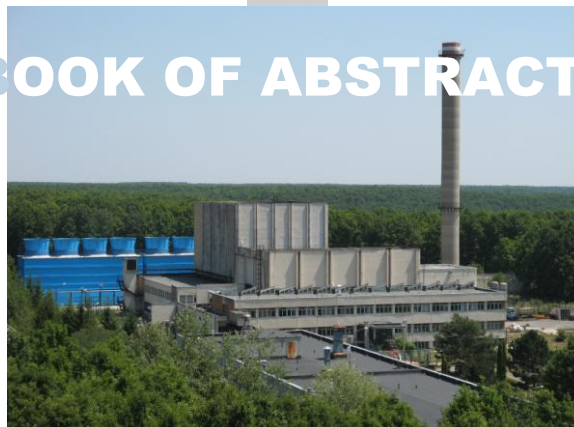




NUCLEAR 2008

Annual International Conference on
**Sustainable Development through
Nuclear Research and Education**

BOOK OF ABSTRACTS



**May 28-30, 2008
Pitești, România**



CONTENT

I.	NUCLEAR ENERGY	1
I.1.	Nuclear Safety and Severe Accidents	1
I.1.1.	Monitoring System in Labview Data Logging and Supervisory Control Module and Labview 8 with Process Analyses of the Cryogenic Pilot Plant for Tritium Removal	1
I.1.2.	Integrated System's Analyses Methods to Increase Performance in Operation.....	1
I.1.3.	Post Irradiation Tests on CANDU Fuel Sheath Irradiated In Reactivity Insertion Accident Conditions ...	1
I.1.4.	Available Post-Irradiation Examination Techniques for CANDU Fuel Performance Evaluation	2
I.1.5.	The Man – Machine – Organization System Analysis for Research Reactor	2
I.1.6.	Investigating Applications of ICARE/CATHARE Code to Air Ingress Issues.....	3
I.1.7.	Overview of Probabilistic Safety Assessment Activity for Romanian TRIGA SSR 14 MW Reactor	3
I.1.8.	Pressurization Events In Primary Side of the CANDU 600	3
I.1.9.	Modeling Human Actions for Loss of Feedwater Initiating Event Using Petri Nets Formalism	4
I.1.10.	Using Failure Mode and Effect Analysis in Identification of Components Sensitive to Ageing	4
I.1.11.	Model Uncertainty in Deterministic Safety Analysis: the Influence of Horizontal Feeders Nodalization on Fission Products Masses, Deposited in CANDU Primary Circuit	5
I.1.12.	Intelligent SSCs as Part of New Era in Nuclear Field and New Generation of Nuclear Facilities	5
I.1.13.	Application of PSA Study at Cernavoda NPP Unit 1	6
I.1.14.	Analyze of Water Hammer Phenomena – Application to Deaerator - Feedwater Pump Node	6
I.2.	Nuclear Reactors	7
I.2.1.	Minor Actinides Transmutation in the Efit Reactor: A Design Challenge.....	7
I.2.2.	Detection Systems of the Failed Nuclear Fuel	7
I.2.3.	Numerical Simulator of the CANDU Fueling Machine Driving Desk.....	8
I.2.4.	A Neutronic Properties Study of an ACR-1000 Shutoff Rod Design Proposal	8
I.2.5.	The Thermalhydraulic Behavior of CANDU 600 Reactor Core Fuelled with SEU 43	9
I.2.6.	SEU43 Fuel Bundles in CANDU 600	9
I.2.7.	Neutron Flux Measurement In The Central Channel (XC-1) of TRIGA 14MW LEU Core.	9
I.2.8.	Full Conversion of Materials and Nuclear Fuel in Research Reactor – TRIGA INR 14 Mw	10
I.2.9.	Modernization of Control-Command System for TRIGA SSR 14MW Main Cooling Circuit.....	10
I.2.10.	Analysis of Scheduled and Incidental Shut-Downs at TRIGA SSR 14 MW (2001 – 2007)	11
I.2.11.	Benchmark-type Modeling of the SCN TRIGA SSR.....	11
I.2.12.	Capsule C5 Operation Modes Analysis	11
I.2.13.	INR Pitesti Triga Reactor Instruments for Structure Analysis.....	12
I.3.	Nuclear Technologies and Materials	13
I.3.1.	Update of ITER ISS-WDS Process Design	13
I.3.2.	Endurance Tests for WDS	13
I.3.3.	Studies and Evolutions about Recovery of Transitionals Metals from Spent Catalysts.....	14
I.3.4.	Conceptual Design of a Tritium Storage Bed.....	14
I.3.5.	History on the Use of Industrial Magnesium -Thorium Alloys in Romania. Technological and Radio-Protection Aspects	14
I.3.6.	Maintaining of Primary and Secondary Circuit Chemistry – The Key of the Minimum Corrosion of Metallic Components	15
I.3.7.	Simultaneous Determination of Dissolved Elements in Water Using ICP-oaTOFMS.....	15
I.3.8.	Physico-Chemical Processes Implied in the Formation of Corrosion Deposits on the Fuel Claddings .	16
I.3.9.	Characterization of Incoloy 800 Oxides Developed Under Simulated Supercritical Water Conditions .	16
I.3.10.	Influence of Aqueous Environment Ph on the Corrosion Behaviour of the CANDU Steam Generator Tubing Material	16
I.3.11.	The Influence of Some Anions Resulted from γ Radiation Action on the Corrosion Susceptibility of Grade 2 and Grade 12 Titanium Alloys.....	17
I.3.12.	Main Decontamination Techniques Included in Decommissioning Activity	17
I.3.13.	Microbiologically Induced Corrosion of Carbon Steel Under Continuous Flow Conditions	18
I.3.14.	The Study of SCC Behavior of Oxidized CANDU Fuel Claddings Using Electrochemical Methods.....	18
I.3.15.	Spectral Dependence of Ultrasonic Attenuation for hydrided Zr-2.5%Nb alloy	18
I.3.16.	Intermetallic Compounds CoAl for γ - Sources.....	19
I.3.17.	Aspects Regarding Non-Destructive Examination of the Surface of the Ring From the Botom of the Welding Seam of two Pipes Having Small Diameter, by Replicating Method	19
I.3.18.	Analysis of G52-28 Carbon Steel Expose in Gs1 Column of Isotopic Change.....	20
I.3.19.	Round Table - Structural Integrity of Nuclear Components.....	20
I.3.20.	Constructive Solutions to Protect Nuclear Objectives Against Earthquakes, Explosions, Shocks and Vibrations	20

II. ENVIRONMENTAL PROTECTION..... 21

II.1. Radioprotection	21
II.1.1. Interference Phenomena in Azeotropic Distillation for Liquid Scintillation Measurement	21
II.1.2. Current Levels of Radioactivity along Romanian Danube Basin	21
II.1.3. Expert Services Emergence From the “Environment Protection” R&D Program	21
II.1.4. Monitoring System of Radioactive Gas Emissions from TRIGA INR 14MW	22
II.1.5. Airborne Tritium Emissions Dynamics at Cernavoda NPP, 1997 - 2007	22
II.1.6. Occupational Exposure to External Ionising Radiation: Personnel Monitoring and Dose Evaluation at Cernavoda NPP	22
II.1.7. Local Radiation Protection Measurement Network for TRIGA REACTOR Building	23
II.2. Radioactive Waste Management	24
II.2.1. Physico-Chemical Cement Matrix Characterization Tests for Ion Exchange Resins from a NPP	24
II.2.2. Treatment of Waste Water Involved in Acid Uranium Ore Processing	24
II.2.3. Possibilities of Using New Technology Materials in Constructing the Radioactive Waste Containers	25
II.2.4. Techniques for Long Term Conditioning and Storage of Radium Sources	25
II.2.5. Regulatory Review of the C-14 Inventory Proposed for Disposal at the Saligny Planned near Surface Repository	25
II.2.6. Dynamics of Radioactive Waste Generation	26
II.2.7. Development of the Conceptual and Numerical Models for Semi-Generic Site to Support Performance Assessment for CANDU Spent Nuclear Fuel Disposal	26
II.2.8. Current Status and New Trends in the Methodology of Safety Assessment for near Surface Disposal Facilities	27
II.2.9. Considerations on the safe and Security of Radioactive Materials Transport in Romania	27
II.2.10. Characterisation of Radioactive Waste at Cernavoda NPP Unit 1 During Normal Operation	28
II.2.11. Non Destructive Assay for Waste Characterization. Is it Enough?	28
II.2.12. Experimental Study for the Assessment of Conditioning Matrices of the Radioactive Aluminum Wastes	29
II.2.13. Treatment of Liquid Radioactive Waste by Membranes Techniques – Needs and Future Trends	30
II.2.14. ¹³⁴ Cs, ¹³⁷ Cs and ⁶⁰ Co Migration in Bentonite	30
II.2.15. Communication Process Related to Lilw near-Surface Repository Siting	31
II.2.16. Radioactive Waste Inventory in Romania – Current Stage	31
II.2.17. Radon and Eurocodes	31
II.2.18. The Waste Management Plan Integration into Decommissioning Plan to the VVR-s Research Reactor from ROMANIA	32
II.2.19. Radioactive Waste Management in Romania	32
II.2.20. Main Issues for Developing the Saligny near-Surface Repository	33
II.2.21. Elaboration Methodology of Determination of Scaling Factors	33
II.3. Air, Water and Soil Protection	34
II.3.1. Modern Wastewater Treatment Methods: Sonochemical Processes	34
II.3.2. Stable Isotope Signature (D/H, ¹⁸ O/ ¹⁶ O, ¹³ C/ ¹² C): Important Element in Paleoclimate Reconstruction	34
II.3.3. Greenhouse Gases Effects on Global Warming	34
II.3.4. Residual Pesticides Analysis in Wines By Gc/Ms	35
II.3.5. THMs Determination by GC/MS	35

III. SUSTAINABLE DEVELOPMENT..... 36

III.1. Strategies in Energy	36
III.1.1. Nuclear Electricity Generation - A Sustainable Energy Resource for Romania Along the next Two Decades	36
III.1.2. Nuclear Power a Viable Alternative in Global Warming Context	36
III.1.3. Perspectives and Constraints in the Romanian Energy	37
III.1.4. Generation-4 Challenges for Research	37
III.2. Education, Continuous Formation and Knowledge Transfer	38
III.2.1. Human Factor as Nuclear Safety Element	38
III.2.2. Nuclear Physics Education Today	38
III.2.3. CPSDN – IFIN-HH Involvement in the Nuclear Education	39
III.2.4. Nuclear Education and Information Technology in the 21 st Century	39
III.2.5. The State of the Art of the Tacit Knowledge Transfer and Preservation in INR	40
III.2.6. The Website – Necessity Or Whim?	40

III.2.7.	Embedded Learning Methods Implemented in Continuous Education and Knowledge Transfer of the Personnel Working in Radioactive Environment	41
III.2.8.	Education and Public Information on Nuclear Power from a Student's Point of View	41
III.2.9.	The Role of the Negative Selection Activities in Securing of Competence from an Organization Belonging to SEN Units	41
III.2.10.	Cernavoda NPP Training Programs.....	42
III.3.	International Partnership for a Sustainable Development	43
III.3.1.	International Cooperation in the Field of Waste Management Activities at IFIN-HH	43
III.3.2.	NULIFE - the European NoE "Nuclear Plant Life Prediction"	43
III.3.3.	The Euratom Programme - a Strategy for the Future.....	44
III.3.4.	ICN's Partnerships in Nuclear Research.....	45

I. NUCLEAR ENERGY

I.1. Nuclear Safety and Severe Accidents



I.1.1.

Monitoring System in Labview Data Logging and Supervisory Control Module and Labview 8 with Process Analyses of the Cryogenic Pilot Plant for Tritium Removal

Carmen Maria Moraru, Iuliana Stefan, Ovidiu Balteanu, Ciprian Bucur, Liviu Stefan, Anisia Bornea, Ioan Stefanescu

Cryogenic Pilot Plant

National Institute of Research-Development for Cryogenic and Isotope Separation Technologies

Str. Uzinei, nr.4, CP 10, OP 4, Code 240050, Rm. Valcea, ROMANIA

mcarmen@icsi.ro http://www.icsi.ro

The system responds to the monitoring requirements of the technological processes specific to the nuclear installation that processes radioactive substances, with severe consequences in case of technological failure, as is the case with a tritium processing nuclear plant. The big amount of data that needs to be processed stored and accessed for real time simulation and optimization demands the achievement of the virtual technologic platform where the data acquiring, control and analysis systems of the technological process can be integrated with a developed technological monitoring system. Thus, integrated computing and monitoring systems needed for the supervising of the technological process will be executed, to be continued with the execution of optimization system, by choosing new and performed methods corresponding to the technological processes within the tritium removal processing nuclear plants.

Key-Words: - monitoring system, process analyses



I.1.2.

Integrated System's Analyses Methods to Increase Performance in Operation

Gheorghe Florescu (gheorghe.florescu@nuclear.ro), Minodora Apostol, Mita Farcasiu, Luminița Bedreagă, Cristina Constantinescu

Institute for Nuclear Research – Pitesti, Romania

Prof. Dr. V. Panaitescu, 'Politehnica' University -Bucharest

The necessity to develop more reliable power systems needs several iterative studies. These studies are many time complexes and imply some specific methods for analysis. To permit system evaluation, using some adequate techniques, must develop specific models that could be deterministic a probabilistic.

A complete modeling means to use both deterministic and probabilistic modeling and models processing methods.

The developed model, for a nuclear facility, includes events associated to the structures, systems and components (SSCs). The modeling and model processing studies suppose specific tasks that are familiar to safety evaluation specialists. Computer codes are today involved in solving several tasks related to system's analyses.

The paper will present the specific methods to integrate and use the deterministic and probabilistic methods, in order to analyze and improve the power systems. Also in the paper are stated the applications of probabilistic and deterministic modeling to perform online and offline monitoring of process, reliability, unavailability, safety or risk associated to the analysed system.



I.1.3.

Post Irradiation Tests on CANDU Fuel Sheath Irradiated In Reactivity Insertion Accident Conditions

Silviu IONESCU, Octavian UȚĂ, Marcel PĂRVAN, Cristian GENTEA, Ovidiu ICHIM

Institute for Nuclear Research – SCN Pitesti

The purpose of this work is to determine by post-irradiate examination, the behavior of CANDU indigene fuel sheath, irradiated in TRIGA ACPR (Annular Core Pulsed Reactor) reactor of SCN Pitești in Reactivity Insertion Accident (RIA) conditions.

In the nuclear reactor the fuel elements endure dimensional and structural changes as well as sheath oxidation, hydriding and corrosion. These changes can lead to defects and even to the loss of integrity. The nuclear fuel sheath performance is determined by the following elements:

- 1 Surface condition (corrosion product deposition, corrosion etc.);

- 2 Sheath integrity;
- 3 Dimensional changes;
- 4 Oxidation and hydriding of sheath;
- 5 Sheath mechanical properties

The paper presents the results of the Post Irradiation Examination (PIE) tests performed in Post Irradiation Examination Laboratory hot cells from SCN Pitesti, which consist of the following:

- 1 Visual inspection and photography of the outer appearance of sheath;
- 2 Profilometry (diameter, bending, ovalization) and length measuring;
- 3 Microstructural characterization by metallographic analyzes;
- 4 Mechanical properties determination.

The obtained data from the post-irradiate examinations are used, on a hand, in order to confirm the security, reliability and nuclear fuel performance, and by the other hand, for development of CANDU fuel.



I.1.4.

Available Post-Irradiation Examination Techniques for CANDU Fuel Performance Evaluation

Marcel PĂRVAN, Marin MINCU, Octavian UȚĂ, Silviu IONESCU

Romanian Authority for Nuclear Activities - RAAN

Institute for Nuclear Research, Post-Irradiation Examination Laboratory

Pitești, P.O.Box 78, Romania

E-mail: marcel.parvan@nuclear.ro

The development of the nuclear energy in our country by commissioning of new units at Cernavoda Nuclear Power Plant (NPP) requires the nuclear fuel safety strengthening and implicitly the development and improvement of the techniques to investigate the processes which take place in the fuel, in order to obtain relevant informations concerning:

- 1 Maneuvre regimes or power rampings;
- 2 Nuclear fuel behaviour in accident conditions generated by reactivity and loss of coolant;
- 3 Fuel element failure kinetics and fission product release.

These informations are needed to evaluate the performance of nuclear fuel and materials in NPP in order to check the concordance with safety criteria.

The Institute for Nuclear Research (INR) of Pitești has a set of nuclear facilities consisting of 14 MW(th) TRIGA SSR (Steady State Reactor), TRIGA ACPR (Annular Core Pulsed Reactor) and LEPI (Romanian acronym for Post-Irradiation Examination Laboratory) which enable to investigate the behaviour of the nuclear fuel and materials under various irradiation conditions. LEPI is an alpha-gamma hot cell facility put into operation in December 1983. It includes two heavy concrete hot cells, three steel hot cells and one lead hot cell.

In order to check and improve the quality of the Romanian CANDU fuel, a significant number of experimental fuel elements has been tested to different power histories in the TRIGA reactors. Most important tests have been performed in conditions of power ramping, overpower, power cycling (load following) and accident (LOCA, RIA). After testing, the fuel elements have been examined in the hot cells using both non-destructive PIE techniques (visual inspection and photography, eddy current testing, profilometry, gamma scanning) and destructive PIE techniques (fission gas release and analysis, metallography, ceramography, burnup determination by mass spectrometry, mechanical testings).

The PIE results concerning the integrity, dimensional changes, oxidation, hydriding and mechanical properties of the sheath, the fission products activity distribution in the fuel column, the pressure, volume and composition of the fission gas, the burn-up, the isotopic composition and structural changes of the fuel enabled to characterize the behaviour of the Romanian CANDU fuel after its testing in the TRIGA reactors both in normal operation and in accident conditions.

The paper describes the available post-irradiation examination techniques and some results.



I.1.5.

The Man – Machine – Organization System Analysis for Research Reactor

M. FARCASIU, M. NITOI, M. APOSTOL, I. TURCU, Gh. FLORESCU

Institute for Nuclear Research – Pitesti, Romania,

The man-machine – organization (MMO) system analysis integrates into the human factor analysis methodology for the complex installation operation. In order to perform such analysis is necessary the interfaces analysis from MMO system by using the Human Reliability Analysis (HRA) methods.

The purpose of this paper is to identify man-machine – organization interfaces that could lead to an accident, types of human interactions which may mitigate or exacerbate the accident, types of human errors, performance shaping factors and to estimate of human error probability as effects of human performance in reliability and safety.

The results of this paper are the interfaces that could have a major contribution to the human error probabilities.

Conclusively, some modifications are recommended in MMO system in order to reduce the human error probabilities and the contribution of the human factor to system unavailability.

The necessary information was obtained from operating experience of research reactor TRIGA from INR Pitesti. The required data were obtained from generic data bases.



I.1.6.

Investigating Applications of ICARE/CATHARE Code to Air Ingress Issues

Mirea Mladin, INR – Pitesti, Romania, mirea.mladin@nuclear.ro

Experimental results demonstrate that in the presence of oxygen the UO₂ fuel and its fission products undergo oxidation reactions, especially under high temperature conditions as those encountered in a severe accident. It is generally accepted that the circumstances in which air can come in contact with irradiated fuel, after a core fusion accident, are related to the reactor lower head failure. Studies indicate that a natural circulation loop can be established between the containment and the reactor pressure vessel through the breaks of the later. In these conditions, the geometry of the core, the fuel and cladding state and the temperatures inside core can be significant for the source term evaluation. In ICARE/CATHARE v2, there is a model for zirconium oxidation by air, taking into account the presence of noncondensable gases, the differences from steam oxidation with respect to the kinetics and energy of reaction and also zirconium nitriding. The paper illustrates the application of ICARE/CATHARE v2 to the post-test analysis of QUENCH-10 air oxidation experiment and to PWR reactor scenarios with air ingress.



I.1.7.

Overview of Probabilistic Safety Assessment Activity for Romanian TRIGA SSR 14 MW Reactor

***Daniela MLADIN**⁽¹⁾, **Mirea MLADIN**⁽¹⁾, **Gabriela RADU**⁽¹⁾, **Cristina CONSTANTINESCU**⁽¹⁾,
Marin PREDA⁽²⁾, **Dumitru CRISTEA**⁽²⁾, **Ion STEFAN**⁽²⁾*

⁽¹⁾ *Institute for Nuclear Research Pitesti, Reactor Physics and Safety Department*

⁽²⁾ *Institute for Nuclear Research Pitesti, TRIGA Reactor Department*

Current international approach of Probabilistic Safety Assessment (PSA) is not only for Nuclear Power Plants but also for Research Reactors. Recently commissioned Research Reactors use PSA in the process of licensing. In case of the Institute for Nuclear Research Pitesti, PSA for TRIGA Research Reactor was developed with a scientific aim joint to the Deterministic Analysis, covering the qualitative and quantitative approach of safety. First the paper presents the PSA activity related to TRIGA SSR 14 MW reactor starting with raw data collection for obtaining a historical view of the reactor operation and to obtain reliability data used in PSA. Further, an overall presentation of the PSA model (Initiating Events, Event Trees and Fault Trees) is made.



I.1.8.

Pressurization Events In Primary Side of the CANDU 600

Camelia Diaconu , Serbanel Marieta

RAAN-SCN-Pitesti

dcamelia7@yahoo.com

Safety assessment of Nuclear Power Plant consists in numerous and different types of analyses, through which we evaluate the response of the important process variables at postulated disturb or initiating event. With this kind of analyses, we can see how behavior of mitigating systems during the postulated accidents. Also, it can assess defense in depth levels of the nuclear power plant.

Important events for operating of CANDU 600 NPP are the pressurization events in primary side of the plant, which we present in this paper. These events are important from point of view of nuclear safety because its affect primary heat transport system.

