization of the liquid nitrogen. The heat and mass transfer, executing continuous between the tube wall and refrigerant ring shaped, encouraging the ice deposit in subsequent layers. For the first phase each technical application has to be treated experimentally, both regarding the ice plugging device and liquid nitrogen gauging during entire ice plugging process and stable keeping in the tube. The time process for stopping with ice plug of the liquid flow through the tube and the liquid nitrogen consumption/inventory depends on the geometrical configuration of the device and by the cooling chamber sizes. Choosing the overall sizes of the device and respectively, of the liquid nitrogen room sizes, are direct related by the nominal tube diameter value and by the liquid capacity affected. The paper contains a constructive description of the experimental technological facilities used, followed by setting off the main moments of some experimental tests carried out for the industrial/service water. Finally, the results analysis and some conclusions outlined that the proper flow stopping process time depends equally (on size and geometry device) by the liquid nitrogen injection method in the ice plugging device and industrial/service water temperature evolution during tests. The paper is dedicated to the specialists working in the research and technological engineering.

I.3.35. The First Experimental Tests to Isolate the Steam Generator Tube by Plugging of the Inlet into Tube Sheet

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The steam generator has to keep its structural integrity during entire operation lifetime in normal condition and design basis event condition. The tubes shall be inspected according to the standard requirements. It is thus performing an assessment of the structural integrity and of the leaks value, the results being compared with the specified performance criteria. The leak rate hasn't to lead to radiological overdose, the consequence required by the control body. The steam generator tube defective detected has to be plugged at both ends following its mechanical separation from the primary heat transport system. The steam generator defective tube isolation assumes plugs assembling which geometrical configuration owning to installing position at the tube end and to the plastic deformation technique applied. The high-speed plastic deformation techniques have application speed less than 100 m/s, develop locally heat, the restraint expansions and, appropriate to them, residual mechanical stresses in the deformed material. The unconventional high-speed plastic deformation techniques used develop locally neither the additional mechanical stresses nor structural changes in the deformed material. In the paper, after a brief introduction and description of the technique developed following by some features of the experimentally application and followed by a description of the tests carried-out. Finally, the results analysis and conclusions outlined that decreasing the mechanical clearance between the plug and tube sheet simulator isn't enough for successfully joint, the swelling socket stiffness being the cause of the failed experimental

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

I.3.36. Effective Use of Corrosion Inhibitors in Concrete Reinforcement

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The design life of existing Nuclear Power Plants (NPPs) was often chosen to be 30-40 years. However, current economic pressures on utilities to extend plant service life (60 years total being a quoted target) and decommissioning strategies that involve use of the containment as a "safe-store" for periods of up to 100 years mean that the containment buildings may have to perform safety functions for a time period significantly greater than their initial design life. Concrete is a durable material and its performance as part of the containment function in NPPs has been good. However, experience shows that ageing degradation of the containment, often caused or accelerated by factors such as improper design, use of unsuitable or poor quality materials, improper construction, exposure to aggressive environments, excessive structural loads, and accident conditions, could impair its safety functions and thus increase the risk to public health and safety. Corrosion occurrence on the surface of the reinforcing steel causes not only decrease of adhesion/adherence between the reinforcing steel and the concrete, as well as the reduction of the reinforcing steel section, but also – due to the big increase of the volume of the corrosion products in relation to the initial reinforcing steel volume – huge pressures, which cause appearance of the fissures, setting apart, cracking and scaling of the concrete protection layer above the reinforcing steel. Corrosion protection may be carried out by the isolation of the concrete reinforcement against aggressive environments through the use of “corrosion inhibitors”. Migrating corrosion inhibitors are chemical compounds on the amine basis which through the process of chemical adsorption, so called chemisorptions, bind/adsorb on the surface of the reinforcing steel/iron (and other metals), making on the surface an firm and resistant micro layer, resistant to many aggressive substances from the environment, primarily to the impact of the chloride, in the nature omnipresent, and simultaneously very aggressive against the iron oxides, which it chemically destroys. MCI-corrosion inhibitors protect the reinforcing steel against the corrosion in both oxidation ranges: cathode and anode range.
These inhibitors can be applied to existing reinforced concrete structures and will then be carried by water into the proximity of the reinforcing steel. MCI-corrosion inhibitors are water soluble, aiding mobility within the cover concrete.
The effectiveness of two inhibitors applied to the mature reinforced concrete was evaluated. The corrosion behavior of the steel rebar and the inhibiting effects of the amino alcohol chemistry in an aggressive environment were monitored using SEM investigations, gravimetric and electrochemical measurements.

I.3.37. Contributions to the Optimization and Simulation of the Processes from the Installation for Tritium Separation in LABVIEW and POWERVIEW Software

