CONTENT

I. NUCLEAR ENERGY

I.1. Nuclear Reactors

I.1.1 Advanced Reactors Lattice Physics.

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In the neutronic design of the reactor core there is not a single well-defined problem that should be solved. The deterministic solution of the neutron transport problem is obtained in a “steps approach” where for each step a specific computer code is used. Because the deterministic codes used for neutronic core design work in the “group-wise” format of the nuclear data, they require a nuclear data pre-processing step leading to this format which is obtained by a condensation of the continuous energy-dependent cross-sections into a discrete energy-group scheme. This pre-processing step is performed by the “lattice code” working at single fuel pin or fuel assembly level. The necessity to use approximations as well as the lack of the neutron coupling leads to a loss of the method validity in the applications. This study highlights that the “classical” deterministic lattice physics codes are not able to analyse correctly at assembly level the fuel assemblies with strong geometric and/or material heterogeneities. Because at core neutrons design level the lattice physics and the core analysis codes equally share the responsibility of an accurate prediction, it is task of the neutron designer to manage the codes in an appropriate way in order to minimize the impact of the approximations and to evaluate the impact of the lattice physics results on the accuracy of the core neutronic design.

I.1.2. Preliminary Analysis of Tantalum Behavior in Fast Reactors.

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One of the key points in the design and operation of the Lead Cooled Fast Reactors is represented by the corrosion of all internal components, including the fuel claddings, due to the lead coolant.
The previous studies performed for ELSY – Lead Cooled Fast Reactor shown that even a corrosion rate on an average of 20 μm/year could lead to a substantial quantity of stainless steel impurities which are transported to the free surface of the reactor lead coolant.
Various materials and technologies are envisaged to be considered as corrosion resistant. Due to its ideal properties (heavy, very hard metal, high melting point) Tantalum (Ta) is considered a fantastic material for a wide range of use.
The preliminary analysis presented in the paper is focused on the Ta impact on the criticality of a fast reactor as well as on its behaviour (activation and burnup) in various neutron spectra. The studies are performed for various thicknesses of the Ta layer deposited on the fuel claddings.
The criticality computations are performed using the Monte Carlo code MCNPX for the actual, detailed three-dimensional geometry of the ELSY type fuel assemblies.

Also, FISPACT code (an inventory code included in the European Activation System EASY-2005) is used in order to evaluate the activation and the inventory due to Ta irradiation in various neutron spectra and at various time steps taking into account a 5 years maximum residence time of the fuel in the reactor.
A comparison with the T91 stainless steel fuel cladding from activation and inventory point of view is also presented. The inventories are of interest in the radioprotection evaluations as well as in the classification as waste of the respective materials.

I.1.3. Study of Advanced CANDU Fuel Radioactivity Evolution with Irradiation by using Burnable Neutron Absorbers in Fuel Bundle Centre Element Composition.

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Atomic Energy of Canada Limited (AECL) is developing the Advanced CANDU Reactor (ACR) to meet customer requirements (industry and public expectations for safe, reliable, environmentally friendly, low-cost nuclear generation) for the emerging nuclear market over the next 20 years of sales.
The use of Low Enriched Uranium (LEU) fuel with a neutron absorbing centre element allows the reduction of Coolant Void Reactivity (CVR) coefficient to a nominally small, negative value, as a response to an important criticism addressed to the positive CVR characterizing the present CANDU reactors. It also results in higher burnup operation than traditional CANDU designs. Burnable neutron absorbing materials are expected to be an integral part of the new fuel design for Advanced CANDU® Reactor, the studies based on using of a Zirconium alloy centre element (no fissile content) surrounded by a thin neutron absorbing material shell being successfully applied to ACR fuel project.
The paper follows to study the fuel radioactivity at the end of irradiation (EOI) characterizing the ASEU fuel bundle (Advanced SEU fuel bundle with 43 elements, developed in INR Pitesti, and similar to Canadian ACR fuel bundle) by considering various centre element shell compositions and fuel burnup. The ASEU fuel bundle elements (except for the centre element) contain LEU pellets, 2.4 wt% enrichment in ^{235}U.
In order to perform the proposed simulations the following shell compositions have been used: pure Hafnium and equal percents mixtures of burnable absorber and Zirconium oxides, the considered burnable absorbers being Gadolinium, Dysprosium, Holmium, Erbium and Hafnium (taken into account according to IAEA WIMS Libraries Update Project).
To obtain a significant and broad range of results, several centre element shell thicknesses (0.5 mm, 1 mm, 2 mm and 3 mm) have been considered. The fuel burnup calculations have been performed for 8000 MWd/tU, 16000 MWd/tU and 20000 MWd/tU, respectively. Fuel elements irradiation was modelled by assuming constant power for entire irradiation period, the considered power being about 50 MW/kg U.
The burnup was simulated by means of ORIGEN-S code, included in SCALE5 programs package. The spectral neutron cross-sections weighting factors used as input in ORIGEN-S data were given by WIMS-D5B multigroup transport lattice code calculations.
The paper provides EOI fuel radioactivity comparisons both for different burnable absorber shell compositions and a specific centre element shell composition but with different thicknesses, by taking into account the considered fuel burnup.

I.1.4. Preliminary Reactor Physics Studies on Using Advanced Fuel Bundles in CANDU

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According to the IAEA predictions on Uranium demand and worldwide resources, improving fuel burnup became a serious issue, especially for natural Uranium fuel cycles. Given the CANDU reactor adaptability for using new fuels, several ideas appeared based on the complex technical, economic and political environment. First of all, using the spent PWR fuel as DUPIC/RUFIC was proposed by AECL, Canada and KAERI, Korea. Converting the weapon grade Plutonium from U.S.A. and/or Russia to MOX and dispositioning it in CANDU was also studied by AECL. Since the Thorium natural abundance is about three times the abundance of Uranium, using Th-based fuels in CANDU was investigated by AECL in the early 1980's and recently by an AECL-supported consortium of Chinese entities (TQNPC, CNNFC and NPIC). Most of the papers presenting the above projects contain extensive details on reactor physics calculations concerning the use of new fuels in CANDU. Our goal was to investigate these projects in the aim to propose a more suitable fuel project for the existing and future CANDU units in Romania based on the CANFLEX bundle geometry and U-Pu-Th MOX. Twelve projects were considered and cell calculations were performed using the WIMS and DRAGON transport codes. The key parameters were the maximum fuel burnup, the radial power distribution, the minor actinides yield, as well as the void reactivity coefficient. The results are encouraging, recommending several project for further reactor core investigations.

I.1.5. Neutron Transport in Fluctuating Stochastic Finite Media with Sinusoidal Internal Source of Energy

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The neutral particles transport problems in an inhomogeneous medium have received a considerable attention over the last decades. These problems become especially interesting if the medium is treated as a stochastic object. It is reasonable to characterize the medium in a statistical sense that requires specifying the probability distribution of the fluctuating property with its spatial and temporal correlations, etc. One assumes that the stochastic equation describing the particles transport can be solved in principle for random medium samples or realizations and that one can carry out the solution averaging over a large number of such realizations. Such stochastic models enjoyed considerable popularity in different applications in several fields of physics, including nuclear technology, astrophysics and climatology, as follows: radiative transfer through mixed regions in inertial confinement fusion pellets, transport of neutrons through burnable poison pellets in nuclear fuels, transport of neutral atoms in turbulent tokamak plasmas or atmospheric radiative transfer in global climate modelling and remote sensing, for instance.
In present work, the neutron transport problem is considered in a scattering and absorbing planar fluctuating random medium with general boundary conditions and sinusoidal internal source of energy. The medium is assumed to be continuously stochastic medium, where the total cross-section of the medium is assumed to be a continuous random function of position,

with fluctuations about the mean taken as Gaussian distributed. The joint probability distribution function of these Gaussian random variables is used to calculate the ensemble-averaged quantities, such as reflectivity and transmissivity, for an arbitrary correlation function. The problem is solved in terms of the solution corresponding to a source-free problem with simple boundary conditions; in order to get the analytical deterministic solution the Pomraning-Eddington approximation was used. The average solution is obtained by using the Gaussian probability distribution function. The average partial neutron fluxes are calculated in terms of the source-free problem albedoes. Numerical results were obtained for pure-triplet scattering considering specular and diffused reflecting boundaries.

I.1.6. Brayton Cycle in Generation IV of Gas-Cooled Nuclear Reactors

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Generation IV reactor systems are designed to achieve durability and to improve economic competitiveness of Nuclear Power Plants. The very high-temperature gas-cooled reactor concept has significant advantages for achieving optimal use of natural resources and minimisation of waste, proliferation resistance, nuclear safety excellence, cost saving and also development of hydrogen production processes and desalination of sea water. This paper presents the advantages of using a direct or indirect Closed Brayton Cycle for energy conversion in gas-cooled reactors. In indirect Brayton cycle the gas transporting heat from the reactor core is used to heat the fluid in a secondary system in this case, the energy conversion fluid is different from the fluid circulating in the reactor core. When using direct cycle, the gas circulating in the reactor core is sent directly to the turbine. The Brayton Cycle is capable of attaining high thermal efficiencies. As a working fluids it can be used helium or different binary mixture working fluids of He–Xe and He–N_2. Using a Brayton Cycle as a power conversion cycle entails a series of advantages, such as: increase of plant efficiency due to very high temperatures in the turbine entry and because the working fluid does not change the phase; smaller sizes for the gas turbine; lower costs; a reduced number of components resulting in a more simplified scheme and shorter installation time and also it is possible the development of modularity concept. In this paper it is also presented a comparison between the direct and indirect Brayton Cycle, highlighting different factors that could influence the thermal efficiency such as reactor exit temperature, pressure ratio or primary loop pressure losses.

I.1.7. Necessary Improvements for CANDU NPP Design in Order to Attain the Performance of the Generation III/III$^+$ NPP Designs

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CANDU 6 NPP design even it is an older design, it have consistently proven to be competitive with other types of NPP designs, while offering unique advantages to their operators due to its design features and unique characteristics as, for example:
- on-power refuelling,
- reactor core design with natural uranium fuel and heavy water for moderator and coolant,
- two fully capable safety shutdown systems.

During the last decades CANDU 6 had an evolution in order to meet industry and public expectations of NPP generation to be safe, reliable and environmentally friendly.
The first enhanced design of CANDU 6 is EC **6 (Enhanced Candu 6)** which is a Generation III design which retains the basic features of the CANDU 6 design and incorporates the improvements resulting from the experience and feedback gained through the development, design, construction and operation of 11 CANDU 6 units operating in 5 countries and the innovative features and state-of-the-art technologies that enhance safety, operation and performance.
ACR 1000 is the next phase of the evolution of CANDU 6 design and it is an evolutionary, Gen III^{+}, 1200MWe class pressure tube reactor with water as coolant and heavy water as moderator. The main improvements of this design are the new passive safety features, the ability to burn alternate fuels, less refuelling and lower spent fuel volume per MWh through use of low enriched uranium (LEU).
This paper analysis all the improvements made to these new enhanced CANDU 6 design in order to select and present the improvements which can be implemented to the CANDU 6 design which will be executed at units 3 and 4 from Cernavoda

I.1.8. Using Q-Space Focusing in Thermal Neutrons Spectrometry to get Improved Resolution Performances

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In neutron spectrometry the optimal use of the available neutrons is of the greatest importance. The general philosophy to do it is to maximize the neutron flux at sample by using neutron guides, supermirrors, or spatial focusing effects involving flexible configurations and curved crystals.
A different approach was developed in INR aimed to obtain the required optimum experimental conditions not by getting focused beams at sample or anywhere else, but only by decreasing as much as possible the scan variable variances. A Q-space focusing configuration is characterized by high-resolution even when no spatial focusing exist at sample position or anywhere else or is used quite open beam without any Soller collimators; only coarse collimators should be used to reduce the background level. This is possible just by decreasing the scan variable variances.
The following steps should be followed:
- to define the scan variable;
- to define the significant spatial variables as are the monochromator and sample width, for example;
- to express the scan variable, by using specific geometry characteristics and the existing correlation between variables, through the spatial variables;
- to cancel the significant contributions to the scans variable variances by canceling the important variables coefficients, from the scan variable expression; when correlation between variable exists, as is the case for these kind of experimental configurations, such a coefficient has more then one term, not all of the same sign, appearing the possibility to be cancelled and therefore to cancel entirely the corresponding contribution of this spatial variable to the line-width.

The paper shows the method, its advantages and peculiarities and also some results obtained in INR.

I.1.9. Out of Reactor Testing Technologies of F/M Heads – Romanian Experience. Functional Testing of Telescopic Cylinder RAM Assembly

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CANDU reactors are PHWR type using natural UO_2 fuel and heavy water as a coolant and moderator. An important feature of this reactor type is it doesn't require shut-down to refuel. The refuelling is automatically performed during the operation by two loading / downloading machines (Fuelling Machines Heads, F/M Hs).
Proper and safe operation of F/M Hs is an important condition that determines the efficient and secure operation of CANDU NPP. For this reason F/M Hs require screening and rigorous testing before their installing into the reactor, in order to guarantee their proper functioning during operation in the fuel channel. This is an implicit condition of nuclear safety.
F/M Hs is an electro-mechanical-hydraulic system composed by the snout assembly, the separators assembly, magazine housing, telescopic cylinders assembly (TCA). It performs the loading/unloading of the CANDU reactor, under operational and radiological safety conditions.
F/M Hs' testing consists of separate tests and adjustments on the F/M head components and also in achieving a number of fuel changes in the automatic mode at the pressure, temperature and flow condition similar to those from the channel of the reactor core.
The F/M head testing can be grouped as:

- the functional testing of the TCA;
- performing of the pre-acceptance and the acceptance tests (cold and hot).

Functional testing of the ram assemblies is about 30% of the entire test comprising the Fuelling Machines Heads.
The functional test for TCA is a checking of all mechanical components of the telescopic cylinders assembly, to certify its safety operation before installing on the F/M head in order to execute the final acceptance test.
The paper shows some results obtained during the TCA testing (tables and charts) aimed to certify the successful testing of F/M Hs.
During these tests, verification of correct execution working by TCA's telescopic cylinders, and nonconformities occurred during testing have been solved according to the functional feature, to their finding out or after completion test.
Were also carried out operations that have demonstrated the safe operation in terms of reliability and mechanical strength of TCA components, ranging between time limits allocated to these operations.
Data about operation obtained during the F/M Hs testing, as well as the experience gained by testing staff formed into a database used to improve testing facilities and team training for co-operation in commissioning of the Fuel Handling System from Cernavoda NPP Units 3 and 4.
Fuelling Machines testing at INR Pitesti is a national and European premier, the potential of testing team and rig capacity from INR Pitesti for such tests, being pointed out by the AECL specialists who attended to INR and supervised the test activities of F/M head no. 4.

I.1.10. Neutronic Studies for Maximizing the Specific Activity Of 99Mo Produced By Activation In TRIGA SSR.

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In current medical practice, the most used radio-nuclide in different types of scintigrafic explorations is ^{99m}Tc, allowing the laboratories to rapidly mark a wide range of pharmaceuticals delivered as 'marking cases'. In this way, one can obtain radio-pharmaceuticals with selective fixing inside the investigated body organs. Molybdenum production can be a solution for the future in the utilization of the Romanian TRIGA, taking into account the international market supply needs.
There are two main methods to produce ^{99}Mo/^{99m}Tc in a nuclear reactor:

-the fission of an uranium target. Producing the ^{99}Mo radioisotope by fission imply high neutron fluxes, expensive processing facilities for handling the fission products of Uranium, and creates important nuclear waste radioactivities;

-irradiation of a natural or enriched (in ^{98}Mo) target, called the neutron activation production method. It leads to low specific activities of ^{99}Mo and also low waste activities, and does not require expensive handling facilities.

The paper analyses two possible irradiation target configurations, with 40 pellets and 181 pellets respectively. The calculations are performed with MCNP, searching the locations of the targets in setups that maximize the ^{99}Mo in the present TRIGA core configuration. It involves the analysis of neutron shielding materials and geometries that would allow making use of the resonance integral of the reaction ^{98}Mo(n,γ)^{99}Mo. Calculations are done both with natural metallic Molybdenum pellets and with enriched pellets.

I.1.11. Diren Burnup Histories' Comparisons for CANDU and Advanced CANDU Fuel Bundle Designs

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The paper aims to offer a comparison between burnup histories for CANDU and advanced CANDU fuel bundle designs, generated with a 3D diffusion code DIREN. The standard CANDU fuel design with 37 rods (NAT-37) and Slightly Enriched Uranium (SEU-43) fuel design with an enrichment of 1.1%U235 were taken into account. Recent enhancements added to the DIREN code along with a graphic interface gave us the possibility to simulate automatic refuelling operations up to 700 Full Power Days (FPD) for 37-NAT fuel design and about 1000 FPD for SEU-43 fuel design in a standard CANDU-6 core. A key parameter in burnup histories generating is the ratio of the outer ring linear power to the average linear bundle power. The obtained values were 0.052 and 0.062 m^{-1} for NAT-37 and SEU-43 fuel bundles, respectively. The NAT-37 refuelling strategy was based on the use of 8 bundle shift scheme, contrarily to 2 or 4 bundle shift scheme used for SEU-43 fuel design.
The calculations revealed that SEU-43 fuel design (based on CANFLEX bundle geometry) allows a better power distribution control over the reactor core through closer values of Power Peaking Factors (PPFs) over the fuel rod rings. Also, the using of SEU-43 fuel design along with 2 bundle shift refuelling scheme determined that the maximum linear powers on the outermost fuel ring being inside of operation margins. Other relevant reactor physics amounts like effective multiplication constant, reactivity, discharge and average burnup are comparatively presented, with respect to the time steps.