We have modeled two events that affect control pressure and inventory system:

- *Loss of pressurizer heaters control* that to have all five heaters turned on in error and

- *Loss of inventory control*, which assumed both feed valves to stay open with a maximum flow capacity of 18.9 kg/s and bleed valves closed.

Using facility of FIREBIRD III MOD1-77 code we have simulated the events assumed that the reactor power is constant at 100% during these events.

The main conclusions of the analyses are: the peak pressure in system that we have obtained can be supported from the primary heat transport system and the maximum fuel temperature is bellow melt fuel temperature and prevention of piping rupture due to overpressure.



I.1.9.

Modeling Human Actions for Loss of Feedwater Initiating Event Using Petri Nets Formalism

Luminița Bedreagă^{*}, Prof. Basarab Guzun^{}**

^{*}*Institute for Nuclear Research, Pitesti, Argeș, Romania*

^{**}*"Politehnica" University of București, Faculty of Power Engineering,
Department of Power Generation and Use, Bucharest, Romania
lbedreaga@yahoo.com*

The action(s) accomplished by operator in the system can analyze and taking into account quantitatively only performing a human reliability assessment for that system. These actions lead to the availability or unavailability of the system.

The human reliability assessment is a procedural activity and consists of identifying the important man-machine interactions, evaluation of these interactions with establishment of their corresponding performance shaping factors and finally deriving of the likelihood to perform correctly these interactions.

The human factor modelling in the probabilistic safety analysis framework can be performed using different methods and models, [1]. The start point of these methods was the **THERP** (Technique for **H**uman **E**rroR **R**ate **P**rediction) method, elaborated in 1983. THERP method has established the underlying of the human reliability field both from conceptual and quantification point of view. The human factor quantification by **THERP** has accomplished using event trees approach.

Logical formalism of Petri nets is applicable in the different science fields because of their specific features. This concept is used in modelling of the dynamic systems where the transfer operations are considered without taking into account time in explicitly manner. In the specific area of reliability studies of systems, Petri nets have applied in modeling of maintenance activity. The analysis of system applying Petri nets concept is a modality of evaluation based on system states. As we have presented in [2] and has ascertained in [3-4], Petri nets can also apply in human factor modeling due to their flexibility.

In this paper we have presented the modality to apply Petri net in modeling of specific human actions for CANDU nuclear power plant at Loss of Feedwater initiating event. This modeling we have continued with another example trying to emphasize the efficiency and easiness to apply Petri net concept in modeling of human actions needed being perform in a specific system.

Bibliografie

- [1] International Atomic Energy Agency - Human reliability in Probabilistic Safety Assessment for Nuclear Power Plant, Safety series 50-P-10, 1995
- [2] Bedreagă L., Guzun B. Modelling of the Human Factor Using Petri Nets, Bulk Power System Dynamics and Control - VII, August 19-24, 2007, Charleston, South Carolina, USA, 1-4244-1519-5/07/@2007 IEEE, IEEE Catalog Number 07EX1897
- [3] Tadao Murata - Petri Nets: Proprieties, Analysis and Applications, IEEE col. 77, No. 4, 1989
- [4] M. Ajmone Marsan, Gianfranco Balbo, Gianni Conte- Modelling With Generalised Stochastic Petri Nets, Università degli Studi di Torino, Dipartimento di Informatica, 1994



I.1.10.

Using Failure Mode and Effect Analysis in Identification of Components Sensitive to Ageing

**Mirela Nitoi, Ilie Turcu, Pavelescu Margarit^{*}, Apostol Minodora, Farcasiu Mita,
Popa Adrian, Florescu Gheorghe**

Institute for Nuclear Research, Pitesti, Argeș, Romania

^{*}*Academy of Scientists, Bucharest, Romania
mirela.nitoi@nuclear.ro*

Ageing represents a concern phenomenon since any degradation that may occur in time could lower a component performance and so reduce its reliability. If the phenomenon is left unchecked and unmitigated, the ageing could increase the risk associated with the facility operation.

To understand the ageing degradation of a component, it is first necessary to identify and understand the ageing processes. Since these processes involve constituent materials, parts and the service conditions of a components, it is necessary to know the design, materials, service conditions, performance requirements, operating experience (operation, surveillance and maintenance histories) and relevant research results for the component of interest.

The purpose of the Ageing Failure Mode and Effect Analysis (AFMEA) is to study the results or effects of item failure caused by ageing, on system operation and to classify each potential failure according to its severity

The paper will present the advantages of using AFMEA in identification of most sensitive to ageing components, as the results obtained for a particular case. For each component analyzed, the stressors will be established, the corresponding ageing mechanisms will be identified, as the failure modes induced by the ageing mechanisms.



I.1.11.

Model Uncertainty in Deterministic Safety Analysis: the Influence of Horizontal Feeders Nodalization on Fission Products Masses, Deposited in CANDU Primary Circuit

M. Apostol, M. Constantin, I. Turcu
Institute of Nuclear Research Pitesti

The deterministic safety analyses, called also accident analyses are very important to confirm the adequacy and efficiency of the provisions in the defence in depth concept to cope with challenges to Nuclear Power Plant (NPP) safety. The deterministic analyses are realized by calculation of plant parameters, using complex computer codes, which solve a set of analytical equations which describe a NPP physical model. The physical models, inputs and assumptions can be realistic (or best estimate) or conservative. A combination between best estimate model and best estimate inputs is called best estimate analysis and is usually combined with uncertainty analysis. Because the safety analyses mandatory include the severe accidents study, in order to represent as realistic as possible the NPP behaviour during this kind of accident, the best estimate analysis is recommended. As in IAEA references, an uncertainty analysis includes "the estimation of uncertainty in individual modelling or in overall code, uncertainty in representation and uncertainty in plant data for the analysis of an individual event". Model uncertainty is one of the three major sources of uncertainty in accident analysis (excepting the user effect) and means uncertainty associated with the models and correlations, the solution scheme, model options, unmodelled processes, data libraries and deficiencies of the computer program.

The paper presents the influence of horizontal feeders' nodalization on fission products masses deposited in CANDU primary circuit, during a postulated severe accident induced by Loss of Coolant Accident (LOCA) followed by Loss of Emergency Core Cooling (LECC). The feeders length has been divided in 5, 4, 3 and 2 nodes; for each situation has been calculated using SOPHAEROS computer code, the masses of the most important (from the source term calculation point of view) fission products deposited in CANDU primary circuit nodes, in the entire circuit and also the masses of the fission products released to the containment. On the basis of the obtained results, the mathematical relations which represent the nodes numbers influence on fission products masses deposited into CANDU primary circuit, have been established. The influence of the nodalization is analyzed in detail and recommendations for users are formulated.



I.1.12.

Intelligent SSCs as Part of New Era in Nuclear Field and New Generation of Nuclear Facilities

Gheorghe Florescu, Ilie Turcu, Constantin Agapi,
Institute for Nuclear Research – Pitesti,
Prof. Dr. V. Panaitescu, *'Politehnica' University - Bucharest*
Ioan-Bogdan Florescu, *University of Pitesti*
gheorghe.florescu@nuclear.ro

Nuclear field, that is in a continuous process of changing and development, offers the opportunity of incorporation of many new designed SSSs and new methods and techniques of study and analysis. Due to the complex items involved in this industrial sector, the nuclear area was always a field of promoting new achievement in hardware science and technique.

A major issue in the nuclear facilities is the very large number of not monitored SSCs and also the very large number of unknown degraded SSCs. Critical SSCs, looking both to operation and safety, is very important to be monitored. Visual contact and inspections are time and resources consuming and access is not always possible.

The evolution of electronics and development of advanced data acquisition devices offers the possibility of use of such equipment to many applications.

Methods of investigations of nuclear facilities operation are diverse, deterministic and probabilistic techniques could be basically shown.

The paper presents the main disadvantages, looking to the possibility of rapid identification of malfunctions, of actual SSC design. A special section is dedicated to intelligent SSCs, description of such equipment and the main advantages of their use.

The characteristics of intelligent SSCs, important for performing their functions and for rapid and correct diagnosis, are presented in the paper.



I.1.13.

Application of PSA Study at Cernavoda NPP Unit 1

SENIOR RELIABILITY ENGINEER **BEATRICE ROSU**
 CERNAVODA NPP, MEDGIDIEI STR. No 1, CERNAVODA
 E-mail: beatrice_rosu@hotmail.com

The core damage frequency calculated as part of Level 1 PSA study for Cernavoda Unit 1 includes contributions from Internal Events, Internal Fires, Internal Floods and Seismic Events, for plant full power operation and shutdown states. Application of PSA study, part of this presentation refers to:

- Risk Monitor EOOS
- Preventive Maintenance Optimization
- Data Collection.



I.1.14.

Analyze of Water Hammer Phenomena – Application to Deaerator - Feedwater Pump Node

Melania Bîgu, Mircea Tenescu, Iulian Pavel Nita, SITON Măgurele
bigum@router.citon.ro nitai@router.citon.ro

The hydraulic hammer adverse effects are presented extended in literature available for those who operates and design installations in which this phenomena's occurs.

There are specialized computational programs which evaluates on divers technical aspects which occurs with this phenomenon are today available for technical personal.

Is to by noticed that not all technical characteristics and not all effective operating modes which are in the work of this paper are not covered by existing computational programs elaborated by specialized software developers. More, even specialized developers of this programs recommends with insistently that computational results offered by specialized programs to be verified by specialized technologist with experience in alternative theoretical computations in order to avoid any misinterpretation of results obtained by computational codes.

After selective exposures of theoretical fundamentals of the problem there are presented a computational calculation obtained using specialized calculation code PIPENET (Sunrise System Limited, Cambridge, Great Britain). The PIPENET calculation is compared with a standard computational calculation using theoretical correlations. An evaluate of the differences between those two computational methods is made in order to the reveal the capabilities of the computational codes to solve hydraulic hammering problems.

The first stage we obtained the elastic characteristic of pipe where the phenomena of hydraulic hammering take place and there are derivative descriptions of differential equations which describe physical phenomena.

In the second part there we fulfill a complete system analyze of water hammer effect due to a fail close of the four level control valves of Steam Generators.

We compared the highest attended pressure with design pressure of the system. We observed that the design pressure is not over-passed.

The analyze concluded that pump's head protection is a very important parameter against overpressure of the feed water system.

I.2. Nuclear Reactors



I.2.1.

Minor Actinides Transmutation in the Efit Reactor: A Design Challenge

**C. Artioli¹, H. Ait Abderrahim², G. Glinatsis^{1*}, L. Mansani³,
C. Petrovich¹, M. Sarotto¹, M. Schikkor⁴**

¹ ENEA, via Martiri di Monte Sole 4, IT - 40129 Bologna, Italy

² SCK-CEN, Institute for Advanced Nuclear Systems, BE-2400 Mol, Belgium

³ Ansaldo Nucleare S.p.A., C.so Perrone 25, IT-16161 Genova, Italy

⁴ Forschungszentrum Karlsruhe (FZK), Postfach 3640, D-76021 Karlsruhe, Germany.

*Corresponding author

georgios.glinatsis@bologna.enea.it

tel. +39 051 6098641, fax +39 051 6098279

In the last years, for a sustainable utilization of the nuclear energy, significant efforts are performed by the worldwide scientific community, with relative financial support, to the purpose of finding a solution for "closing of the cycle". Both technical-scientific and social-economic aspects must be satisfied, including hopefully significant reduction of Minor Actinides (MA) and High-Level radioactive Waste (HLW) by Partitioning and Transmutation techniques (P&T). The various proposal cover a wide variety of potentialities offered by both thermal and fast systems, and for these last particular attention is lent to the sub-critical systems: Accelerator Driven System (ADS). A such attention is motivated by the potential of the ADS to reduce volume and radio-toxicity of the nuclear waste by transmuting and/or burning the MA.

In the European 6th Framework Program and within the EUROTRANS Integrated Project, funded by the European Union (EU), the conceptual feasibility of high rate MA transmutation in an ADS has been demonstrated: EFIT Concept Reactor (European Facility on Industrial scale Transmuter), fuelled by the innovative U-free oxide fuel at high content in Minor Actinides: Cer - Cer type with MgO inert matrix and cooled by lead, satisfying the design constraint of the cladding and fuel maximum temperatures, through the radial flattening techniques of the power distribution. The conceptual design of the EFIT sub-critical core is based on the so called "42 - 0" concept characterized by zero Pu net balance, with reduced BU reactivity loss (~200 pcm/year), and maximum MA transmutation rate of 42 kg/TWh_{th}. Result of the neutron design consists in a core of three fuel assembly typologies at constant Pu/MA (= 0.8416) mass ratio, being different for pin diameter and/or inert matrix volume fraction (VF_MgO/VF_AnOx = 0.57/0.43 and 0.50/0.50 to maximize the average power density flattening the radial power distribution). To assure the K_{eff} = 0.97 target value of sub-criticality, a total of 180 FAs are needed producing a thermal power of 384 MW requiring a proton beam (E_p = 800 MeV) current (almost constant) never exceeding the 15mA. From the neutron design point of view, the EFIT/Pb core is also characterized by hard neutron spectrum, by high average neutron flux, by a low β_{eff} fraction, by the missing of the Doppler prompt reactivity feedback, by a strongly positive reactivity worth for the (whole active core flowing) coolant void effect. At the same time, the T/H analysis show acceptable maximum temperatures and respected the cladding and fuel temperature failure limits due to the high natural convection coolant flow rate after the pumps coast-down. The hard neutron spectrum involves relatively high neutron irradiation damage at some components, such as the Spallation Module. The neutron performances can be achieving acting on the irradiation "residence time". Moreover after 20 years of full power irradiation the lead coolant activity rises (in average) at about 10¹¹ Bq/kg, with dominant isotopes (as far as activity is concerned) being ^{207m}Pb, ²⁰⁹Pb, ¹¹⁰Ag and ⁷⁶As. Finally the EFIT/Pb reactor, from reactor design point of view, must be also characterized by decay heat removal components of "long term" reliability.

Keywords: Core Design, Minor Actinides, Transmutation, Monte Carlo, Nuclear Data.



I.2.2.

Detection Systems of the Failed Nuclear Fuel

**cs III G. Matei, cs II M. Crucean,
cs I D. Dobrea, cs I Csaba Roth, cs III L. Aioanei**

In this paper are presented two detection systems of the gaseous fission products in the Primary Heat Transport Systems of the CANDU and TRIGA reactors.

Both systems (one analogical and the other digital) are designed on base the Analog to Digital Converter and Memory (ADCAM) Architecture. The differences between these systems consist in the structure of the MCBs (Multichannel Buffer): TRUMP in the case of GFP-Monitor and digiBASE in the case of DGFP-Monitor.

Also, in the case of GFP-Monitor is necessary to build a collimator system of the gamma-ray, because the maximum amount of the gross gamma rate is about 10⁶ cps.

The movement of the collimator is provided by a ball screw actuator powered by a stepping motor. Its position is sensed by an encoder.

The two systems belong to the group of auxiliary safety systems of the CANDU and TRIGA reactors.

Based on the information obtained from these systems the operator can propose starting of the localization and replacing procedures of the defective fuel rod or, in the case of a major fault, even the shutdown of the reactor.



DGFP-Monitor



GFP - Monitor



I.2.3.

Numerical Simulator of the CANDU Fueling Machine Driving Desk

Cezar DOCA, PhD

Institute for Nuclear Research Pitesti

Mioveni, Str. Campului Nr.1, Arges, ROMANIA

cezar.doca@nuclear.ro; cezar_doc@yahoo.com

As a national and European premiere, in the 2003 – 2005 period, at the Institute for Nuclear Research Pitesti were successfully tested two CANDU Fueling Machine Heads, no.4 and no.5, for the Nuclear Power Plant Cernavoda – Unit 2.

To perform the tests of these machines, at the Institute for Nuclear Research Pitesti a special CANDU Fueling Machine testing rig was built and was (and is) available for this goal. The design of the CANDU Fueling Machine test rig from the Institute for Nuclear Research Pitesti is a replica of the similar equipments operating in a CANDU 6 type Nuclear Power Plant.

High technical level of the CANDU Fueling Machine tests required the using of an efficient acquisition and data processing Computer Control System. The challenging goal was to build a computer system (hardware and software) designed and engineered to control the test and calibration process of these fuel-handling machines. The design takes care both of the functionality required to correctly control the CANDU Fueling Machine and of the additional functionality required to assist the testing process.

Both the Fueling Machine testing rig and staff had successfully assessed by the AECL representatives during two missions.

At same the time, at the Institute for Nuclear Research Pitesti was/is developed a numerical simulator for the CANDU Fueling Machine Operators' training.

The paper presents the numerical simulator – a special PC program (software) which simulates the graphics and the functions of and the operations at the main desk of the Computer Control System. The simulator permits "to drive" a CANDU Fueling Machine in two modes: MANUAL or AUTOMATIC.

The numerical simulator is dedicated to training of the Operators who operate the CANDU Fueling Machine in a Nuclear Power Plant with CANDU Reactor.

Keywords: numerical simulator, CANDU Fueling Machine, training.



I.2.4.

A Neutronic Properties Study of an ACR-1000 Shutoff Rod Design Proposal

Iosif Prodea¹, Andrei Rizoiu¹, Ilie Prisecaru², Nicolae Dănilă²

¹RAAN-Institute for Nuclear Research Pitești

²Politehnica University of Bucharest

The AECL's Advanced CANDU Reactor (ACR) project has developed in two versions known as ACR-700 and ACR-1000. The world nuclear energy market requests determined the abandon of ACR-700 and promotion of ACR-1000 during the last years. This latest AECL Generation III+ nuclear reactor design is shortly characterized by a reduced elementary cell to 24 cm (as opposed to 28.575 cm of CANDU-6), the using of both slightly and low enriched Uranium dioxide fuel with a target burnup of 20 MWd/kgU, light water as coolant, burnable poison (Dysprosium) in the central fuel pin to reduce the coolant void reactivity and a net output power of 1200 MW. The aim of our paper is to perform some cell and supercell calculations using dedicated computer codes like WIMS, DRAGON and PIJXYZ for a proposed ACR-1000 Shutoff Rod (SOR). Rectangular and circular transversal rod shape sections were taken into account. The materials' compositions were basically those from CANDU-6 SOR with slightly variations from the standard values. The supercell calculations are needed to provide the "incremental" cross sections, i.e. the slight perturbations of the neutronic flux shape due to the presence of the SOR and its guide tube. The supercell incremental cross section for different analyzed configurations were compared. The ACR-1000 SOR project were also compared to CANDU-6 SOR one. The main result of the paper consists in the ACR-1000 SOR design itself.



I.2.5.

The Thermalhydraulic Behavior of CANDU 600 Reactor Core Fuelled with SEU 43

Serbanel Marieta, Diaconu Camelia

RAAN-SCN-Pitesti

marietaserbanel@yahoo.com

CANDU 600 nuclear reactors are usually fuelled with Standard 37 rods fuel bundles, and natural uranium (NU) dioxide (UO_2), is used as fuel composition.

A new fuel bundle geometry is proposed with 43 rods and slightly enriched uranium fuel (SEU 43 with 0.96% enrichment U_{235})

In this paper a comparative analysis of the behavior of the primary circuit during a LOCA 35% RIH accident is performed for two core types (normal core- 37pin/bundles and a 43pin/bundles proposed core).

This kind of accident is considered to be a severe accident for CANDU type fuel elements. This analysis uses FIREBIRD code coupled with bi-point kinetics module.

The bi-point kinetics module includes the models for the neutronic measurement instrumentation (the platinum detectors and the ion chambers) and the RRS (Reactor Regulating System) module.

For the large LOCA, the RRS is of no effect, therefore the RRS option not be used for this analysis.

The main conclusions of the analysis are the following:

- in this case of 35% RIH, LOCA, the thermalhydraulic behavior of the 43 pin/bundles core is better than the normal core.
- the fuel and the sheath temperature not reached the melt temperature.



I.2.6.

SEU43 Fuel Bundles in CANDU 600

Alexandru Catana, Nicolae Danila*, Ilie Prisecaru*, Daniel Dupleac*, Prodea Iosif

RAAN, Institute for Nuclear Research,

Str. Campului Nr. 1, Pitesti, Arges, Romania

Tel:40-248-213400 (633), Email: calex_2002@yahoo.com

* University POLITEHNICA of Bucharest, Romania

Cernavoda 1 & 2 are pressure tube 650 MWe nuclear stations moderated and cooled with heavy water, of Canada design, located in Romania. Fuelling is on-power and the plant is currently fuelled with natural uranium dioxide. Fuel is encapsulated in a 37 fuel rod assembly having a specific standard geometry (STD37). In order to reduce fuel cycle costs programs were initiated in Canada, South Korea and at SCN Pitesti, Romania for design and build of a new, improved geometry fuel bundle and some fuel compositions. Among fuel compositions, which are considered, is the slightly enriched uranium (SEU) fuel (0.96 w% U_{235}) with an associated burn-up increase from ~7900 MWd/tU up to ~15000 MWd/tU. Neutron analysis show that the Canadian-Korean fuel bundle geometry with 43 rods called SEU (SEU43) can be used in already operated reactors. A new fuel bundle resulted. Extended, comprehensive analysis must be conducted in order to assess the TH behavior of SEU43 besides the neutron, mechanical (drag force, etc) analyses. In this paper, using the sub-channel approach, main thermal-hydraulic parameters were analyzed: pressure drop; fuel, sheath and coolant temperatures; critical density; critical heat flux. Some significant differences versus standard fuel are outlined in the paper and some conclusions are drawn. While, by using this new fuel, there are many benefits to be attained like: fuel costs reduction, spent fuel waste minimization, increase in competitiveness of nuclear power generation against other sources of generation, etc., the safety margins must be, at least, conserved. The introduction of a new fuel bundle type, different in geometry and fuel composition, require a detailed preparation, a testing program and a series of neutron and thermal-hydraulic analysis and the results of this paper is part of this effort. The feasibility to increase the enrichment from 0.71% U_{235} (NU) to 0.96% U_{235} , with an estimated burn-up increase up to 14000 MWd/tU is evaluated. Some thermal-hydraulic and neutron distribution consequences are analyzed.