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Paper aim is to set up contributions to the development and improvement of information system for control, supervising and ensuring safety in operation to an experimental pilot installation for the separation of tritium from tritiated heavy water, resulting in nuclear reactors used for electrical energy. Also, the paper shows the development of some integrated systems for monitoring and control of instant power consumption in normal and abnormal operation mode of tritium separation technological installations, with the view of reducing energy losses, supports mainly the development of new efficient technologies and software methods on power consumption in nuclear processes, experimenting on an existing installation for tritium separation. The system allows the recording of events from the tritium separation plant, with the view of conducting analyses on the quality of the electricity. In the paper presented, briefly, some solutions developed by the authors for the information system to accomplish the optimizing and control of a pilot plant installation for the tritium separation from heavy water produced in nuclear reactors. In terms of hardware, the solutions aimed to the increasing the number of sensors and transducers and to the introduction of data acquisition systems of the last generation that provide more benefits in monitoring of performance parameters and also of the installation equipment's. A special attention was given to the software issues, in this respect is emphasized the introduction of LabVIEW graphical programming that has proven to be particularly advantageous, having adequate facilities for applications as monitoring and control of processes, also including capabilities for modelling, simulation and optimization. The authors consider that by implementing solutions, partially illustrated in this paper, there are satisfied at an appropriate level the requirements of a pilot installation to ensure the flexibility of operating modes and to increase the quantity and quality of information necessary to perform complex research programs in the field of tritium separation. Energy management of an installation (consumption in different sectors) is the process of monitoring, coordination and control of these inputs in order to ensure efficient consumption of energy by reducing energy losses and thus minimize energy costs. Also, the research in the field of monitoring power for important consumers within a technological installation represents an important contribution to the economic development by means of increased energy efficiency of the entire energy chain.

I.3.38. Tritium gas handling/storage system proposal for the TRF at ICIT Ramnicu Valcea

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Tritium gas resulting from cryogenic distillation process in the pilot plant for separation of tritium and deuterium is extracted and stored with tritium handling and storage system coupled to cryogenic distillation plant located in the Tritium Removal Facility (TRF) at ICIT Ramnicu

Valcea. For flexibility in tritium storage strategy for long term and medium term the proposed tritium handling/storage system for TRF is designed on two types of getters bed, one based on titanium and the other one based on depleted uranium. The titanium getter bed has the main objective to store the tritium gas for long terms. For adsorption/desorption processes the depleted uranium getter bed is use because of its lower temperature needs in gas diffusion. Another advantage of the depleted uranium getter bed is the complete recovery of tritium from gas mixtures that may result from purging/cleaning plant or for temporary storage of tritium in normal use. From technical aspect the storage system consists of a series of valves, a 1 litter calibrated vessel, technological pipelines, fittings, pumps, instrumentation and two vessels (tanks) tritium storage bed metal getters, sized appropriately for performance requirements. Both getters, one with titanium bed the other with uranium bed together with the whole system, are placed in a vacuum chamber called glove box. Glove box will have a controlled atmosphere by the following parameters: oxygen level, the level of hydrogen (tritium), humidity, pressure and temperature.

I.3.39. Proposal of a Dynamic Hydro-Gas Redistribution Pan for Isotopic Exchange Process Optimization in Columns with Catalytic Mixt Packings

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The paper presents a technical solution for isotopic exchange process optimization in the Tritium Removal Facility (TRF) at National Institute for Research and Development for Cryogenics and Isotopic Technologies Rm.Valcea. The proposal of a dynamic hydro-gas redistribution pan for the catalytic exchange column resulted as a consequence of the peripheral preferential flow development inside the isotopic exchange column, having undesired effects in the isotopic exchange process. The designed and manufactured redistribution pan will be able to provide a uniform redistribution of fluid flow, gas flow and also vapour flow at the interface of the catalytic mixt packings, between each of them. Tests of the redistribution pan inside the column have been carried out proving that the pan is suitable for applications of liquid flow from 1l/h to 30l/h depending of the geometry of liquid holes and gas nozzles. For easy filling of the isotopic exchange column of the TRF, mechanical characteristics of the catalytic mixt packing have been studied and a clamping system with metal hooks for the dynamic hydro-gas redistribution pan was developed. This clamping system allows the redistribution pan to be attached to each of the packings. Experimental data obtained from the tests carried out in the isotopic exchange module of TRF at Rm.Valcea proved that the proposed solution significantly contributes in obtaining uniform dynamic hydro-gas flow distribution improving the catalytic activity and hold-up for each packing and optimizing the isotopic exchange process.

II. ENVIRONMENTAL PROTECTION

II.1. Radioactive Waste Management

II.1.1. How Important is the National Context in Planning Romanian Geological Disposal?

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There is a general consensus that the national context, often named as the national specificity, is recognized as an important factor to be taken into account when approaching the geological disposal. Documentation on the international standards and reports or the presentations from international forums, conferences or seminars indicates clear recommendations that when planning geological disposal activities such as research, siting, licensing or stakeholders' involvement, the international knowledge and developments should be considered but these should be adapted by considering the national context. The matter of what effectively the national context does mean and how important it is when geological disposal is initiated is not simple. Each country should consider a detailed and complete answer to this matter as early as possible in the decision making process for embarking a geological disposal program. The national context induces issues that easily could evolve in significant risks and some of them might be already be triggered. These represent potential causes for incipient delays or even failures in the geological repository program. An engineering research study on the national context was approached by authors for the case of the Romanian National Geological Repository (NGR) program. Basically, the national context was considered consisting of inputs, actions or inactions from outside the NGR program developer organization, over which the management of the developer organization has no control, but which may have significant impact on the program. The preliminary results of a risk analysis clearly indicated that the study of the national context is not simple and it should be systematically approached through a risk management process. The paper presents in brief the results of the risk identification and analysis and some information on how national context might impact on the NGR program.