I.1.12. Modern Control, Monitoring and Diagnosis Solution for Spent Fuel Bay Cooling System.

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Due to the complex technological processes involved, and because of functional interdependencies which appear between subsystems within ensembles of high complexity lead to the implementation of modern techniques of monitoring, control and on-line diagnosis, which provides process robustness in case of a failure.
The paper deals with various modern control methods of spent fuel cooling system, which has the primary role of removing heat generated by the discharge bays, reception bay, defective fuel storage bay and spent fuel storage bay of CANDU NPP.
These solutions enable remote monitoring and control of equipments or parts of them, the operator using a human-machine interface displayed as synoptic diagrams form, which are characterized by clarity and concision providing information about the process involved, in order to avoid confusion.
By implementing this solution system operation has a high level of freedom regarding psychological or physical conditions of human experts (a human expert may decide otherwise, e.g. stress conditions) and reasoning behind such a system are more consistent and reproducible than the human expert.
Any event which occurres during operation may be described in a post failure analysis, the system maintain a logged history of monitored processes, providing full information necessary for a relevant analysis of the events occurred.
In addition, this solution increases the level of safety in operation of the investigated nuclear system, as well as meeting recently imposed cost optimization requirements, with both personnel and maintenance, by reducing the allotted time and thus helping reduce the radiation doses received by qualified personnel.

I.1.13. Implementing CANDU Advanced Control and Instrumentation in Cernavoda NPP future units.

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Advanced control and instrumentation (C&I) concept starts from the base of proven CANDU reactor technology coupled with demonstrated new features in order to enhance economics, safety, operability and maintenance. A key strategy for implementing advanced C&I and information technology systems at future Units 3 and 4 at Cernavodă NPP is to integrate operations and maintenance (O&M) functions. Hence, in this paper, is presented an approach which consists of a particular emphasis on enhancing the control center and the systems that support it. The control center integrates automated and manual operations for the various process systems, with operators handling information related to: nuclear instrumentation (measurements from flux detectors that allow reactor power determination), process

instrumentation (system and component information), radiation monitoring, and other specialized monitoring systems. All of this is driven by the innovative “control from the console” which provides the human system interface (HMI), in addition to plant supervisory control functionality. Implementing this advanced instrumentation and control will be significantly improved and the plant monitoring information will be used in order to enhance the maintenance and technical support areas of the plant. The advanced C&I systems can provide the platform for final integration of the process and safety system controls, both with each other and with the operating and management team. This integration is a key factor in achieving the operating and safety goals for the plant.

I.1.14. Long-Term Operation for Generatinon II/III NPP.

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In the period 2010-2050, the successful operation and management of Generation II beyond their originally foreseen lifetime, will be vital to the security of supply of electricity in Europe. The Strategic Research Agenda (*SRA*) vision is to move towards 60 or more (60+) years of safe and economic operation of nuclear power plants. The next evolutionary design of LWRs, Generation III/III+, will be deployed over many decades and will represent a large part of the worldwide fleet throughout the 21st century. At the horizon 2040-2050, the nuclear fleet will consist of a mix of technology: GEN-II, GEN-III/III+ and GEN-IV prototypes.
Recently, a Technology Working Group (*TWG*) was constituted within *SNE-TP* in the area of R&D on Generation II and III Reactors. The TWG will coordinate, prioritise, monitor and report on Gen II/III R&D activities described in the SRA. The *NULIFE* Network of Excellence (NoE) which is proceeding to permanent entity in 2011 (*“NULIFE – Nuclear Plant Life Management International Nonprofits Association”*) will play an important role inside the TWG in proving the operation mode and in implementing some key areas of the Gen II and III of the SRA. *The Road map for the safe and economic operation of current and future GEN II and III reactors (OPERA)* was prepared within *NULIFE* to support *SNE-TP TWG on Gen II/III* industrial initiatives. *OPERA* defines the priorities, targets, project portfolio and future scenarios.
The INR Pitesti is partner both in *NULIFE* and in the Technology Working Group *SNE-TP TWG on Gen II/III.*
In Romania, the Cernavoda NPP was originally conceived as a 5 units Power Generating Station, out of which only 2 units are now operational. For another two (U3 and U4) a turn-key contract for construction and commissioning was signed in 2008. It was discussed a long-term program for nuclear power development. The main reactor concepts for the next 2–3 decades have been determined. They are heavy water-cooled reactors of Generation II (CANDU 6 – Cernavoda NPP Units 3&4) and probable Generation III/III+ NPP.
The electricity annually generated by the Cernavoda NPP Units 1 and 2 represents approximately 18% of the overall electricity production of Romania. Shortly after Cernavoda NPP Unit 1 was put into commercial operation, on December 1996, the management became well aware of the advantages of implementing a strong Plant Life Management Program in the early stages of the plants life, (http://www.neimagazine.com/story.asp?storyCode=2022147).
The safe and reliable long term operation (LTO) of a Nuclear Power Plant requires the systematic and proactive identification of degradation and aging mechanisms of critical SSC that are not addressed by the normal plant Preventive Maintenance strategies, and mitigation of these inherent major problems which will otherwise impact on the Plant assets. The LTO

program, proposed to be applied at the Cernavoda NPP, was developed based on WANO recommended best practices detailed in INPO-AP-913 and it is supported both by the experience of CANDU 6 owners and by the results of research conducted within INR Pitesti.
The objective of this paper is to present the R&D activities to support LTO for Generation II / III Nuclear Reactor.

I.2. Nuclear Safety

I.2.1. Moderator Flow Analysis Inside Calandria Vesel of a CANDU6 Nuclear Reactor.

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In CANDU6 reactors the main moderator system functions are to slow down fast neutrons and to continously remove heat from calandria vessel by moderator circulating. It maintains a constant moderator temperature in the Calandria vessel, typically between 60ºC to 80ºC. The target of Computational Fluid Dynamics (CFD) analyses are both normal operating conditions as well as accident conditions including severe accident conditions. In this paper, in a first step, is analyzed moderator flow inside Calandria vessel for steady-state normal operating conditions. We consider a thermal-hydraulic CFD analysis considering thermal phenomena starting from a 3D distribution of generated power which are transferred to main moderator system. We consider that this steady-state analysis is a good start for future transient analyses both in Design Basis Accident (DBA) and in severe (SA) accidents. A good understanding and detailed information regarding moderator fluid flow and temperature distribution inside Calandria vessel is significant in Nuclear Reactor Safety (NRS) analyses. Boundary conditions are established by modeling moderator flow through inlet nozzles. Velocity profile at entrance in main moderator system is provided. For 3D power distribution we have a theoretical approach. Flow turbulence model is RANS k-ε, mono-phase. Once the boundary and operating conditions are setup the problem of moderator flow inside calandria vessel is solved. The results are the spatial 3D distribution of some representative field variables like temperature, enthalpy, velocity and density. In paper the importance of inertial, bouyancy (Archimedic) and gravitational forces are emphasized. The results were obtained using an Open Source chain of codes: Salome-MECA, Discretizer, OpenFOAM and Paraview .

I.2.2. An Enhanced Nuclear Security Culture-A Challenge.

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The problem of security and physical protection of nuclear installation and nuclear material accountancy, the security of radioactive sources, as well as the security in transport of radioactive material and other nuclear materials have to be approached as an integrated system to protect against nuclear terrorism or other malicious acts. IAEA Vienna can help to identify threats and vulnerabilities related to the security of nuclear materials and other radioactive material.
Taken into consideration the above statement, an enhanced nuclear security culture will provide a much greater assurance that an integrated nuclear system will fulfill its main functions: to detect, to delay and responding to potential theft, sabotage, unauthorized access, illegal transfer of other malicious acts involving radioactive material and the associated facilities and transport.

The paper will present some consideration to be made on the enhancement of nuclear security measures of the Romanian Nuclear facilities and a presentation of some security aspects taken during transportation of radioactive material in Romania, the vulnerabilities of such transportation routes.
This paper presents certain results of the IAEA Scientific Research Contract on The State Management of Nuclear Security Regime (Framework) where the main author of this paper is the Chief Scientific Investigator of this contract concluded with IAEA Vienna on Safety and Security Regime (Framework).

I.2.3. Extrapolation of Monotonous Functions using the Linear Correlation Coefficient

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Given the real functions, continuous and double derivable $y = f(t;a,b,c...)$, of variable *t* and parameters *a*, *b*, *c*… describing the evolution in time of a phenomenon (physical, economical, social etc.) one ascertains, related to prognosis problems, that $y' = df(t;a,b,c...)/dt$ offers information regarding the short-duration tendency, while $y'' = d^2 f(t;a,b,c...)/dt^2$ deals with the long-term tendency.
Knowing the analytical expression $f(t;a,b,c...)$ and with the help of a set of measurements (t_i, y_i), $i = 1,2,...,N$, the most probable values of parameters *a*, *b*, *c*… are determinate, for instance, by the method of smallest squares, solving the system of equations: $\partial S/\partial a = 0$, $\partial S/\partial b = 0$, $\partial S/\partial c = 0$…, where $S = \sum_{i=1}^{N} [y_i - f(t_i;a,b,c...)]^2$. Now, with $f(t;a,b,c...)$ thus explained one can extrapolate the function at the moment $t = t^*$.
The paper presents a possible method of extrapolation / prognosis for monotonous functions based on the utilization of the linear correlation coefficient. While having the advantage of not requiring knowledge of the analytical (explicit) expression of the function that gave initial values, the computing formulae proposed by the authors can be implemented in computer programs for automatically processing and interpretation of experimental data or of those taken over from various computing tables.
More exactly, knowing the set of *N* measurements (t_i, y_i), $i = 1,2,...,N$, for a unknown function $y = f(t)$, and the linear correlation coefficient $R_{y,t,N} \in [-1,1]$, then the extrapolated value $y^* = f(t^*)$ can be approximated as the solution y_{N+1} of the equation $R_{y,t,N+1} = R_{y,t,N}$, where $t_{N+1} = t^*$. The paper also offers the calculation formula of the relative error dy_{N+1}/y_{N+1} for the evaluated value y_{N+1}.
In order to check and sustain the above assertions, the paper presents the numerical results obtained in the case of extrapolation by means of the linear correlation coefficient for four monotonous increasing functions, strongly non-linear, i.e.: $f(t) = t^2$, $f(t) = t^3$, $f(t) = exp(t)$ and $f(t) = ln(t)$, the inevitable errors being more than acceptable.

I.2.4. Laminar Flow with Phase Exchange for Real Liquids – A Simplified Theoretical Model.

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In a Nuclear Power Plant with CANDU Reactor the losses of heavy water (D_2O steam, at the pressure p = 100 bars and the temperature t = 300 ^oC) from the fuel channel must by less of 10 g/24h.
Can a qualitative characterization be made for the liquid losses through the imperfect tightness of a technological facility under high pressure and temperature? May a quantitative evaluation be made for the masse/volume of the losses? Fundamental answers to these question cannot be given only after the detailed analyses of the physical phenomena which happen to the imperfect tightness level (pore, fissure etc.), subject with which this paper deals.
More exactly, the paper presents a simplified theoretical model of the laminar flow for real liquids through the imperfect tightness (pore, fissure etc.) of a technological installation working in high pressure and temperature conditions, taking into account the liquid-steam phase exchange. The physical-mathematical model allows establishing the formulae to evaluate the masse/volume flow of the fluid losses, i.e.: the average masse flow of the steam at the saturation pressure, the average masse flow of the steam to the going out in atmosphere and the “hydraulic diameter" of the imperfection.
The model was verified and validated at the Institute for Nuclear Research (SCN) Pitesti, Romania, during some adequate tests with light water (H_2O). Based on this theoretical model, at the SCN Pitesti has been developed an original method to detect and measure the steam losses through the imperfect tightness of a technological installation working in high pressure and temperature conditions, method which is the object of the Romanian Invention Brevet RO 118230 B8 “METHOD AND EQUIPMENT FOR STEAM LOSSES DETECTION” and which was used in the "Channel closure testing" activities for the Nuclear Power Plant U2 Cernavoda, Romania.

I.2.5. The Evaluation of the Human Performance Analysis Methods in Nuclear Facility Operation.

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The studies of the major accidents from nuclear installation denote the fact that these result not only from failures of the equipments, but for the most part result from a combination between human and the organizational errors and equipments failure. In the last years it have been agreed that many technical issues from nuclear field were resolved but the issues of the human performance are yet in the study and research phase.
In this paper a study of the human performance analysis methods (THERP, ASEP, HCR, SPAR-H, MACHINE, WPAM, BORA, i.e.) is performed. The main objectives of this study were the following: to identify the analysis tools; to establish the principles of the methods, the necessary data for analysis and framework; to identify the mod of the applicability; to determine the advantages and disadvantages of each method.

After the evaluation of these methods from point of view of the applicability in the investigation and quantitative and qualitative analysis on the human action, the deficiencies and weakness of these methods were indentified. Using these results, the justifications were obtained and a framework was created to develop new methods of human performance analysis.
A general conclusion of results of the paper is that the effects and causes of the human performance are not included explicitly in safety assessment.

I.2.6. The Influence of the Safety Systems Activation Instant on the Containment Failure Probability.

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Accident management is the overall capability of the plant to respond to and to recover from an accident situation. Accident management measures or strategies may be preventive or mitigative or both. The accident management measures or strategies to mitigate the consequences of a severe accident are referred to as severe accident management. These actions mitigate core damage and protect fission product barriers. Guidance is provided in the Severe Accident Management Guidelines (SAMG).
SAMGs contain set of strategies for all major challenges due to severe accident phenomena and a guided method for choosing optimal mitigation strategy under current plant conditions.
The Level 2 PSA (Probabilistic Safety Analysis) is an approach which provides a structured assessment of the possible accident sequences that could occur following the onset of core damage and gives insights into which of the phenomena that could arise have the greatest potential to lead to containment failure or bypass resulting in a release of radioactive material to the environment. One of the main aims of the Level 2 PSA is to provide a technical basis for the identification of the plant specific severe accident management measures and quantify their effectiveness.
This paper presents a fine time analysis of the risk of containment loss of integrity due to hydrogen combustion in function of the safety systems activation times. This calculation represents a solution of the step 3 of a benchmark exercise for dynamic reliability methods applications. The exercise concerns hydrogen combustion risk assessment in case of water injection during in vessel core degradation phase for a French 900 MWe PWR. This problem is considered relevant and representative for nuclear safety.
By using this example, the paper intends to give an idea of the type of information that could be obtained from a probabilistic safety analysis applied for SAMG developing.

I.2.7. Modelling of Ageing Effects in Initiating Events Frequencies.

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A frequent method used for investigation of ageing problems (besides analysis of operating experience and experts' opinion) is the probabilistic technique. It is very well known that a standard Probabilistic Safety Assessment (PSA) supposes the failure rates of the components, frequences of initiating events (IE), sysyems' unavailabilities and the frequency of core damage

as constant in time. The merit and advantage of an ageing PSA (APSA) is that can explicitely modell the ageing effects in these level 1 PSA important parameters. In order to introduce the ageing impact in PSA models, the unavailability at component level (on reliability law basis) for which the failure rate is time dependent, must be calculated. For estimation of aged component unavailabilities, several mathematical models have been developed: linerar model, Weibull model, stress-resistance models, etc. The linear ageing model supposes that the total failure rate of the component consists of two independent failure rates: one without ageing and one which describes the ageing linear part.
An important level 1 standard PSA parameter, that can be affected by ageing of plant components is the IE frequency. This is derived by using: the fault tree method, the method based on piping specific design data and generic data, the Bayes method that combines the station specific experience with the IE frequencies appeared to other similar station, non-informative prior distribution, etc. To see the ageing effects in estimation of IE frequencies, this paper will apply the linear ageing model for estimating the frequencies of some initiating events. These have been included in Cernavoda Unit 1 level 1 PSA, study performed at INR Pitesti. The paper presents the assumptions used in analysis, values of IE frequencies without ageing and with ageing for age of components at 10, 20, 30 and 40 years. An increase of IE frequencies (as expected) has been observed, the bigger values being obtained for age of components at 40 years.

I.2.8. Database Applications in Support of Risk Analysis at Nuclear Reactors.

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As the complexity of safety assessments for nuclear reactors increased continuously, there has been a need for managing the information, on the one hand on systems and components properties and behavior, and on the other on operation data and events. Various types of databases for nuclear reactor safety are possible and actually exist, created by reactor vendors, by operating and research organizations, as well as international agencies. The focus of the paper is database management in areas connected with safety analysis. It deals with the database creation, organization and data retrieval with software systems designed to support Probabilistic Safety Assessment (PSA) studies. PSA is a tool that can be used to assess the nuclear risk of the plant but can also target system design, configuration decisions and improvement of operation. Ideally, failure data used for safety and reliability analyses should be based on site-specific data although the creation and maintenance of extensive databases with generic information on components failure is very helpful in starting a PSA project. The paper offers an overview of how this task is approached in PSA Level 1 for nuclear reactors. Also, it treats the topic of data managements systems that deal with severe accident information for risk evaluation and mitigation strategies, in support of PSA Level 2 applications.