KEYWORDS: CANDU, FUEL BUNDLE, SEU43, SUB-CHANNEL, CHF, PRESSURE DROP, WASTE REDUCTION



I.2.7.

Neutron Flux Measurement In The Central Channel (XC-1) of TRIGA 14MW LEU Core.

D. Bărbos, P. Busuioc, C. Păunoiu, Cs. Roth

Institute for Nuclear Research, Pitesti-Mioveni, Romania, *dumitru.barbos@nuclear.ro*

The full conversion of the 14MW TRIGA Research Reactor was completed in May 2006 and each step of the conversion was achieved by removal of HEU fuel, replaced by LEU fuel, accompanied by a large set of theoretical evaluation and physical measurements intended to confirm the performances of gradual conversion. Following the core full conversion, a program of measurements and comparisons with previous results of core physics and

measurements is performed, allowing data acquisition for normal operation, demonstration of safety and economics of the converted core.

Neutron flux spectrum measurements in the XC-1 water-filled channel were performed using multi-foil activation techniques. Activation reaction rates were obtained by absolute measurement of the induced activity using gamma spectroscopy techniques. The neutron spectra and flux are obtained by unfolding from measured reaction rates. The guess spectrum used for unfolding was generated using SAND II energy mesh according to the formula:

$$\Phi(E) = \Phi_{th} \frac{E}{(kT_n)^2} e^{-\frac{E}{kT_n}} + \Phi_{epi} \frac{\Delta(E/kT_n)}{E},$$

and matched to fission spectrum at 0.5 MeV. The $\frac{\Phi_{th}}{\Phi_{epi}}$ ratio is estimated from measured cadmium ratio of

$Au^{197}(n,\gamma)$ reaction.

The integral flux measured value for LEU core at 14MW reactor power is $4.66 \cdot 10^{14}$ n/cm²s. For reference, the measurement of the integral neutron flux performed at the commissioning time (standard core) also for 14MW reactor power yielded a value of $4.22 \cdot 10^{14}$ n/cm²s.



I.2.8.

Full Conversion of Materials and Nuclear Fuel in Research Reactor – TRIGA INR 14 Mw

M. Preda, E. M. Ana, D. Cristea, C. Iorgulis, M. Ciocanescu, S. C. Valeca*
Institute for Nuclear Research Pitesti, Mioveni, ROMANIA, office@nuclear.ro
**Pitesti University, Pitesti, ROMANIA*

During 1952-2005 General Atomics built and commissioned 62 TRIGA research reactors.

Almost all reactors built by General Atomics use Low Enriched Uranium (19,9%). One of the exceptions was the TRIGA reactor from ICN Pitesti.

The transition from HEU to LEU utilization is called as *conversion* and it is supported by Department of Energy – USA and Member States in the project “Reduced Enrichment in Research and Testing Reactors (RERTR)” and by IAEA and Member States through Technical Cooperation projects.

The activities related to full core conversion are approached and analyzed as refueling operations. Since TRIGA SSR 14MW commissioning this type of operation was performed at least six times under the full control of qualified personnel. The workers are licensed by National Commission for Nuclear Activities Control and periodically reinstructed for this type of activity.



I.2.9.

Modernization of Control-Command System for TRIGA SSR 14MW Main Cooling Circuit

M. Preda, D. Cristea, E. M. Ana
Institute for Nuclear Research Pitesti, Mioveni, ROMANIA, office@nuclear.ro

Modernization activities of control-command system from TRIGA INR main cooling circuit consists in replacement of actual system for

- parameter measurement
- safety
- reactor external scrambling
- protection, command and supply for electrical elements of the system.

This modernization project is designed to assure the necessary feature for reactor external safety and for technological parameter measurement.

The new safety system of main cooling circuit is completely separate from its operating system and is arranged in a panel assembly in reactor control room.

The operating system has the following features:

- data acquisition
- parameter value and state of command elements displaying
- element command on hierarchical levels
- operator acknowledgement through visual and acoustic alarm



I.2.10.

Analysis of Scheduled and Incidental Shut-Downs at TRIGA SSR 14 MW (2001 – 2007)

Doru Oprea, Ion Stefan, Marin Preda
Institute for Nuclear Research Pitesti

The paper is a statistical assessment of the scheduled and incidental shut-downs at TRIGA SSR 14 MW, as function of the following :

- the system or installation that determined the shut-down
- the true / false character that determined the shut-down
- class of nuclear safety of the system or installation that determined the shut-down (QA classification of the reactor structures, systems and components)

An analysis of the operation and shut-off periods is performed, for the interval 2001 – 2007.

Also, the authors analyze the relation between the true / false unscheduled shut-downs and the wearing of the reactor systems or components.

The paper shows in form of tables and graphs the operation history during the above-mentioned period.



I.2.11.

Benchmark-type Modeling of the SCN TRIGA SSR

Stefan Covaci
Institute for Nuclear Research Pitesti

Due to the increased RAM capacity and computing speed, the present PC generation allows the precise modeling of the geometry, materials and fine representation of the energy distribution of the neutrons in the reactor core.

The paper presents

The modeling of the experimental data for LEU fuel burnup measurements

The programs for isotopic concentration update of the input file in CSAS module, and the main geometry and material units required by the KENO V section in the same input file, both programs being included in the specific nuclear software SCALE 5

The computation showed a reactivity reserve of \$6.41 compared to the observed experimental value of \$7.23.

The difference between computation and experimental data is fully acceptable taking into account the uncertainty of the burnup measurement, temperature, and hence of the impossibility of an “absolute” precise representation of the reality



I.2.12.

Capsule C5 Operation Modes Analysis

Gh. Neguț¹, Mirela Ancuța², Violeta Stefan²

1. ANDRAD

2. *Institute for Nuclear Research Pitesti*

The paper is part of the research program entitled “ TRIGA reactor performance increase” , which has as an objective the engineering of the fuel and structure materials irradiation in SCN TRIGA reactor.

The paper is based and summarizes the results of previous works within this research program, which resulted in upgrading of the irradiation device C5 in order to increase the operational performance.

The paper presents :

Evaluation of the operation modes for the material testing device C5, during testing of Zirconium – Yttrium samples.

These tests have been proposed by AECL, and an evaluation of the capsule C5 has been performed in the conditions specified by AECL.

Also, the capsule C5 modeling and computation input data for this model are presented.

The paper results will be used for irradiation testing of new types of structure materials of the CANDU reactor, to point out the factors that put a limit to their operational life.



I.2.13.

INR Pitesti Triga Reactor Instruments for Structure Analysis**I. Ioniță***INR Pitesti, Mioveni, Romania, ionionita@lycos.com*

Two instruments for structure analysis are realised at INR Pitesti, a crystal neutron powder diffractometer in focusing configuration and a SANS instrument.

The principle to obtain focusing for a given experimental configuration is to compensate the scan variable variances by using the cross-correlation between variables. The main characteristics of this instrument are: the monochromatic beam take-off angle of 83° , The wave-length of 1.3855 Å, silicon perfect crystal monochromator of 200mm diameter and 3mm thickness cut upon the (100) plane, the reflexion plane (511), "the cutting angle" of $\chi_m = -15.8^\circ$. The distances source-monochromator, monochromator-sample and sample-detector are 5200, 2800 and 1200 mm respectively. The sample-detector distance can be modified in the range 1200-3000 mm and the detector window width can be chosen in the range 0-10mm.

The SANS instrument has the following components: The mechanical monochromator, shielding, the Bi filter, the sample table and holder, the detecting system formed by two rows of 40 He^3 detectors each, 2 flux monitors, the beam stop. Paraffine blocks and recipients with water form the shielding. The flux monitors are positioned one before the sample (in front of the monochromator window) the other in front of the beam stop. The sample can be rotated using a step by step motor. A cadmium slit system actioned by a step by step motor allows for the determination of the monochromatic beam center at the beam stop position.

Key words: (Crystal neutron diffractometer, SANS instrument, focusing configurations, structure analysis, high resolution)

I.3. Nuclear Technologies and Materials



I.3.1. Update of ITER ISS-WDS Process Design

* **A. LAZAR, S. BRAD, N. SOFALCA, M. VIJULIE**

** **I. CRISTESCU**

*National Institute for Cryogenics & Isotope Technologies (ICIT)

**Forschungszentrum Karlsruhe-Tritium Laboratory (TLK)

The general objective of a process for tritium recovery from heavy water (Water Detritiation System (WDS) coupled with Cryogenic Distillation (CD)) is to produce a stream sufficiently enriched in tritium so that it becomes economical in terms of storage or disposal. On the other hand, sometimes a stream sufficiently depleted in tritium is desired so that it can be either recycled through the reactor system or discharged directly into the environment.

The ITER Isotope Separation System (ISS) and Water Detritiation System (WDS) will be integrated in order to reduce potential chronic tritium emissions from the ISS. This is achieved by routing the top (protium) product from the ISS to a feed point near the bottom end of the WDS Liquid Phase Catalytic Exchange (LPCE) Column. This provides an additional barrier against ISS emissions and should mitigate the memory effects due to process parameter fluctuations in the ISS.

The objective of this task is to update the designs of the ITER ISS and WDS as documented in the 2001 FDR (Final Design Report) taken into account the result and the recommendation of the FMEA report and experimental results from ongoing R&D tasks. Already during the preparation of the Design Description Document package for the final report of ITER 2001 a number of trades off between the Tritium Plant subsystems have been identified.

The CATIA V5 software was chosen to create layouts of plant sites by defining the buildings, the major areas, all the way down in the plant area, the path to the equipment and so on. Subareas for facilities and technological lines have been created within the plant. The system allows a hierarchical approach including true partition of space with shared boundaries, areas with multi patches, and so on.

A 3D layout of the WDS and ISS systems in the building has been developed based on the FDR 2001 report and the recommendation from the reports presented at Tritium Plant Project Team (TPPT) Meeting in Cadarache, 8-10 October 2007. The designing work involved spaces reservation for the major equipment of ITER ISS-WDS system, analyze area/volume allocations and optimize the general 3D layout of plants and equipment or piping lines placed in them. The arrangement of the constituent process systems has been optimized in terms of minimizing the length of interconnections, and has taken into account provision of adequate space for operation and maintenance, separation of areas of the building into zones. The 3D layouts are organized in structure called product tree (skeleton) comprising assemblies, sub-assemblies and parts.

From the 3D WDS and ISS layouts, plot plans were generated with equipment arrangement for each floor. The plot plans have been sent for review to the EFDA CSU Garching / ITER.



I.3.2. Endurance Tests for WDS

* **M. Vladu, S. Brad, F. Vasut, M. Vijulie, M. Constantin**

*National Institute for Cryogenics & Isotope Technologies (ICIT)

For ITER, the WDS is one of the key systems to control the tritium content in the effluent streams, to recover as much tritium as possible and consequently to minimize the impact to the environment. The most suitable process for water detritiation is the CECE process, a combination of electrolysis and catalytic exchange of hydrogen isotopes in a Liquid Phase Catalytic Exchange (LPCE) column. The separation duty of the CECE process for fusion reactors is very high (tritium concentration in the released H₂ from the WDS for ITER shall be < 700 Bqm⁻³ air at a typical activity of tritium in the water feed of 370 GBqm⁻³). As the separation performances are given by the efficiency of the catalyst/packing mixture from the LPCE column, there is an on-going R&D activity devoted to develop and characterize high-performance catalyst/packing mixtures for a wide range of operating parameters.

During the project, the research efforts were focused on *long-term catalyst endurance tests with tritiated water*. Due to the long time necessary for the experiments (approx. 1 year for a set of input parameters), to the end of the project information will become available whether the tritium decay has an influence on the catalyst, altering its efficiency. The efficiency of the regeneration methods will also be determined.

The paper presents the improvements in the manufacturing process for catalysts to be used in the water detritiation process:

- *New catalysts.* Based on the experience in the field of hydrophobic catalysts used in isotopic exchange reaction, ICIT will produce and characterize new types of catalysts (e.g. titanium, cerium and zirconium oxides).

- *New catalyst manufacturing techniques.* Taking into account the properties of the proposed metallic oxides (hydrophobic and binding properties), new manufacture techniques will be designed and developed.
- *Catalysts testing.* The testing activity will be focused on endurance tests to evaluate the influence of tritium decay. Then, a suitable catalysts regeneration procedure shall be developed. In particular, catalyst packing will be improved and tested.



I.3.3.

Studies and Evolutions about Recovery of Transitionals Metals from Spent Catalysts

**Elena David, Marius Constantinescu, Felicia Bucura,
Claudia Sandru, Claudia Sisu**

National R&D Institute for Criogenics and Isotope Technologies, Rm. Valcea
Uzinei Street, No. 4, Zip Code 1000, Rm. Valcea, Romania
E-mail: maicsi@yahoo.com; marius@icsi.ro

The project proposes the improvement of processing methods of wastes and extraction of transitional metals from these ones as well as establishing new performance technologies in this way.

Principal objectives of this project are:

- evaluation of environmental impact ;
- development of new technologies at European standards;
- optimization of the efficiency of metals recovery and purification;
- dissemination of results at special conferences.

Key words: recovery, leaching process, spent catalyst.



I.3.4.

Conceptual Design of a Tritium Storage Bed

**Nicolae Sofilca¹, Nicolae Bidica¹, Mariea Deaconu², Ciprian Bucur¹,
Sorin Soare¹**

¹ INC-DTCI – ICSI Rm. Valcea

² RAAN – SCN Pitesti

This paper presents an engineering solution for a tritium storage bed that is suitable for performing *in-situ* and *real-time* measurement of the tritium content.

In case of commercial storage beds, the usual method is to measure directly the heat generated by tritium decay (0.324 W per g tritium), using a tritium calorimeter. This method is very accurate, but the storage bed to be assayed must be small enough to fit inside the calorimeter chamber, and a tritium calorimeter is very expensive.

In comparison with commercial tritium storage beds, the proposed design includes some different technical, structural and functional features, because the method proposed to estimate the tritium content is different. It consists in determination of the decay heat of tritium by simply measuring of the temperature increase of the storage bed.

The present work proposes a storage bed based on titanium powder (used as storage material) uniformly distributed over a number of copper fins which have the role to assure a good distribution of the heat inside of the bed. The heat generated during absorption of tritium is removed by nitrogen gas cooling coil wrapped around the outer wall of the storage bed. To calibrate the system, an alternative source of heat (electrical resistance) will be used to simulate tritium radioactive decay. For optimum heat transfer, all components of the storage bed are brazed together.

The proposed design allows filling, removing or replacing the storage material after all parts has been assembled and brazed.



I.3.5.

History on the Use of Industrial Magnesium -Thorium Alloys in Romania. Technological and Radio-Protection Aspects

A. Buzaianu¹, A.F. Olteanu², I. Rusu,³ V. Manoliu⁴

¹METAV- R&D S.A., Bucharest, C.A.Rosetii 31, P.O.Box 22; Cod 020011, Romania.

²Research and Development National Institute for Metals and Radioactive Resources, Bucharest, B-dul Carol I ,nr.68-70; Cod, 020917, Romania.

³"Gh. Asachi" Technical University, Iasi, Romania.

⁴ Research Aeronautical Institute, Bucharest, Iuliu Maniu Str.220; Cod 061126, Romania

The development of the aerospace industry at the beginning of the '90's involved the assimilation of new materials capable to bring reliability and safety in the operation of certain components of the Viper - Rolls-Royce turbopropeller engines assimilated by the national industry.

In specialized foundrys for aeronautical cast components, a special category of magnesium alloys exists in which the

main alloying element is thorium. Alloys based on Mg-Th-Zn are used in the manufacture of jet engine components. Magnesium alloys cast components are required to operate in creep conditions, which determined the elaboration of distinct technological specifications for the use of Mg-Th-Zn alloy systems and of certain restrictions imposed to ensure radiological protection in the handling of pre-alloys and alloys but also of cast and machined parts that exhibited good operation characteristics at temperatures reaching 350 °C.

This paper is aimed at presenting some of the general technological prescriptions and measurements performed during the technologic flow involved by the manufacture of thorium alloyed magnesium base alloys, at the Romanian Metallurgical Factory for Aerospace Components in Bucharest.



I.3.6.

Maintaining of Primary and Secondary Circuit Chemistry – The Key of the Minimum Corrosion of Metallic Components

Dr. M. Rădulescu, Dr. I. Pîrvan

Institute for Nuclear Research

POBox 78, 1 Campului Str., Mioveni Pitesti code 115400

ROMANIA

The objective of this paper is the presentation of some chemistry regimes available in primary and secondary circuit of a CANDU Nuclear Power Station (NPS) simultaneously with the establishment of some correlations between the corrosion behaviour of some representative metallic materials and the specific parameters of the respective aqueous environments.

To attain this aim, first we reviewed the main chemical parameters of the aqueous environments from the primary and secondary circuit of one CANDU NPS. The second step was the establishing of some correlations between these chemistry variations and the quality of superficial films formed on some representative alloys from primary and respective from secondary circuit such as: zirconium alloys and carbon steels.

The main parameters implied in environment chemistry are: the pH value, the substances used to regulate it and respectively the oxygen concentration dissolved in the aqueous environment. To study the films formed on the samples made from the respective above-mentioned alloys in the NPS operation conditions, the respective samples were exposed in autoclaves at the operation parameters (temperature, pressure) and in respective aqueous environments specifically to both circuits.

To characterize the filmed samples were used the following methods: the gravimetric, X-rays diffraction, metallographic microscopy method and some electrochemical methods such as: potentiodynamic and electrochemical impedance spectroscopy (EIS).

Using these methods, there were established the corrosion kinetics and the characteristics of the films formed on the samples in the operation conditions of both circuits that were correlated with the chemistry regimes of same circuits.



I.3.7.

Simultaneous Determination of Dissolved Elements in Water Using ICP-*oa*TOFMS

M. Dragomir, D. Ohai, I. Dumitrescu, I. Furtuna

Institute for Nuclear Research, POBox78 115400 MIOVENI

ROMANIA

The aim of the paper is devoted to the developed a strategy that permits determination of trace elements in water, using inductively coupled plasma orthogonal acceleration time-of-flight mass spectrometer (Optimass 8000, ICP-*oa*TOFMS).

The instrument used, the Optimass 8000, allow simultaneous acquisition of the entire mass range from the light ions to the heavy ions, resolution ranged being from 600 (⁷Li) to 2300 (²³⁸U). Also, as all masses can be displayed simultaneously, the user can view possible contaminants and elements previously not considered in real time. This allow a user to investigate what a particular peak is and then, if important, to quantify these.

The instrument has been successfully used to analyze fifty-seven dissolved elements in water samples: aluminum, antimony, arsenic, barium, beryllium, bismuth, boron, cadmium, calcium, cerium, cesium, chromium, cobalt, copper, dysprosium, erbium, gallium, gadolinium, germanium, hafnium, holmium, indium, iron, lanthanum, lead, lithium, magnesium, manganese, mercury, molybdenum, neodymium, nickel, neodymium, platinum, potassium, praseodymium, rubidium, rhodium, samarium, scandium, selenium, silicon, silver, sodium, strontium, tantalum, tellurium, thallium, tin, titanium, thulium, uranium, vanadium, ytterbium, zinc and zirconium.

Key words: ICP-*oa*TOFMS, elements.



I.3.8.

Physico-Chemical Processes Implied in the Formation of Corrosion Deposits on the Fuel Claddings

I. Pirvan, M. Radulescu, L. Velciu, M. Mihalache, A. Dinu

Institute for Nuclear Research, Pitesti-P.O. Box 78, Mioveni, ROMANIA

Fuel claddings from Zircaloy-4 can be covered, in principally with deposits of iron oxides (Fe_3O_4 , $\text{Fe}(\text{OH})_2$, Fe_2O_3), which have precipitated from the coolant. The process of crud deposition on cladding surfaces depends on various chemical control and reactor system parameters.

To determine Zircaloy-4 samples oxidation influence on corrosion deposits formation an experimental programme, in laboratory, was developed:

- 1 Oxidation experiments on Zy-4 samples in water and steam at high temperature and pressure, in static autoclaves, for formation of different oxide films and characterization of Zircaloy-4 samples oxidation;
- 2 Formation and characterization of corrosion deposits (different iron oxides) on Zircaloy-4 samples covered with different oxides films.

By oxidation of Zircaloy-4 samples in LiOH solutions ($\text{pH}=10.5$) and steam at high temperature and pressure, ZrO_2 films with different thicknesses and physical-chemical properties were formed. The corrosion kinetics took place only pre-transition, in pre-transition and post-transition, or in steps, as a function of testing conditions. The substoichiometric or stoichiometric oxide films with thickness under $5\text{-}5.5\mu\text{m}$ were continuous, uniform, adherent and relatively compact. These oxides had a monoclinic structure or monoclinic and tetragonal structure. Fraction of tetragonal/monoclinic structure of ZrO_2 increased with thickness of oxides. The stoichiometric and suprastoichiometric amorphous ZrO_2 films with thickness over $7\mu\text{m}$ were relatively uniform, less adherent and had a porous character, at exterior interface, being present and lateral cracks. The interior substrate, from interface with metal, remained compact.

Formation of the corrosion deposits on Zircaloy-4 samples surface was dependent on thickness and physical-chemical characteristics of initial zirconium oxide films presented on their surface.