II.1.2. Safety Requirements on the Radioactive Waste Pre-disposal Activities and Facilities

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The paper has intention to present the regulation which will be issued by nuclear regulatory authority. The regulation contains the safety requirements on the radioactive waste pre-disposal activities and facilities as well as on the storage of spent nuclear fuel. The regulation is structured in six chapters on safety requirements and one chapter which

contains licensing requirements. The first chapter is dedicated to the scope, objective and definitions. The second chapter refers to the safety area, safety management and detailing requirements on the responsibilities, organisation structure, management system as well as record keeping. The next three chapters detail the safety areas siting, design and operation. The operation safety requirements are related to the characterisation and classification of radioactive waste, pre-treatment and treatment of radioactive waste with focus on the different kind of radioactive waste: liquid and solid waste, spent sealed sources, high activity sources, biological waste, as well as conditioning and storage of radioactive waste. The sixth chapter is dedicated to the safety area safety verification and contains requirements on the safety case, safety assessment and safety analysis as well as on the verification and periodic safety review. The licensing requirements are focused on the licensing phases which need licence as well as on the documents which shall be submitted by licensee in support of licence application.

II.1.3. Development of In-Situ, Long-term, Inspection and Monitoring Programs for Metallic Components

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The couple service environment / metallic wall of any technological equipment or component is prone to several damage phenomena in aqueous environments. The development of an inspection and monitoring program is an extremely complex task for any design engineer. Depending on the type of metallic component to be evaluated, on technical expertise and available resources, could be proposed a methodology for the in-situ evaluation of long-term behaviour of a metallic component in a real service condition. The philosophy presented in this paper could be further extended for inspection and monitoring of metallic components in nuclear disposal systems or for transport/technological pipeline networks. This viewpoint is based on the well-known procedure developed by material engineers in order to analyse the remainder technical capability of a technological component in real service condition. In case of disposal systems, monitoring results will permit to forecast interactions between metallic surfaces and sealing/backfilling materials. Based on designer experience related to in-service failure analyse of industrial components an inspection and monitoring program adapted for real in-situ condition could be developed. This program will consider available applicable codes and standards, scientific & technological development in the area, ability to achieve and reliability of the proposed technical solution for the inspection and monitoring program. The aim of this paper is to highlight difficulties related to the implementation of in service inspection and monitoring programs (considering as an example storage / disposal systems) and, also, the need for a cautious approach of the records and of the results of these programs to avoid misinterpretations or questionable decisions.

II.1.4. Assessment of the Long-Lived Radioactive Waste generated from Cernavoda NPP Operation and Decommission for Geological Disposal

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A good knowledge on the types, inventory and quantities of radioactive waste generated during Nuclear Power Plant life time is essential in planning and implementing a proper management strategy to avoid any undue burden on future generations. In this paper the types and quantities of long-lived radioactive waste (including spent fuel) generated during Cernavoda U1&2 life time were estimated based on existing CANDU reports and IAEA technical documentation. Also, based on Origen simulations the inventory of the spent fuel assembly at their discharge as well as in different steps of the wet and dry storage were assessed. The long-lived radioactive waste (LL-LILW) are generated both in the operational and decommissioning phases of Cernavoda NPP. Furthermore, since it is intended to extend the life time of the two CANDU units, part of the metallic components dismantled during the refurbishment will also contain long-lived radionuclides exceeding the WAC for Saligny repository. The *operational waste* consisting mainly in non-fuel contact resins (SIER) and part of the spent cartridges with significant amounts of ^{14}C and ^{129}I might not meet the WAC for surface repository and have to be disposed of in a geological repository. The quantity of non-fuel contact SIERs was estimated to be around 272 m^3 for an operational period of 40 years (from the U1&2). The total amount of spent cartridges was estimated to be around 4 m^3 but only a small fraction will exceed the WAC for Saligny repository. The LL-LILW generated from the *refurbishment* of one CANDU unit consist mainly of pressure tubes (380 pieces weighting 23.2 tonnes) and Calandria tubes (380 pieces weighting 8.74 tonnes). Also due to the ^{14}C and ^{36}Cl content the end-shieldings (720 pieces weighting 113 tonnes) could exceed the WAC for Saligny repository. From *decommissioning*, the majority of the radioactive waste resulted from the reactor dismantling (Calandria components and control rods with drivers and ion chambers, as well as the steel balls) are intermediate level waste and have to be geological disposed. Since in Romania the geological repository is foreseen to be operational in 2055 a proper storage solution for the LL-LILW has to be established and implemented. Also the national programme for the implementation of geological disposal shall be developed.

II.1.5. Method for Gamma Emitters Measurement in Spent Ion Exchange Resins

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In order to establish the measurement method for gamma emitters present in the spent resins arising from the purification systems at Cernavoda NPP, a mixed bed resin was simulated, consisting of a mixture 1:1 anionite – cationite (eq). To obtain the liquid samples consistent with gamma spectrometry analysis, the resin was prepared using the most effective methods. To realise the ionic exchange with the cations from the IRN 77 resin, the resin previously prepared was loaded onto chromatographic columns simulating similar conditions as those of Cernavoda NPP purification systems. Solutions with known concentrations containing cations such as Cs, Co, Mn and Ba (that could be found in the real spent resins) were passed through the resin layer. The chemical decomposition process (transition into a liquid-aqueous phase) of the resin charged with stable elements was performed by microwave digestion, in a microwave digestion system SpeedWave4. The process takes place in the presence of a HNO_3 - H_2O_2 mixture, in TFM™ pressure resistant (100 bars) vessels and the resulting liquid samples were analysed by inductively coupled plasma - optical emissions spectrometry (ICP-OES), excepting the caesium. This element couldn't be measured by the emission spectrometry due to its low intensity characteristic line and therefore, this element has been measured by inductively coupled plasma mass spectrometry (ICP-MS).The recovery efficiency was calculated for each element on the basis of the obtained results and these will be used to correct the results of the measurement obtained by gamma spectrometry on the real resins.