I.2.9. On-Line Diagnosis Procedures for Electromechanical Equipments from Nuclear Power Plants

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The paper contains a general presentation concerning potential faults and on-line diagnosis of basic electromechanical equipments (power transformers, induction motors, power electric sources), from nuclear plants.
The principal electromechanical equipments which ensure Nuclear Safety of a nuclear power plant are induction motors, which drive the pumps for cooling of the nuclear reactors, respective independent energy sources like accumulator batteries and diesel generators.
Power transformers are one of the most expensive components in a transmission and distribution electrical system and the failures of such transformers can result in serious power system issues of nuclear plants, so faults' diagnosis for power transformers is very important to ensure the power supply system run normally.
The main available techniques for on-line diagnosis of the power transformers are: a) Vibration signals from the transformer tank, widely used for analysis and extraction of the fault characteristics. Signal processing advanced methods, including Wavelet Transform (WT) have been presented to extract vibration features in recent years; b) Oil temperature monitoring for diagnosis. Oil temperature increase can give the information about the condition of power transformers; c) Partial discharges measurement of transformers. Electrical measurements of individual discharges are made with sensors on the bushing tap, neutral or inserted into the tank. Also partial discharges studies uses UHF interference detection; d) On-line monitorization and dissolved gas analysis (DGA) from insulation oil. The most often DGA analysis systems are based on different types of neural networks (NN), like MLP-NN (Multi-Layer Perceptron NN), BP-NN (Back-Propagation NN), GR-NN (General Regression NN), etc., and other techniques like the fuzzy logic, chromatographic analysis method, expert systems, decision trees, support vector machine, evolutionary programming and many other techniques.
The on-line diagnosis of induction motors is provided by electrical signature analysis. Potential faults like dynamic eccentricity, stator-winding faults, rotor bars damage, mechanical faults (shaft and bearings fault), load problems and unbalanced voltages of a system may be diagnosed from electrical signature analysis. Various analysis techniques of motor's current signals include study with Fourier transform, wavelet analysis. Other preventive diagnosis methods include acoustic noise analysis, temperature measurement, infrared measurement, radio frequency emission monitoring and partial discharges measurement.
The stationary accumulator batteries can have two main faults, low charge or low capacity. These are from charge system or degradation of accumulator cells. The state of charge of a battery is still very important. Battery pilot cell testing, on-line monitor systems of accumulators' temperature and voltage and of charge-discharge current are the main diagnosis techniques.
The faults of Diesel-electric generators are classified in two categories: faults of electric generator and Diesel engine. Preventive diagnosis of Diesel engine faults consists in on-line monitor and analysis of engine operating parameters.
In nuclear plats, to ensure Nuclear Safety very important is faults' diagnosis in incipient phase. The advantages of electrical signature analysis for electromechanical equipments are:

- On-line, no stoppage required;
- Remote monitoring, no need to approach the motor procedure of capturing a motor's current & voltage signals and analyzing them to detect various faults;
- Accurate detection of electrical & mechanical problems;

I.2.10. RIH 35% RIH 35% Large Break LOCA Analyses for the RU43 Fuel Core of CANDU 6 Reactor.

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The recently developed advanced CANDU fuel bundle, RU 43(Recycled Uranium) has major design improvements over the standard 37 elements fuel bundle with natural uranium (NU 37).
The RU-43 design is a 43-element fuel-bundle assembly offering improved operating and safety margins, compared with the standard natural uranium CANDU-6 bundle with 37 elements (NU-37), for operating CANDU-6 reactors.
The RU-43 bundle consist of 2 fuel element sizes: small-diameter elements in outer and intermediate rings, and larger diameters elements in the inner and centre rings. The small-diameter elements in the two outer rings allow the peak element ratings in the bundle to be reduced by 20% in comparison to the standard NU-37 bundle. The larger-diameter elements in the inner rings of the bundle compensate for the fuel volume lost due to the smaller-diameter outer ring elements.
The isotopic specification of RU fuel is very similar to slightly enriched uranium fuel, but with higher concentrations of 234U and 236U than the concentration found in enriched fuel derived directly from natural uranium. To maintain compatibility of the new bundle with the existing CANDU-6 reactor systems, the basic overall dimensions of RU-43 fuel bundle were designed to the same as those of the NU-37 bundle.
Effect of these design changes of RU 43 bundle on CANDU 6 reactor safety has been assessed for the RIH 35% LB LOCA.
CATHENA models for a PHTS circuit and fuel channel analyses have been developed for a CANDU 6 reactor loaded with RU 43 fuel bundles, and a demonstration analyses results for a RIH 35 % large break LOCA are presented in this paper.
The results of RU 43 core thermal-hydraulic behaviour in RIH 35% LB LOCA are discussed comparatively with the results of NU37 core thermal-hydraulic behaviour.
In general, the results of the CATHENA circuit analyses presented on this paper fall within the foreseeable expectations with reasonable trends.

I.2.11. Reliability Analysis For Nuclear Safety Adaptive Redundant Systems

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An approach have been developed addressing to the problem of optimum fault tolerant philosophy using fault detection and isolation function for redundant electronic safety systems.
The motivation of our study is the desire to develop a reliability analysis for a redundant safety system, having fault tolerant dynamic redundancy channels using self evaluation, reconfiguration and adaptation functions.
We present the working diagram and Analytical Reliability relations between process inputs X_m and system outputs Y_r for dynamic redundancy operating electronic channel, having software and hardware reconfigurations skills.
Also we present in this paper Analytical Reliability relations for a safety redundant electronic system having adaptive dynamic redundancy channels.

The paper describes thus a system working in a distributed control structure having local reliability resources management with dynamic redundancy, software and hardware reconfigurations skills, information's exchange between redundant channels and flexible voting decision blocks for safety. System performance indicators, safety indicators, reconfiguration indicators can be provided by every local station to all other redundant stations and to a central analyzer/monitoring stations.
The contribution of this paper is an interdisciplinary research in the direction of reliability modelling for modern redundant adaptive systems, working with input errors, operating errors and hardware faults. The analysis technique used can be important in distributed control critical electronic diagram implemented in nuclear applications, for increasing safety and stability parameters.

I.2.12. Methods for On-Line Monitoring and Risk Evaluation for NPPs Pipeline Systems

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The recent nuclear research issues need new technologies and continuous progress in finding different and efficient solutions for sustained and upraising energy demand. The increasing trend in energy consumption and occurring of new and large consumers, especially from Asian countries, imposes finding of new means for energy clean, large scale and sustained production.
NPPs pipeline systems availability is necessary to be continuously monitored and supervised; in the same time the safety of the nuclear energy production was under surveillance. Actual development of the new technologies and discoveries of new materials and efficient technological processes offers the opportunities for appropriate implementation and use of them in the NPPs system configuration and functioning/operation.
The new technologies and scientific discoveries, also the international cooperation, offers the opportunities to mitigate the actual barriers in order to cumulate and use advanced energy production processes for finding new energy sources and to build improved, reliable and safety power plants.
The monitoring systems, intelligent sensors and SSCs, nanotechnologies and new/intelligent materials constitute the main reasons for improvement of the NPP's systems configuration and processes.
The paper presents:
- The state of the art in the level of actual and useful technologies for nuclear power systems development;
- The actual technological limits that need to be over passed for NPP's systems improvements;
- The main systems that need improvement and reconfiguration for actual NPPs development and increase in efficient operation, appropriate availability and total safety;
- The actual energy production issues;
- The key arguments in sustaining the R&D new NPP's systems development;
- Future trends in NPPs development;
- The limitations in industrial processes knowledge and use.

Appropriate R&D in the field of NPP's systems is a specific characteristic that is used for paper completion. Adequate interface between actual technologies, corresponding redesigned NPP's systems and the environment are also focused in the paper.

I.3. Nuclear Technologies and Materials

I.3.1. Modern Graphite Characterisation Techniques and their use in Supporting a Small-Scale 'Nuclear Battery' Concept.

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Small-scale reactor designs are becoming an increasingly popular option for future reactors, following increasing costs of generation and grid infrastructure. The ability to have an unattended 'battery-type' reactor, utilizing passive safety systems and the minimum of mechanical moving parts as well as a sealed coolant loop is an attractive idea, providing years of electrical generation on-site with no reliance on a national grid or foreign fuel suppliers.
As a starting point for such a design, the VHTR reference design has been used, including TRISO fuel and a prismatic core design, intended to be small enough for the core components to arrive on a single vehicle. In order to reduce the number of mechanical parts however, carbon dioxide has been considered as a coolant instead of the more usual helium gas – it is cheaper, less leak prone, and more capable of naturally circulating with less pumping.
The introduction of carbon dioxide in such a sealed system has not been considered before, and nor has it's use with modern graphite and graphite matrix materials. In order to test these irradiation experiments are currently underway at the *Technical University of Delft (Netherlands)* to provide sealed, irradiated samples for characterisation. These samples will then be returned to *The University of Manchester (UK)* where their behaviour will be compared with thermal oxidation data currently being produced in Manchester.
Laser Confocal Microscopy will be used to examine the macro-structure of the samples – providing high-resolution optical images combined with a 'height map' of the sample surface this allows a 3D image to be produced at up to 100x optical zoom. X-Ray tomography will be used to examine the pore structure of the samples and it's pore volume distribution completely none destructively and be compared to and supported by helium pycnometry. Traditional SEM will also examine the microstructural changes within the samples.
Each technique will be explained by the author, along with it's role in supporting or refuting the material case for the use of modern graphites in a sealed carbon dioxide reactor.

I.3.2. Thermogravimetric Method to Evaluate the Average Compressive Stress Evolution During Zy-4 Fuel Cladding Isothermal Oxidation.

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The development of an oxide on zirconium alloys is governed by the diffusion of the oxygen ions through the oxide scale and the apparition and growing of compressive stress at the oxide – metal interface. It is admitted that the diffusional driving force is the compressive strain gradient in addition to a chemical potential gradient across the oxide scale. Each of this two contribution to the kinetic curve can be treated separately, allowing the evaluation of their dependency of the oxide scale thickness and the evolution in time of both the flows given by the two potential gradients.

In the first part of our work we present the theoretical treatment of the kinetical curves in both pre and post-transition regions. The equations obtained where used in the treatment of the kinetic data obtained experimentally on isothermal oxidation of Zy-4 cladding in 873-1673K temperature range. The contribution of chemical potential gradient to the kinetic curve is linear for all range with different kinetic coefficients for the pre-transition and for the post-transition regions. The average compressive stress exponentially increases in time, in the pre-transition region which means a linear increase with the oxide scale thickness. For the post-transition region the shape of stress evolution curves became more complicated due to the cracks and pores formation. The periodical stress relaxation can be related to the kinetical behaviour over the transition and explain the multilayered structure of the oxide scale.

I.3.3. Investigation of High Temperatures Tensile Properties on Cold-Worked and Thermal Affected CANDU Pressure Tubes.

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In a postulated Loss of Cooling Accident (LOCA), the CANDU pressure tubes temperature may increase up to high values. The mechanical properties alter with increasing of temperatures over operating temperature. At higher temperatures the pressure tube will suffer plastic deformation at lower stresses.
Furthermore, if the pressure tube remains in reactor after a LOCA temperature peak at high temperature, its mechanical properties has changed and thermo mechanical behaviour has not the same as received state.
This study summarizes the results from tensile tests performed on tensile specimens from unirradiated, cold-worked Zr-2.5%Nb pressure tube and also from same alloy thermal affected. The thermal affected samples were obtained by exposure at transient thermal treatments where temperature highest point was 1000^{o}C and it could be hit with constant and controlled rate of temperature increasing and decreasing of 10°C/min and 20°C/min. The maximum temperature of thermal treatment is located in □-Zr phase region.
The tensile tests have been performed at temperatures in a large range between room temperature and 700^{o}C, in isotherm conditions using a servo hydraulic “Walter + bai ag” mechanical testing system equipped with an electrical furnace. The temperature range is located in □-Zr phase and mixed (□+□)-Zr phase regions. The yield stress, ultimate tensile stress and modulus of elasticity as functions on temperature were investigated for both microstructure types. Moreover, the results were compared to examine the effect of pre-existing thermal transients on the mechanical properties of Zr-2.5%Nb alloy. Also, effect of temperature on the microstructure was examined.

I.3.4. Influence of Hydride Precipitates on Crack Growth by DHC Mechanism.

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In normal operation of CANDU pressure tubes (Zr-2,5% Nb), it was found that one of the important phenomena that affects its structural integrity is deuterium absorption, which may

finally lead to the DHC phenomenon (delayed hydride cracking) under certain thermo-mecanical conditions. This can lead to the pressure tube cracking and further to the leaking of the coolant with dramatic consequences.
This paper aims to develop techniques for obtaining the concentrations of hydride and its reorientation at the flaw tip region in the pressure tube samples, under specific thermomechanical conditions.
To check the influence of thermal transients on the reorientation of hydrides on flaw region has been set-up a specific experimental test, on pressure tube samples with high levels of hydrogen content, representative for the pressure tube at the end of its life in reactor.
The samples were tested at different numbers of thermal cycles, between 250-90-250^{0}C, under constant load of 295N (30Kgf). The thermal transients were similar to those caused by the planned or unplanned shutdowns of the CANDU reactor.
The optical micrographs revealed that the density and size of the reorientation of hydrides become significant with increasing number of thermal cycles test. The hydride platelets are reorientated form the circumferential direction to normal at the crack tip.
The results from present work will be used in the analytical modeling the DHC phenomenon.

I.3.5. Improving the Corrosion Resistance and Creep Strength of Steels Exposed to Heavy Liquid Metals by Surface Modification Using Pulsed E-beams.

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A key problem in development of heavy liquid metal - cooled nuclear energy and transmutation reactors is the corrosion of structural and fuel cladding materials in contact with heavy liquid metal. Lead and lead alloys attack unprotected steel surfaces by dissolution of the metallic components into the liquid metal. It is a common understanding that dissolving a small amount of oxygen into the liquid metal leads to the formation of oxide scales on the steel surface, providing the best protection against dissolution attack. However, at temperatures above 500°C, austenitic steels suffer from severe dissolution attack, while martensitic steels form thick oxide scales, which hinder heat transfer from the fuel pins, and which may break off, eventually blocking the coolant channels.
Above 500°C, steels have to be protected by stable, thin oxide scales. A well understood protection measure is the alloying of stable oxide formers, like aluminium, into the surface. With a concentration of 4–10 wt% Al at the steel surface, a stable thin alumina scale is formed, by Al diffusion to the surface and selective oxidation, during the exposure to oxygen-containing heavy liquid metals. The alumina scale grows very slowly and prevents diffusion of oxygen into the steel, as well as migration of steel components to the surface.
Alloying Al into the surface can be done by diffusion processes or by melting, using pulsed electron beams, of a thin surface layer, on which Al or Al - containing alloys were previously deposited. The alloying of steel surface with aluminium, using microsecond-pulsed intense electron beams, was developed and optimized at Karlsruhe Institute of Technology in order to be used for improving the corrosion resistance of steels, exposed to heavy liquid metals, like Pb and Pb-Bi-eutectic. The procedure consists of two steps: (i) coating the steel surface with an Al-

containing alloy layer and (ii) melting the coating layer and the steel surface layer using intense pulsed electron beam.
Using the mentioned procedure, the corrosion resistance and creep strength of the ferritic-martensitic 9Cr “T91” steel, exposed to Pb and Pb-Bi-eutectic, with different oxygen concentrations and under different temperatures, was considerably improved due to the formation of a thin alumina layer, whose thickness was lower than 1µm for all the tested temperatures and durations. This slowly growing alumina layer acts as an anti-corrosion barrier and is less susceptible to crack formation and, therefore, to lead alloy enhanced creep.

I.3.6. Diagnosis and Analysis of Structural Materials Corrosion Degradation in NPP

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The experience accumulated during operation of some NPP showed that even, in normal operation conditions could appear no-desired effects especially, due to corrosion, erosion, hydriding and deposition of corrosion products on heat transfer surfaces. Diagnosis of corrosive phenomena in nuclear installations implies investigation of the structural materials corrosion processes, in different conditions of water chemistry and temperature, for understanding of the corrosion degradation phenomena that conduct to failure of some components from PHTS of CANDU reactor. In this aim, a chemical control of the water and structural materials corrosion testing program, as well as a system for data acquisition and processing, were established and are presented. The principal objectives of structural materials corrosion testing program are: elaboration of the testing and evaluation methodology of the structural materials corrosion, development of some databases regarding corrosion behaviour and modelling of the oxidation processes, and establishment of the causes that conducted to failure of some components and possible remedies. For this, corrosion experiments in different conditions of water chemistry and temperature and well as corrosion analyses on some corroded components in nuclear installations were performed and are presented. For corrosion phenomena analysis and material-environment interaction modelling, on the basis of the experimental data obtained in corrosion experiments, on structural materials, a programme on computer was developed.

I.3.7. Correlation Between the Efficiency of Some Corrosion Inhibitors Used in Acidic Descaling Solutions and Their Structures.

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To remove the deposits formed on the carbon steel pipes is currently used a descaling process executed using some solutions containing citric acid, tetrasodium EDTA salt, hydrazinhydrate and several types of inhibitors. A method of maintaining of the solutions corrosivity under some limits consists in the use of those inhibitors which can maintain between the reasonable limits the rates of two concurrent processes: the dissolution of deposits and the attack of base metal.
Our experiments were executed on the samples from the following types of carbon steel: SA106, SA508 and SA516, filmed by autoclavization in an environment and at specific parameters of the Nuclear Power Station (NPS) secondary circuit.