Gravimetric and microscopical analysis, as well as XRD analysis evidenced that the adherence of the corrosion deposits (Fe_3O_4 and Fe_2O_3) increased with the porosity of oxides; a significant quantity of these deposits incorporated in pores and lateral cracks of thicker oxides, evidenced and by EIS. The highest quantities of deposits formed on initial oxidized samples with oxides over $7\text{-}8\mu\text{m}$ thickness.

Experimental results showed that the formation of corrosion products on Zircaloy-4 fuel claddings depends beside water chemistry and on by initial surface state, respectively by oxidation condition and physico-chemical characteristics of zirconium oxides.



I.3.9.

Characterization of Incoloy 800 Oxides Developed Under Simulated Supercritical Water Conditions

M. Fulger, D. Ohai, D. Lucan, M. Mihalache, M. Pantiru

Institute for Nuclear Research Pitesti – Romania

For a correct design of SCWR components, are necessary data regarding the behavior of candidate materials in supercritical water. Corrosion has been identified as critical problem because the high temperature and the oxidative nature of supercritical water may accelerate the corrosion kinetics.

The paper presents results after Incoloy 800 oxidation in autoclaves under supercritical water conditions up to 1440 h. The exposure conditions (demineralised and dezaerated water at temperatures of 450°C , 500°C , 550°C , and pressure of 250 bar) have been selected as relevant for supercritical power plant concept.

The corrosion performances of Incoloy 800 in the above mentioned experimental conditions were analyzed from the point of view of morphology and chemical composition of the oxide layers developed on the samples surface.

Results from the characterization of the corrosion products with optical microscopy, scanning electron microscopy (SEM), X-ray diffraction (XRD) and electrochemical impedance spectroscopy (EIS), are presented and discussed.



I.3.10.

Influence of Aqueous Environment Ph on the Corrosion Behaviour of the CANDU Steam Generator Tubing Material

D. Lucan, M. Fulger, L. Velciu

*Institute for Nuclear Research, Mioveni, Romania
dumitra.lucan@nuclear.ro*

The second side environment is known to be affected significantly by several factors dependent on the balance of plant conditions (condenser leaks, condensate polishing, and coolant system materials) and operational conditions, particularly through their thermal-hydraulic effects. The presence of tube-tubesheet crevices and restricted flow areas within sludge or surface deposits provide for local concentration sites for various impurities, including acidic.

The generalized corrosion is an undesirable process because it is accompanied of the deposition of the corrosion products which affect the steam generator performances. It is very important to understand the generalized corrosion mechanism in the purpose to evaluate the quantities of corrosion products which exist in the steam generator after a determined period of operation. The purpose of the experimental research consists in the assessment of corrosion behavior of the tubes material (Incoloy-800) at normal secondary circuit parameters (temperature-260°C, pressure-5.1MPa). The testing environment was the demineralised water without impurities, at different pH values regulated with morpholine and cycloheylamine (all volatile treatment – AVT). The results are presented like micrographies and graphics representing loss of metal by corrosion, corrosion rate, the total corrosion products, the adherent corrosion product, the released corrosion products and the release of the metal.

Key words: (steam generator, corrosion, pH, corrosion products)



I.3.11.

The Influence of Some Anions Resulted from γ Radiation Action on the Corrosion Susceptibility of Grade 2 and Grade 12 Titanium Alloys

L. Popa, M. Rădulescu, A. Dinu, L. Velciu

Institute for Nuclear Research, POB 78, 1 Campului Street, 115400, Mioveni, ROMANIA

E-mail: claur_popa@yahoo.co.uk

The use of titanium as a material for the construction of nuclear waste containers is being considered in many countries since it is an obvious candidate for use in hot saline environments. Grade 2 titanium was chosen as a reference container material because of its resistance to chloride containing environments although other corrosion resistant alloys are also being investigated. Grade 12 titanium was identified as the preferred material for containers in the concentrated brine environment.

The study of the influence of γ radiation on the corrosion process is important because the the presence of radiation can influence the corrosion mechanism. Thus, in presence of radiation is produced the radiolytic dissociation of solutions containing chloride ions, resulting some anions such as: ClO^- , ClO_3^- , ClO_4^- , wich are aqble to accelerate or inhigate the corrosion processes.

Key words: titanium alloys, nuclear waste, γ radiation , anions, corrosion



I.3.12.

Main Decontamination Techniques Included in Decommissioning Activity

D.Stefanescu

Institute of Nuclear research Mioveni Pitesti

E-mail: doina.stefanescu@nuclear.ro

Decommissioning is defined by the International Atomic Energy Agency (the Agency) as the administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a nuclear facility. The actions taken in decommissioning must be such as to ensure the protection of the work force and long-term protection of the public and the environment and typically include reducing levels of residual radionuclides so that material and buildings can be safely released and reused. Decommissioning activities also create radioactive waste which needs to be appropriately managed so that the public is protected from the associated radiation hazards. A large number of nuclear facilities worldwide will ultimately require decommissioning. The tasks associated with decommissioning cover a wide spectrum; for nuclear facilities they can include large-scale decontamination and destruction of massive concrete structures.

The decommissioning process can present many safety challenges and before a decommissioning programme is started, these challenges must be anticipated, evaluated and satisfactory solutions found. In all cases, the decommissioning process must be well planned and arrangements must be made to ensure that sufficient resources will be available when needed. Over time, without proper arrangements being made for decommissioning, shutdown facilities would deteriorate and ultimately, unless they are properly maintained, could constitute a radiological hazard to persons in their vicinity from direct exposure to radiation or as a result of radioactive material that may be released to the public environment.

Decontamination is a major decommissioning activity that may be used to accomplish several goals, such as reducing occupational exposures, limiting potential releases and uptakes of radioactive materials, permitting the reuse of components, and facilitating waste management. The decision to decontaminate should be weighed against the total dose and cost.

Some form of decontamination is required in any decommissioning programme, regardless of the form of the end product. As a minimum, the floor, walls, and external structural surfaces within work areas should be cleaned of loose contamination, and a simple water rinsing of contaminated systems may be performed. The question will arise, however, whether to decontaminate piping systems, tanks and components.

Several decontamination techniques have been developed to support maintenance work in nuclear installations. With relative success, the same techniques have also been adopted when decommissioning nuclear installations and components is made.

This work presented a comprehensive list of examples for various decontamination techniques and processes applied in decommissioning.

Based on the information gathered, some specific characteristics of selected decontamination techniques for segmented components are discussed.

In addition, some critical elements of choosing techniques for a practical decontamination problem are mentioned.

Key words: decommissioning, decontamination.



I.3.13.

Microbiologically Induced Corrosion of Carbon Steel Under Continuous Flow Conditions

Mariana Tunaru, Maria Dragomir, Anca Voicu

e-mail : mtunaru@yahoo.com

Microbiologically induced corrosion is the label generally applied to corrosion involving the action of bacteria on metal surfaces.

While different combinations of bacterial species, materials and chemical constituents are interrelated factors, stagnant water is the factor most often mentioned in reported cases.

This paper presents the results obtained regarding the testing of microbiologically induced corrosion of carbon steel under continuous flow conditions in the presence of iron-oxidizing bacteria.

The tests were performed on coupons of SA106gr.B exposed both in stagnant conditions and in flow conditions.

The surfaces of these coupons were studied by metallographic technique. The developed biofilms were analysed using both the microbiological technique and EIS techniques.

The correlation of all the results which were obtained emphasized that the minimizing the occurrence of stagnant or low-flow conditions can prove effective in reducing the risk of microbiologically induced corrosion in plant cooling-water systems.

Key words: SA106gr.B carbon steel, microbiologically induced corrosion, iron-oxidizing bacteria



I.3.14.

The Study of SCC Behavior of Oxidized CANDU Fuel Claddings Using Electrochemical Methods

A. Dinu¹, M. Abrudeanu², M. Rădulescu¹, L. Velciu¹

¹*Institute for Nuclear Research Pitești, Câmpului Street No. 1, Mioveni, Argeș*
alice.dinu@nuclear.ro

²*University of Pitești, Târgu din Vale Street, No.1 Pitești, Argeș*

With the aim to study the behavior of oxidized zircaloy-4 fuel claddings at SCC in iodine environment, it was prepared some samples with different thickness of oxide layer (2.7, 4.5 and 8 μ m) using a thermo-analyzer type SETARAM SETSYS EVOLUTION 24.

The oxide layers were characterized by EIS spectroscopy and XRD technique. It was put in evidence that the monoclinic oxide ratio increase with the growth of the oxide thickness and that the oxide porosity increasing with the increases the oxide thickness.

The oxidized C-rings were electrochemically tested in following conditions: 0.55g/l KI solution (pH=10.5); applied stress 480 MPa; temperature 85^oC.

By optical and scanning electron microscopy it was observed that the pits appear on the inner surface of the all tested samples. The number of these pits increasing according to the increase of the thickness of oxide layer. In the case of thicker layer, some incipient SCC cracks were put in evidence. It was observed the exfoliation of the external monoclinic under layer, too. In this way, in the inner tetragonal oxide layer appears micro tensions which conducted to breakage of tetragonal oxide. Consequently, the corrosive agent one may penetrate until the zirconium matrix and SCC mechanism can be initiated.

Keys word: oxidized zircaloy-4 fuel claddings, stress corrosion cracking, iodine environment.



I.3.15.

Spectral Dependence of Ultrasonic Attenuation for hydrided Zr-2.5%Nb alloy

V. Ionescu, M. Mihalache, S. Florea

Institute for Nuclear Research - Pitesti ROMANIA,
viorel.ionescu@nuclear.ro

The cold-worked Zr-2.5%Nb alloy is used for the pressure tubes of CANDU nuclear reactors. During the service life in reactor, diffusion of hydrogen and/or deuterium in the pressure tubes wall will occur. Below a certain temperature, stable hydride will form as a brittle phase, which could lead to catastrophic failure.

For this reason, it is very important to be able to investigate the hydrogen effect on the elastic properties of zirconium alloys.

In the present paper, is used a non-destructive testing technique, known as ultrasonic spectral analysis. With this technique was determined the frequency dependence of ultrasonic attenuation from the amplitude spectra of any two consecutive back wall echoes. The algorithm described in this paper requires only one transducer emission – reception (pulse-echo technique).

At material examination, a narrow response in time range is required for good definition of pulse. This is reflected in the frequency response which is relatively broad band.

This algorithm works in the frequency domain by determining attenuation versus frequency.

The measurements were applied to Zr-2.5%Nb samples cutted from the same pressure tube and hydrided at different hydrogen concentrations.

Keywords: ultrasonic spectral analysis.



I.3.16.

Intermetallic Compounds CoAl for γ - Sources

Paula Carlan*, D. Benga*, Maria Mihalache*, C. Ducu**

*Institute for Nuclear Research, Câmpului 1, 115400 Mioveni, POB 78, Pitesti Romania

** University of Pitesti, Research Center Advanced Materials, Targul din Vale, Pitesti, Romania

In order to optimize and enhance the activity of Co- γ sources there was prepared intermetallic compounds CoAl as targets of irradiation. It is known that aluminum is transparent to flux of neutrons. Its activation cross-section is 0,24b. One of the most common superlattices occurring in binary intermetallic phases is the B₂ type (or CsCl type). The disordered structure of B₂ type consists of two identical lattice points, one at the corner position and the second at the body center position. On the average, each lattice points represents 50% of an A and 50% a B atoms. The structure has

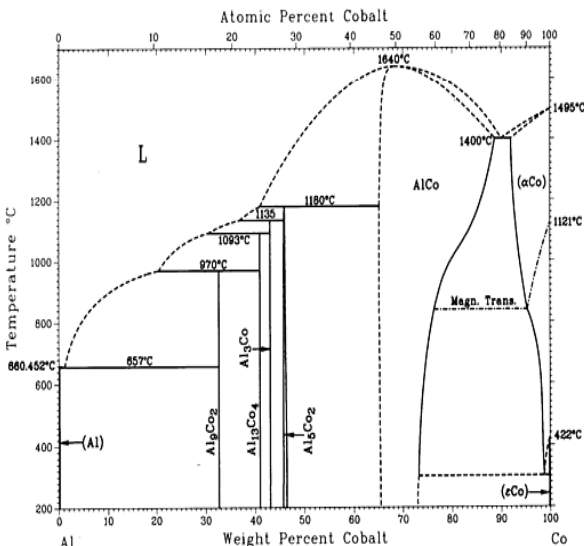


Figure 1 Phases diagram of Co-Al system

the symmetry of a body-centered cubic A₂ type lattice. On the other hand the ordered structure also has two atoms per unit cell with one kind (say, B atom) occupying the body centered position and the other (say A atom) occupying the corner position. The symmetry is no longer of body centered cubic but one of simple cubic with each lattice point associated with two different atoms (B₂ type). The structure may also be considered as consisting of two simple cubic sublattices interpenetrating each other. A perfectly ordered structure with all lattice sites occupied can be achieved only at the stoichiometric composition and at absolute zero. These materials must have a good behavior in all time of service at temperatures up to 600°C, and irradiation with neutrons for activation and λ auto irradiation. The method was to obtain a powder of intermetallic compound by direct reaction of components, Co and Al, in shape of powder and pressed at little density. The composition was those of CoAl and the thermal treatment followed the equilibrium diagram presented in figure 1. The powder was obtained by grinding the compacts. Then the powder was conventionally processed. The powder and the sintered samples were analyzed by currently methods.

The results show a micron powder with a medium size of grains of 30 μ m and the apparent density of 0,35g/cm³. The XRD

analyze relieves the CoAl compound and small quantities of compounds depleted in Al or Co (about 5%). The shape of grains show in electronic microscopy is approximately spherical, keeping anyway, in small percent, a shape of rings following the shape of Co powder.

The sintered samples had densities about 93%TD. The fracture is brittle.

To enhance the ductility of material there will be prepared targets with enriched count in compound with Co₃Al and controlled microstructure.



I.3.17.

Aspects Regarding Non-Destructive Examination of the Surface of the Ring From the Botom of the Welding Seam of two Pipes Having Small Diameter, by Replicating Method

Tiberiu Gyongyoși,

Institute for Nuclear Research

The method has been generated by a real case of breaking of the steel tape from the position control system of the ram "C" (Fuelling Machine).

The paper contains a brief description of the replicating device and of the replicating method proposed. The results obtained and some conclusions are showed at the end of paper.



I.3.18.

Analysis of G52-28 Carbon Steel Expose in Gs1 Column of Isotopic Change

Lucian Velciu, Alice Dinu, Dan Doanta, Stefan Dragomir
 RAAN/Institute for Nuclear Research-Pitesti, Romania; PO BOX 78
 1 Campului Str. Mioveni 115400
 lcvclciu@yahoo.com

This paper presents some analysis performed on G52-28 carbon steel samples exposed in GS1 column of ROMAG-PROD Heavy Water Plant. The samples were maintained in isotopic change column on all period its continuous working (approximate 2,5 years).

Analysis consisted in the quality evaluation of structural material and layer formed, after exposed period, using following methods: optic microscopy (metallography), scanning electronic microscopy (SEM), traction and adherence tests.

Keywords: isotopic change column, carbon steel samples, microstructure, traction strength, adherence test.



I.3.19. Round Table - Structural Integrity of Nuclear Components

**Maria Roth⁽¹⁾, Dan Mihai Constantinescu⁽²⁾,
 Sebastian Brad⁽³⁾, Cătălin Ducu⁽⁴⁾, Viorel Malinowski⁽⁴⁾**

⁽¹⁾ Institute for Nuclear Research, Pitesti, Romania

⁽²⁾ University "POLITEHNICA" of Bucharest, Romania

⁽³⁾ Institute for Cryogenics and Isotopes Technologies, Rm Valcea, Romania

⁽⁴⁾ University of Pitesti, Romania

The round table aims to join specialists working in the research area of the Romanian R-D Institutes and Universities involved in structural integrity assessment of materials, especially those working in the nuclear field, together with the representatives of the end-user – NPP Cernavoda.

This scientific events will offer the opportunity to disseminate the theoretical, experimental and modelling activities, carried-out to date, in the framework of the National Program "Research of Excellence", Module I 2006-2008, managed by the National Authority for Scientific Research. Entitled "Integrated Network for Structural Integrity Monitoring of Critical Components in Nuclear Facilities", acronym RIMIS, the project has two main objectives:

- 1 to elaborate a procedure applicable to the structural integrity assessment of critical components used in Romanian nuclear facilities (CANDU type Reactor, Hydrogen Isotopes Separation installations) ;
- 2 to integrate the national networking in a similar one at European level, and to enhance the scientific significance of Romanian R&D organisations as well as to increase the contribution in solving major issues of the nuclear field.

The topics of the round table will be focused on:

- 1 Development of a Structural Integrity Assessment Methodology applicable for the nuclear facilities components;
- 2 Experimental investigation methods and procedures;
- 3 Numeric simulation of nuclear components behaviour;
- 4 Further activities to finalize the assessment procedure;

Participations and contributions to sustain the activity in the European Network NULIFE, FP6.



I.3.20.

Constructive Solutions to Protect Nuclear Objectives Against Earthquakes, Explosions, Shocks and Vibrations

Viorel Serban, Adrian Panait, Marian Androne, George Ciocan, Ioana Florea
 SITON - Subsidiary of Technology and Engineering for Nuclear Projects

The paper is a presentation of a constructive solution for to protect nuclear objectives against earthquakes, explosion, shocks and vibration employing new SERB 800x800 devices developed by SITON.

The new SERB 800x800 devices can overtake permanent compression loads (up to 6000 KN) over which dynamic compression and thrust loads up to 1500 KN may overlap. On vertical direction, the devices have a very high stiffness in order to avoid the occurrence of different settlements that may lead to additional loads in the isolated supra-structure. On horizontal plane, the devices are sliding with a very low friction coefficient (ranging between 0.03 – 0.08) allowing displacements up to ± 225 mm while limiting the range through a non-linear stiffness which starts operating after 2/3 of the displacement has consumed.

The advantages of the isolation against dynamic actions such as earthquakes, explosions, shocks, vibrations and plane crash impact with nuclear objectives are presented for one PHWR 700 NPP standard unit but the solution may also be applied to any type of nuclear objective including detritation installation.

II. ENVIRONMENTAL PROTECTION

II.1. Radioprotection



II.1.1. Interference Phenomena in Azeotropic Distillation for Liquid Scintillation Measurement

C. Varlam, I. Stefanescu, D. Dobrinescu, I. Popescu, I. Faurescu

National Institute for Cryogenic and Isotopic Technologies - ICIT Rm. Valcea

The Cryogenic Pilot is an experimental project in the national nuclear energy research program, which has the aim of developing technologies for tritium and deuterium separation by cryogenic distillation. The process used in this installation is based on a combined method for liquid-phase catalytic exchange (LPCE) and cryogenic distillation. There are two ways in which the Cryogenic Pilot can interact with the environment: by atmospheric release and through the sewage system. In order to establish the base level of tritium concentration in the environment around the nuclear facilities, we investigated the sample preparation treatment for different types of samples: spinach, spring wheat, onion, hay, grass, apple, garden lettuce, soil, milk, and meat.

For the azeotropic distillation of all types of samples were used two solvents, toluene and cyclohexane, and all measurements for the determination of environmental tritium concentration was carried out using liquid scintillation counting (LSC), with ultra-low liquid scintillation spectrometer Quantulus 1220 specially designed for environmental samples and low radioactivity. Sample scintillation cocktail ratio was 8:12 ml and liquid scintillation cocktail was UltimaGold LLT. The background determined for the prepared blank samples was between 0.926 CPM and 1.002 CPM and the counting efficiency between 25.37% and 26.10%. The counting time was 1000 minutes (50 minutes/20 cycles) for each sample, and the minimum detectable activity according to ISO 9698 was 8.9 TU, respectively 9.05 TU, with a confidence factor of 3.



II.1.2. Current Levels of Radioactivity along Romanian Danube Basin

R. Dobrin, C. Dulama, Al. Toma

*Institute for Nuclear Research, Pitesti
relu.dobrin@nuclear.ro*

The paper contains the most important conclusions and some correlations and interpretations of the results of the radioactivity monitoring of the Romanian Danube basin, performed in the period August 2006 – September 2007, within the framework of the government funded research project: “Research network for integrated monitoring of the radioactivity and environment related isotopes throughout the Romanian Danube basin”. The measurements and samples collection were made within a monitoring plan including locations along the Danube River and on the main of its tributaries or other interesting locations. The targeted radionuclides were Cs-137, K-40, U-235 and the gamma emitting radionuclides of the natural series of the U-238 and Th-232. Results included in this study concern the dose rate measurements, the gamma emitting radionuclides concentrations measured by gamma-spectrometry in soil, sediment and water samples, the in-situ gamma-spectrometry and the gross beta and gamma concentrations measured in sediment and water samples. A series of interesting interpretations of these results was made, i.e. the correlation between the gamma dose rate and its main natural radioactivity contributors, which might serve as support for the representativeness of the sampling method or comparative analyses of the gamma in-situ measurement results and the results obtained by laboratory gamma spectrometry measurements of the soil samples taken from the same locations.



II.1.3. Expert Services Emergence From the “Environment Protection” R&D Program

Alexandru TOMA, Cristian Nicolae DULAMA

Institute for Nuclear Research Pitesti

The recognition by society of the radiation associated risks is commonly conditioned by the fair gain that everybody can get by using nuclear technologies and/or radioactive materials. As far as nuclear technologies penetrated areas vital for the development of human civilization, such as energy or health, the issue of protecting people and environment became more and more meaningful.

The success in operating a nuclear facility depends on several factors, among which an important one is the settlement and maintenance of efficient radiation protective measures. In this respect, the SCN approached two primary objectives:

- insurance of radiation and environment protection for its own nuclear facilities, and
- sustaining an R&D program destined to increase proficiency in human and environment protection against radiation-associated risks.