II.1.6. Estimation of the Flow and Transport Parameter for Column Experiments in Limestone

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Proper site characterization is a critical component in assessing the safety of radioactive waste repositories. The purpose of this paper is to determine the flow and transport characteristics of radioactive contaminants, such as H-3 and C-14, in limestone. The experimental results collected from flow-through experiments in two limestone columns were analyzed by using inverse modelling with HYDRUS-1D code. Old samples of limestone from the depth of 71 m from an existing borehole, located in the Dobrogea Berriasian Platform, in the vicinity of the Saligny site were crushed, sieved, packed into two columns of different particle sizes and saturated with synthetic water having an ionic strength of 26.5mM and a pH of 8.7. Two-stage pulse experiments at different flow rates were conducted, to investigate the effect of the flow rate on contaminant transport. The eluted fluid was collected at the top of the columns, and the concentrations were measured by liquid scintillation spectrometry. Tritium transport was investigated both by physical equilibrium (LEA) and non-equilibrium (mobile-immobile, MIM) models, with the later showing better agreement with the experimental data. Flow parameters (diffusion constant, longitudinal dispersivity, mobile and immobile porosity, mass transfer coefficient) obtained from H-3 transport, were further used for estimation of the transport parameters for C-14. Sorption of C-14 was assessed considering physical non-equilibrium and three promising mechanisms: (1) equilibrium non-linear sorption and kinetic sorption in the mobile liquid region, (2) equilibrium non-linear sorption, kinetic sorption in the mobile liquid region and irreversible sorption, respectively (3) attachment-detachment model with irreversible sorption. The third model gave the best results, with a very good match of the breakthrough curves, including the tailing part. Nevertheless, a better understanding of the sorption mechanisms implies the need for additional information, concerning mineralogical composition of the rock, water chemical composition, measurement of the adsorbed concentrations in the rock and a better documentation of the experimental works.

II.1.7. Chlorine Separation by Extraction Chromatography for ^{36}Cl Evaluation

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Evaluation of ^{36}Cl in nuclear wastes is important because this radioisotope is a problematic β emitter for radioactive waste disposal. It has a long half-life (3.01E05 years) and is very water-soluble, with a high soil-plant transfer factor, making it a potential long-term hazard to the environment and people if it is not disposed properly. Measurement of ^{36}Cl radioactivity is based on detection of its beta particles emission by liquid scintillation cocktails technique (LSC). In order to obtain accurate and precise results, the samples have to pass certain preparation steps such as chemical separation and purification of the ^{36}Cl nuclide from other β interfering radionuclides. This paper describes an analytical method for separation and evaluation of Cl from an aqueous matrix. To remove interfering elements (Co, Mn, Eu, Ni, Sr and Cs), the extraction chromatography method is used. The Triskem International CL Resin has an extraction system that is selective for silver, gold and platinum group metals. To selectively extract the chloride ions, the resin is preloaded with silver ions from 1M H_2SO_4 solution. The maximum uptake for chloride ions strongly depends on the amount of silver ions loaded onto the resin. Retention of chloride ions is tested in 1M H_2SO_4 solution, and elution conditions are achieved using 0.1M KSCN solution. The interfering elements are efficiently removed from the sample with a very high decontamination factor. The recovery of chlorine is very good in the CL Resin experiments; therefore this method can be applied for the separation of Cl from radioactive waste before beta spectrometry measurements.

II.1.8. Scale Models Package Use for Demonstrating Compliance with **Requirements of the Transport Regulations**

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Because it takes place on the public domain transport of radioactive materials involves potential radiological risks and therefore needs to be conducted under full safety and security conditions. The requirements are assured primarily by the quality of the transport packages, which must be designed, executed and tested according to national and international regulations. These packages need to be capable to retain the radioactive content under normal conditions of transport, in different situations that may occur during transport or in conditions of possible accident during transport in order to be approved in accordance with IAEA regulations. Demonstrating the compliance of packages with current regulations for approval can be done by several methods that are specified by the IAEA TS-G 1.1 (rev.1) 2008 such as rigorous testing on prototypes, experimental models on the scale (representative model copies) on parts or components of packages using calculation methods (computer codes), data collected from sensors or combinations of them. Experience has shown that the testing of *scale models* may be very useful for demonstrating compliance with certain specific requirements of the Transport Regulations, particularly the mechanical tests. This paper will present the results obtained in ICN on experimental scale models testing for type B(U) transport package in order to demonstrate

compliance with the transport regulations. It will be emphasized *the mechanical test* due to the conditions of similitude that are relatively simple to create, because it is possible to study the relationship between package orientation and the overall deformation of the package and to obtain information concerning the deceleration of package parts.