The inhibitor types used in above-mentioned descaling solutions were organic substances containing the following specific groups: alcohol, double and triple bonds simultaneously or not and one or more amine groups. These inhibitors have been used in concentrations comprised between [5.10^{-4}- 3.10^{-3}]M. From the tested inhibitors used in our descaling solutions, remember: triethanolamine, ethylendiamine, triethylamine, propargylalcohol, propargylamine and others. To establish the inhibitors efficiency were used the following two methods: gravimetric and potentiodynamic.
Finally on the basis of the molecular structure – activity relationships, we confirmed a reaction mechanism of some inhibitors type amine.

I.3.8. Stress Corrosion Cracking Initiation on Hydrided Zircaloy 4 Alloy.

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The mechanical properties of Zircaloy-4 alloy with different hydrogen content between 100 and 350ppm, established by tensile tests, were correlated with the SCC susceptibility in iodine vapours at 320^0C. In this paper we chose to use C-ring method. The maximum values of stress and deformation in C-rings, induced by a zircaloy-4 block with 18mm length, were estimated with the ANSYS code. At room temperature the maximum values for equivalent stress and local deformation are 480MPa, respective 2.4%. When the temperature increases at 320^0C, the equivalent stress decreases until 300MPa and the local deformation grows at 2.7%. In the case of one hydride considered as an initiator of a SCC crack, the maximum value of the factor of intensity of stress was 4.51 $MPam^{-1/2}$, attained at the border of hydride. Taking in account the experimental results obtained in this work, we can affirm that the maximum values of the factor of intensity of stress, computed with FEA-CRACK program, represent the threshold value over that the SCC mechanism is initiated. Must not omitted that these represent the mean values, being computed starting from the mean dimensions of hydrides. By metallographic method and SEM it was put in evidence the transgranular character of SCC cracks.

I.3.9. Studies regarding oxidation of nuclear grade materials in supercritical water environment

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A primary challenge to the Supercritical Cooling Water Reactor concept is degradation of alloys in supercritical water so, the main viability issue of this type of nuclear reactor is the selection of suitable materials that provide sustained corrosion resistance in this aggressive environment. As in other high temperature environments, corrosion resistance in SCW is given by quality of the oxide layers developed on the materials surface. This paper presents a preliminary study regarding oxidation behaviour of an austenitic stainless steel (304L SS) and a Ni base alloy (Incoloy 800HT) in water at supercritical temperatures in the range 450 - 600ºC under a pressure of 25 MPa for up to 1680 h. After exposure to deaerated supercritical water, the samples were characterized using gravimetry, scanning electron microscopy/energy dispersive

X-ray spectroscopy (SEM/EDS), X-ray photoelectron spectroscopy (XPS), and electrochemical impedance spectroscopy (EIS).
Corrosion films on the studied materials are bilayer structures with an outer layer consisting of a mixture iron oxide/ iron-nickel spinel oxides on both alloys a chromium oxide inner layer on Incoloy 800HT and a nickel–chromium spinel oxide inner layer on 304L SS. The mass gains for Incoloy 800 HT at all temperatures were small, while 304L SS exhibited the highest oxidation rate. In the same time the results obtained by EIS indicate the best corrosion resistance performance of oxides developed on Incoloy 800 HT surface comparatively with 304L SS.

I.3.10. Gamma Spectrometry Characterisation of the Sodium Molibdate Solution Produced by Cintichem Method

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An experimental process for the production of sodium molibdate, based on the Cintichem process, was practically demonstrated by the specialists of INR Pitesti.
In order to assess the process itself and the feasability of the economical production of sodium molibdate, a careful gamma spectrometric evaluation was carried out. Hence, the activities of Mo-99 and other gamma-emitters (‘impurities’) present in the solutions produced along the process (five solutions) were measured. The gamma spectrum of the final solution QC5 indicated the presence of only Mo-99, no gamma-emitter impurity of any kind being present.
Based on the activities of Mo-99 measured for each step of the process, the yields were calculated for each of the steps of the process and for the whole process itself. The yield for the whole process was found to be 45.3 %. The yield calculated after the precipitate was dissolved and before the first purification is 53.5 %. The yield calculated right after the first purification is 94.2 %, whereas the yield calculated after the second purification (for the final solution) is 89.75 %.
Four out of the five solutions studied contained ‘impurities’ which were identified and measured. This kind of data, together with results from alpha spectroscopy measurements, could be used in the future for the assessment of scaling factors ‘gamma’/’alpha’ for the characterisation of the radioactive waste produced along the process.

I.3.11. Monitoring and Controlling Instant Power Consumption in Normal and Abnormal Operation of Tritium Separation Technological Installations

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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 use of the system allows the recording of events from the tritium separation plant, with the view of conducting analyses on the quality of the electricity.
The analysis of some undesired events after shutdowns helps finding the causes of some events or abnormalities of the electricity networks, as well as the possibility of localizing them.

I.3.12. Uranium Supply from Domestic Resources in Poland.

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The Institute of Nuclear Chemistry and Technology (INCT) coordinates the project, financed by EU structural funds and Government of Poland, from Operative Programme - Innovative Economy, POIG, entitled “Analysis of the possibility of uranium supply from indigenous resources”. The project is realised by consortium formed by Institute of Nuclear Chemistry and Polish Geological Institute. The project’s programme presents comprehensive solution of raw material for nuclear fuel production including identification of resources with extensive characterisation of the ores and the ways of exploitation of selected depositions, as well as the methods of uranium extraction form different, available in Poland, sources.
This project makes an effort to meet recent plans of national electricity supply development. The document „Polish Energy Policy until 2030“ takes into account the nuclear option to ensure the national energy security and to enable economic development of the country. According to this plans the first nuclear power plant in Poland is expected to be put in operation around the year 2020.
There is a need of precise analysis of uranium domestic resources in Poland and the assessment of their exploitation possibilities, which is the aim of discussed project. According to tentative researches performed by Polish Geological Institute the Lower Triassic sandstones from Peribaltic syneclise are the most perspective formation in Poland with high uranium concentration, locally reaching 1.5%U. The other sources are Dictyonemic shales from the east part of Poland (Rajsk Deposits) spread out on the large area, which, however, have lower uranium concentration (100 ppm). In these ores uranium is accompanied by vanadium and molybdenum, locally by other metallic admixtures, like copper, zinc or lead. To enhance the economy of the extraction process the recovery of these metals is taken into account.
The best available technologies will be employed to obtain yellow cake - U_3O_8. The technological part of the project will be completed with techno-economic analysis that will include assessment of the costs of the exploitation and the production of pure U_3O_8 from domestic resources.
The secondary materials and by-products like phosphoric acid and post-leaching copper concentrates from copper industry will be tested as a potential uranium source.
New approach to the uranium supply from different sources including multistage, counter-current system for uranium leaching, application of contactors with high-tech membrane for uranium concentrate purification and selective extraction of uranium from copper concentrates is going to be considered for the purpose of reduction of waste amount, minimisation of reagents end energy consumption as well as enhancement of uranium extraction efficiency.
New technology will be elaborated for uranium recovery from phosphoric acid using membrane process which is favourable for environment protection causing less uranium concentration in phosphate fertilizers and in the waste generated during their production.

I.3.13. The Reducing Emissions of Gases with Greenhouse Effect using the Metallic Catalyst Supported. Production Technology, Preparation, Physical and Chemical Characterization

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The reduction of emissions of gases with greenhouse effect it is conditioning by production of selective adsorbents, zeolites and carbonic molecular sieves type. The increasing of purification and separation degree of gases in the modern and nonpolluting technology it has become a social-economical modern problem which, in the advanced industrial countries, but even in Romania too become a special importance, which accompany to realizing the new materials and technologies. This new application has to be achieved with low costs of preparation, distribution and exploitation. The selective adsorbents have application special in separation and purification processes of gases but in the same time they could be utilized as support for incorporation of active metals, which achieved to obtain of catalysts with application on reducing of gases with greenhouse effect by catalytic chemical reactions.

I.3.14. Study OF THE Filled Skutterudites Yb-(Fe, Ni)$_4$Sb$_{12}$ prepared by Mechanical Alloying AND Spark Plasma Sintering (MA-SPS)

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Ternary intermetallic compounds with the general formula RE-M_4-Sb_{12} (where RE is Er or Yb, M is Fe, Ni) crystallises in the cubic skutterudite structure, filled by rare earth atoms. Skutterudite compounds have been shown to possess good thermoelectric properties.
Samples of the filled skutterudites $YbFe_3NiSb_{12}$, $YbFe_4Sb_{12}$ were prepared by mechanical alloying and spark plasma sintering.
We report results concerning preparation and chemical properties of the skutterudites Yb-(Fe, Ni)$_4$Sb$_{12}$.
The skutterudite structure of Yb-(Fe, Ni)$_4$Sb$_{12}$ were investigated by means of X-ray diffraction (XRD) experiments, energy-dispersive X-ray (EDX) measurements under a transmission electron microscope. The sinterized fractured samples were investigate using scanning electronic microscope (SEM).

I.3.15. CANDU Steam Generator Tubesheet Degradation in Water Containing Impurities

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Steam Generator (SG) is one of the key components for a Nuclear Power Plant (NPP) because this equipment assures the separation boundary between the primary and secondary circuit and its unavailability suppose the NPP shutdown. The synergetic action of the high pressures and temperatures, constraints (stresses, vibrations) and the chemical parameters of the cooling agents make the steam generator susceptible for more types of degradations.
The feedwater that enters the steam generators under normal operating conditions is extremely pure, but nevertheless contains low levels (generally in the µg/l concentration range) of impurities such as iron, chloride, sulphate, silicate, etc. When water is converted to steam and exits the steam generator, the non-volatile impurities are left behind. As a result their concentration the bulk steam generator water is considerably higher than these in the feedwater. When the silicon compounds concentration is higher the localized corrosion appears and can have a catastrophic result consisting in the loss of components integrity and the development of a large quantity of soluble corrosion products and deposits. For low concentration the effect of the silicon compounds on the behaviour of steam generator materials has never been considered adequately. The information available is contradictory, and it is possible to encounter in the literature old results that indicate an inhibition or an aggressive effect of silicon compounds in caustic environment.
The purpose of this work consists in assessment of localized corrosion behaviour of the tubesheet material (carbon steel SA 508 cl.2) at the normal secondary circuit parameters (temperature = 260^{0}C, pressure = 5.1MPa). The testing environment was the demineralised water containing different silicon compounds, such as silica (SiO_2) and sodium metasilicate (Na_2SiO_3) used separately or together, at pH=9.5 regulated with morpholine and cyclohexilamine (All Volatile Treatment – AVT). The paper presents the results of metallographic, X-Ray Diffraction and Scanning Electron Microscopy examinations as well as the results of electrochemical measurements.
The results of this paper contributes in the establishing of causes that produced components degradation, the knowledge of mechanisms degradation and can make possible the evaluation of corrosion evolution in time by extrapolation of obtained data and estimation of remaining safe operation life for the nuclear power plant key-components.

I.3.16. Corrosion Tests of Incoloy 800 at the Specific Parameters of the Steam Generator Primary Circuit

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For long period of steam generator (SG) operating, the heavy water conduces to tubing degradation, especially the corrosion degradation. Accordingly to this, the study of steam generator material tubes (Incoloy 800) at primary circuit specific parameters (t=310°C, p=110 atm, pH=10.5, O_2<20ppb) has a large requirement.
The goal of this paper is to obtain data about the Incoloy 800 corrosion process in primary circuit conditions. With this purpose the samples were tested in CANDU SG primary circuit conditions and dynamic regime using the recirculating loop. After autoclaving, the samples were analyzed by X-ray diffraction and optical microscopy. The results of testing program consist in the assessment of the composition, the structure of the oxide layers and the grain size in relationship with the testing period and these will be exposed as micrographies and X-ray emission spectra.

The recirculating loop for corrosion tests offers the possibility of testing the behavior of structural materials belong to primary and secondary circuit from a Nuclear Power Station at uniform and localized corrosion, reproducing the technical specifications of the two circuits (temperature, pressure, pH, content of O_2).
The experimental results can be used for a future data base related to corrosion processes of the principal SG components.

I.3.17. Steam Generator: Incoloy 800 AND SA 508 Tested of Corrosion in Dynamic Regime

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This paper presents some analyses performed on Incoloy 800 and carbon steel SA 508 grade 2, materials used in the Steam Generator (SG), tested of corrosion in dynamic regime.
The tubes from CANDU Steam Generator are manufactured from iron, nickel and chrome alloy (Incoloy 800), and shell material is a carbon steel (SA 508 grade 2). The tests were realized in dynamic conditions in demineralised water (pH 9.5), temperature 260˚C and pressure 5.1 MPa using two different installations: Testing Loop in Dynamic Regime and CORMET Testing Systems. The samples were maintained in these installations 10 and 130 days. After tests, the samples were prepared metallographic (cutting, mounting, polishing, and etching) for the microscopic examination. The metallographic analysis was performed using OLYMPUS GX 71 metallurgical microscope, at magnification x200- x1000 for surface morphology, microstructure, corrosion and thickness oxide layer. The macrographs aspects were performed with a photo camera. The determination of Vickers microhardness was performed at a micro-durometre OPL (x500; micro load 0.1 Kgf) on tested and as received samples, in cross sections. The analysis consisted in the quality evaluation of structural material and layer formed, after tested periods, using optic microscopy.
Results and conclusions are related to surface morphology, thickness oxide layer, microstructure, micro hardness and corrosion.
Some results of these tests can be used in the study of the degradation properties materials (mechanical and structural) of the Steam Generator.

I.3.18. The Decontamination Impact on the Corrosion of Structural Materials in CANDU Primary Circuit

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The decreasing of the radioactive field from the transport systems of cooling agent in the aim to minimize the doses received by operators during reactor operation, routine maintenance and before closing of a nuclear plant, is achieved by chemical decontamination.
To obtain an efficient decontamination of the metallic components is necessary the partial or total removing of the superficial films that implies some corrosion processes at the respective surfaces. Hence, results the necessity of the surfaces examination before and after the execution of decontamination cycles.

The aim of this paper is devoted to the study of behavior of SA 106 gr.B carbon steel, 403m steel, 304L steel and Iy-800, in CAN-DEREM decontamination solutions.
The paper presents the results achieved after application of CAN-DEREM decontamination procedure on the above-mentioned oxidized steels. The oxidation has been performed in autoclaves in similar conditions of CANDU primary circuit; afterwards, the autoclavized samples were decontaminated in a laboratory loop with dynamic circulation.
The corrosion kinetics of these steels in the specific aqueous environment of primary circuit was established using the gravimetric method. Their losses weight and corrosion rates resulted after decontamination were established using the gravimetric method.
The examination of the superficial films formed on the samples after autoclavization and decontamination have been performed using the metallographic microscopy, ESCA and electrochemical impedance spectroscopy (EIS) techniques.

I.3.19. Corrosion Evaluation of Reinforced Concrete from Reactor Containment

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Nuclear safety-related concrete structures are composed of several constituents such as: concrete, conventional steel reinforcement, pre-stressing steel and metallic or non-metallic liners materials.
Corrosion of steel in concrete is a well known problem with large consequences in terms of economics. In appropriate conditions (e.g., high pH), the steel from concrete will be protected against corrosion, but the protective film formed on the steel surface in concrete under specific conditions, can be destroyed by lowering the alkalinity (e.g. by carbonation and / or the presence of chloride).
The paper discusses results of experiments performed in conditions simulating steel state in concrete, for both protective and corrosive conditions by adding salt. The experiments were performed on carbon steel samples immersed for a longer period of time in solutions having the following compositions: demineralized water, 0.02M $Ca(OH)_2$ (pH ~12.5) and solutions simulating pore water of concrete [0.6M KOH, 0.2M NaOH and 0.001M $Ca(OH)_2$] with a pH value of ~13.
Electrochemical behavior of steel in alkaline environment is complex and not well understood. We used a series of electrochemical methods for evaluating the corrosion behavior of reinforcing steel concrete and the metallographic microscopy examination to detect the possible corrosion attack produced by testing solutions.

I.3.20. Sample Preparation for Mechanical Tests, Using the Numerical Controlled Milling Machine

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The objective of this paper is to present the method of preparation of pressure tube samples for mechanical tests. For sample preparation a DM 1007 Dyna numerically controlled milling machine is used.
This milling machine has a cutting feed rate of 0 ÷ 5000 mm/min, an automatic (pneumatic) tool changer for six tools, simultaneous XYZ axis control, and a positioning precision of 0.001 mm.
So far, the milling machine has been used to prepare C-shape non-irradiated CANDU pressure tube samples for mechanical testing.
The C-shape sample must have a notch for an initiation of delayed hydride cracking (DHC) along the radial direction; this notch plays a key role in the determination of the threshold stress intensity factor (K_{IH}).
To perform the C-shape CANDU pressure tube sample machining, some devices were manufactured: the device to fasten the pressure tube during machining, the device to fasten the sample during cutting its both ends and the device to fasten the sample during notch machining.
The pressure tube is fastened to the milling machine table, the ensemble of both being allowed to move in the XY plane, whereas the milling tool is moving independently along the Z axis. The programming of sample machining needs the size measurement of the pressure tube and needs also the movement coordinates of the milling machine table. A Point Finder is used for size measurements of the pressure tube and a similar tool, fixed on the milling table, is used for dimensional measurements of the milling tools that will be used.
The C-shape sample machining needs six operations for which six different tools are used. All tools are mounted on the tool changer prior to begin the machining and are changed automatically as needed.
This machine will be installed into a heavy concrete hot cell and will be used to prepare C-shape irradiated CANDU pressure tube samples for threshold stress intensity factor (K_{IH}) and delayed hydride cracking (DHC) velocity mechanical tests.