The paper presents the technical and scientific objectives of the R&D program *Environment Protection* and its main results. Many of these results have an important potential for applications, by increasing the competence and capabilities of the laboratory in support of nuclear industry, nuclear forensic activities, oil, gas, metal processing, cement production and fertilizers industry for NORM, radioactive materials shipment etc. Relevant examples of this kind of applications are presented in the paper.



II.1.4.

Monitoring System of Radioactive Gas Emissions from TRIGA INR 14MW

Ion Oprea, Stefan Covaci, Tudor Galan*, Radu Vasilache**

Institute for Nuclear Research Pitesti, Mioveni, ROMANIA, office@nuclear.ro;

**SC Global Nuclear Services SA, Constanta, ROMANIA*

***SC Canberra Packard SRL, Bucharest, ROMANIA*

In site monitoring activity of radioactive gas emissions is mandatory for every nuclear facility. This activity must comply with National Regulatory Body requirements and international recommendations.

The paper present the monitoring system of radioactive gas emissions from TRIGA INR 14MW, related Engineering and Quality Assurance issues and human – machine interface optimization aspects.



II.1.5.

Airborne Tritium Emissions Dynamics at Cernavoda NPP, 1997 - 2007

I. Popescu, E. Bobric

ipopescu@cne.ro, ebobric@cne.ro

Health Physics Department, CNE Cernavoda, Romania

Most of the tritium in a CANDU reactor is formed by thermal-neutron-capture reactions, ${}^2\text{H}(n, \gamma){}^3\text{H}$, which occur both in the Moderator and Primary Heat Transport System.

Very small amounts of tritium may escape from heavy water circuits and auxiliaries of a CANDU reactor during normal operation and outages. Tritium emissions of Cernavoda NPP have been monitored since 1996, when the plant became operational. In present HTO is continuously sampled on molecular sieves and measured daily.

Time series statistical methods were applied to study tritium emissions in air in order to estimate trends, to develop criteria for identifying abnormal emissions, and to predict further emissions. Tritium releases evolves stochastically around a slowly increasing trend but PHT and Moderator events with significant leaks could have an important contribution to tritium releases to the environment

The standard deviation of time series residuals distribution, after detrending, could be used to establish investigation criteria for the emission levels.



II.1.6.

Occupational Exposure to External Ionising Radiation: Personnel Monitoring and Dose Evaluation at Cernavoda NPP

Catalina Chitu, cchitu@cne.ro, Ion Popescu, ipopescu@cne.ro,

Vasile Simionov, vsimionov@cne.ro

"CNE Cernavoda" NPP, No. 2, Medgidiei Str. Cernavoda 905200, Romania

phone: 4-0241-239-340; fax: 4-0241-239-269

Cernavoda NPP has two CANDU 600 reactors in commercial operation, first since December 1996, the second since November 2007. For a CANDU type reactor the major contributor (95%) to the external dose is gamma radiation.

Individual dose monitoring is provided by an accredited dosimetric service, approved by the Romanian regulatory body, National Commission for Nuclear Activities Control (CNCAN), at CNE-PROD.

For all the persons entering radiological controlled areas (CNE Cernavoda employees, short-term atomic radiation workers, contractors and visitors) Health Physics Department provides individual dosimetric surveillance.

During Unit 1 new fuel loading activities in 1995 individual dosimetric surveillance was provided for 30 individuals using film dosimeters.

Since Unit 1 "Radiation Island" in effect, on February 26th, 1996, individual monitoring for external gamma radiation exposure is performed using thermoluminescent dosimeters (TLDs).

When entering / working in areas where approved dose rates could be exceeded (variable or heterogeneous gamma radiation fields) beside TLD an electronic, direct reading, Personal Alarm Dosimeter (PAD) is used.

When entering / working in areas with significant neutron dose rates an integrating portable neutrons monitor is used (both as field instrument and personal dosimeter).

When contact beta-gamma dose rate exceed 10 times the dose rate at the level of the chest, thermoluminescent extremities (hands or / and feet) dosimeters are used.

The main purpose of design and implementation of a "Monitoring, Evaluation and Recording of Individual Doses Program" (Individual Dosimetry Program) is to measure, assign and record all the significant radiation doses ($H_p(10)$, $H_p(0.07)$ and E_{50}) received by an individual during activities performed at CNE Cernavoda NPP and ensure that all the exposure are kept ALARA.

The actual levels of individual and collective effective doses due external exposures 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.

During nearly 12 years of operation, most of the exposures (76%) were below the Recording Level and the majority of recordable doses (60%) were less then 1 mSv. Any legal or administrative individual dose limit has not been exceeded.



II.1.7.

Local Radiation Protection Measurement Network for TRIGA REACTOR Building

*ing. Corneliu Mihail TALPALARIU, ing. Jeni Elena TALPALARIU,
ing. Corina MATEI, ing.fiz. Valentin STOICA , fiz. Nicolae VÂJA, fiz. Ion OPREA
Institute for Nuclear Research Pitesti, Câmpului no. 1, 115400, Mioveni, România
talpalariu_cornel@yahoo.com*

This system is intended for the protection of personal working in those areas of the Reactor Building where high gamma radiation fields are expected.

A detector, sensitive to gamma radiation, is installed in each of the areas to be monitored. The detector will send a signal proportional to the radiation level in the area, to a corresponding electronic module (Alarm Unit), where the signal will be amplified and checked against alarm set-points for possible alarming condition.

In case the field exceeds the alarm set-point, the Alarm Unit will produce a signal that will trigger the field alarms (Horn and Beacon), located in the area where condition occurred.

Each Alarm Unit will send a numerical input to central computer command.

The system is required to accomplish of the following:

- Monitor the level of gamma radiation in those areas of the Station where high radiation fields are expected.
- Provide a continuous and centralized display of the radiation level in each of the monitored areas. The display shall be in exposure rate units (R/h).
- Provide a visual and audible alarm in each monitored area.
- The control room operator shall be able to check at any time the radiation levels and alarm conditions in each of the monitored areas.
- The control room operator shall be alerted of any alarm conditions that occurs in the Station.

A typical monitoring loop is composed of the following components:

- Detector Assembly type: CI-MA-522 two channels, two range;
- Horn and Beacon Assembly;
- Remote Indicating Meter;
- Remote Monitoring Computer.
- Central computer; common for all loops.

Technical Features

Measurement range: 0,1mR /h÷100R/h(1µs/h÷1s/h).

Measurement Accuracy :± 20%.

Maximum channels numbers 40.

Response time 2 sec.

4π geometry response for detectors.

100 K/rad high level radiation rezistent.

Some significant features:

- substantial reduction in panel complexity many of the fixed indicators and controls can be eliminated in favor of interactive CRT console;
 - fault detection and diagnosis;
 - system performance monitoring;
- intelligent alarm handling.

II.2. Radioactive Waste Management



II.2.1. Physico-Chemical Cement Matrix Characterization Tests for Ion Exchange Resins from a NPP

E. MAGRO, G. PIÑA, E. RUIZ, E. BENAVIDES*, P. LÓPEZ**

L&I Radioactive Waste Unit, CIEMAT, Madrid, Spain, enrique.magro@ciemat.es

**NUCLIN, Madrid, Spain, ebenavides@nuclin.com*

***SOCOIN, Madrid, Spain, plopezf@socoin.es*

The characterization of materials used for immobilization of radioactive waste is a fundamental task to demonstrate the suitability, security and viability of the locations used for its interim or final disposal. This report covers the results obtained in the experiments carried out in the L&I Radioactive Waste Unit of CIEMAT aimed to obtain the better mixture to immobilize the operation spent ion exchange resins from a VVR NPP according to particular technical specifications.

The assays was planned and carried out in two phases. The first phase to decide the optimal formulation and the second one in order to attest the qualification of the selected formulation according to the technical specifications proposed. It has been developed tests of resin humidity, consistency and aspect of the mixture, cement mechanical properties after different curing periods and storage conditions, cement setting time and leaching behavior of Cs-137 and Co-60.

Different mixture proportions were tested in order to find the most appropriated one within the specification. The following characterization studies were tried with a number of prismatic specimens and prepared with the formulation who presented better characteristics in terms of mechanical strength, aggregate stability, fluency and handy of the mixture.

Assays of Setting time, density, setting temperature

The reostatic properties and its relative ratios were tested as well as dimensional controls of the specimens were carried out at different conditions

- ✓ after a curing period of 28 days in environmental climate storage,
- ✓ after 90 days of storage in water immersion
- ✓ after a curing period of 90 days in ambient conditions,
- ✓ after 30 cycles (-40 °C to +60 °C) in a humid chamber control system

Leaching test behaviour of different samples traced with Cs-137 and Co-60 (according to specifications) during 90 days was performed.

The completed tests show that selected mixture fulfils the specifications in terms of waste minimisation, matrix durability and nuclides leaching resistance.



II.2.2. Treatment of Waste Water Involved in Acid Uranium Ore Processing

Antoneta Filcenco – Olteanu, V.T. Vacariu, Eugenia Panturu,

Rozalia Radulescu, Carmen Diaconu

Research and Development National Institute for Metals and Radioactive Resources, - Bucharest

The uranium ores lixiviation is one of the hydrometallurgical processes which generate waste waters that need to be treated in accordance with the quality requirement of NTPA - 001/2002. The uranium ores acid lixiviation process generates two types of waste waters: (I) an alkaline solution out of which uranium was extracted by an ionic exchange process and (II) a mixture of mother - water and washing waters from DUNa precipitation. In order to treat wastewater (I) a flow sheet is proposed that includes: the uranium removal to the CMA level using the ion exchange process; the effluent's neutralization with sulphuric acid and reverse osmosis, which recovers more than 80% of water; the treatment of the mineral salts concentrate by crystallization - evaporation. Wastewater (II) is processed using its recirculation in the elution industrial process. The excess is treated by neutralization with sulphuric acid, molybdenum adsorption on active carbon, sulphate removal using calcium chloride; the sodium chloride solution is recycled to add up to the elution of uranium recovered from (I).



II.2.3.

Possibilities of Using New Technology Materials in Constructing the Radioactive Waste Containers

Itu Răzvan Bogdan – C.N.H. S.A. Petroșani

The paper will consider using the latest technologies in material science for building the radioactive waste containers. A new amorphous steel has been discovered by the scientists from the University of Virginia, a material three times stronger than conventional steel and non-magnetic. Scientists shown that this steel, DARVA – Glass 101, has superior anticorrosive proprieties.

The paper will also consider using Para-Aramid in protecting the radioactive waste containers. Chemical and physical proprieties of these materials shown a great tensile strength and the inter-chain bonds make these materials extremely strong.



II.2.4.

Tecniques for Long Term Conditioning and Storage of Radium Sources

Gheorghe Dogaru, Felicia Dragolici, Mihaela Nicu

*National Institute of Reserch&Development for Physics and Nuclear Engineering – Horia Hulubei, Magurele, Ilfov, Romania
dogaru@nipne.ro*

The National Institute of Research&Development for Physics and Nuclear Engineering-Horia Hulubei developed its own technology for conditioning of radium spent sealed radioactive sources. In the mean time, the institute benefited of IAEA assistance in order to set up a laboratory dedicated to radiological characterization, identification of radium sources as well as the encapsulation of spent sealed radioactive sources. The laboratory has been arranged into a room used for storage of scrap metal and other metallic pieces resulted from demolish or decommissioning of some non nuclear devices. The room was equipped with a local ventilation system, welding devices, tightness test devices as well as radiometric portable devices. The wells were painted and the floor was covered by plastic sheets in order to prevent the spreading of contamination.

Two types of capsules have been designed for conditioning of radium spent sealed radioactive sources. For these kinds of capsules different type of storage packaging was developed. Data on the radium inventory will be presented in the paper.

The paper contains the description of the process of conditioning of spent sealed radioactive sources as well as the description of the capsules and packaging. The paper describes the equipment used for the conditioning of the radium spent sealed sources.

Keywords: radium sources, spent sealed radioactive waste, radioactive waste, conditioning, long term storage



II.2.5.

Regulatory Review of the C-14 Inventory Proposed for Disposal at the Saligny Planned near Surface Repository

Daniela Dogaru¹, Petre Ilie², Gheorghita Jinescu³

¹*National Commission for Nuclear Activities Control, 14 Libertatii, 5 Bucharest*

²*Institute for Nuclear Research, 1 Campului, Mioveni, Arges*

³*"Politehnica" University of Bucharest*

According to the "National Strategy for long and medium term concerning the management of spent fuel and radioactive waste, including the final storage", it is anticipated that in 2014 a near-surface disposal for radioactive waste generated by the operation and decommissioning of Cernavoda Nuclear Power Plant will be commissioned. Since 1992 an investigation program was started to select an appropriate site for the near surface repository, and Saligny site, adjacent to Cernavoda Nuclear Power Plant, was declared as the preferred site. The first attempt to assess the safety of repository was in 1997 when a safety assessment report has been submitted to the Regulatory Authority-CNCAN. At that time, the Regulatory Authority considered the report as incomplete, with poor information on the radioactive inventory and poor characterization of the site. In the mean time, new site investigation and characterization works have been performed and characterization of the radioactive inventory has been started. In 2007 a technical documentation for Saligny repository was submitted by the National Agency for Radioactive Waste-ANDRAD to the Regulatory Authority for obtaining the partial siting license. The radioactive waste from operating CANDU reactors contains high activity concentration of C-14 compared to light-water reactors. The major contribution of the total effective dose summarized for all exposure pathways after the post closure period is associated with C-14. In the technical documentation, ANDRAD developed a particular approach for estimating the C-14 inventory in the repository. In the first step, intrusion scenarios were analyzed using the C-14 inventory as a variable parameter. In the next step, a dose constraint of the 10 mSv/year

was imposed to derive the activity of C-14. Finally, the value of the inventory was slightly reduced to provide a factor of safety, such that the overall calculated intrusion dose was about of the 7 mSv/year. Based on this estimation, the Regulatory Authority imposed dose constraint for C-14 of 1 mSv/year, which leads to the need to reassess the inventory disposed into repository. The Regulatory Authority self assessment of C-14 associated with non-fuel contact and fuel contact resins generated by first unit of the Cernavoda Nuclear Power Plant will be compared with the C-14 inventory proposed by ANDRAD for disposal. This paper contains a brief description of the Regulatory Authority review process for licensing of a near surface repository, as well as a Regulatory Authority self assessment of dose associated with C-14. Also, the paper includes the regulatory review of human intrusion scenarios presented by ANDRAD in the safety assessment report, and the Regulatory Authority justification for dose constraint of 1 mSv/year associated with C-14 imposed by license.



II.2.6. Dynamics of Radioactive Waste Generation

Daniela Dogaru, Cornelia Virtopeanu, Alexandrina Ivan

National Commission for Nuclear Activities Control

14 Libertatii, 5 Bucharest, Romania

In Romania there are in operation three facilities licensed for collection, treatment and storage of radioactive waste resulted from industry, research, medicine, agriculture named institutional waste. One of them belongs to National Institute of Research&Development for Physics and Nuclear Engineering-Horia Hulubei and another two belong to Nuclear Research Institute from Pitesti. There is in operation the National Radioactive Waste Repository which is a near surface disposal facility. The repository is located in Baita Bihor and it is designed to dispose of this kind of radioactive waste. The institutional radioactive wastes generated are allowed to be disposed into repository according to the waste acceptance criteria defined for the disposal facility. The radioactive wastes which are not allowed for disposal are stored on the site of each facility above mentioned.

The paper describes the dynamics of generation of institutional wastes in Romania both for radioactive waste which are allowed to be disposed into repository and for radioactive waste which are not allowed to be disposed of. The dynamics of generation of radioactive waste is based on annual operating reports submitted to regulatory authority by each treatment, storage and disposal facility from Romania.



II.2.7. Development of the Conceptual and Numerical Models for Semi-Generic Site to Support Performance Assessment for CANDU Spent Nuclear Fuel Disposal

Petre Ilie¹, Alexandru Danchiv², Alice Ionescu³, Viorel Deaconu¹, Gabriel Ghita¹

¹*Subsidiary for Nuclear Research Pitesti*

²*University of Bucharest Faculty of Geology and Geophysics*

³*National Agency for Radioactive Waste*

The aim of this paper is to give essential information on the current status of the studies developed in the framework of Research Contract "The development of numerical models and computer codes in support of siting and performance assessment studies of geologic repository in crystalline and igneous rocks", part of the Agency's coordinated Research Project "The use of numerical models in support of site characterization and performance assessment studies of geologic repositories". As a result of the discussion during the first RCM (Beijing, 2006), the Versania (Ukraine) site, for which a preliminary characterization was available, was selected as a viable "semi-generic site", to conduct the comparative process modelling studies. The disposal concept for CANDU fuel in granite is adapted following a Canadian concept. The repository consists of 40 disposal rooms in which 11280 containers are disposed representing the spent fuel resulted from four nuclear CANDU units after 40 years of operation. The paper describes the disposal concept and repository layout, and the conceptual model for radionuclide release. The near-field transport code GRAPOS (a component of the EMOS7 code) was used for source term calculation. The code GRAPOS distinguished between different parts of the inventory, and considers radioactive decay, solubility limits, diffusion through the bentonite buffer, and the diffusive release into the advection-dominated flow in the excavation disturbed zone and interconnected fractures. Sensitivity analyses were performed considering variations in repository layout, the presence or absence of fractures, and matrix dissolution time. For the same repository layout and simulation time, it has been shown that the contaminant release rates for the fractured medium are higher than for the porous medium. The modelling of the near-field did not consider any optimized repository layout, based on temperature considerations, nor the temperature dependencies of certain parameters. The modelling should be refined in the future to account for temperature effects. A far-field model was developed using the conceptual model provided by Ukraine, with a discrete horizontal and vertical fracture embedded into a porous matrix. The computation was restricted to a 2D cross section of the disposal site. The dimensions of the cross section were 5000 m along the horizontal and 1200 m along the vertical. The model is made up by a sedimentary cover containing two aquifers separated by an aquitard, and the granite layer which is separated from the lower aquifer by a second aquitard. Two layers can be distinguished in the granite: an upper part, the fractured granite and a lower one, the monolithic granite – the repository host. Sensitivity analyses were performed with respect to

the head gradient imposed across the model. Moreover, the impact of horizontal and vertical fractures was evaluated. Hydraulic heads and concentration distributions were calculated for the considered variants. As a first step to assessing coupled THM effects, conductive heat transfer was calculated using rather detailed model of the repository system. The analysis, which did not include coupled effects, was performed using the general-purpose finite element code ANSYS. Heat output from spent nuclear fuel was specified as a time-dependent function, and temperatures throughout the model domain were calculated.



II.2.8.

Current Status and New Trends in the Methodology of Safety Assessment for near Surface Disposal Facilities

Petre Ilie¹, Liana Didita¹, Alexandru Danchiv²

¹*Subsidiary for Nuclear Research Pitesti*

²*University of Bucharest Faculty of Geology and Geophysics*

The main goal of this paper is to present the status of the safety assessment methodology at the end of IAEA CRP "Application of Safety Assessment Methodology for Near-Surface Radioactive Waste Disposal Facilities (ASAM)", and the new trends outlined at the launch of the follow-up project "Practical Implementation of Safety Assessment Methodologies in a Context of Safety Case of Near-Surface Facilities (PRISM)". In the present paper we describe our consideration based on participation in the ASAM project by IAEA Research Agreement "Modelling Subsurface Transport at the New Romanian Near Surface Repository Site" and upon published information and the results of a Common Application Aspects Working Group questionnaire. The ASAM project has successfully tested the applicability of the ISAM methodology to a range of near-surface radioactive waste disposal facilities, and also addressed the review of safety assessments, confidence building and the safety case, and has examined selected issues that need to be referred in safety assessment whatever the facility type. Over the duration of the ASAM project, the ISAM methodology was confirmed as providing a good framework for conducting safety assessment calculations. In contrast, ASAM project identified the limitations of the ISAM methodology as currently formulated. The major limitations are situated in the area of the use of safety assessment for informing practical decisions about alternative waste and risk management strategies for real disposal sites. As a result of the limitation of the ISAM methodology, the PRISM project is established as an extension of the ISAM and ASAM projects. Based on the outcomes of the ASAM project, the main objective of the PRISM project are to: (1) develop an overview of what constitutes an adequate safety case and safety assessment with a view to supporting decision making processes, (2) provide practical illustrations of how the safety assessment methodology could be used for addressing some specific issues arising from the ASAM project and national cases; and (3) support harmonization with the IAEA's international safety standards. Finally, the paper includes some authors' suggestions for the new project formulated tasks during the meeting on the outcome of the ASAM project and proposals for follow-on work that took place at the IAEA's Headquarters in Vienna from 10-14 December 2007.



II.2.9.

Considerations on the safe and Security of Radioactive Materials Transport in Romania

Gheorghe VIERU, Ph. D

Head, Reliability and Testing Laboratory

Institute for Nuclear Research, Pitesti, P. O. Box 78, ROMANIA

The safe and security during transport of dangerous goods class 7-Radioactive Materials (RAM) is a very important problem taking into consideration its potential risks and radiological consequences due to carrying-out this activity.

Romania as a Member State of the International Atomic Energy Agency has implemented national regulations for a safe transport of RAM in accordance with the IAEA's recommendations as well as of other international specialized organizations.

Based on the IAEA's Safety Standard-TS-R-1 (ST-1), Romanian National Nuclear Regulatory Body – CNCAN adopted and implemented, by order nr. 357/December 2005, the safe regulations for the transport of the radioactive materials in Romania.

The paper will present the main sources of radioactive materials in Romania its transportation routes with a particular interest paid to the radioactive wastes (low level and mixed low-level radioactive materials), and NORM materials (natural uranium ore).

Starting from the fact that the risk, safety and security in the transport of radioactive materials is dependent on packaging appropriate for the contents being shipped, rather than operational and/or administrative actions required for the package, the paper present, very briefly, the main packages used for transport and storage of such RAM in Romania.