II.1.9. LSC Method for C-14 Measurement in Solid Radioactive Waste

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C-14 is one of the main radionuclide in the solid radioactive waste generated at Cernavoda NPP that has to be measured to adopt proper treatment/conditioning technologies for this waste category. These radioactive wastes are classified into compactable (such as paper, textiles, plastics and rubber) and non-compactable (such as tools, metallic parts, wood pieces, glass and molecular sieves). In this paper a method for C-14 measurement from simulated solid radioactive waste was adapted and implemented in order to be applied in real waste characterisation. To measure a beta emitter by LSC method homogenous solution has to be obtained and for this the simulated waste samples were oxidized using 307Sample Oxidizer equipment. 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 scintillation (PermaFluorE+) and the resulted solution is ready to be measured by LSC. The liquid samples were counted with liquid scintillation counter TriCarb3110TR, in Normal Count Mode (typical count rate > 500CPM). Tests for assessing the fraction of C-14 recovered in the combustion stage have been performed. The recovery yield was between 95% and 100%, with an average value of 97%. This value was applied to correct the measured values of ^{14}C in the simulated solid waste. The very good results for the recovery yield obtained applying this method confirms the applicability of the two techniques in the characterization of solid radioactive waste.

II.1.10. Conditioning Tests of Used Cartridges Containing Active Charcoal

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To minimize the radioactive releases and the environmental impact the potentially contaminated air is circulated at Cernavoda NPP through four ventilation systems and all potentially contaminated exhausted air is routed to the exhaust stack. As well as the liquid releases, the gaseous releases are under the limits approved by the Regulatory Body. In the ventilation systems a complex decontamination is achieved using a pre-filter or a combination of pre-filters, HEPA filter and activated charcoal filter. The filters from the iodine and noble gases monitoring system comprise of active charcoal impregnated with *Tri-EtyleneDi-Amine* (TEDA); the retention process consists in a chemical reaction between TEDA and iodide and the resulted ammonium quaternary salt is irreversible fixed on the active charcoal. In the gaseous effluents, tritium might be released in form of tritiated water vapors, C-14 as carbon dioxide and iodine in form of I, I_2, organic iodine (CH_3I) as well as important quantities of hypoiodous acid (HOI). To establish a proper technology for spent active charcoal cartridges tests on their embedding in Portland cement, at a ratio of about 6.5-7 % (w/w) waste were performed. The efficiency of the resulted waste form was assessed by means of leaching tests, measuring the losses of H-3), the pH and conductivity during the whole period of leaching test, and the variation in the weight of the samples. Tritium is released from the waste form with rates in 10^{-7}-10^{-6} kg/m^2s range. The conductivity values of the leachant solution decreased in time and the pH variations are low and, generally, within 10.5-11.5 range.

II.1.11. Method of I-129 Separation by Extraction Chromatography

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I-129 is a long-lived radionuclide (t1/2=1.6x10^7 y) with a lower specific activity (1.73x10^{-4} Ci/g). For this reason, the separation and the purification of I-129 from the radioactive waste samples are the key stages in the I-129 determination. The iodine (I-127 and I-129) separation is usually performed based on the redox reactions where the iodine is converted in the desired oxidation state, this being combined with the iodine ability to be soluble in organic or inorganic media depending on the its oxidation state. Recently, Triskem International proposed a new separation method for I-129 and Cl-36relying on the halogens ability to selectively combine with Ag^+ ions fixed on a specific resin, named Cl-resin, which has a great affinity for gold and silver, in acidic conditions. The elution of I^- and/or Cl^- from Cl-resin is performed using the different solubility of the AgI and AgCl in different eluent solutions. The Cl-resin extraction chromatography was applied at the separation of I-127 from simulated aqueous liquid samples. The simulated samples, prepared in H_2SO_4 1M solution, were passed through the Cl-resin that was pre-loaded with Ag^+. The iodine adsorbed on the resin was eluted with a small volume of K_2S. The concentration of iodine in the collected effluents in each stage of the separation process was measured by ICP-OES. The obtained results show that the iodine was strongly adsorbed on Cl-resin from H_2SO_4 1M solution, and more than 85% from the initial concentration of iodine was found in 5 ml K_2S 0.35M.

II.2. Radioprotection

II.2.1. Determination of ^{3}H and ^{14}C in Organic Samples after Separation through Combustion Method

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Cernavoda NPP has designed and implemented Environmental Radioactivity Monitoring Program to provide an independent assessment of the effectiveness of source control, effluent control and monitoring based on measurements in environment. The purpose of the routine environmental radiation monitoring program is to provide reliable and accurate data, which comprise statistically valid data sets per nuclide/ environmental media combination on an annual basis. The Environmental Program is based on the analysis of the following types of sample: airborne, surface water, drinking water, depositions, soil, grass, sediment, fish, milk, meat, eggs, grains, fruits and vegetables, using determination of H-3 and C-14, gross beta and gamma spectrometry measurement techniques. The Environmental Control Laboratory of the Cernavoda Nuclear Power Plant, located in Cernavoda town, is equipped with modern preparation equipments and analyzing systems of last generation in nuclear field, operates at international standards with specific procedures and trained personnel, to determine the natural and artificial radionuclide content in the environmental samples around the NPP. In the Sample Oxidizer 307, water is condensed in a cooled coil column and then washed into a vial where it is mixed with an appropriate LSC cocktail. The CO_2 is trapped by vapour-phase reaction with an amine and the resulting product is mixed with an appropriate LSC cocktail. At the end of the combustion cycle, two separate samples (a ^{3}H sample and a ^{14}C sample) are trapped at ambient temperature, thus minimizing cross contamination. Since sample combustion is suitable for any organic and even some inorganic samples, the selection of this method is usually governed by the number of samples that need to be processed. When the sample load exceeds 50 per day, then sample oxidation becomes the method of choice for many sample types. Separation of tritium from organic samples at Sample Oxidizer 307 is very useful when investigate results from different preparation methods, providing accurate data.