I.3.21. Destructive Post-Irradiation Examination of an Experimental CANDU Fuel Element Irradiated in a Power Cycling Test in the TRIGA Reactor

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Institute for Nuclear Research Piteşti disposes of facilities which allow the testing, manipulation and post- irradiation examination of nuclear fuel and structural materials from Cernavoda NPP CANDU reactors. These facilities include TRIGA material testing reactor and Post-Irradiation Examination Laboratory (PIEL).
The aim of this paper is to determine by destructive post-irradiation examination, the behaviour of CANDU fuel element, in a power cycling test in the TRIGA reactor using “Capsule C9” device.
A CANDU nuclear fuel manufactured by INR Piteşti was irradiated in a power cycling test in the TRIGA reactor. The testing was intermittently performed in several irradiation cycles.
The results of post-irradiation examination consist of:

- Metallographic and ceramographic examination performed with LEICA TELATOM 4 microscope, revealed the following:
    - cracking fuel appearance;
    - fuel microstructure;
    - grain size;
    - porosity distribution;

- sheath hydring;
- macroscopic and microscopic image analysis areas of interest to determine: the fuel element diameter; the thickness of the sheath; fuel- cladding gap the oxide layer thickness.

- Mechanical properties determination: tensile tests performed on ring samples taken from the fuel element irradiated with Instron tensile machine model 5569.
- Examination by scanning electron microscopy: were studied macroscopic appearance of fracture surfaces and fracture topography.
- Burn- up determination: by uranium 235 depletion method.

The data obtained by post-irradiation examinations are used first of all to confirm the safety, reliability and performance of nuclear fuel, and then for CANDU fuel development.

I.3.22. Determination of Impurities Content in Uranium Dioxide Powder Using Time of Flight Mass Spectrometry

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In the nuclear industry, the determination of impurities content in the nuclear fuel is of high importance. In this paper, the Time of Flight Mass Spectrometry is applied to the impurities content analysis of uranium dioxide powders used for nuclear fuel manufacturing. The level of these impurities is limited and depends on reactor characteristics. For monitoring the contamination level during fuel pellets manufacturing process is necessary to determine the impurities content in the beginning and in the end of the process. The analysed impurities were: Cd, Gd, Dy, B, Na, Mg, Al, Si, Ca, Cr, Mn, Fe, Ni, Cu and Mo.
Choosing an optimum calibration technique is essential because the measurement precision depend on it. The calibration techniques most used by ICP-MS are: standard addition, internal standard and external standard. In this case, external standard addition was used.
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 uranium dioxide powder samples were dissolved in High Purity Nitric Acid and then the uranium matrix was extracted with a TBP solution to avoid interferences. The samples were, then, diluted and analysed.
The best mass resolution obtained in this study is ~2300 for Dysprosium.

I.3.23. Testing of Some Organic Inhibitors for the Protection of Grade 2 Titanium Alloy in Aggressive Media

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The corrosion failures of titanium alloys are inevitably associated with the non-observance of environmental limits prescribed in corrosion handbooks, e.g., their use in solutions containing chloride salts. When in a corrosive substances flux some oxidant species are present, it is

necessary to add very small quantities of inhibiting adequate species to respective environment with the aim to protect the Grade 2 Ti components.
In this paper we followed the determination of protection efficiency against corrosion in the case of some systems consisting in titanium alloys/ solutions containing Cl- (NaCl) ions and the following organic inhibitors: resorcinol (REZ), 5-nitrobenzimidazole (NBZD) and benzimidazole (BZD). Some tests on Grade2 Ti samples in solutions containing Cl- (NaCl) in presence, respectively, absence of some organic inhibitors using the potentiodynamic polarization were performed.
To choose the optimum inhibitor type for the passivation of tested titanium alloys, some comparisons of cathodic branches of potentiodynamic curves recorded in inhibited/ uninhibited solutions of NaCl were made As result of electrochemical measurements, we concluded: the resorcinol proved very efficient in a concentration of 0.02M; 5-nitrobenzimidazole had far smaller in this case of Grade 2 Ti; benzimidazole is not adequate for this system Grade-2 Ti / NaCl solution 0.083M.

I.3.24. The Involvement of Sulphate –Reducing Bacteria in the Failure of Brass Tubes of Heat Exchangers in a Nuclear Power Plant

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In this paper the failure due the microbiologically induced corrosion of brass condenser tubes in a nuclear power plant was investigated. Copper and its alloys are most commonly used in the fabrication of heat exchangers in cooling water systems.
In aqueous environments microorganisms are attracted towards surfaces and form biofilms. The implications of microfouling are energy losses due to increased fluid frictional resistance and increased heat transfer resistance.
An example quoted in the literature reports the failure of the alarming number of 4000 condenser tubes in a short span of 7 years of operation of a power plant (Marchwood, Southampton, UK).
The brass samples were exposed for 270 days in inoculated media with sulphate reducing bacteria(SRB). On the surfaces of the exposed samples in inoculated medium appeared the characteristics of a bacterial attack and some thick, brown and friable deposits developed.The thickness of the biofilms varied point by point along the surfaces achieving a variable covering of their.
At the end of experiments the following aspects were monitored: macroscopic and microscopic (metallographies made on a Neophot microscope), aspect of surfaces samples and the morphology of microbial cells and colonies using the scanning electron microscopy (SEM).
The achieved experiments emphasized a strong dezincification of the surfaces of exposed samples in sulfate-reducing bacteria media with involvement in the failure of heat exchangers from cooling water systems.

I.3.25. Internal State Surface Investigation of Steam Generator Demage Pipe (CANDU 6)

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The tube ageing of CANDU 6 steam generators has been and remains marked by the effects in operation of the damaged mechanisms: chemical corrosion mechanism, stress corrosion mechanism, the fretting wear mechanism and fatigue. The steam generators (SGs) tubing failures are generally specific to secondary circuits and are mostly located on outer walls surfaces and have resulted in ageing process following NPP operation. The internal tubing failures are less from aging process; they are in general induced by the manufacturing process, by the assembly and the pickling process, or could be the in-service process results.
The paper contains geometrically restrictions and some details need to be followed in initiation of the steam generator tubes replication devices for both tube sheet rolled joint areas and beyond them, as well as some possible constructive solutions. The defect shape and geometrical restrictions imposed the choice of variant constructive solution. Thus, we need a solution for the first 500 mm of SG tube, including the rolled joint in tube sheet areas and another one for the straight tube, until the curved zone of the SG tube beam. The defect shape requires on its turn, different solution to carry out replication. The sampling mechanism simplicity for each of the advanced solutions is mandatory being a guarantee to maintain its functionality whatever the application conditions are. The paper has original contributions in the development of non-destructive testing method and its area of application and is addressed to specialists and employees working in research and engineering technology.

I.3.26. Forming of the Ice Plug in a Large Diameter Horizontal Pipe Simulator, Process Marked by Demin Water Flow Asymmetry

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Local insulation of a pipe section, horizontally placed in a facility, can be accomplished with an ice plug resulting in a controlled freezing process inside of the duct. The method is applied in order to repair or replace the components without stop and drain the flow agent from installation. This insulation is a special process technology developed continuously by the specialized company in the field, according to the requirements of each application.
Basically, the method is simple and involves applying a special designed freezing sleeve for each case. Sleeve represents the freezing device, fixed or removable. It is mounted on the pipe outside surface in the chosen area for plugging the flow and forms an annular space with the pipe surface area or in its immediate neighbourhood (open to the outside). This space is continuously injected with liquid refrigerant (CO_2 or N_2).
The heat and mass transfer continuously conducted between the pipe wall and the formed refrigerant ring, leads to lower wall temperature, increasing local deposit of successive ice layers. Thus, the duct is gradually developing a growth of local hydraulic resistance resulting in slight reduced flow rate until flow stops. The ice plug development process is strongly marked by the required application, which is characterized by the circulated fluid, by the operating fluid temperature, by the real freezing point (pressure, temperature), by the pipe inner diameter, by the pipe position in space and by the distance to the first flow restriction.

The agent flow asymmetry in horizontal pipe makes the application to be special. Article contains a functional and constructive description of technological experimental facilities. The experiment, the first results obtained are presented and some conclusions also. The work is addressed to specialists employed in research and engineering technology.

I.3.27. High Performances of PANalytical X'Pert Pro X-Ray Diffraction System of INR Pitesti

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X-ray diffraction is the science that studies the microstructure of materials and alloys by X-ray scattering. In the Laboratory of Physico-Chemical Analysis of ICN Pitesti it is available a diffractometer which is utilised for nuclear materials characterization. The PANalytical X'Pert Pro X-ray diffraction system can be used for a wide variety of applications in analytical X-ray diffraction, in scientific and industrial research. These applications are: qualitative phase analysis, quantitative phase analysis, crystallography and Rietveld analysis, residual stress analysis, texture analysis, analysis of small spots. The system can be configurated from one application setup to another within a few minutes without additional alignment of the system. Also, PANalytical X'Pert Pro X-ray diffraction system can be upgraded for in-plane diffraction on thin films, transmission measurements through foils, analysis of changes in the crystal structure in changing environmental conditions. The system includes the X-ray tube cooling system, signal processing and the dedicated softwares applications: X'Pert DataCollector V2 LTU for record data, X'Pert HightScore Plus LTU for crystallographic phase identification and quantitative analysis, X'Pert Stress LTU WIN for residual stress analysis and X'Pert texture for texture analysis. The data base used in the analysis is ICDD PDF4+ updated in 2010.
The X-ray diffractometer is very utile in the characterization of materials used in nuclear industry and their corrosion behaviour in the extreme conditions of a nuclear reactor (temperature, pressure, water chemistry...).
In this paper we will present the type of analysis that we can perform and the additional softwares. Also, some experimental results data are given.

I.3.28. Thermo-Gravimetric Analysis (TGA) as Condition Monitoring Technique of the Power Cable in Nuclear Power Plant

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Development of the condition monitoring techniques for cables made a considerable progress among last years. There are some current techniques, which are already of practical use and are being used in plant. TGA test is representative of volatile molecules and plastifiers amount and in the main properly for PVC materials loosing the plastifiers.
In the technical literature are presented the theoretical and practical results regarding use of Thermo-Gravimetric Analysis (TGA) as ageing evaluation technique due to thermal and/or radiation exposure of the power cables. Due to influence of many factors over TGA results is too difficult to compare data resulted in different laboratories. Each plant operator may apply his

own method and compare the results with his own reference data. From this reason it would be necessary for each plant to generate own baseline data.
Prospective of development by the Institute for Nuclear Research of the condition monitoring techniques for the power cable, as a component of the ageing management programme for NPP cables, performed thermo-gravimetric tests on the samples for this purpose. The experimental results (determination of the TGA parameters) have been gained for PVC jacket cables being accelerated thermal aged. The initial weight, density and degree of settling of the samples were different from one test to another, influencing the results. The samples were singular for each ageing period chosen, missing of the statistic calculus being a factor that affects the result accuracy.
The operation lifetime for insulation is less than the jacket one. To establish the degradation value and the real residual operation lifetime of the power cable on duty may correlate the jacket and insulation conditions.
The paper contains a TGA parameters description, the experimental results of the thermo-gravimetric tests performed, description of the sample and installation characteristics used, the equation describing the TG5% parameter value evolution versus ageing period, the procedure followed to correlate the jacket and insulation power cable condition and the conclusions. It is dedicated to the specialists from research and management of the power cable ageing.

I.3.29. Control of Oxygen Impurification during TRIGA Fuel Manufacturing.

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TRIGA LEU fuel, $U(ZrH_{1.6})Er$, is a fuel rod type cermet of zirconium hydride. It can be obtained by classical or by powder metallurgy. Powder metallurgy involves the conversion of the metallic components in powdery form followed by their mixing up and their sintering.
It is very important to use in this manufacturing the primary metals of high purity. It is possible that the zirconium sponge, used as starting material, contain more oxygen than the permissible limits. Its presence can block or limit the zirconium hydriding reaction. The oxygen can also appear from some incidents occurring during manufacturing process. The oxygen traces existent in the working area can induce some undesirable effects, influencing the quality of the final product.
To solve this problem, a series of changes in the automation of argon purification process was achieved. In this context, the continuous argon purification and the oxygen concentration monitoring in working area are very important, too. Thus, the maintaining of the oxygen content under the permissible limits in boxes chain provides optimum working conditions.
The aim of this paper consists in the presentation of some consequences of the contamination with oxygen of argon atmosphere and to establish the modalities to avoid them. Therefore, during fabrication of TRIGA LEU fuel, a permanent control of the oxygen concentration is necessary to avoid the fuel impurification.

I.3.30. Experimental Study on Titanium Powder and its Application to Tritium Storage Bed Designed

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The Nuclear Power Plant Cernavoda, equipped with a Canadian reactor, is the most powerful tritium source, in Europe. A TRF (Tritium Removal Facility) will be constructed to remove the tritium from heavy water, and this will necessitate the storage of tritium in a special vessel. The titanium metal powder supplied by Alfa Aesar GmbH&Co KG, Karlsruhe (100 mesh, purity 99.4%) was used in this study as a suitable material for tritium immobilization. The properties of titanium hydride for hydrogen storage include the ability to bind isotopes at normal temperature and pressure, and also being easily prepared. The samples of titanium powder were tested to examine their hydriding/dehydriding performances. The characteristics of the hydriding/dehydriding reaction over titanium powder have been investigated by means of thermal analysis. The morphology of the powder was obtained by scanning electron microscopy and the present phases were detected with X-ray diffraction experiments. The purpose of this paper is to describe the experimental data which supplies the information regarding the practical use of titanium powder for the storage of hydrogen isotopes under different conditions. Titanium powder showed a high hydrogen absorption capacity. Since, the chemical properties of tritium are virtually identical to those of hydrogen, this work has been conducted using hydrogen. The results will be used to evaluate storage vessel design loading limits. Using metal hydrides for storage of heavy isotopes in a tritium containing system, can solve many problems arising in the nuclear fuel cycle.

I.3.31. Mechanical Simulator of the CANDU Fuelling Machine.

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Fuelling Machines (F/Ms) in CANDU type reactors are high accuracy robots that operate in maximum safety and reliability conditions. Fuel bundles are pushed into one reactor channel by a remotely operated fuelling machine. Simultaneously, the irradiated fuel bundles are discharged into another fuelling machine, at the opposite end of the same reactor channel.
Before to be installed in the nuclear power plant, the F/M(s) are subjected to complex checking, adjusting and testing operations in order to demonstrate their performances and reliability.
To achieve this goal, at INR Pitesti has built a special F/M(s) test rig for CANDU reactors. By its structure and designation, it is unique in Europe, the AECL Canada having a similar installation.
The F/M test rig has performed by a Romanian design and is entirely complying with testing requirements of AECL – Canada. The installations and equipment assembly of the test rig and the afferent control and checking systems are similar to those from CANDU reactor, and perform the identical operating parameters (temperature, pressure and flow rate of water) to those from reactor to actuate the F/M for refuelling in nuclear fuel channel.
The operating personnel and the test rig has licensed by AECL Canada representatives. The licensing activities have performed using the Mechanical Simulator of F/M.
The paper describes the independent equipment Mechanical Simulator (MS) of F/M designed and performed as original variant at Institute for Nuclear Research (INR) from Pitesti.
MS is dedicated to develop commissioning activities - checking and functional tests, adjustments, settings – of the F/M(s) test rig, for effective testing of them. Together with the

other F/M simulators, MS constitutes the basic structure for periodic training of the test rig operators.

I.3.32. CFD Analyses in Open-Source Environment for a Lead Test Facility.

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The purpose of the paper is to simulate melted lead flow conditions inside a scaled vessel in an experimental installation at temperatures between 450-500 C. The heat source consists of seven electric heaters which provide the necessary power to maintain the melted lead in the mentioned temperature range. Natural circulation inside the installation is considered. The experiments are due to study corrosion and erosion of the structural materials to be used in the Generation IV nuclear reactors of LFR (Lead-cooled Fast Reactor) type. Experimental assembly is designed to reproduce at a 1:10 scale the vessel of a LFR Generation IV nuclear reactor producing 300 MWt.
CFD (Computational Fluid Dynamics) three dimensional analysis is performed using OpenSource tools: for geometry and mesh generation – Salome-Meca modules, the Code_Saturne solver for CFD calculations and the Post-Pro module of Salome-Meca is used for analysis and post-processing of numerical simulation results.
Three dimensional thermal-hydraulic parameter distributions (i.e. pressure, temperature, density, velocity, k and epsilon turbulence) were obtained with high degree of accuracy and detail. The results are presented primarily in a graphic manner capable to synthesize a huge amount of information.
CFD analysis results provide valuable data regarding the thermal-hydraulic conditions in which the corrosion/erosion experiments for materials are conducted.

I.3.33. Strategies for Maintenance, Testing and Evaluation in Nuclear Energetics.