There are presented also specific problems related to the identification and evaluation of the risks and potential radiological consequences associated with the transport of radioactive materials in Romania, for all those three situations: routine transport (without incidents), normal transport (with minor incidents) and during possible accidents.

As a conclusion, it is stated that the evaluated annual collective dose for the population due to RAM transportation is less than those received by natural radiation sources. In the same time it is concluded that Romanian built packages are safe and prevent loss of its radioactive contents into environment.

Background

This paper is a part of the Romanian National Nuclear Programme where the author is responsible for the approach of risk and safety of RAM materials in Romania.

References:

1. **G. Vieru**, "*Ten years of Romanian experience on risk assessment owing to radioactive materials transport in Romania*", Packaging, Transport, Storage & Security of Radioactive Materials Journal, Maney Publishing, London, UK, Volume 17, Number 4, (2006)
2. **Romanian Nuclear National Programme**, (2002);
3. **G. Vieru**, "*Risk and Safety Evaluation in Radioactive Waste Transport in Romania*", RAMTRANS, Vol. 10, No. 2, pp. 105-112, London, UK, (1999);
4. **G. Vieru**, "*The Identification and the approach of the risk and safety problems associated with the transport of radioactive materials in Romania*", Romanian Nuclear Program, Internal Report, **INR Pitesti, Romania**, (2002);
5. **G. Vieru**, "*Safety criteria for the transport and storage of the Radioactive Materials, in Romania*", Romanian Nuclear Program, Internal Report, INR Pitesti, Romania, (2001);



II.2.10.

Characterisation of Radioactive Waste at Cernavoda NPP Unit 1 During Normal Operation

M. Iordache, L. Bujoreanu, I. V. Popescu
Institute for Nuclear Research, Pitesti, ROMANIA

During the operation of a nuclear plant results significant quantities of radioactive waste that have a very large diversity. At Cernavoda NPP the important waste categories are non-radioactive wastes and radioactive wastes, which are manipulated completely different from which other.

For a CANDU type reactor, the appearance of radioactive wastes is due to contamination with the following types of radioactive substances:

- fission products resulted in the nuclear fuel burning;
- activated products of materials which form part of the technological systems;
- activated products of process fluids.

Radioactive wastes can be in solid, liquid or gas form.

At Cernavoda NPP the solid wastes represent about 70% of the waste volume which is produced during plant operation and as a consequence of maintenance and decontamination activities. The most important types of solid wastes that are obtained and then handled, processed (if required) and temporarily stored are: solid low level radioactive wastes (classified as compactable and non-compactable), solid medium radioactive wastes, spent resins, used filters and filter cartridges.

The liquid radioactive waste class includes organic liquids (used oil, scintillator liquids and used solvents) and aqueous wastes resulting from process system operating, from decontamination and maintenance operations.

Radioactive gas wastes occur subsequent to the fission process inside the fuel elements as well as due to the process fluids neutron activation in the reactor systems. As result of the plant, iodine, noble gases, tritium and radioactive particles occur and are passed to the ventilation stack in a controlled manner so that an exceeding of the maximum permissible concentrations of radioactive material to the environment should not occur.



II.2.11.

Non Destructive Assay for Waste Characterization. Is it Enough?

C. Dulama, R. Dobrin, Al. Toma
Institute for Nuclear Research, Pitesti, ROMANIA

Non destructive assay is the most common technique used for categorization and characterization of radioactive waste. It involves direct analysis of materials by observing their spontaneous emission of nuclear radiation. NDA may be used as control method for several stages of a waste management program. To be successful NDA requires a good definition of the waste streams, a better knowledge of the waste producing processes, an accurate description of the waste forms and a skilled technical personnel.

A common misperception is that a waste package has not been properly characterized unless it has been subjected to NDA. What is required for proper characterization is determining an appropriate level that will be specific to the wastes, the generation process and regulatory regime. NDA is good for some applications but severely limited for others.

NDA does not by itself provide conclusive characterization. It always requires a minimum amount of process knowledge to interpret the results.

It is advisable to perform the waste characterization as close to the time of waste generation as possible by using for measurement the simplest methods designed to match the given waste streams. This seems to be the most cost effective and accurate approach unless there is necessary to characterize historical wastes.

Even the simplest and cheapest NDA methods may be used with high degree of accuracy when there is an abundance of process knowledge and the characteristics are stable and reproducible. Otherwise, if there is little prior knowledge or the package characteristics exhibit a high degree of variability, more sophisticated and complex methods must be employed, and accuracy will be severely limited.

The paper presents the rationales of the waste characterization program employed at the Institute for Nuclear Research as part of the radioactive waste management activities. The approach is a combination of destructive and non destructive analytical techniques which are applied on the main waste streams.

Author's conclusions are related to the increasing need for the proper characterization of the waste streams by focusing the analytical effort on the quantification of difficult to measure radionuclides and development of site specific scaling factors which should be used for further complete characterization of the waste. In this perspective NDA technique becomes the central point of the waste characterization program being the main analytical tool in a well structured knowledge based management system.

Key words: NDA, waste characterization, waste stream



II.2.12.

Experimental Study for the Assessment of Conditioning Matrices of the Radioactive Aluminum Wastes

**Nicoleta Deneanu¹, Ion V. Popescu¹, Mirela Dulama¹,
Jon Teoreanu²**

¹ Institute for Nuclear Research Pitesti –Campului Street, No. 1, 115400, Mioveni -Arges, ROMANIA

² University "POLITEHNICA" Bucharest, Gheorghe Polizu Street, No.1, sect.1, Bucharest ROMANIA

At national level, there are aluminum radioactive wastes produced in the decommissioning of the research reactor VVR-S Măgurele and neutron irradiation installations from this reactor, from Institute for Atomic Physics platform and from decommissioning of the multi-area zero power reactors, from SCN platform.

Internationally are considered two basic options regarding the management of the aluminum radioactive wastes: (1) decontamination of the aluminum and reuse of the aluminum release under governmental authority and (2) treatment and conditioning in security conditions for final disposal. National Strategy on intermediate and long term regarding management of the spent nuclear fuel and of the radioactive waste, including the final disposal and decommissioning of nuclear and radiological installations is part "National strategy for Development in Nuclear Field and of the afferent action plan, approved through H.G. 1259/2002. In Romania the main objective regarding the activities of radioactive waste management is to ensure reasonable achievable minimum impact over the population and environment, established by the national regulating authority through the CNCAN order 56/2004 "Fundamental norms for safe management of radioactive waste".

Specific property of the aluminum, for example, high reactivity with alkaline environments of inorganic binders with yielding hydrogen, may be produce destruction of the conditioned waste forms or produce fire end explosion, conduct of the major risk in the storage, transportation and repository.

A method of conditioning of aluminum radioactive wastes [3], comprising the steps of reacting metallic aluminum with an alkali solution to generate a hydrogen gas and mixing the resulting reaction liquid with a solidifying material containing a latent hydraulic material as a main component to the conversion into a *conditioned waste form*¹ which minimizes the probability of radionuclides release to the environment during interim storage, transportation and final disposal.

LiNO₃ addition to the cement solidified miscellaneous wastes has been proposed for preventing hydrogen gas generation caused by the corrosion of aluminum materials contained in the wastes [4].

Results of a series of experimental tests performed to determine the influence of matrix characteristics on the properties of aluminum immobilized into cement matrices are presented.

REFERENCES

1. *-* Norma fundamentală pentru gospodărirea în siguranță a deșeurilor radioactive, CNCAN, Nr.56, Monitorul Oficial, Partea I nr. 393 din 4 mai 2004 (in Romanian)
2. *-* INTERNATIONAL ATOMIC ENERGY AGENCY, Radioactive Waste Management Glossary, Technical Reports, IAEA, Vienna (2003)
3. Method of disposal of metallic aluminum-containing radioactive solid waste_US Patent Issued on October 26, 1999 MATSUO T. ; NISHI T. ; MATSUDA M. ; IZUMIDA T. LINO[3] addition to prevent hydrogen gas generation from cement-solidified aluminum wastes J. Nucl. Sci. Technol. ISSN 0022-3131 CODEN JNSTAX 1995, vol. 32, No.9, pp. 912-920

¹ Waste form- Waste in its physical and chemical form after treatment and/or conditioning (resulting in a solid product) prior to packaging. The waste form is a component of the waste package [2];



II.2.13.

Treatment of Liquid Radioactive Waste by Membranes Techniques – Needs and Future Trends

M. Dulama¹, I.V. Popescu¹, N. Deneanu¹, E. Dumitru¹, M. Pavelescu²

1 – Institute for Nuclear Research, Pitești, Romania, mirela.dulama@nuclear.ro,

2 - Academy of Scientists - Bucharest, Romania

The nuclear industry generates a broad spectrum of low and intermediate level liquid radioactive wastes (LRWs). The treatment methods used for liquid radioactive waste are related to the conventional processes used for the treatment of the industrial wastewater, like chemical treatment, adsorption, filtration and ionic exchange. The use of such of techniques is limited due to their decreased capacity to remove the radioactive contaminants, the increased operational costs (e.g. evaporation) or the yielding potential for significant quantities of secondary solid waste.

After development of suitable membrane materials and their long-term verification in conventional water purification fields, these membrane processes have been adopted by the nuclear industry as a viable alternative for the treatment of LRWs. The most utilized processes in the field of water and wastewater treatment are those utilizing pressure gradient as the process driving force. These processes include reverse osmosis, ultrafiltration and microfiltration, allowing the selective removing both dissolved and particulate contaminants.

The membrane separation processes can be used alone or as part of complex treatment schemes, which combine conventional technologies and membrane techniques. These combined systems appeared to have superior treatment performances and to be capable of producing high quality treated effluents, bearing an acceptable level of residual radioactivity, for discharge. In addition, the volumes of secondary radioactive waste residues are minimized and can be suitably conditioned to meet the waste form criteria for disposal.

Membrane systems are rarely acquired off the shelf but must be designed and then built only after extensive on-site testing for each specific application. Selection of proper membrane materials and the membrane module configuration is a prerequisite for the successful application of a membrane system.

Knowledge of the characteristics of the feed water is mandatory for understanding of any changes that may occur in the plant's performance. It is the non-radioactive components of the feed water that will determine the overall throughput of the plant, with considerations of radioactivity provided for in the design and layout of the plant.

The purpose of the experimental study was to elaborate and evaluate an adequate technology for treatment of low salt liquid radioactive waste, by using semipermeable membrane. The wastes of concern are those produced during the POD, CAN-DECON, CAN-DEREM and AP decontamination processes and their treatment should assure: the effluents with activity concentration below the prescribed limits, radioactivity confinement in a volume as low as possible and an acceptable cost.

References

1. *** Technical Reports Series no. 431, Application of Membrane Technologies for Liquid Radioactive Wastes Processing, International Atomic Energy Agency, Vienna, November 2004,
2. T.D.BROCK, Membrane Filtration: A User's Guide and Reference Manual, Science Tech Inc., Madison, WI, 1983.



II.2.14.

¹³⁴Cs, ¹³⁷Cs and ⁶⁰Co Migration in Bentonite

Carmen Arsene

Institute for Nuclear Research Pitesti, Romania

The radionuclide migration through the hydrological medium is of primary importance with respect to the establishing the emplacement medium of the low and intermediate radioactive waste repository. To know and understand the migration process represents one of the key elements for understanding the possible impacts resulted from potential releases of the radionuclides from repository.

The waste management strategy includes their conversion into insoluble forms and available for final disposal into geological formations, approaching the disposal leading to the developing of more concepts depending the nature of the geologic shelf selected. A concept model for a repository emplaced in a clay soil and developed by the Centrum for Nuclear Research of the European Community are based on the analysis of all the processes could produce radioactive leakages from repository and on the assessment of the barriers are present or might be interposed between the waste and habitants (technological barriers - immobilization matrix, drums, fill material and natural barriers - geological isolation, geochemical barrier, environment).

Every barrier has be analyzed on the basis of available information and of some sets of experimental studies necessary for getting the best hypotheses about the control mechanisms that could provide date for the model assessment. These studies have to direct a special attention to radionuclide migration in geosphere and to the retardation phenomena during the transport of the radionuclides that might be solubilized from the repository.

The experimental studies consist in determination the results of the processes that take place between the dissolved radionuclides (^{134}Cs , ^{137}Cs and ^{60}Co) and bentonite. Laboratory experiments were carried out in active, by batch method, and had in view:

- determination the retention of the radionuclides by the bentonite - by determination of the distribution coefficients (K_d) and the retardation factors (R);
- the determination of the emplacement medium conditions

In the experiments the bentonite was used in its raw form, it did not stand to any process, drying, pulverizing, sieving. As contacting solutions, the liquid waste from TRIGA reactor, containing Cs-137, Cs-134 si Co-60, have been used.

The activities were measured by a multichannel spectrometer γ with detector of Ge hyper pure.

The distribution coefficients obtained as a result of experiments are of $2\text{-}5 \times 10^2$ ml/g magnitude for Cs-137, Cs-134, and up to 800 ml/g for Co.

For all the studied radionuclides it was showed the values of retardation factors are very close by the values of distribution coefficients, according a direct proportionality between these two parameters, given by the calculation formula, and also a dependence on the characteristics of the material, density, porosity (whose ration is close by 1).

Comparing with the literature data, we can admit that, at least the type of the bentonite used in the experiments has a much higher retardation capacity.



II.2.15. Communication Process Related to Lilw near-Surface Repository Siting

Stela Diaconu

National Agency for Radioactive Waste (ANDRAD), Mioveni, Romania stela.diaconu@andrad.ro

The main responsibility of the National Agency for Radioactive Waste (ANDRAD) is the safe disposal of the radioactive waste, including spent nuclear fuel. At present, we are focused on the development of a near surface repository which will accommodate short-lived low and intermediate level waste (LILW) with certain quantities of long-lived radio-nuclides generated by the operation and decommissioning of the four units equipped with CANDU-6 Canadian type reactor at Cernavoda Nuclear Power Plant. In order to get the siting license for this repository, ANDRAD has coordinated a lot of studies and analysis and is still hardly working for the technical solution. On the same time, ANDRAD is committed to provide efficient information to the population, to all partners with responsibilities in the process of radioactive waste management and to create the necessary framework to consult the population within the neighboring areas and involve them in the decision making process. The paper deals with the most important communication activities developed and foreseen to be developed by ANDRAD within the siting process of LILW near-surface repository.



II.2.16. Radioactive Waste Inventory in Romania – Current Stage

Cristian Litescu

ANDRAD, Pitești, România, cristian.litescu@andrad.ro

According to the Governmental Ordinance No. 11/2003 on the safe management of spent nuclear fuel and radioactive waste, including final disposal, with the subsequent modifications and completions, ANDRAD provides the constitution and the annual updating of a **national database** regarding the quantity and types of radioactive waste which is generated, including those resulting from decommissioning of the nuclear and radiological facilities.

ANDRAD started the activity to constitute the national **inventory of radioactive waste** in 2005. This activity continued in 2006-2007 within the PHARE project RO 5812.06.02 by selection and adaption of a radioactive waste tracking system (**WTS**) adequate to ANDRAD's necessities.



II.2.17. Radon and Eurocodes

Traian MAUNA, Andrieșica MAUNA

65 Polona, Buchrest – 1, Romania

tmauna@nuclearelectrica.ro, traianmauna@rdslink.ro

Radon, a heaviest noble gas is produced by radioactive decay of radium, which is widely distributed in soils and rocks, being a member of natural radioactive decay chains. The natural radioactivity are coming from all type of solid, liquid and gaseous substances as results of decay of any primordial heavy elements like uranium (U-238, U-235), thorium (Th-232) series, potassium (K-40) and others with no series. This is known as Naturally Occurring

Radioactive Material (NORM) and it is enriched by any applied technology coming to Technologically Enhanced Naturally Occurring Radioactive Material (TENORM). Radon is chemically inert, atomic gas alpha ray emitter with no other radiation type accompanied. Radon is easily dissolved by hydrocarbons and water can be as good transport vector. Easily diffused in air, radon can also concentrate in a closed air tight spaces, as cave or dwelling rooms. Alpha radiation emitted in inhaled radon by humans from air process have a destructive effect on the lung cells as well as radon decay daughter so the direct effect irradiate sensitive cells in the airways breathing, attack the DNA double helix thereby enhancing the risk of lung cancer. Many studies regarding radon from different countries in the last 3 decades demonstrate without any doubt that radon is guilty as first natural cause of lung cancer. Based on the set of radon studies by EPA sustained, the VI Committee of Biological Effects of Ionizing Radiation (BEIR) dedicate his report to radon, so giving a synthesis of collected data to all specialists involved in the health protection and radioactive risk assessment against radon aggression. After that IAEA Vienna included the radon into a Safety Report Series and radioprotection issue. In the last few years was began some actions in order to harmonized the national codes and standards by European codes. The set of building EuroCodes (EN1 to EN9) planned to adopt in unitary mode using the same symbols and methodologies including radon. To establish by EuroCode the radon intervention level in terms of concentration taking into account a wide range of their variation is not easy. The researchers have find maximum radon concentration reaching 4 MBq/m^3 in pores soil air, 110 kBq/m^3 in indoor air, 1.1 kBq/m^3 in outdoor air or 110 MBq/m^3 dissolved in underground water. By the other part no consensus are regarding the intervention level, by different country varying in range from 150 Bq/m^3 to 800 Bq/m^3 . These actions involve a lot of activities for all European countries in order to have a better knowledge about natural radioactivity, radon gas properties, behavior and correlation with construction design, erection work requirements, procedure, quality program, building materials and codes of practice,. Until now only the few countries have planed actions to modify all codes, regulations and standards in order to harmonize with EuroCodes. Unfortunately in Romania the consistent program regarding the radon mapping, information center, education or other actions connected to are not running or not in preparation. The ERRICA program of European Community tries to collect and process the data from every country and elaborate the recommendation and procedures for the future.



II.2.18.

The Waste Management Plan Integration into Decommissioning Plan to the VVR-s Research Reactor from ROMANIA

Gheorghe Barariu , Theodor Oprescu, Mihaela Filip, Sociu Florin,

National Authority for Nuclear Activity-Subsidiary of Technology and Engineering for Nuclear Projects, ROMANIA

The paper presents the progress of the Radioactive Waste Management Plan which accompanies the Decommissioning Plan for research reactor VVR-S located in Magurele, Ilfov, near Bucharest, Romania.

The new variant of the Decommissioning Plan was elaborated taking into account the IAEA recommendation concerning radioactive waste management. A new feasibility study for VVR-S decommissioning was also elaborated.

The preferred safe management strategy for radioactive wastes produced by reactor decommissioning is outlined. The strategy must account for reactor decommissioning, as well as rehabilitation of the existing Radioactive Waste Treatment Plant and the upgrade of the Radioactive Waste Disposal Facility at Baita-Bihor. Furthermore, the final rehabilitation of the laboratories and reusing of cleaned reactor building is envisaged.

An inventory of each type of radioactive waste is presented. The proposed waste management strategy is selected in accordance with the IAEA assistance.

Environmental concerns are a part of the radioactive waste management strategy.



II.2.19.

Radioactive Waste Management in Romania

Gheorghe Negut, Gheorghe Ionita, Ortenzia Niculae, Ion Durdun, Stela Diaconu

National Agency for Radioactive Waste (ANDRAD)

1, Campului Str., 115400 Mioveni, ROMANIA

e-mail address gheorghe.negut@andrad.ro, gheorghe.ionita@andrad.ro, ortenzia.niculae@andrad.ro, ion.durdun@andrad.ro, stela.diaconu@andrad.ro

In Romania, the radioactive waste results from nuclear industry and from the applications of the nuclear energy in research, medicine, industry and agriculture. The main producers of radioactive waste are:

- Nuclear Power Plant - Unit 1 & 2 of Cernavoda Nuclear Power Plant;
- Nuclear Research Reactors –VVRs IFIN-HH and TRIGA SCN-Pitesti;
- The Factory which produces nuclear fuel Nuclear Fuel Plant(FCN-Pitesti)
- Mining facilities and uranium processing facilities - The Uranium National Company;

- Hospitals which use applications of the radioisotopes in medical field (radiology, oncology);
- Classical industry, as a consequence of the industrial applications (the use of radioactive, sources in weld testing, leak detection, wall thickness measurement, etc.).

According to the Romanian legislation in force, the licensees who produce radioactive waste are responsible for the safe management of the radioactive waste up to the moment of disposal.

National Agency Radioactive Waste ANDRAD was created on the basis of the Governmental Ordinance No.11/2003 on the 28th of August 2004.

ANDRAD is responsible for the disposal of the radioactive waste and the spent nuclear fuel. In order to achieve this objective ANDRAD has to develop a lot of activities, defined in the Governmental Ordinance No. 11/2003 modified and completed in 2007.

The paper deals with the most important aspects of radioactive waste and spent nuclear fuel management, the ANDRAD responsibilities in this area. Last year was approved by the Governmental Ordinance the charge for waste produced nuclear power stations that will be used for nuclear waste management. There are finalized safety studies for LILW final repository and is in the process licensing procedures.

Key words: Radioactive Waste, ANDRAD.



II.2.20.

Main Issues for Developing the Saligny near-Surface Repository

O.NICULAE, G. GHERGHE, V. ANDREI and A.PETRESCU

ANDRAD – National Agency for Radioactive Waste, Mioveni, Arges, Romania,
 ortenzia.niculae@andrad.ro, geanina.gherghe@andrad.ro, veronica.andrei@andrad.ro,
 alin.petrescu@andrad.ro

Safe disposal of the radioactive waste represent one of the main objectives of nuclear policy and, in the same time, a theoretical and engineering challenge, both for national and international involved organizations. In a society which promote, develop, sustain and manage the safe nuclear installations, this purpose is achieved based on some interdependent and important elements, namely: adequate policy and legislative/regulatory framework, adequate administrative procedures, development of disposal structures, acting as barriers against the spread of radionuclides in the environment and demonstration of the safe evolution of the disposal structures.