II.2.2. Simultaneous Determination of Total Alpha and Beta Activity by Liquid Scintillation Spectrometry

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For radiation protection, determination of total alpha and beta activity in various samples is a sensitive and useful tool, giving the possibility to categorize materials and/or sites and thus, to optimize further monitoring activities. Usually, the measurement of samples' activity is done by counting beta/alpha particle emission with a detection system using one of the following types of detector: gas flow proportional counter, solid state particle detector, plastic scintillator. In this case the sample is prepared such as the radioactive material or the sample itself is deposited on a flat metallic counting planchet. The method proposed in the present paper is based on the possibility offered in liquid scintillation spectrometry to distinguish between contributions of alpha and beta radiations in the spectrum. The preparation of samples is often, simpler that in case of planchet counting, requiring just solubilisation of radionuclides and mixing of sample solution with an appropriate scintillator. The authors developed a data processing methodology based on a de-convolution algorithm. For total beta activity determination, the *Efficiency Tracing* method is indicated and used, since it can give an accurate indication for beta emission rate in a LSC cocktail, for radionuclides emitting beta particles having maximum energy over 70 keV. Experimental tests have been performed on different sample solutions, containing mixtures of radionuclides with combined alpha, beta and gamma emission. The accuracy of the method proposed in the present paper, was evaluated by comparing the results obtained for testing solutions with the activity concentrations determined by gamma spectrometry. Due to the fact that the performance of the proposed method is fitting the purpose of the radioactivity monitoring, the implementation of it, at the level of laboratory working procedures, was proposed.

II.2.3. Management of Tritium Exposures at CERNAVODA NPP

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Situated at 180 km east of Bucharest, Cernavoda Nuclear Power Plant is a CANDU 6 type NPP. CANDU (CANadian Deuterium Uranium) is a Canadian design power reactor, which employs natural uranium as fuel and heavy water as a neutron moderator and as the thermal agent.

The Cernavoda Nuclear Power Plant is required to provide a dosimetry service to assess and document all significant doses of ionising radiation received by workers and visitors at Cernavoda NPP as a result of the presence of radioactive sources or the operation of radiation devices. The purpose of the dosimetry program is to measure, assign and record all significant radiation doses received by an individual over a known period of time, whether these result from exposures of the whole body or of only a limited region, and to retain these records in useable form for comparison with the appropriate regulatory limits.

CANDU occupational dose is assessed as a combination of internal and external dose. External dose is the sum of gamma and neutron doses. Internal doses include tritium and the effective dose assigned from whole body counter scans but the tritium dose practically accounts for the whole internal dose.

Tritium doses are estimated for the critical group defined as a self-sustaining agricultural community that obtains no food and water from outside sources and is located in the area of highest potential concentration.

Tritium emissions of Cernavoda NPP were continuously monitored. The tritium concentrations in environmental samples were monitored as part of routine program and public doses were calculated. The supplementary tritium doses for a member of public, is estimated using HTO concentrations in effluents and in environmental samples. For better modeling OBT contribution to public doses is estimated based on OBT/HTO ratio in food samples.

II.2.4. Experimental Tests for Determination of Po-210 in Aqueous Samples

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Polonium-210 is a naturally occurring alpha emitter and exists in the environment as a result of the Pb-210 decay within the U-238 decay chain. Po-210 is the most interesting radioactive isotope from environmental impact point of view. The activity concentration of Po-210 is usually determined by alpha-particle spectrometry. Experimental tests were focused on the implementation of Po-210 determination method in aqueous samples, according IAEA/AQ/12 procedure. To simulate real samples of known activity certified reference materials were used. The measured samples covered a relatively wide range of values for Po-210 concentrations and a variety of matrices (water, soil, sediment, phosphogypsum). The purpose of the conducted tests was to compare the performance of polonium separation methods based on extraction with organic solvents or chromatographic resin. The following activities were conducted for implementation of IAEA/AQ/12 procedure: reagents and equipment selection, reagents and electrodeposition discs preparation, reference materials selection, tracer and solutions of known activity for Po-210 preparation, by reference materials mineralization. The tracer solution was prepared using certified reference material, SRM 4326 (Po-209 chloride in 2M HCl). The samples were subjected to analytical procedures in three variants as follows: option 1 - the extraction with organic solvent in liquid phase, option 2 - spontaneous deposition on transition metal in acid solution, option 3 - extraction using strontium resin. The chemical separation step improves the reliability of the procedure in terms of both Po recovery and spectrum peak resolution by removing interfering elements present in the sample. Test results have shown that the implementation of the method is favorable. The discrepancies between determined and the certified concentration values for Po-210, were within the experimental error. Although the separation yields of polonium were modest, they were stable and the results were accurate, proving the robustness of isotopic dilution method.