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If initially, the research and development efforts from nuclear energetics were concentrated on design features, improving the operability or ageing mitigation for the main components, later it was tried establishing of some programs for testing/ evaluation and management of ageing for all structures, systems and components (SSC) which are important for plant safety.
The paper highlights the issues related to the objectives pursued under the ageing management programs, the limitations (constraints), and recommended programs.
Such a program involves the following objectives:
- identification of the components/ systems whose ageing could influence safe operation;
- determination of the ageing mechanisms and their consequences on safe operation;
- identification of criteria for tracking, testing and evaluation for early detection of the ageing mechanisms;
- establishment of procedures for testing and replacement of components/ systems for safe operation during the plant designed lifetime, despite the appearance of ageing phenomenon.

Maintenance, testing and evaluation involve aspects related to following issues: environmental conditions, routine maintenance, periodic testing, optimizing maintenance practices or implementing procedures for optimal testing, management of spare parts, components obsolescence, as well as replacement policies of components for equipments or systems.
Developing a strategy for maintenance, testing, evaluation and ageing management of equipments requires the following aspects:
- information from maintenance and inspection teams;
- history of nonconformity or failures;
- significance in terms of safety for systems and components with safety functions;
- the failures effect on the plant availability.

Qualification of the personnel as an essential component of the maintenance strategy, an example of a method of evaluation/ extension of the equipment lifetime, and a case study on sensors, transducers, detectors, their performances affected by ageing, together with test methods for performance estimation/ verification, are also included in the paper.

II. ENVIRONMENTAL PROTECTION

II.1. Radioactive Waste Management

II.1.1. New Directions in Treatment Options for Irradiated Graphite Waste

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The United Kingdom has the largest quantities of irradiated graphite from its commercial programme of Magnox and Advanced Gas-Cooled Reactors, along with its early production and development reactors. In common with most of the world, planning for the disposal of this material, which world-wide totals around 250,000 tonnes, is limited to good intentions for repository disposal, either in planned deep repositories or (for France) in a shallow site. In most other countries which have employed graphite-moderated technology for its nuclear programme, a similar state of unreadiness exists.
The EU CARBOWASTE project, for which the Romanian contribution will be presented elsewhere in this conference, has considered a number of new treatment options for carbonaceous wastes which can be foreseen. An IAEA collaborative research programme initiative, 'Treatment of Irradiated Graphite to meet Acceptance Criteria for Waste Disposal', aims to enhance and extend the work undertaken within CARBOWASTE by looking at further innovative methodologies for removal, pre-treatment and subsequent processing of such graphite wastes, with a view to minimising the volumes of such material which must ultimately be sent to repositories. The potential benefits include avoiding the need for expensive removal of intact graphite components, the potential to recover useful isotopes, and the potential to treat waste materials in such a manner as to reduce their waste category, and even the potential for recycling graphite as carbonaceous products for the future nuclear industry, thus saving very large amounts of money.
The author, as Chief Scientific Investigator for this IAEA programme, will review the options which are being considered against the 'default' option of creating safe and secure repository storage for a quarter million tonnes of material.

II.1.2. Characteristics of Irradiate Graphite from Material Testing Reactors and Management Approaches

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Most of the radioactive waste already had found final technical solutions and clear strategies for their management have been defined: near-surface repositories for LIL waste and geological disposal for the high level waste, spent fuel and long-lived waste.
Irradiated graphite was not included in any of these strategies, despite the large volumes existing or expected to come after the gas cooled reactor (GCR) decommissioning. Since 2008, a collaborative work at European level was initiated under CARBOWASTE project having as main objective the identification of a suitable management of carbonaceous waste. Dismantling,

characterisation, treatment and conditioning, and disposal or re-use are the key aspects followed by this project.
A very small part of the whole irradiated graphite arises from the thermal column of material testing reactors (MTR). Even if initial characteristics of the virgin graphite used in both CGR and MTR are similar, the irradiation history and environmental conditions lead to significant differences in the structure and radioactivity of the two types of irradiated graphite. The simple geometry and lower content of radionuclides determine a simplified approach of the management of MTR irradiated graphite compared to GCR type, but it requests the same level of knowledge and understanding of irradiated graphite as waste.
The paper presents the experimental works and results achieved under CARBOWASTE project by ICN in the characterization of the MTR irradiated graphite, using a large range of techniques (DRX, OM, SEM, EDS, XRF, gamma spectroscopy, etc.) in order to define the structure, impurities content and localization, mechanical behaviour and radioactive content. Structural properties of virgin and irradiated graphite revealed by DRX, OM and SEM indicated minor irradiation effect in a MTR thermal column.
Considerations on the potential radiologic impact of its final disposal as LIL waste are also included in this paper and based on preliminary safety assessments, possible management approaches are discussed.

II.1.3. Decontamination Factors for Irradiated Graphite from Thermal Columns of TRIGA Reactor

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The irradiated graphite in Romania arises mainly from the thermal columns of the two research reactors TRIGA 14 KW (around 3.5t) and VVR-S (around 4.5t).
It was found that the radionuclides content of the graphite irradiated in the TRIGA research reactor is mainly dominated by Co-60 (50-150 Bq/g), Eu-152 (600-1400 Bq/g), Eu-154 (50-100 Bq/g) and C-14 (1-10 KBq/g) but also it may contains Cs-137 (600-1400 Bq/g) and H-3 (100-200 Bq/g) depending on the position in the thermal column and irradiation history.
Co-60 and Eu-152, the most important radionuclides measured in the irradiated graphite are activation products of the impurities existing in the virgin graphite.
Eu-152 could also arise from the aluminum claddings and in this case it represents a contamination.
Both elements (Co and Eu) need strong solutions to react and to be removed either from the surface or from inside the graphite block.
The experiments for removal of Co-60, Eu-152 & Eu-154 consisted in contacting the i-graphite with different acids, HNO_3 65%, HNO_3 32%, HCl 37%, HCl 18%, H_2SO_4 98%, H_2SO_4 49%, H_3PO_4 85%, H_3PO_4 42% and mixtures of acids HNO_3 65% + H_3PO_4 85%, HCl 37%+ HNO_3 65%, HCl 37%+H_2SO_4 98%, HCl 37% + H_2SO_4 98%, oxalic acid 10% +citric acid 10% in different concentrations and/or different ratios.
In the experiments i-graphite samples with known activity of Co-60, Eu-152 and Eu-154 have been used.
By gamma ray spectrometry the activity of radionuclides eluted has been determined.
The decontamination efficiency (for radionuclides contented and global beta activity) was determined.

II.1.4. Membrane Distillation as a Method for Liquid Radioactive Waste Treatment

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There is a variety of methods for the treatment of liquid radioactive wastes prior to disposal. The treatment technologies employed in nuclear industry include chemical precipitation, ion exchange, evaporation, solvent extraction and also membrane processes.
There are many examples of application of membrane techniques in nuclear technology. Such methods like reverse osmosis, ultrafiltration or microfiltration can be used for liquid radioactive waste processing, cleaning the reactor waters, and separation of isotopes.
The pressure-driven membrane processes are very effective, but they also have many limitations as fouling phenomena often resulting in cleaning operations which decrease membrane resistance and produce secondary wastes. Some of these disadvantages can be avoided by membrane distillation, non-isothermal process employing hydrophobic porous membrane.
Tentative experiments were performed using non-active solutions of pure inorganic salts like NaCl or $NaNO_3$. The experiments were conducted at different temperatures and flow rates of circulated solutions. The influence of feed concentration on the distillate flow rate was also investigated. Afterwards some tests using model solutions consisting of selected radionuclides usually present in the radioactive wastes were performed. Finally, real wastes originating from institutions, which apply radioactive substances in their ordinary activity, were treated using membrane distillation process.
The experimental work performed has shown that membrane distillation can be used for a concentration of radioactive solutions. The results of experiments have proved the statement that membrane distillation can be applied for liquid low-level radioactive waste treatment. In one stage installation, radionuclides are retained by the membrane and they can be concentrated resulting in high volume reduction of the waste. The distillate obtained in the process was pure water which can be re-utilized or safely discharged into the environment.

II.1.5. Unsaturated Hydraulics and Reactive Transport Fully Coupled for the Simulation of Atmospheric Carbonatation of Concrete

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We propose a complex model of atmospheric carbonatation taking into account the full chemistry of concrete and all chemical exchanges, and the feedback of chemistry on unsaturated hydraulics. Chemistry can be both at equilibrium or kinetically-controlled (if only partially). Temperature, Saturation and CO_2 partial pressure are treated as chemical concentration in a global nonlinear system. To solve this system we use a classical Sequential Iterative Approach (SIA), with the Fixed-Point algorithm, or the Nonlinear Conjugate Gradient algorithm using an analytical jacobian.
We adapt the Sequential Iterative Approach to the possible presence of partial kinetics by creating a chemical sub-system containing only the species controlled by reactions at thermodynamic equilibrium.

Taking into account the action of the CO_2 on the reactive transport implies a strong coupling between the equation of the gas density evolution and the reactive-transport system of aqueous and mineral species. We propose here an original approach in which we allow during the inner loop of the sequential method intermediate values of the gas density to raise above the boundary condition limit in order to simulate the capture of the CO_2 by the minerals on large time steps, the final values of the gas density calculated at each time step remaining within physically acceptable values.
The use of a complete chemical model instead of a simplified one allows us to have a direct access to the chemical outputs such as the pH, which are very important for the studying of concrete durability, for example in the context of nuclear waste storage.

II.1.6. Alliances Case Study: Flow Field in Unsaturated Zone of the Saligny Site

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This paper targets to describe the flow field - through variables such as pressure head, saturation, Darcy velocity – in the geological layers of the unsaturated zone at Saligny. Saligny site is currently the location chosen in Romania for a LLW repository which is foreseen to be built beginning with 2017.
The simulation has been performed with the Alliances platform, developed by CEA, ANDRA and EDF and licensed to INR in the framework of the project NSRAWD – Numerical Simulations for Radioactive Waste Disposal. Alliances provides overall graphical user interface, geometric modeling, mesh generation and post-treatment visualization, supplemented by powerful capabilities such as choice and coupling of different numerical components and simulation of multi-physical and multi-scale phenomena, making it a valuable tool for simulations associated with nuclear waste repositories.
The model describing the geological environment of this case study contains seven media in their natural state: silty loess, fossilized silty loess, upper clayey loess, fossilized clayey loess, lower clayey loess, red clay and prequaternary clay, stratified until a depth of 54 m, above the water table level. A mesh with a 0.5 m discretization was generated for the computation. Data on porosity, density, moisture content, suction curves were previously determined through on-site tests and gathered in a database. RETC code was used for obtaining the residual water content and the fitting curves regarding volumetric water content vs. pressure on the existing data. The Van Genuchten parameters - α, m and n - were then determined according to the relationship between the effective saturation and the suction which was implemented in Alliances.
The unsaturated hydraulics problem has been solved with the numerical component CAST3M of Alliances until steady state regime was reached. The boundary conditions imposed were a water inflow of 20 mm/yr at the top boundary, saturation of 1 at the bottom boundary and a zero flux on side boundaries. The initial conditions considered a saturation of 0.8 in the red clay and prequaternary clay layers and 0.5 in all the other layers.
The saturation field obtained through simulation was compared with the profile from data experimentally recorded, showing a good correlation.

II.1.7. Determination of Alpha Radionuclide Concentration in Different Types of Radioactive Waste

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In order to reduce radioactive risk on short term and long term for the environment and population, the radioactive waste has to be treated, conditioned and stored in appropriate conditions. In every country the regulatory body for nuclear activities establishes acceptance criteria for storage or disposal of radioactive waste package. As time goes on, the acceptance criteria become more and more restrictive and the number of radionuclides required to be quantified increases. Nondestructive and destructive characterization methods are used to check whether the radioactive waste meet the acceptance criteria.
Quantification of alpha emitters requires a special attention because of their long decay time. In order to measure the alpha emitter content of a sample, alpha spectrometry techniques are used. Our experimental approach, which is a destructive analysis method, consists basically of the following steps: preliminary treatment of the sample (sample dissolution, adding of spike solutions), chemical separation (ionic exchange, precipitation) and thin layer source preparation. Once these steps were performed, the activity of alpha source previously obtained is measured using an ORTEC dual alpha spectrometric system.
The paper contains a description of the method, a presentation of the system for alpha spectrometry and recent experimental results. The experiments presented were done mainly on i) radioactive waste resulted from the operation of Post Irradiation Examination Laboratory and TRIGA research reactor of Institute for Nuclear Research Pitesti and ii) liquid radioactive waste from operation of Cernavoda Nuclear Power Plant.
This experimental work reflects the current interests of the “Laboratory for characterization of spent nuclear fuel and high/medium level radioactive waste- LABORAD” within which it was done.

II.1.8. Radioactive Waste Management in Bulgaria

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Bulgaria supports the development of nuclear energy, its nuclear program started in 1970 by the construction of the first block unit of Kozloduy Nuclear Power Plant (KNPP). In 20 years on the same site were built 5 more power units.
The future and the major priority of SERAW at the moment is building a national repository for radioactive waste disposal. In 2015, it is expected to finalize the construction of a national repository for low and medium law radioactive waste (NDF).
Long-term target is to provide safe, effective, safe and socially - acceptable radioactive waste management in the context of sustainable development of nuclear energy and nuclear applications in the country.

II.1.9. The Behaviour of TRISO Fuel Matrix Material in a Sealed CO_2 Atmosphere, related to a Small-Scale Reactor

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The matrix material used to produce fuel compacts or pebbles containing TRISO particle fuel is not true graphite, and is instead composed of graphite powder and semi-graphitic material bound with resin. As a result the materials behaviour in a CO_2 atmosphere is not well understood, despite extensive work on nuclear grade graphites in similar environments. In order to examine the possibility of using TRISO fuel within a CO_2 atmosphere for a small-scale battery-type design, its oxidation behaviour must be well understood from both thermal and radiolytic mechanisms. Work is underway between TU Delft in the Netherlands and the University of Manchester to initially investigate both mechanisms effects on A3-3 matrix material, and compare reaction rates to NGB-18, a standard modern nuclear grade graphite, and Gilsocarbon, a graphite currently used in the UK AGR fleet. Irradiation rates will be investigated using TU Delfts test reactor, alongside gas chromatography at Manchester, providing data on gas composition as well as weight loss. Thermal oxidation work will be conducted at The University of Manchester, and will be based around the maximum proposed operating temperature of the reactor of 750°C.

II.1.10. A Comprehensive Overview of Upgrade of Waste Department from INR for Ensuring High Quality in Safety, Maintenance and Processing Activities of Radwaste Resulting from Fuel Cycle with Natural Uranium

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The Institute for Nuclear Research Pitesti (INR) has its own Radwaste Treatment Plant (STDR) that was commissioning in 1984. The object of activity is to treat and condition the radioactive waste resulted on the site of INR and some other nuclear facilities.
In STDR are also treated, with uranium recovery, the liquid and burnable solid radioactive wastes resulted from the production of CANDU type fuel in Nuclear Fuel Plant (FCN) Pitesti, which is sited in the same place as INR Pitesti.
In order to reach a full compliance with the International Safety standards that evolved over the last three decades, INR decided to upgrade the STDR, with the main objective to bring the installation to a safety level compatible with current international practices.
The implementations of these modifications are in compliance with the recommendations of the chief designer from Subsidiary of Technology and Engineering for Nuclear Projects (CITON), following the Governmental Decision No.626/2009 for upgrading of the Radwaste Treatment Plant.
This paper presents of the implementation measures associated with the advanced technologies and the conditions required by the National Commission for Nuclear Activities Control (CNCAN) for plant operation extension above the designed life-time.

II.1.11. A Critical Analysis on Catalyse of Hydrogen-Water Isotopic Exchange Process

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The isotopic exchange between hydrogen and water represents one of the key point of tritium removal technology from tritiated heavy water resulted during of CANDU reactors' operation. At normal reaction conditions, the rate of isotopic exchange it's very low and therefore in order to increase the efficiency of the process, the use of an appropriate catalyst with high catalytic activity and long lifetime it's necessary.
Based on the long experience of the authors in preparation, testing and evaluation of the performances of the catalyst and of the isotopic exchange process, the paper presents a critical analysis on up-dated scientific information, obtained by the authors or communicated by the others researchers from abroad.
Critical analysis it's focused on:

- selected types of catalysts;
- methods and conditions for preparation and shaping;
- utilization ways and results;
- improvement of the performances of the proposed catalysts for industrial nuclear applications;

As result of this assessment, a new improved catalyst based on platinum deposited on carbon moulded with poli-tetra- fluor- ethylene (PTFE), obtained in new conditions and with improved physico-structural parameters has been proposed by the authors. The tests concerning the main characteristics of the catalyst are underway and preliminary results are very promising.

II.1.12. Control System Architecture Applied to a Storage Tritium Environment

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A critical issue, in direct connection with the tritiated heavy water processing in a detritiation plant, is the safe storage of the obtained tritium, both for environment and operating personnel. The properties of the tritium storage used materials refers to the ability to high "connect" the tritium, to have a large storage capacity, to be easily achievable and, if necessary, to allow its easy recovery. The metals and intermetallic compounds are the most used materials. Since the reaction of metal hydride formation is, in most cases, sever exothermic and for many materials almost spontaneous, a new control system architecture applicable to tritium storage media is presented. This architecture uses the resources of a DCS (Distributed Control System) which is based on i-processor technology (able to integrate all typical control functions, friendly display options, an alarm management system, historical data bases and advanced control tools). These resources usage allows us to operate a Tritium storage system under automatic control and to use an advanced Operator Interface too. The DCS closed loop algorithm is used for manual and automatic closed loop control for monitoring the corresponding operating conditions.