For the low and intermediate radioactive wastes resulted from the Cernavoda Power Plant operation and decommissioning, the National Strategy established the development of a near surface repository, with multiple barriers, which will be located in the Cernavoda area (Saligny site). According the Romanian legal provisions, ANDRAD – Romanian National Agency for Radioactive Waste is in charge to develop and implement the new repository.

The paper presents an overview on planning of new repository, which is the main activity performed in order to identify the milestones and the main issues. The paper describes the milestones, the assumptions and some proposed corrective actions. Approval procedure of repository, as well as the identification and integration of requirements of CNCAN – Romanian Regulatory Body and all authorities responsible for legal approvals seems to be the main challenge in order to develop the new repository. It is necessary, also, to develop a specific database for repository, including the data collected during the early stage of Saligny site characterisation process, and to obtain new data necessary to complete the site characterization, and to support the safety assessment and performance assessments of repository.

At the same time, the basic elements of Saligny repository plan regarding the data collection, waste acceptance procedures and criteria, the waste matrix and other engineered barriers performance and degradation process, gas migration, repository impact on non-human biota, time periods of interest, etc. are described.



II.2.21.

Elaboration Methodology of Determination of Scaling Factors

S. Valeca and M. Valeca * I.V. Popescu and M. Preda **

* University of Pitești, Romania, www.upit.ro

** Institute for Nuclear Research Pitesti, Mioveni -Arges, Romania

By analyzing the technical literature, it was considered relevant the study realized at Energetic Power Research Institute of USA (EPRI) which demonstrate that the values of radionuclides concentrations which don't release gamma can be sometime correlated with the values of radionuclides concentrations which release gamma radiations, thus using the scalar factors, any radionuclide can be evaluated. This study has defined a correlation between transuranic radionuclides (TRU) and gamma release radionuclides, made by fission, for example, Ce-144.

Key words: (Scaling factors, transuranic radionuclides)

II.3. Air, Water and Soil Protection



II.3.1.

Modern Wastewater Treatment Methods: Sonochemical Processes

Iordache Ioan¹, Iordache Mihaela², Nichita Mircea Teodor³, Patularu Laurentiu¹

1. ICSI, 4 Uzinei Str., 240050 RM. Valcea, Romania

2. INCD-ECOIND, 1 Uzinei Str., 240050 RM. Valcea, Romania

3. Technical University "Gh. Asachi", Chemical Engineering Faculty, D. Mangeron 71A, Iasi

Sonochemistry - the application of ultrasound to chemical systems - is known to produce localised cavitations, acoustic streaming and other energetic phenomena. The enormous local temperatures and pressure together with extraordinary cooling rates and free radicals formation can provide a unique medium for leading chemical reactions under the extreme conditions. The researches related to physic-chemical effects of the ultrasound had known a powerful development in the last decade with important impact in different area of science: chemistry, medicine, environmental protection and others. Nevertheless, the physical, chemical and biological consequences of ultrasound irradiation as well as the mechanism of acoustical cavitations phenomena are still not completely elucidated.

Since the beginning of the investigation on sonochemistry processes a lot of works were appeared on ultrasounds treatment of wastewater. Now, thanks to the large number of practical application, ultrasounds become usual tools for wastewater processing in order to clean it from non-biodegradable pollutants.

It is already known that ultrasonic irradiation of aqueous solutions cause acoustic cavitations: the formation, growth and implosive collapse of bubbles. The cavitation generates sites of locally high temperatures and pressures for extremely short periods of time, which are responsible for unusual sonochemical effects. The mechanism proposed and generally accepted of the degradation of organic pollutants under sonication is based on the formation of short-lived radicals generated in violent cavitation events.

In this paper, the authors present some original results of the pollutants degradation by combination of ultrasound irradiation with some of the other classical and modern methods (oxidation with H₂O₂, O₂ and electrochemical oxidation). Hence, the purpose of this presentation is to emphasise the potential use of the combination of the ultrasonic action with some oxidative technologies of pollutants remediation.

Keywords: sonochemistry, sonoelectrochemistry, pollutants, wastewater treatment.



II.3.2.

Stable Isotope Signature (D/H, ¹⁸O/¹⁶O, ¹³C/¹²C): Important Element in Paleoclimate Reconstruction

Costinel Diana, Iordache Andreea, Vremera Raluca,

National R&D Institute of Cryogenics and Isotopic Technologies – ICSI Rm. Valcea, Research & Development Department – ICSI Rm. Valcea, Str. Uzinei, nr. 4 OP 4, CP 10, Rm. Valcea, cod 240050, Romania, e-mail: diana@icsi.ro

The method of stable isotope analysis is an important element in paleoclimate reconstructions. Stable isotopes help to distinguish between types of ice of different geneses. The stable water isotopes ($\delta^{18}\text{O}$ and δD) in precipitation are the basis for the reconstruction of paleotemperatures due to the empirical linear relationship between mean annual air temperature and the mean isotopic composition of precipitation. These values are used to reconstruct climate variations using different archives such as glacier ice and ground ice (e. g. ice wedges).

The present paper emphasize the isotopic analysis for D/H, ¹⁸O/¹⁶O, ¹³C/¹²C from glacier ice and ground ice using a new generation Isotope Ratio Mass Spectrometer in Continuous Flow (CF-IRMS) Delta V Plus, coupling with a three flexible preparation device (GasBench II, TC Elemental Analyser and GC-C/TC).



II.3.3.

Greenhouse Gases Effects on Global Warming

DIANA FLORESCU, GILI SAROS, ROXANA IONETE, VASILE STANCIU

National Institute R&D for Isotopic Separation and Cryogenic Technology – ICSI Rm. Valcea, Department of Research and Development – ICSI RM. VALCEA, Str. Uzinei, nr. 4, OP 4, CP 10, E-mail address for author: dianaf@icsi.ro

Monitoring of air quality must take an essential place in the environment monitoring system, atmosphere represent the proper environment to spread the pollutants.

In this paper, we present a theoretical study about atmospherically pollutants levels and the negative effects under global warming.



II.3.4.

Residual Pesticides Analysis in Wines By Gc/MS

M. Culea, A. Iordache, E. Culea, O. Cozar
University Babes-Bolyai, 1 Kogalniceanu, 400084 Cluj-Napoca, Romania
 e-mail: mculea@phys.ubbcluj.ro

The analysis of some pesticides in different Romanian wines by gas chromatography-mass spectrometry (GC-MS) was performed in scan and selected ion monitoring (SIM) modes. Solid phase extraction (SPE) and solid phase micro extraction (SPME) methods were used. The incorporated Rtx-5MS capillary column, 30mx 0.32mm, 0.25µm film thickness, was used in a temperature program from: 60 °C for 2 min, then increased to 140 °C with a rate of 12 °C /min, and then 6 °C /min to 280 °C kept 30 min. Helium flow rate was 1ml/min for compound separation and determination.. The transfer line was maintained at 270°C and the ion source at 250°C. Electron energy was 70eV and electron emission 300µA. Scan EI mode in the mass rage 30-550 Da was followed.

References:

- [1] M. Culea, L. Fromondi, C. Gherman, R. Podea, O. Cozar, Aroma compounds and antioxidants from red wine by GC/MS, *Studia*, 343-350, 2002.
- [2] M. Culea, M. Oros, O. Cozar, Red Wine characterization by GC/MS, *Analele Universităţii de vest din Timişoara*, vol. 44, 2004 p 125-128
- [3] M. Culea, O. Cozar, GC/MS characterization of wines, Proceeding "Agricultural and food sciences, processes and technologies", 2007, p..175-180.



II.3.5.

THMs Determiation by GC/MS

M. Culea, A. Iordache, E. Culea
University Babes-Bolyai, 1 Kogalniceanu, 400084 Cluj-Napoca, Romania
 e-mail: mculea@phys.ubbcluj.ro

Chlorinated by-products in swimming waters and indoor air were measured by gas chromatography–mass spectrometry (GC–MS) in the selected ion moniroting (SIM) mode. Trihalomethanes (THMs) have been determined by headspace extraction method in water and by collecting air samples in indoor air. The quantitative methods were validated to demonstrate the ability to identify and to measure reliably the yields of the toxic compounds. The separation chromatograms by SIM-GC-MS of the four THMs studied: chloroform, bromodichloromethane, dibromochloromethane and bromoform were characterized using a GC-MS system that incorporated a Rtx-5MS capillary column. The capillary column was a 30mx 0.32mm column, with 0.25µm film thickness used in a temperature program from: 31°C, 1min, 1°C/min to 33°C, 100°C/min to 200°C with helium flow rate of 1ml/min. The GC/MS interface line and the ion source were maintained to 250°C. Electron energy was 70eV and electron emission 300µA. Good validation parameters were obtained for each method: linearity, precision, accuracy and LOD.

References:

- [1] Monica Culea, Onuc Cozar, Dumitru Ristoiu, Methods validation for THMs determiation in drinking water, *JMS* 41, p.1594-1597.

III. SUSTAINABLE DEVELOPMENT

III.1. Strategies in Energy



III.1.1

Nuclear Electricity Generation - A Sustainable Energy Resource for Romania Along the next Two Decades

**Ioşif Prodea¹, Cristina Alice Mărgeanu¹, Corina Aioanei¹,
Ilie Prisecaru², Nicolae Dănilă²**

¹RAAN-Institute for Nuclear Research Piteşti

²Politehnica University of Bucarest

The main goal of the paper is to evaluate different electricity generation costs inside of the National Romanian energy sector along the next two decades. The IAEA (International Atomic Energy Agency)-MESSAGE code (**M**odel for **E**nergy **S**upply **S**trategy **A**lternatives and their **G**eneral **E**nvironmental **I**mpacts) will be used to accomplish these assessments. Due to the natural gas crisis started at the beginning of 2006, Romania has adopted a courageous energy policy based on nuclear electricity share increasing. Since then, the second nuclear Unit was commissioned at Cernavoda in 2007 and the other two CANDU-6 (700 MWe) were scheduled to be operational in 2015. On the other side the European integration of Romania asks for doubling the indigenous gas price during this year, 2008 and also for reducing the atmospheric gaseous emissions from the fossil fuel technologies. Therefore, we evaluated the economical competition between all electricity technologies in the Romanian energy sector in the next two decades for which our MESSAGE model was developed. We focused on coal, gas and, of course, nuclear electricity technologies. Some representative energy scenarios centered on nuclear share electricity growing were considered and MESSAGE results were analyzed from the energetic sustainability point of view. The results showed a low sensitivity on mined Uranium price for the nuclear electricity generation technology. This peculiar feature along with low greenhouse gases emissions and proven security along many decades persuaded us to consider the nuclear technology (CANDU type in our case) to be a sustainable energy resource for Romania at least in the next two decades.



III.1.2

Nuclear Power a Viable Alternative in Global Warming Context

Creţu Ileana, Bălan Ioşif Bogdan, Ionilă Maria, Petra Nicoleta Mihaela

Institute for Nuclear Research, POB 78, 115400 - Mioveni, Argeş County, Romania, e-mail:

ileana.cretu@nuclear.ro

Energy sources available in the world include: coal, oil, gas, biomass, nuclear, hydroelectric, wind, solar, refuse-based, and hydrogen. In addition, fusion had been originally proposed as the long-term source. Every form of energy generation has advantages and disadvantages.

Burning fuel for energy requirements represent approximate 88% from total emission of NO_x and CO₂, respective about 90% from SO₂ and about 72% from suspension powder evacuate into atmosphere.

The global warming represent a real threat and it is the most visible sign of climatic changes who take place in the entire world. To reduce the emission of greenhouse gases: Carbon dioxide (CO₂), Methane (CH₄), Nitrous oxide (N₂O), Hydrofluorocarbons (HFCs), Perfluorocarbons (PFCs), Sulphur hexafluoride (SF₆), in 1997 it was adopted "Kyoto Protocol to the United Nations framework convention on climate change".

According to this protocol European countries must reducing their overall emissions of greenhouse gases by at least 5% below 1990 levels in the commitment period 2008 to 2012.

In this context, because the natural resources for generation the fossil fuel are decreasing and their prices are rising, the nuclear power has become a real alternative for classical energy sources. It is indicated by:

- Fuel is inexpensive because the uranium is a very small part of nuclear power cost and uranium sediment is propagating on a large scale all over the world.

- No greenhouse emission or acid rain effects during a normal operation. Nuclear power is also named "clean energy".

- Wastes are more compact than any source of energy and were deposited in underground and secure storages. Nuclear energy has a number of advantages that warrant its use as one of the many methods of supplying an energy-demanding world. Even with conservation efforts, energy demand has been and will continue to increase. Using each and every one of these forms of energy production, we need to be sure we conserve the environment as much as we can, so we can leave resources for future generations.

This paper will demonstrate why the nuclear energy represents a response for world demanding energy and why it is considerate a "clean" source of energy.



III.1.3.

Perspectives and Constraints in the Romanian Energy

Rodin Traicu

*Autonomous Company for Nuclear Activities,
Drobeta Tr. Severin, Romania*

Romania, as well as the entire world, faces the major problem of the energy supply on short, medium and long term. In Europe, the energy demand for 2030 was estimated to be around 50% larger than today, while its own oil and gas resources will dramatically decrease, enhancing the dependence of import.

On the context of an increasing concern on the climate changes as result of the anthropogenic activities, the energy policies are more and more oriented toward an energy mix based on clean energy production and renewable resources, with an important contribution of the nuclear energy.

The energy policy of Romania, stated in the Energy Strategy for 2007 – 2020, is oriented to accomplish the safety of supply, the sustainable development and competitiveness.

The paper presents the current status of the energy sector in Romania, in terms of production/consumption, energy efficiency, available resources, and discusses the major problems of the energy system.

Based on the most probable evolution scenario of the Romanian society and economy, the paper shows up the predictions on the energy consumption until 2020, the contribution of each component of the energy mix (coal, oil and gas, hydro, nuclear and other renewable resources) and identifies the major constrains that Romanian energy system will encounter in the future European context.



III.1.4.

Generation-4 Challenges for Research

G. Cognet, C. Renault, F. Carré

CEA France

The rising awareness of a fast growing world primary energy demand at the beginning of the 21st century, together with considerations of energy security and concerns about greenhouse gas emissions led to acknowledge that all energy sources are to be used. In this context, whatever the scenario for future, the use of nuclear energy will take part of the energy mix in countries where it can be developed safely and competitively.

This energy strategy, called today nuclear renaissance, leads to anticipate an installed capacity of nuclear power of the order of 1000 to 1500 GWe by 2050, which is about four times the current installed capacity (370 GWe). Such a nuclear power capacity would require about 15 millions tons of natural uranium, if realized only with current or even generation-3 reactors which use only a small part of the uranium energy content (less than 2 %). This consideration, together with the accumulation of spent fuel, has incited scientists and research organizations to propose a new generation of reactors, generation-4, which would achieve a more efficient use of uranium and minimize the ultimate long lived radioactive waste.

Within this context, prospective studies carried out by the CEA and industrial partners led to elaborate for France a R&D strategy on future nuclear energy systems for the medium and the longer terms (> 2040), that aims at three complementary objectives:

- The development of fast neutron systems with a closed fuel cycle (sodium or gas cooled generation-4 reactors) for a sustainable energy supply through breeding in the long term, and for managing actinides from light water reactors spent fuel in the medium term;
- The development of key technologies for the nuclear production of hydrogen or the supply of high or very high temperature heat for the industry in close collaboration with industrial partners;
- Innovations for light water reactor fuels, systems and high conversion cores to further optimize LWRs until 4th generation fast neutron systems are mature for industrial deployment around 2040.

These objectives require breakthroughs with current reactor generations (2nd and 3rd). They pose real technological challenges for nuclear fuels, systems materials and technology, spent fuel treatment processes and non-conventional applications. Their application in recycling spent fuel for efficient use of the uranium and burning long-lived radioactive waste, leads to consider these 4th generation reactors as “nuclear systems” consisting of reactor, fuel and fuel cycle, optimised as a whole.

After reminding the stakes, the French R&D programme and associated scientific challenges will be addressed. The prospects opened by international cooperation (GIF, FP-7 and SNE-TP) will also be outlined.

III.2. Education, Continuous Formation and Knowledge Transfer



III.2.1.

Human Factor as Nuclear Safety Element

S.C. Valeca, M. Preda*, M. Valeca, D. Popescu, E. M. Ana***

Pitesti University, Pitesti, ROMANIA

**Institute for Nuclear Research Pitesti, ROMANIA, office@nuclear.ro*

***Nuclear Agency, Bucharest, ROMANIA*

National nuclear power system is based on western technology, covers almost 20% from national need and could be briefly described by:

- Safety and economic performances of Cernavoda NPP Unit 1,
- Reduced influence on environment, population and workers,
- Excellent ranking (place 4) between CANDU units from all over the world.

Also national nuclear power system plays an major role in Romanian power policy accomplishment:

- Energy safety and independence assurance,
- Decrease the green house effect gases production,
- Preserve the stability and adequacy of energy cost.

“Nuclear Safety” concept covers all the activities resulted from nuclear fuel cycle. By taking account of international experience, the related activities are estimated to last around 70 years in Romania:

- 10 years for site description and selection, design, manufacturing and commissioning activities,
- 40 years for Nuclear Power Plant operation, maintenance and modernization activities
- 20 years for preservation and decommissioning activities.

The above mentioned activities requires human resources, qualified and specialized in the following areas:

- research and development
- equipment design, manufacturing operation,
- components construction and assembly, operation and maintenance.



III.2.2

Nuclear Physics Education Today

N. Ghiordanescu, V. V. Grecu, A. Jipa

Physics Faculty, University of Bucharest

There are any problems concerning nuclear physics education today? Yes, there are!! A few of them will be presented here with our opinion about them. Different opinions can exist in other institutes.

The level of education must be changed. The young generation is less interested in sciences. They must be attracted to physics by a more concrete and interesting way of teaching it; probably a simpler way, less mathematics, more experimental and oriented to newer and accessible facts. In the same time the basic knowledge should not be affected. The “Bologna system” has reduced the general education to 3 years. In our opinion it is not sufficient time for a profound assimilation of the basic knowledge, and great care must be given to the content of general courses. We shall present the measures taken in the Physics Faculty, the continuous change of educational plan and of the content of general course. Comments on the master degree educational plan will be made,

In our days the influence of nuclear techniques, especially the nuclear energy, upon the society is more and more important. The society is not prepared to understand it properly. We consider that the educational system must provide means for a continuous information of a large public, including non-specialized authorities, on the most disputed problems. A few measures have been taken in our faculty concerning the radioprotection and radioactive pollution of environmental medium. A short presentation of the “Research Center of Radioprotection – Physics Faculty” will be made.



III.2.3.

CPSDN – IFIN-HH Involvement in the Nuclear Education

Camelia AVADANEI

Head, Nuclear Training Centre

National Institute of R&D for Physics and Nuclear Engineering “Horia Hulubei”

407 Atomistilor Street, Magurele City, Ilfov County, P.O. Box MG 6, 077125

Tel./Fax No. +40-21-457 42 79

E-mail address: cpscdn@ifin.nipne.ro

Nuclear Training Centre (CPSDN) activity consists in the qualification and/or specialization of personnel for the nuclear applications or related fields. CPSDN training is focused on the professional exposed workers using ionizing radiation sources, developing activities in the medical field, industry, customs, army, air force, post offices, education, research, agriculture, as well as activities related to the management of radioactive waste, uranium and thorium mining and milling, transportation of nuclear materials.

During time, CPSDN has developed various training curricula taking into account applicants training necessities, knowledge level, participants' background and specific requirements of their working place, as well as the national legal requirements. Each training programme is previously certified by the National Commission for Nuclear Activities Control, the Romanian regulatory body in the nuclear field and/or other Romanian dedicated bodies, such as State Inspection for the Control of Boilers, Pressured Vessels, Elevating Devices (ISCIR), Romanian Physicians College, etc., if necessary.

CPSDN activity is carried out by a team of specialists from the National Institute of R&D for Physics and Nuclear Engineering “Horia Hulubei” and other physics research institutes from Magurele Platform. For the practical aspects of training CPSDN is cooperating with specific laboratories adequately equipped.

CPSDN activity is carried out in compliance with a Quality Management System certified in accordance with EN ISO 9000:2000 by TÜV HESSEN through TÜV CERT (Technischen Überwachungs-Vereine) certification body.

In order to respond to the education market requirements, CPSDN envisages to develop and implement the e-Learning method for major radiation protection programmes. In this respect an important goal recently achieved was the development of Centre dedicated web site, <http://cpscdn.nipne.ro>.

Being aware of the actual tendencies consisting in the decreasing of youth interest for the nuclear field, CPSDN is interested not only in knowledge transfer, but also in knowledge management practices focused on knowledge preservation. In this regard, another important task is the development of a virtual library.

CPSDN is a qualified flexible training provider, ready to adapt its programmes curricula to the specific requirements of the applicants and to perform training both at its classrooms from Magurele and at the beneficiaries units Headquarters, if necessary.

As a result of Romania's recent accession to the European Union, CPSDN has to face new challenges in the field. In this context CPSDN is decided to become European competitive training provider for the nuclear field according to its EUTERP (European Training and Education in Radiation Protection Platform) and ENEN (European Nuclear Education Network) status. In this context CPSDN is open to any offer of cooperation both for national or international partnership in the field.



III.2.4.

Nuclear Education and Information Technology in the 21st Century

Marin Constantin

Institute for Nuclear Research

Campului 1, 115400 Mioveni, Arges

In the last two decades extensive changes in the information technology occurred. Amazing development of computer both in hardware and software, globalization of communication technologies have involved changes in educational tools, methodologies, curriculum, learning and teaching processes, interaction between persons.