II.2.5. Application of polymeric membranes to reduce flue gases

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The separation of radionuclides from contaminated soils is a complex and still unsolved problem. The paper presents experimental data on the testing of sequential leaching method of radionuclides from the soils contaminated during accidents or malfunctions of various nuclear installations. The research was performed with soils from the area of Institute for Nuclear Research Pitesti contaminated long time ago with ^{137}Cs and ^{60}Co. The soil specimens were sampled at two depths: 15 cm and 30 cm, respectively. The leaching tests were performed for ^{137}Cs and ^{60}Co removing from the soil samples, using various chemical reagents (mineral acids, ammonium and potassium salts, acidified ammonium oxalate and hot water). The objective was to achieve desirable decontamination of the soil even with considerable loss of its biological potential. Also, it was analysed the influence of soil grains size, soil radioactivity, leaching agents and their concentration, on the decontamination efficiency. The results obtained show the high efficiency of the method developed for ^{60}Co (96%) and 10% decontamination level for

^{137}Cs. The aggressive reagents used were found generally unsuccessful for leaching of ^{137}Cs and thus, more advanced methods should be planned. The Cs^{+} sorption/desorption process is important for evaluation of the environmental risks associated with the radiocesium migration and the feasibility of remediation strategies.

II.2.6. Communicating Radiological Risks to the General Public: How to Get Properly Understood

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This is a story about physics, politics, and public naivety. As we all know, nuclear incidents and accidents do unfortunately happen from time to time. As soon as such an incident / accident happens, the mass media grinding machine starts to present apocalyptic scenarios, most of them going, nowadays, back to Chernobyl. While nobody denies the gravity of accidents like Chernobyl or Fukushima, it turns out that the aftermath is nowhere near the apocalypse presented by the media and the fundamentalist eco-organisations. No one should underestimate the dangers, but no one should overestimate them, either. The problem is that the general public has a tendency to believe more in what media presents than in what the experts in the field are saying. The results are quite dramatic in terms of industry costs: to give only one example, soon after Chernobyl, chancellor Merkel decided to shut down all German NPPs and to pull Germany out of the production of nuclear energy. This was a purely political decision but a quite surprising one for a politician with a strong background in physics and radiochemistry. The present paper analyses how such decisions are taken, what is their emotional impact compared to their real impact. In order to avoid public hysteria and misunderstanding, proper channels for communications must be set up and, for each instance, constant, clear and transparent communications with the general public must be maintained. Also, this must not be done only after an incident, but ways to educate the public in general should also be set up, as it does not help to communicate with an un-educated public. To exemplify, we will present some results on how Chernobyl affected Romania and how it was and still is reflected in the general media and the conscience of the public.

II.2.7. EAGLE – Public Perception on Ionizing Radiation Communication in Europe

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Education, training and information to the general public are key factors in the governance of ionising radiation risks. Communication about ionising radiation with the general public has to be further improved, as highlighted also by the 2011 accident in Japan. An effort is needed to analyse the state of the art and the existing needs in education, training and information, and to coordinate the information and communication about ionising radiation at European level.

The EAGLE project funded by the European Union intends to address this issue by identifying and disseminating good practices in information and communication processes related to ionising radiation. The project aims at coordinating the information and communication strategies related to ionising radiation for the general public, in order to get a better understanding of the effects of ionising radiation, taking also into consideration the lessons learnt from the 2011 accident in Fukushima. EAGLE dedicates activities to each of the three segments involved in the communication process: sources creating the information, channels transferring it to the public, and public itself.
Romania is leading the efforts to define and propose best practices and recommendations for the responsible parties to improve communication in the near future. In order to fulfil this goal, the EAGLE partners must understand what is wrong today in this process, what is missing in the current practice, what was not considered related to the nature and quality of information or in the way it was provided.
There are already available the results of a Belgian survey recently performed by SCK.CEN, which will be completed by the French and Slovenian polls, but to be representative at the EU level, a review of the Eurobarometer surveys measuring the Europeans attitudes towards nuclear activities is also considered. In addition a mental model protocol is prepared to capture the “mechanism” leading to this perception and attitudes related to ionizing radiation.
But Europe has a large diversity of cultures that must be captured to be sure EAGLE recommendations will be appropriate for each country, for each EU citizen. A large network was built in this regard.
Project approach, its activities, an overview of the Eurobarometer results and the major expectations of the project will be in details presented by this paper.

II.2.8. Cherenkov Counting of ^{90}Sr-^{90}Y in a Liquid Scintillation Counter — Instrument Performance Data

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Cherenkov radiation, when produced at significant levels, can be employed for the efficient measurement of radioactivity. Although the production of Cherenkov radiation does not involve the scintillation phenomenon, a conventional liquid scintillation counter can detect and count Cherenkov photons emitted from a given sample in a standard counting vial. Cherenkov counting can be a powerful technique for the radioassay of relatively high-energy beta emitting nuclides, such as ^{32}P, ^{89}Sr, ^{90}Sr-^{90}Y, etc. The main advantages of the technique, compared with liquid scintillation analysis are (i) extreme simplicity of sample preparation, (ii) the ability to count in aqueous systems without use of any organic fluors or reagents that could destroy the sample, leaving it suitable for further tests and (iii) no interference is caused by other radionuclides in the sample with decay emissions that cannot produce the Cherenkov effect, such as ^{3}H, ^{14}C, ^{35}S, etc. The main factors affecting the Cerenkov counting efficiency are color quenching and photon scattering. Here we investigated systematically the reliability of beta activity determination by Cherenkov counting using a TriCarb 2100TR Liquid Scintillation Analyzer from Packard, Meriden, CT. The results obtained for a series of water samples loaded with ^{90}Sr-^{90}Y, with different known emission rates and with varying color quench levels both in glass and plastic vials are presented. The influence of various factors on the counting efficiency was studied, such as: sample volume, vial type, color quench level and sample turbidity. All results showed that better performances are obtained with plastic counting vials,, these being preferred over glass vials for Cherenkov counting.