II.1.13. Measurements of Hydrogen Isotopes Mixtures Using a Quadrupole Mass Spectrometer of Low Resolution

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To determine the concentrations profiles of hydrogen isotopes species along the cryogenic distillation column for hydrogen isotopes separation, in the Pilot Plant for Tritium and Deuterium Separation from ICIT Rm. Valcea, a GSD 320 quadrupole mass spectrometer with unit resolution will be used. The major issues in the analysis of hydrogen isotopes mixtures by low resolution mass spectrometer are the isobaric interferences due to ionic species, other than those being examined, that are produced inside the ion source by dissociation of molecules, or by reactions between ions and neutral molecules. High precision and reliability measurements of hydrogen isotopes by mass spectrometry require an extensive knowledge of instrumental effects and understanding their underlying causes. In this respect, we performed several experiments in order to characterize the dependence between the ion current intensities and the partial pressures of the gas components, to identify the sources of deviations from linear behaviour, and to specify and determine the isobaric interferences at masses of interest. High purity gases (Hydrogen and Deuterium), and different mixtures of them, prepared in our laboratory, were used in our experimental procedures. Collection of experimental data was done over a relatively wide range of pressures in the ion source. This work presents the experimental results, theoretical interpretation of experimental data and the correction algorithms of isobaric interferences that are needed for an appropriate calibration of the mass spectrometer.

II.1.14. Advanced WATER-HYDROGEN ADSORPTION SYSTEMS for Pilot Plant for Tritium and Deuterium Separation

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ICIT Rm. Valcea developed the cryogenic technology of tritium separation from tritiated heavy water. Equipment is composed of an catalyzed isotopic exchange module how the tritium is extracted from tritiated heavy water. An important process in this way is ripping extract moisture from wet hydrogen isotope exchange column. This paper studies the possibility of increasing the efficiency of adsorption systems of cryogenic pilot plant for separation of tritium and deuterium. In the present configuration, the cooling is done by passing hydrogen gas through the adsorber where we find within the molecular sieve 13x. Tests were made for both heating phase (adsorption) and for the regeneration (desorption). The conclusions presented in this study show the possibility of using nitrogen cold for the stage of adsorber regeneration. Cold nitrogen derived from liquid nitrogen station, which has a daily rate of loss of cold gas, thus making the recovery of energy dissipated into the environment and an important environmental factor. The conclusions presented in this paper reveals that by using dual cooling systems (inside and outside) can increase the efficiency of adsorption systems, reduces cooling time required and increase the use of molecular sieve. Monitoring of process parameters was done by two methods, infrared thermography and thermocouple type k with the computer data acquisition system via USB 2.0 interfaces.

II.1.15. The Influence of the Contaminant Interaction with Geological Media on the Subsurface Transport

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Near surface disposal is considered a suitable option for low and intermediate level (LIL) radioactive waste disposal containing mainly short lived radionuclides that decay up to insignificant radioactivity level in few decades or centuries. Long term performance of this kind of radioactive waste disposal facility depends on the radionuclide sorption on technical barriers (conditioning matrix, filling materials, the liner material, the engineered foundation and so on) and also in geosphere (geological layers finding below repository).
C-14 is one of the radionuclide of concern presented in significant quantities in the LIL waste generated from Cernavoda NPP operation and decommissioning that will be disposed of on the Saligny site. For this radionuclide an extensive experimental program consisting of batch and column experiments were performed to determine its sorption characteristics on the technical and natural barriers that will form the Saligny LIL waste disposal facility.
Using the experimental data obtained in characterization of the C-14 sorption, FEHM simulations were performed to estimate the performance assessment of the Saligny site. These simulations showed that even the sorption of C-14 on the Saligny natural barriers is low it has a beneficial influence on the rate and concentration at which C-14 will enter in the saturated zone of the disposal site (Beriasian aquifer). But significant influences on the rates and concentration of the C-14 releases have its interaction with the cement matrices and with the improved loess layer that will comprise the repository foundation.

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

II.2.1. Radon Emanation and Uranium Concentration In Phosphate by Passive and Active Methods

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The objective of this work was to assess the radiological risks to human health since it is important to have the knowledge of radioactivity levels of the phosphate industry. Because of their geological history, phosphate ores may contain an important amount of natural radioactivity from ^{238}U and minor amounts from ^{232}Th and ^{40}K. Knowledge of the concentrations and distributions of these radionuclides is of interest since it provides information on environmental radioactivity. To assess the impact of the phosphate industry on the environmental radiation level of a region, we evaluate radon emanation and uranium content of phosphate samples from the Djebel Onk mine situated in eastern Algeria. Solid state nuclear track detectors LR115 and CR-39 and an AlphaGUARD ionization chamber were used in tandem for ^{222}Rn detection in the same container. Specific activities of ^{238}U and ^{222}Rn for different phosphate grain sizes were performed using LR115 and CR-39. The radiological impacts of the phosphate sample were also measured by □-ray spectrometry. Results for uranium content and radon emanation obtained by the different methods show excellent concordance.

II.2.2. CNE Cernavoda - Management of Internal Tritium Exposures

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During normal operation of a CANDU nuclear power plant significant tritium quantities are generated. Through design solutions that have been implemented we manage to control the tritium losses from the reactor systems and keep them as low as achievable.
Special dryers are designed and are used to remove moisture from different ventilation systems of a CANDU reactor in order to maintain tritium in air concentration and gaseous tritium emissions below the limits established by the national authorities. Vapor Recovery System is designed to control tritium in air concentration and to recover heavy water loss from PHT and Moderator Systems and to control the air circulation, providing atmosphere separation between different areas of the Reactor Building.
CNE Cernavoda developed a special strategy in order to control workers' internal exposures to tritium and dedicated programs are running to implement this strategy: improvement of radiation protection practices, increasing equipment performances, leakages prevention through maintenance program and finalize the de-tritiation facility.
This paper presents the actual situation of workers internal exposures and emphasize the results of the ALARA policy promoted by Cernavoda NPP management.

II.2.3. Cernavoda NPP – Using Dose Constraints as ALARA Instruments

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At Cernavoda NPP, the 20 mSv/y individual legal dose limit is the only dose constraint required by the national regulatory body:
Nevertheless, since the beginning we established an administrative individual annual limit of 18 mSv, 2 mSv lower than the legal limit, supported by:
- the Dose Control Point (DCP), representing half of the available individual dose until reaching the administrative limit of 18 mSv;
- 2 mSv investigation level for external single exposure
- 1 mSv committed dose, investigation and removal level for internal exposure (single intake).

Since 2007 when the ALARA program was implemented more individual and collective dose objective were established as ALARA performance indicators:
- maximum individual dose at the end of the year – 7 mSv for 2011 (planned exposures)
- maximum individual internal dose due to tritium intake at the end of the year – 4 mSv (planned exposures).

For the specific jobs in the RWP are established individual (total, internal and external) and collective dose limits. If these limits are reached before finishing the job the RWP is re-evaluated and the individual dose limits can be exceeded only with the approval of Station Health Physicist.
The actual levels of individual and collective effective doses reveal the effectiveness of implementation of the Radiation Safety Policies and Principles established by the management of the Cernavoda NPP, based on the ALARA principles.

II.2.4. A New Passive Collector for Airborne Tritiated Water

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Tritium is one of the radionuclides with the highest weight in the composition of the radioactive emissions from a CANDU plant. Most of the tritium, which is in the plant's circuits, is in the form of a molecular compound, DTO, thus being closely linked to the movement of heavy water within the plant. The presence of inevitable heavy water leaks within technologic rooms results in exposure hazard for personnel. Therefore airborne tritium monitoring is mandatory, especially during interventions and outage activities, being performed with both real-time monitoring systems and portable devices.
The paper presents the results obtained during the development and testing phase of a passive collector for airborne tritiated water using calcium chloride as desiccant. Tests were done to establish the loading capacity of the collector and the accuracy of the tritium concentration determination. Dynamics of the sorption was studied and a model was proposed for the sorption mechanism. The model parameters for a particular configuration and testing conditions were obtained and a brief description of the working procedure is included. The minimum detectable concentration for standard temperature/pressure conditions and saturated atmosphere was estimated to less than 10 Bq/m^3 which means four order of magnitude lower concentrations than the derived air concentration limit for tritium. The method's performances recommend it as a rapid and inexpensive way to determine the average concentration of tritium in air, for both indoor and outdoor applications.

II.2.5. Environmental Radioactivity Monitoring Results at INR Pitesti after Fukushima Accident

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The paper presents the results of environmental radioactivity monitoring at Institute for Nuclear Research Pitesti site, following the nuclear accident from Fukushima, Japan. Two weeks after the accident the regular monitoring program at the institute was supplemented by including continuous air sampling for iodine with impregnated charcoal filters and increasing of the deposition sampling frequency. When I-131 was identified in wet depositions (about three weeks from the accident) in situ gamma spectrometry measurements were performed to estimate soil deposition of radionuclides. In the paper measurement results are presented for the ambient equivalent dose rate, gross beta activity concentration in air, I-131 concentration both in air and wet depositions and also, the results of the in situ gamma spectrometric measurements. The meteorological data were included, for the reported period, serving as basis for the interpretation of the radiological measurement results.
One estimates that starting from March, 27 in the area of Mioveni, Arges, the I-131 concentration in air reached measurable values laying around 1 mBq/m^3. This concentration is meaningless from the point of view of public supplementary exposure as compared to the natural background.

II.2.6. Integrating Radiation Monitoring Systems for Nuclear Facilities

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Nuclear safety – and especially radiation monitoring - has always been tantamount licensing the nuclear installations, and even more so after the unhappy event from Fukushima. Therefore, the tendency is to integrate RMS systems into a unique system that can give the operator at-a-glance insight into what is going on in various points of the installation. The best way to do this would be to have systems as similar as possible doing various things. As an example, gamma portal monitors, whole body contamination monitors and hand foot monitors should all use the same software and use the same components (this would allow a single set of spare parts being used for all types of monitors, shortening very much the downtime for service) – one example being Canberra's GEM5 gamma portals, ARGOS W/B monitor series, SIRIUS H/F monitor series and CONDOR tool monitor series. Furthermore, such systems should be connected to an overhead that would allow them to send the data to an RMS software that collects not only what these systems are measuring, but also what the hand held instruments are measuring, what the FAGM is measuring and all other RMS from the installation.
The present paper will present an example of fully integrated RMS and the way the components are communicating with the system

II.2.7. Role of the Radionuclide Metrology in Quality Assurance of the Enviromental Physics Measurements

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This paper presents some aspects of the technical support offered by the Radionuclide Metrology laboratory from IFIN-HH in the field of environmental physics measurements.
-The development of the basis of installations and methods for absolute standardization of radionuclides;
-The assurance of the equivalence between our primary activity standard and other national standards;
-The continuity of the traceability chain, assured by the development of secondary activity standards.

II.2.8. Radioactivity of Some Underground and Ground Waters

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In this paper we present the results of radioactivity measurements on some underground and ground water flows in four locations where a cleaning process has been previously applied. The measurements have been performed using a Polyradiation Meter MIP21 equipped with a SG-2 Gamma scintillation counter. We identified the main sources for water pollution:

- Mining galleries bring to surface, by pumping or direct drainage, many dissolved ores that have a very high toxic content. Likewise, through characteristic works, the galleries cross clean underground water layers.
- The mining waste dumps are continuously being washed by the waters in the gallery, by the rivers in the area or by the meteoric waters. Due to physical characteristics of the water (incontrollable flows, freezing, pH changing) they can influence the stability of the waste dumps.
- The mining decantation devices are strong pollution sources even after their stabilization. The material in the decantation devices can end up in the surrounding water in more than one way.
- The primal processing mining projects are pollution generators even after they are finalized. The soil remains contaminated at a significant depth for an undefined period of time. The migration of the radionuclides and heavy metals are difficult to control.

The samples were sent to some collaborators for further analyses. For some radionuclides the measured values of the radioactivity are in the legal limits while for others these limits are exceeded.

II.2.9. Tritiated Effluents Releases in Danube Black Sea Canal

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Tritium is produced in reactors by neutron activation of ^{2}H, ^{3}He, ^{6}Li and ^{10}B, and is thus formed in the different reactor components. Typical tritium discharge rates in liquid effluents (IAEA, 2004) varies from 3.73×10^{3} GBq $Gw(e)^{-1}a^{-1}$ for BWR (Boiling Water Reactor) to 1.85×10^{5} GBq $Gw(e)^{-1}a^{-1}$ for HWR (Heavy Water Reactor).
Nuclear Power Plant (NPP) Cernavoda is situated 2.6 km from Cernavoda town on the border of Danube Black Sea Canal, and it has two HWRs (CANDU type) fully operational. During October – December 2009, the liquid effluents from NPP were evacuated in Danube Black Sea Canal (DBSC). Beside the monitoring program of NPP Cernavoda, a sampling campaign was conducted in November 2009 in order to evaluate the tritium level in DBSC water, which is water supplier for different human communities (Medgidia, Constanta, etc.). The water was sampled from surface water one meter from the left border, and from the DBSC middle (from surface water and 4 m deep and each 5 km along the canal). The paper present a synthesis of the tritium measurements made over 70 samples from mentioned area. The average tritium activity concentration was 6.7 Bq/l with the maximum tritium activity concentration measured in Agigea of 13.3 Bq/l. The recorded values prove the compliance of NPP Cernavoda practice to Council Directive 98/83/EC on quality of water intended to human consumption, where tritium level is accepted to a maximum level of 100Bq/l.

II.2.10. The Cernavodă NPP Environment Radioactive Controled Releases Limiting, the Key to Increase the Public Confidence in Nuclear Power Energy in Romania

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The objective of the management of radioactive waste is to protect human health and the environment from the harm associated with these contaminants due to direct exposure and bioaccumulation in the food chain. Environmental protection includes the protection of living organisms other than humans and the protection of natural resources, including land, forests, water and raw materials, together with a consideration of nonradiological environmental impacts.
In assessing the impact of radioactive releases, the possible exposure of an individual in the immediate vicinity outside an exclusion boundary area, and of the public at large, should be considered. Appropriate Programmes on Effluent and Environmental Monitoring are therefore required for each nuclear facility to ensure protection of the public from radioactive discharges and in this paper are presented some aspect of these Programmes at Cernavodă NPP.
Cernavodă Nuclear Power Plant is dedicated to generate electrical and thermal power, safely and efficiently for at least 40 years. Cernavodă site has 2 operating units, each of 706 MW and 2 units in construction. CANDU nuclear power stations have consistently proven to be competitive with other types of nuclear power plants, while offering unique advantages to their operators. Some of the design features and unique characteristics of the CANDU-6 reactor are presented in this paper.
The Cernavoda NPP Units site is delimited by a physical protection fence, outside of this being permitted a maximum annual limit of 1 mSv (annual effective dose for public provided by Radiological Safety Fundamental Norms).

These maximum permitted limits which regulate the normal radioactive emissions at Cernavoda NPP are called Derived Emission Limits (DEL).
DEL is the maximum permissible concentration for a single radionuclide or a group of radionuclides, so that the annual dose limit for an individual in the most heavily exposed group (the critical group), is not exceeded.
All the regulatory commitments included in both Operating and Maintenance License and Environmental License were fulfilled at Cernavodă NPP.
Radionuclide discharge rates were continuously monitored and controlled.
No accidental release, which could produce the increase of radiation levels above limit values, was produced from the beginning of operations at Cernavodă NPP.
In these condition the protection of public and environment objectives are accomplished and in this case the public confidence in nuclear power generation increase. Thus, nuclear energy will become in Romania more attractive and construction of Units 3&4 at Cernavodă NPP will be of paramount importance. In this situation the production of energy in Romanian NPP will increase at about 40% from total amount of energy produced.

II.2.11. Considerations about Iodine 131 Concentrations in Different Environmental Samples after the Fukushima and Chernobyl Nuclear Accidents

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Beginning with the middle of nineteen century several nuclear atmospheric releases have been recorded, but the nuclear accident from Chernobyl remain significant through the high amount of radionuclides ejected into the environment. Following this, many European countries were affected. Principal radioactive elements measured in the air at the beginning of May 1986 were ^{131}I, ^{132}Te, ^{103}Ru, ^{137}Cs, ^{134}Cs and ^{140}Ba. The higher ^{137}Cs deposition reached 80 kBq/m^2 in Transylvania area. Iodine inventory of a roof sediment collected from Cluj-Napoca in May 17, 1986, showed a value of 172 kBq/m^2 and consumption milk had a maxim activity of 1370 Bq/l before May 10.
The radioactive releases from the Fukushima reactors, beginning with March 11, 2011, could be detected in different environmental samples collected from Cluj county and surroundings. All the samples measured between 25 of March and April, showed the presence of radioactive ^{131}I, but in much smaller concentrations that the ones founded in 1986. The ordinary measurements made in rainwater give us reasons to believe that the maximum iodine activity in Cluj area was reached around the 5th of April, when a value of 2.09 Bq/m^2 had been calculated after the rain from 4th and 5th of April. The highest activity has been found in sheep milk collected in 5th of April from Cluj area. The measured concentration was about 9.22 Bq/l, higher that the maximum ones reported in France, 2.12 Bq/l in sheep and goat milk, but close to the ones reported in Italy, 5.24 Bq/l. Measurements have been also made for cow milk, goat milk, grass, pollen and eggs, by the reason of the plants-animals-food chain in nature. As reported before, iodine has a strong affinity for organic matter, thus the highest concentrations have been measured in sheep milk. ^{134}Cs has been observable in air, rainwater and grass, but not in milk.
Radioiodine is considered as one of the most hazardous radionuclides after accidental releases of radioactivity into the environment, the main concern being radioiodine accumulation in the

thyroid gland inducing thyroid cancer, especially in young children. The radioactive levels measured in Cluj county and surrounding areas showed insignificance levels, yet the measurements must continue for tracking any change in iodine concentrations. For the moment there are no signs of worry about the Fukushima radioactive pollution in territory.