Nuclear education is characterized by a long time formation consisting of graduate background and a period of 4-6 years of direct and specialized formation by community of practice, or other knowledge transfer methods. Nuclear education involves experimental and theoretical activities. For example in Reactor Physics the measurement of reactor rates, particle fluxes, currents, reactivity etc. introduces the necessity of dedicated trainings, experimental abilities and a good interaction with the member of experimental team. Both theoretical and experimental formation needs long time formation. For experimental part significant financial resources should be involved. Fortunately, important tools for simulation of the main phenomena were developed and may lead to an optimal process of education with reasonable funds.

Another important problem is the decreasing of the interest of society for nuclear education. Moreover, the problem is connected with the predictable loss of experienced personell in nuclear industry and research for the next 10 years, mainly by retirement. Countermeasures from the point of view of knowledge preservation and transfer are required. Obviously the harvesting of the tacit knowledge of the experts and converting it into a form that is available and useful should be the crucial point. Software applications development and dissemination should be encouraged by the management of the nuclear organization. At the same time in the building up of a new and younger staff process we need to fight with the idea

'balast is discharged', very present in the mind of the short term retiring experts. On the other hand a direct support for educational activities of the universities should be permanently encouraged.

The paper is intended to analyse the influence of the technology and globalization on the teaching techniques and pedagogy with a focus on Romanian aspects. Some aspects of teaching research connected with the actual changes (such as "what to study", "relevance", "methods and traditions", "where to start", "reference discipline" etc.) will be discussed. Also, the networking methods and the sustainable partnership creation in the Romanian context taking into account the harmonization of the Universities, Research Institutes, Nuclear Industry and other nuclear organizations will be analysed.



III.2.5.

The State of the Art of the Tacit Knowledge Transfer and Preservation in INR

V. Balaceanu, M. Constantin, M. Apostol

Email: victoria.balaceanu@nuclear.ro

The recent trends in the whole nuclear domain are based on the sustainable development, both for the nuclear power and the nuclear applications. Starting from this reality, it is necessary to admit that the success of all these applications of nuclear knowledge depends upon being sufficient competent and well-qualified personnel for their implementation and that depends in turn on knowledge management. The knowledge management has been most visibly introduced to the nuclear research and industry as a response to the aging nuclear domain workforce. In this context and without to minimize the importance of the explicit knowledge management, this paper intends to highlight the stringency that the nuclear tacit knowledge must be transferred and preserved and, also, appropriate methods and tools to perform it should be identified/selected/developed. The paper shows a detailed analysis of the state of the nuclear knowledge in INR. The starting point in this work were two investigations: a questionnaire-based investigation regarding the motivations of the INR retiring experts for knowledge transfer and preservation and an IAEA general questionnaire applied in INR. The necessity of a such analysis is motivated on the fact that in the next 5-10 years many experienced researchers should be retired and the INR staff must very quickly find the best methods and tools for an optimal transfer and preservation of the knowledge, mainly the tacit one because this is embedded in experts' minds. One of the most important conclusion of this analysis is that a short-medium term strategy must be developed, regarding the knowledge management, based on the following actions: **obtaining national funds for knowledge transfer and preservation**- as a partial support for the preparing of young graduate newcomers in the nuclear research and nuclear industry activities; this requirement is based on the importance of the activity for science and industrial applications; **intensify the collaborative processes**- this includes the implementation of rotation principle, community of practice and mentoring; **collaboration with other institutes/universities**- this area includes researchers collaboration, INR researchers teachers in universities, student internships in INR, using community of practice at interorganizational level (communication and knowledge sharing, preparing newcomers, learning/teaching collaboration); **invest in people and knowledge**- this area includes the assessment of future market needs, trainings, involving in exercises and experiments, participation in conferences/seminars (including the conduct of the events) combining the communities of practice with building the expertise by learning/teaching/coordination; **implement the best practices**- this consists in the development of informal networks of personal contact with NPP/universities/regulatory body/research institutes/etc. personnel.



III.2.6.

The Website – Necessity Or Whim?

Bogdan BALAN, Ileana CRETU, Maria IONILA, Mihaela PETRA, Silviu RUSU

Institute for Nuclear Research, POB 78, Postal Code 115400 - Mioveni, Arges County, Romania, e-mail: bogdan.balan@nuclear.ro

When asked „What do you know about the nuclear field?” an unformed person belonging to the civilian population would surely start by noticing: “I think it’s a dangerous field”

As the number of people having an Internet access increases every day, a valid and relatively cheap solution would be posting on the Internet informations from the nuclear field presented in a professional manner, written in an accessible language, in order to keep the population well-informed.

A site can be regarded as a way of informing people as well as a branding instrument, a means of technological transfer and marketing for the companies and firms working in this field. A site is the easiest way to keep in touch with the population and the clients. Through it, one can achieve many of the communication goals of the nuclear domain.

The present work aims at pointing out a few aspects related to web in the Romanian nuclear field and to the way it is used, it also provides an analyses of the Institute for Nuclear Research and suggestions for improving its activity.



III.2.7.

Embedded Learning Methods Implemented in Continuous Education and Knowledge Transfer of the Personnel Working in Radioactive Environment

Dumitru CHIRLEȘAN, Șerban Constantin VALECA,
Monica VALECA, Georgeta CHIRLEȘAN

University of Pitesti intends to develop an education platform which shall be able to offer professional formation services in the field of radio-analytical techniques. Course materials will be elaborated which will be structured considering various levels of professional competence.

The theoretical training materials will be embedded in software applications which will be offered as on-line courses. We will seek for the regular deployment of practical training sessions containing demonstrative exercises.



III.2.8.

Education and Public Information on Nuclear Power from a Student's Point of View

Students: **Ruxandra Cristina Toma**
Radu Cristian Arsene
George Cristian Nistor
Bogdan Florin Tiba

Professor **Ilie Prisecaru**
University Politehnica of Bucharest

Through this paper we try to answer the frequently asked question among the people related to the nuclear field on how to improve actual communication ways with the general public and how to prove them that nuclear power is a component of the sustainable development.

The paper is divided in two parts. The first one refers to the actual situation we meet world-wide, the lack of information concerning the nuclear field among population. The second part presents possible solutions which in our opinion could help people understand how nuclear works.

The complexity of this domain makes it more difficult to be understood by people who have no connection with it. We need to use all the possible means of communication to spread out our message.

There are many organizations which promote nuclear energy. Only the people involved are mainly interested, so these organizations don't achieve their goals entirely. Another fact is that people have lost their faith in nuclear power due to some incidents that occurred in the past and they became skeptics. Even if some countries like France and other occidental ones adopted the concept of "communication and total transparency" there are many left to embrace this concept. Most of the population is afraid of radiation as a result of not being given the opportunity to be informed.

These are the most encountered issues in promoting nuclear nowadays. To change this situation and promote this domain we made some suggestions like working with children, students, teachers, scientists and mass-media.

The acceptance of nuclear power may not be easy to achieve but with appropriate education and means of information we hope people will realize that nuclear is the key to their continuous need of energy, even if they don't see it yet.



III.2.9.

The Role of the Negative Selection Activities in Securing of Competence from an Organization Belonging to SEN Units

Boruga (Marian) Carmen
Romanian Authority for Nuclear Activities
Dr. Tr. Severin
Romania
carmenboruga@yahoo.com

The transformation and the accommodation of SEN units to a new management system, own to a decentralized economy and having financial support both native capital and foreign capital, in uncertain proportion at the moment and being in a continuous adjustment determined by the reasons and the motivations first of all political, generated conceptual changes in the so-called modern management theory.

The accommodation of this theory to the concrete situation from SEN units imposed the concept of negative selection. Is a concept which represents my desire to be created in an original contribution to a the contemporary management theory. Unlike the selection at the hiring (or positive selection), the negative selection is achieved on personal expense already employee, through activities of decrease of personal below different forms, on aside, but also redistribution of personnel under decrease of personnel, on the other side, this rearward side being complementary of the reorganization activity, in the context of an unitary activity based on the cybernetic principle of feedback. The negative selection, much more than the positive selection activity in this period in the SEN units (proceeded to a drastic decrease from 90.000 employee counted by the energetic system at the end of 9th decade of the past century, to a little over 30.000 which at

the moment work in the societies became self-governing), determines the specialist in human resources to have a judicious training and high competence, as far as what concern the knowledge of the standard system and also relating to the method of knowledge and valuation of behavior of the organizational employee.

Develop this subject in the content work on the strength of the documentation and the analysis of these documents but also to a set of studies and researches that were made during 1996-2006 in the SEN units.

I analyze during work methods and models of negative selection used in the SEN units. Will notice that they don't significantly differ from the methods used till 1989. They are still specific rationalist management, the forms of management own to centralized economics. They are, on the other side, a devoted mirror of the mode and the rhythm of evolution of the management in the SEN units. And an analysis in depth shall put in evidence and the appearance of innovative new methods according with the management specific informational revolution.



III.2.10.

Cernavoda NPP Training Programs

Cornelia Valache,

Senior Nuclear Engineer at Training Department Cernavoda NPP

The paper presents a general assessment of Cernavoda NPP personnel training programs, highlighting the role of training in human performance improvement.

Cernavoda NPP Personnel Training and Authorization Department (PTAD) is responsible for the training of CNE Cernavoda NPP personnel and its contractors.

PTAD is structured in such a manner in order to support and respond to all plant training, qualification and authorization requirements.

The training of personnel is continuously adapted based on IAEA Guides and INPO/WANO recommendations, to keep with world standards, based on the internal and external reviews.

At Cernavoda NPP the Training Concept and the Training Programs are based on SAT – Systematic Approach to Training.

The Training Concept is established on a set of training documents (RD's, SI's, IDP's), which address all the SAT phases: Analysis, Design, Development, Implementation and Evaluation.

The Training Programs are structured on the initial and continuing personnel training. Their content and goals are responding to the training specific needs for each plant major job family.

In order to successfully support NPP training programs, CNPP training center has upgraded classrooms with new presentation facilities and there are plans to expand the space of the building, to develop additional operator and maintenance skills facilities.

By responding in a timely and completely manner to all plant training requirements PTAD will help in rising human performance of Cernavoda NPP personnel, supporting the safe, efficient and cost effective production of power.

III.3. International Partnership for a Sustainable Development



III.3.1.

International Cooperation in the Field of Waste Management Activities at IFIN-HH

F. Dragolici, C. Turcanu, Gh. Rotarescu, I. Paunica, L. Ionascu, M. Nicu, Gh. Dogaru, C. Cristache, M. Toma, L. Done, G. Matei, R. Rotaru, D. Matei
 "Horia Hulubei" National Institute of R&D for Physics and Nuclear Engineering

In order to face the needs related to the waste management activities, in the last years, the Waste Management Department from IFIN-HH started a number of research and development international projects in some specific areas: up-grading of the infrastructure, development of new technologies and researches on the conditioning matrices, engineering barriers, others.

The implementation of these projects will significantly improve the safety of radioactive waste treatment and conditioning in Romania, assure the treatment capabilities for decommissioning of nuclear installation and will increase the public acceptance and confidence in the national nuclear power program.



III.3.2.

NULIFE - the European NoE "Nuclear Plant Life Prediction"

Mihail COJAN

Institutul de Cercetări Nucleare – Pitesti, ROMÂNIA

E-mail: mihail.cojan@gmail.com

ICN Pitesti has become on the 29th September 2006 a partner in the European **Network of Excellence Nuclear Plant Life Prediction (NULIFE)** coordinated by Technical Research Centre of Finland (VTT) <http://nulife.vtt.fi> [1]. The vision of this NoE is to create a Virtual Institute with: *An integrated RTD platform embracing all European stakeholders within a completely new structure with improved and efficient use of public and private RTD funding; Sustainable forum for realising harmonised technical procedures giving impact for Nuclear energy industry, National regulators and European Regulatory Working Groups; and Service provider and sustainable source of qualified expertise for all customers in Nuclear energy field.*

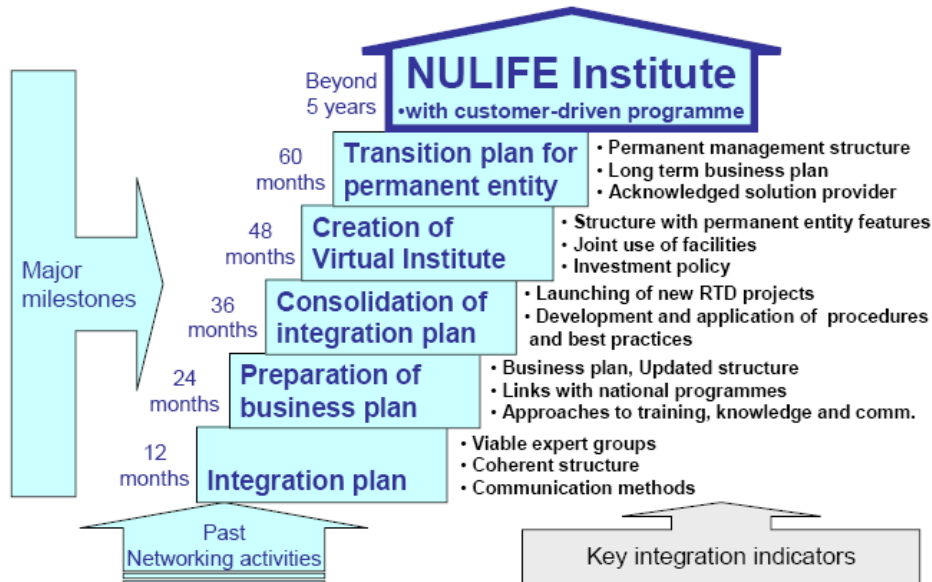


Figure. 1 The major milestones in evolution of the integration towards achieving the vision

ICN Pitesti is involved in **NULIFE** for the following Work Packages:

- **WP IA-1 "Mapping of partner RTD expertise and competences"** to provide the ICN expertise and competences in application of ageing management in CANDU 6 PLIM / PLEX programs;
- **WP IA-2-3 "Lifetime evaluation"** to provide advice on/develop the lifetime evaluation tools used by the ICN and to provide benchmarking of specific lifetime evaluation tools (experimental and analytical procedures, evaluation criteria);
- **WP IA-2-4 "Safety, risk information and reliability"** to review the practical applicability of various probabilistic methods for assessment of structural reliability, ageing and residual life of NPP components, identify R&D needs in this area and assess the limitations of structural component modeling in PSA and their importance for risk-informed decision making.

NULIFE will be the future focal point and umbrella for ICN R&D activities in support to CANDU 6 Lifetime Management Program [2].

References:

[1] I. Aho-Mantila, R. Rintamäa, L. Heikinheimo, N. Taylor, "European research network aiming at harmonized nuclear plant life prediction procedures", IAEA-CN-155/001, Second International Symposium on Nuclear Power Plant Life Management, Shanghai, China, 15-18 October 2007.

[2] M. COJAN, GH. FLORESCU, M. ROTH, I. PIRVAN, D. LUCAN, "Considerations related to CANDU 6 lifetime management", IAEA-CN-155/035, Second International Symposium on Nuclear Power Plant Life Management, Shanghai, China, 15-18 October 2007.



III.3.3.

The Euratom Programme - a Strategy for the Future

S. Webster

DG-Research

European Commission

The year 2007 marked the 50th anniversary of the original signing of the Euratom Treaty and also saw the start of the 7th Euratom Framework Programme (FP7), representing an important continuity in Community-funded research in the field of nuclear science and technology. The Euratom Treaty states that "in order to perform its task, the Community shall, amongst other things, promote research and ensure the dissemination of technical information...". As in previous programmes, Euratom FP7 (2007-11) shares its budget between fusion energy research (about 70%) and fission-related activities (30%), the latter being split between the programme of indirect actions in "nuclear fission and radiation protection" managed by DG-Research and the JRC's programme of direct actions. The research priorities of the "fission" indirect programme are radioactive waste management, including both geological disposal and partitioning and transmutation, nuclear installation safety, advanced nuclear systems, radiation protection (in particular the effects of low doses), and related support to infrastructure and education and training. The first FP7 contracts were signed in December 2007.

Last year also saw the launch in September of the first European Technology Platform in nuclear fission. The Sustainable Nuclear Energy Technology Platform (SNE-TP; all details on www.snetp.eu) is a major initiative to foster effective integration and cooperation between leading players sharing a common vision for the future of nuclear research, and marks the natural progression in European collaboration beyond the integrative effect of the major FP6 projects at a time when nuclear technology is increasingly seen as a key component of the response to the world's energy problems. SNE-TP is structured around three main pillars: continued safe operation of current LWRs; very high temperature reactors for the production of process heat for use in hydrogen production and other industrial applications; increased sustainability of nuclear technology through operation of fast reactors and full actinide recycling. The platform also covers key cross-cutting issues such as research infrastructures and human resources and knowledge management. Now that the platform's governance structure is in place, and the various working groups established, the first key milestone will be the adoption of an agreed "strategic research agenda" later this year. Platform partners will then implement the needed R&D activities by committing their own resources to the platform's operations, with the Euratom FP also contributing via the normal calls for proposals.

International cooperation will also play an important part in the development of a new generation of nuclear technology, and the Euratom membership of the Generation-IV International Forum (GIF), together with the contribution to the GIF activities being made through the FP6 and FP7 projects, are clear indications of the determination of Europe to play its rightful role in this process.

On the policy side, 2007 also saw in January the publication of the Commission's long-awaited "energy package" – collectively known under the banner "Energy for a Changing World" – framing the debate over an effective energy policy for Europe and the measures needed to counter the increasingly urgent problems of climate change, security of supply and competitiveness. Nuclear energy was clearly seen as part of this process, and two of the documents made public on 10th January 2007 were of particular importance in this regard. One was the first PINC (Community illustrative nuclear programme) to be published since 1997, and the other was a communication on the planned Strategic Energy Technology Plan (SET-plan). Following endorsement of this strategy at the spring Council Summit in March 2007, at which the EU Member States also signed up to ambitious 20% CO₂ reduction targets by the 2020, the SET-plan strategy was further refined during 2007 in a broad stakeholder consultation exercise and formally adopted by the Commission in its communication to the Council and European Parliament of 22 November, which was subsequently endorsed by Member States at the 2008 March Summit.

The SET-Plan covers the R&D strategy for a range of low carbon energy technologies, the aim being to accelerate innovation and push EU industry to exploit the opportunities, thereby enabling the EU to reach its ambitious 2020 and 2050 targets. At the core of the SET-Plan strategy is a limited number of European Industrial Initiatives in key technologies considered essential to ensure these targets are achieved. Crucially, one of these initiatives is in the area of Generation-IV fission technology. This is a real opportunity for the EU nuclear research and industrial sector, and though details of financing and timing are yet to be clarified, it is clear that this initiative will be closely integrated with the R&D

effort being coordinated by SNE-TP, which has already, in its vision document (available on www.snetp.eu), set out a tentative schedule for research and eventual deployment of Generation-IV systems.

However, Generation-IV technology cannot make any impact over the 2020 timeframe, and for this reason the SET-Plan also covers competitiveness of current nuclear technology and the importance of implementing long-term waste management solutions, which are considered to have a shorter-term impact. This is a reflection of long-standing concerns in EU Member States on safety and waste issues, clearly stated in the conclusions to the March 2007 European Council that mentioned the need to continue R&D on waste management and signalled the creation of a High-Level Group of senior regulators covering nuclear safety and waste management regulatory issues. This HLG was established by the Commission's DG-Energy & Transport in the autumn of 2007. Another initiative stemming from the March 2007 European Council is the European Nuclear Energy Forum (ENEF), subsequently launched by DG-Energy & Transport in November 2007. The ENEF is hosted alternatively by Slovakia and the Czech Republic in the cities of Bratislava and Prague, and is a broad stakeholder forum looking at all issues related to opportunities and risks of nuclear energy. These shorter-term issues are also being addressed by the Euratom research effort, for which nuclear safety and radioactive waste management have been priority areas for many years and which remain so in FP7.

One thing is clear: political and societal acceptance of any nuclear renaissance must go hand in hand with an integrated, effective, well-funded and long-term European research effort. The Euratom programme is being reoriented to the future needs and boundary conditions, and will ensure to the extent possible that the strategy for continued Community support in this crucial area extracts the maximum added value for Europe's citizens.



III.3.4.

ICN's Partnerships in Nuclear Research

Daniela Diaconu

*Institute for Nuclear Research - Pitesti
Campului 1, Mioveni 115400, Romania
daniela.diaconu@nuclear.ro*

Nuclear energy is seen by more and more states of the world as a viable solution of the energy demand increase on medium and long term. Europe talks today about the renaissance of the nuclear energy and considers it as a component of the sustainable development. The expected reactor systems Generation IV, the geological disposal, the continuous increase of the nuclear safety are challenges that request a better focus of the human efforts and funds. The way is the international cooperation.

Framework Programmes of the European Commission, Coordinated Research Program of IAEA, and Generation IV International Forum (GIF) are only few examples of large partnerships in which the participants are working together for common objectives, avoiding research fragmentation and duplication, increasing competitiveness and favoring the transfer knowledge in nuclear energy field.

Aware on the role of the international cooperation in maintaining a high scientific level of its personnel, in building and promoting the researchers competence and in contributing to the most important problems of the nuclear energy, the Institute for Nuclear Research - Pitesti increased continuously the participation in international research projects and extended the bilateral cooperation with important nuclear research centers in the world: AECL – Canada, DoE – USA, KAERI – Korea and recently CEA - France.

The paper presents the most important contributions of the ICN scientists in the FP6 and FP7 projects, the main achievements in the CRP projects and illustrates the most successful cooperation activities developed under the Sister laboratory Program of DoE.



Institute for Nuclear Research Pitești

Câmpului Street, 1, 115400, Mioveni

România

Phone: +40 248 213 400; fax: +40 248 262 449

www.nuclear.ro