II.3. Air, Water and Soil Protection

II.3.1. Advances in Gas Purity Investigation by Chromatographic Techniques

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An accurate gas chromatography (GC) method for the determination of gas purity that competes with mass spectrometry is presented. Due to its high impact on gas applications and costs, the purity of gases is a very important parameter. For instance, He 6.0 is significantly more expensive than He 5.0, the difference between them being 10 ppm of impurities, a low content but essential for the quality of pure gases and its specific applications. The pure gases are used in various applications as: carrier gas for gas chromatography (He 5.0, He 6.0), special weldings (Feromix Ar + 18%Vol CO_2), instrument calibration gases (N_2, Ar, H_2, O_2). Classical GC methods using a single column to measure low levels impurities in pure gases is ineffective. Our proposed solution is a configuration of three analytical channels with three different capillary columns (Molsieve 5A for H_2, O_2, N_2, CO, Ar, CH_4; Q-Plot for O_2+N_2, CO_2, H_2O; and Alumina/Na_2SO_4 for C_nH_m) and three detectors (two Pulsed Discharge Helium Ionisation Detector PDHID1 and PDHID 2*;* and a Flame Ionisation Detector - FID). The goal was to analyse impurities under 1 ppm detection limit and this was accomplished. All impurities had eluted with peak shapes which allowed to be measured and over 7 consecutive runs of each mixture the retention times remained the same. The GC system is very important for the determination of trace gases in different mixtures because all components are separated and quantified.

II.3.2. Environmental Protection by monitorisation of impact factors

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The purpose of this work is to highlight the capabilities of ICSI Rm. Valcea laboratory in the field of environmental protection. Several air quality monitoring techniques are summarized, along with a description of the emission and oxidation factors to emphasize the potential protection.
Environmental monitoring describes the necessary processes and activities to characterize and monitor the quality of the environment. Its results are useful to assess the environmental impact, as well as in many other circumstances in which human activities carry a risk of harmful effects on the natural environment. All monitoring strategies and programmes have reasons and justifications which are often designed to establish the current status of an environment or to establish trends in environmental parameters. In all cases the results of monitoring will be reviewed, analyzed statistically and published. The

design of a monitoring programme must therefore have regard to the final use of the data before monitoring starts.
Currently, the fossil fuels characterization laboratory from ICSI Rm. Valcea, has gained an important role on the testing/analysis market, because in addition to the analysis services from energetic point of view, it provides a monitoring of the environmental impact by determining those representative factors for environmental impact.

II.3.3. Application of polymeric membranes to reduce flue gases

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In this work an efficient solution to reduce emissions from burning biogas combustion engine inside a municipal landfill is presented. For the environment, of particular interest are carbon monoxide (CO), nitrogen oxides (NOx) and sulfur dioxide (SO_2). The engine uses a technology that has grown in Romania in the recent years that of using gases resulted from the anaerobe fermentation of waste in the combustion process to produce heat and electricity. Biogas from landfills varies in composition and is a challenge to have an efficient and clean burning. In this gas certain amounts of hydrogen sulfide and water are contained that may damage the producing installations of electricity and heat. Therefore, the too high concentration of H_2S has to be released from biogas. Once treated, the biogas passes into simple heat generating or cogeneration systems to simultaneously produce electricity and heat, releasing pollutants to different amounts. The polymeric membrane will be placed on the engine gas discharge flow path. It should also be placed near the sampling point, computed orifice based on the hydraulic diameter, due to gas homogeneity, relatively low rate and temperatures higher than 200 °C in this area. By placing the membrane above the orifice of sampling, it deteriorates due to the turbulences in the output section of chimney exhaust, the increased gas temperature of 500 °C and the speed of 65m/s.

III. SUSTAINABLE DEVELOPMENT

III.1. Research Infrastructure

III.1.1. Modelling a Linear Ironless PMSM and the Control Structure

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Sensorless control of PMSMs (Permanent Magnet Synchronous Motors) has occupied scientists for a long time. The result of this research is becoming widely accepted by the industry due to its low cost and reliability. However, the majority of today's motor drives are still equipped with some kind of position sensor. The reason is that sensorless controls still have several limitations and is usually more complex than a traditional motor control.
Worldwide energy-saving emission has stimulated extensive application of permanent magnet synchronous motor in industry. PMSMs are of great interest, particularly for industrial applications in the low-medium power range, since it has superior features such as compact size, high torque/weight ratio, and high torque/inertia ratio. Moreover, compared with induction motors, PM synchronous motors have the advantages of higher efficiency; due to the absence of rotor losses and lower no-load current below the rated speed, its decoupling control performance is much less sensitive to the parameter variations of the motor.
The typical construction of a PMSM consists of a three phase stator winding and a solid iron rotor with magnets attached to its surface or inserted into the rotor body. Permanent magnet synchronous motor control system mainly consists of two parts, the main drive circuit and the control circuit. The main drive circuit topology remains basically unchanged, while the study of the control system focuses on the control circuit and control strategies.
The construction of the PMSM results in a magnetic field fixed to the rotor position. Since such machines are not capable of directly starting from the mains, excitation by voltage source inverters (VSI) controlled by field orientation is required. Control techniques such as vector control or direct torque control (DTC) are standard for this type of drives.
The modelling process is focused on techniques which are suitable for observers such as Extended Kalman filters. The linear ironless motor is used to demonstrate that it is possible to achieve good and robust position control without using a direct position sensor.

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.