II.2.12. Establishing optimum ratio water: cocktail and calibration curves for low level liquid SCINTILLATION measurements

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In this paper we describe a method used to establish calibration curves for low level tritium concentration in aqueous samples. The cocktail used was one of new liquid scintillation cocktail commercially available, ULTIMA GOLD uLLT. Using an optimization function we established optimum ratio water:cocktail based on maximum volume of the counting mixture stated by the manufacturer. To obtain the most accurate tritium results we calculated the counting efficiency and best counting window; these parameters were used for all measurements. We also established two calibration curves of efficiency as a function of quenching parameter, t-SIE. t-SIE is independent of sample isotope and of the activity in the vial, it's very reproducible means of tracking quench in the cocktail. Quenching agents used were carbon tetrachloride (CCl_4) and water, which closely approximate the chemical environment in the samples. We also evaluated the accuracy of each calibration curve for tritium measurement. All measurements were carried out using liquid scintillation analyzer TRICARB 2800TR from PerkinElmer.

II.2.13. The Preoperational Monitoring Program for Environmental Radioactivity around the Pilot Plant for Tritium and Deuterium Separation ICSI Rm. Vâlcea

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The goal of this paper is to present the results of the environmental radioactivity monitoring program, in preoperational stage, for the Pilot Plant for Tritium and Deuterium Separation ICSI Rm. Vâlcea, from 2006 until 2010.
This preoperational monitoring program for environmental radioactivity is a requirement of Romanian nuclear regulatory body, CNCAN, through NSR 22 – regulation regarding the monitoring of the environmental radioactivity around a nuclear or radiological plant.
The objective of this environmental radioactivity monitoring program consists of measuring the concentrations of some specific radionuclides in environmental factors, to underline the changes, on short or long term, in the level of radioactivity in the food chain specific to the area, as a result of the operation of the Pilot Plant.
Between 2006 and 2009 all these measurements were made by the Radiation Protection Laboratory from SCN Pitesti, and for 2010, all the measurements were made by Radiation Protection Laboratory from ICIT Rm. Valcea, which has been notified by CNCAN, in January 2010, as a laboratory for environmental radioactivity measurements.
The results of measurements of the following samples, each with its specific frequency and measurement type, will be presented in this paper:

- air humidity, rain water, drinking water, milk, meat, fruits, vegetables, cereals in order to determine the tritium concentration;
- water, fish, aquatic vegetation and sediment from Olt river, soil and spontaneous vegetation from several locations in order to determine the concentration of tritium and gamma-emitting radionuclides, and the beta-global activity;

For tritium measurements in water samples we used a liquid scintillation counter (LSC) of low level Quantulus 1220. For gamma measurements we used a high resolution gamma-spectrometric chain with HPGe detector, and for global beta activity we used a low background radiometer, model MPC9300.

II.2.14. Measurements of low activity levels of ^{3}H and ^{14}C using a TriCarb 2100TR Liquid Scintillation Analyzer

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For the assessment of exposed radioactivity coming from nuclear facilities, sensitive and economic methods are needed for measuring radionuclides. For many radionuclides – especially those with very low beta particle energy emission like H-3 or C-14 and all type of alpha particle emitting radionuclides – liquid scintillation counting (LSC) is a simply accomplishable and up-to-date activity measurement technique.
Radiation Protection and Environmental Protection Laboratory from the Institute for Nuclear Research – Pitesti operates a Tri-Carb Liquid Scintillation Analyzer (LSA), model 2100TR. Initially it was intended to be used for the determination of beta-emitting radionuclides concentrations, mainly tritium and carbon-14, in different types of samples collected in the activities developed within the institute, especially at the Radioactive Wastes Treatment Station. Generally, these samples have activity concentration levels sufficiently high to allow accurate measurements using this model of analyzer. According to the manufacturer recommendations, the Tri-Carb 2100TR LSA is typically used for samples with activity above 750 dpm or, equivalent, 12.5 Bq. However, often it is necessary to measure other kind of samples, such as biological or environmental samples, which are characterized by significantly lower activity concentration levels. For example, effluents and environmental radioactivity monitoring programs from a nuclear plant require sampling and measurement of a large number of samples, whose activity is generally lower by over an order of magnitude than the regulatory levels. Accurate determination of this level of activity would involve a considerable consumption of resources and time. In such situations, the regulation and control authorities does not require performing activity measurements with a very good accuracy, most important being to ensure that the specified reference levels are not exceeded.
Considering the above mentioned, we set up an experiment to evaluate how accurately our analyzer measures low levels of activity, even in the vicinity of its statistically determined detection limit. Experiments were focused on measurement of two of the most important β-emitting radionuclides in nuclear energy, tritium and carbon-14. The results showed that the TriCarb 2100TR LSA can provide accurate results (±5%) for activities of tritium and carbon-14 higher than 0.5 Bq/sample, this activity level being more than twenty times lower than the recommended level for this type of analyzer. The instrument provides results with not so good accuracy (±35%) for activities of tritium and carbon-14 of about 0.1 Bq/sample (more than two orders of magnitude lower than the recommended activity level). However, this accuracy can be

good enough when a sentence like "activity must be lower than a threshold value" is to be satisfied. It is not recommended to perform measurements near lower limit of detection (LLD) of the device. Reliable results were not obtained for H-3 samples; also, although maximum deviation from the real activity of the C-14 samples was less than 46% and this result could be interpreted as acceptable for an activity level just 1.5 times higher than the LLD, not enough samples and measurements were made to draw this conclusion.

II.2.15. INR'S Technical Capabilities in the Field of Radiation Protection and Environmental Radioactivity Monitoring

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The development of nuclear power and other applications of radionuclides are associated with unavoidable releases of radioactive materials into the environment. In order to assess the radiological impact or consequences of environmental release of these radioactive materials is necessary to have a comprehensive monitoring program.
The *Radiation Protection, Environment Protection and Civil Protection Laboratory* within the INR has the responsibility to insure the radiation and environment protection for our own nuclear facilities and also, to sustain an R&D program destined to increase proficiency in human and environment protection against radiation associated risks.
The aim of this paper is to present the main INR's technical capabilities in the field of radiation protection and environmental radioactivity monitoring. To insure the radiation and environmental protection for our own nuclear facilities, the *Radiation Protection, Environment Protection and Civil Protection Laboratory* is fully equipped with high-efficient devices that allow:

- radiation survey (gamma, neutrons, contamination);
- "in situ" characterization of gamma radiation fields and gamma emitter radionuclides identification and determination;
- air samples and different environmental samples contamination measurements;
- sampling preparation for environment radioactivity monitoring and radiochemical measurements;
- gamma emitter radionuclides concentration measurement in environmental samples;
- beta emitter radionuclides concentration measurements in liquid samples.

Also, The CROWN (**C**haracterization of **R**adi**o**active **W**astes and **N**uclear Materials) Laboratory was created within the *Radiation Protection, Environment Protection and Civil Protection Laboratory* to provide services in the field of radiological characterization of materials from licensed nuclear practices for clearance or classification as radioactive waste.

II.2.16. Aspects Related to the Security of Radioactive Sources

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Nuclear radioactive sources are extensively and commonly used in a wide range of applications, such as: medical, industrial, agricultural and scientific research. They vary widely in physical size and properties, their level of radioactivity, and ease of access, being used under well regulated conditions. However, if the control over a small fraction of those sources is lost (accidentally or unintentionally), sometimes resulting in accidents of which could have serious consequences.

To ensure ***the safety*** of radioactive sources it requires controlling exposure for radiation in two ways: directly and as a consequence of incidents, so that the likelihood of harm attributable to such exposure is very low.

To ensure ***security*** of sources requires that measures to be applied in order to prevent unauthorized access to radioactive sources at all stages of their use and life cycle, as well as loss, theft, and unauthorized transfer of nuclear radioactive sources.

Safety and Security aspects are linked to many of the measures designed to be addressed to protect persons, property and the environment from the effects of radiation exposure. For this reason, the safety measures and procedures already in place for some sources may already meet the basic security needs. Security of radioactive sources has become an issue of serious public concern after the terrorist attacks of September 11, 2001.

The purpose of this paper is to set out the concepts of security and the appropriate measures related to the radioactive sources (security culture and functions, the graded approach, the threats and vulnerability assessment). These measures have to be implemented at national level according to the guidelines of *Code of Conduct on the Safety and Security of Radioactive Sources* and *Guide No. 11*, *Security of Radioactive Sources.*

Also, the paper present an example of a possible security plan and some aspects related to the security of radioactive sources in Romania.

II.2.17. Extraction, Transformation and Accumulation of the Energy

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The actual humanity industrial development is indestructible linked by the energy use. The primary energy from the nature is extracted from different sources and then transformed by different processes and technologies in other kind of energy that is adequate to be used for particular application.

Sun power seems to be infinite. Sun energy is sent every time to the Universe without rest. Recent energy production state needs new technologies and continuous progress in finding different and efficient solutions for sustained and upraising energy demand. The necessity for energy is in continuous increasing evolution also due to new and large consumers, imposing stable sources for environment friendly energy and industrial production processes.

The actual energy sources are based on fossil or nuclear ones. But environment change and destruction imposes to use those technologies for energy production that permit no changes on short and log terms in nature and adjacent life.

Energy production needs continuous monitoring; actual development of the new technologies and discoveries of new materials and efficient technological processes offers the opportunities for appropriate implementation and use of them in the energy systems configuration and operation.

Storage of large energy is very difficult; there are only few options in large hydro lake, in primary fuels, in converting the energy in different kind of energy or products that could be reconverted in the first one. Until our days, in electrical accumulators a low amount of energy could be accumulated or let say stored.

Large accumulation of energy is in the lake's water, in the deep soils, in environmental air. Many time this energy is extracted by using different means: air conditioning devices, heat pumps.

The paper presents:

- The state of the art in the level of actual and useful technologies for energy source finding and use;
- The actual energy production issues;
- The actual technological limits that need to be over passed for energy extraction, transformation and storage;
- The important devices that exist in order to extract, convert and storage the energy;
- The use of energy extraction, conversion and storage devices for industrial application.

III. SUSTAINABLE DEVELOPMENT

III.1. Polices and Strategies in Nuclear Research

III.1.1. International Trends in Establishing Regulatory Requirements on the Design of New Reactors

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The paper provides an overview of the trends in establishing regulatory requirements on the design of new reactors, at international level. It also gives insights on the requirements and guidance issued by international organisations in this area, such as those developed by the International Atomic Energy Agency (IAEA). The paper is focused on the so-called Generation III and III+ reactors and highlights the requirements and expectations which are to be applied to the new reactors built worldwide. Among the main differences from the requirements applied to the previous generations of reactors are the consideration, from the design stage, of specific provisions for coping with severe accidents, the increased role of the probabilistic safety assessment in optimising safety, more attention given to human factors engineering, as well as to the interface between safety and security. Consideration is given also to the numerical safety objectives or quantitative safety goals, established in various jurisdictions for quantifying the level of protection offered by the new reactors. The goal-setting and the prescriptive approaches to regulation are also discussed, together with their advantages and disadvantages. Based on the study of the various regulatory frameworks that currently include requirements specifically addressing new reactors, the paper identifies a set of good practices which should be taken into account by countries developing or revising their nuclear safety regulations.

III.1.2. Analysis of the Approaches for Public Participation in Radioactive Waste Disposal - a Romanian Perspective

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Two available deposition options – geological repositories and surface storage repositories - are analysed in specific context by different countries with nuclear programmes. Other alternatives such as extraterestrial disposal, participation&transmutation are not considered taken into consideration the lack of the maturity of technologies and also long-term safety aspects.
Most European countries recognize the importance of a continuous process of dialogue among stakeholders and encourage the construction of national arenas that encourage participation and transparency. Some approaches are developed and used in different contexts. The paper identifies and discusses approaches and techniques used in European countries and their adaptation to Romanian context and local circumstances. The methods and techniques may be grouped into four classes as

- inform, educate, share or disseminate information to public;

- discuss through two-way dialogue; the use of varied consultation and deliberation formats;
- collaborative research with volunteer communities to obtain equitable implementation;
- sharing the decision power among nuclear authorities, national and local representatives;
- empowering local communities; creation and functioning of local committees;

All the methods are aimed to move away from Decide ‐ Announce ‐ Defend classical method to public participation in decision making process method. Depending on national and local context methods adapted to low level of public involvement (or influence) to high level of public involvement may be used. The paper discuss the methods in Romanian context both used in COWAM2 CIP, ARGONA, RISKOM projects and intended to be implemented in IPPA project.

III.1.3. The Role of the Institute for Nuclear Research Pitesti in the Romanian Nuclear Education and Training

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Romania has a dynamic development of nuclear power in the last two decades. First CANDU 6 unit started operation in 1996 and the second unit in 2007. Two new CANDU 6 units are scheduled till 2017. For the next 5-10 years the needs for new qualified personnel are very high. The personnel ageing effect, both in nuclear industry, research, and support organizations, as well as the decreasing interest of the young generation for nuclear field also contribute to the scarcity of nuclear specialists. Therefore, at high level of decision the authorities should target developing, implementing and improving the appropriate and efficient nuclear knowledge management strategies (both organizational and national) through the national strategy regarding human resources.
All nuclear organizations- universities, R&D, regulatory, TSO's, industry etc. - should develop in cooperation the appropriate strategies for saving their organizational knowledge. Networking at national and international level will strengthen each partner's capabilities and will smooth the introduction of European Credit System for Vocational Education and Training.
This paper intends to underline and to analyze the knowledge management issue in the Institute for Nuclear Research, (INR), as well as it's involvement in the nuclear education and training in Romania.
This analysis tries to establish if INR has still enough trainers and quality staff for sharing their knowledge (mainly the tacit one), who are the experts and how to motivate them in the knowledge transfer and preservation.
Based on the existing tradition of cooperation with universities involved in nuclear education, INR accumulated a valuable and appreciated experience in organizing courses and course modules, printing books, sharing laboratories and researchers, offering subjects for thesis and dissertation, and accepting internships. This also enhanced the Universities' capabilities to attract top-quality students by offering competitive curricula and international exchanges
The appropriate ways for further developing and improving the involvement of the INR in the Romanian nuclear education and training are stressed.

III.1.4. The Institute for Nuclear Research Strategy on Nuclear Knowledge Preservation

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In the context in which the concern about the future human resources in the Romanian nuclear field related with the expected development of nuclear power in our country, the Institute for Nuclear Research Pitesti can be a real vector in the nuclear education and training, besides other activities.
To reach this goal, in parallel with the efforts for a valuable involvement in the national nuclear education and training, the Institute must find out the appropriate ways to improve the management of its accumulated nuclear knowledge.
This paper intends to highlight a recent started activity in our Institute related with the organizational knowledge management, namely the development of a specific strategy on the Institute knowledge preservation. This strategy was developed especially for the explicit knowledge and will be focused on the entire knowledge accumulated in the Institute from its beginning. It was presented and approved in the Scientific Council of the Institute.
It is important to mention that this developed strategy will be implemented in the near future, and if necessary it will be improved by choosing the appropriate methods and tools to preserve the organizational knowledge. It was fixed both the steps which must be performed for reaching this goal and the responsible persons for each activity developed or in progress.
We want to underline that this activity was started in December 2010 will be the base for a real, coherent and consistent knowledge management accordingly with the international efforts from NKM section of IAEA Vienna to develop the “Guide on Nuclear Knowledge Management”.

III.1.6. NUCLEAR POWER GENERATION ALTERNATIVE FOR A CLEAN ENERGY FUTURE IN ROMANIA

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World Energy Council stated that is huge scope to raise the efficiency in which energy is provided. Over 60% of primary energy is, in effect, wasted. At present 63% of the world’s electricity comes from thermal power (coal, oil and gas) 19% from hydro, 17% from nuclear, 0.5% from geothermal and 0.1% from solar, wind and biomass. Nuclear power almost completely avoids all the problems associated with fossil fuels: no greenhouse effect, no acid rain, no air pollution with sulfur dioxide, nitrogen oxides, no oil spills etc. Its impact on health and environment is related to radiation and is relatively minor. Without pretending a high accuracy of numbers, if the Romanian nuclear power reactors will be replaced by a coal plant of equivalent capacity, about 10 millions tons of CO_2 and large quantities of associated sulfur and nitrous oxides, would be discharged to the atmosphere each year. However the acceptance of nuclear power is largely and emotional issue. In all activities in which nuclear industry is involved, it takes care of the environment. Nuclear energy can have an important contribution for the future of mankind regarding the sustainable supply of energy. Security problems are universal nuclear technology management is not risk free. The nuclear industry acknowledges responsibilities and has a unique security culture. Security is not only a technical problem, but also an emotional one.

Based on the environmental monitoring program this paper tries to demonstrate that the routine radioactive emissions of Cernavoda NPP, which are limited by National Competent Authority, gives an insignificant risk increase. For assessing the environmental impacts and damage costs from exposure, IAEA's Model SIMPACTS. SIMPACT is powerful and convenient tool for evaluating external costs of human health and environmental impacts for nuclear power and other energy sources.