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III.1 EDUCATION, TRAINING AND KNOWLEDGE MANAGEMENT
AN INFORMATICS SYSTEM FOR TRAINING, EXAMINATION AND KNOWLEDGE EVALUATION OF THE FHS PERSONNEL

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ABSTRACT

The paper presents the way to implement a Fuel Handling System (FHS) data base in order to carry out an informatic system for training, examination and knowledge evaluation. The sessions, are organized as "ebooks" represent a way of modern learning and thoroughness, examination and assessment of the professional knowledge. The use of these lessons for personnel training, working in the FHS area, leads both to the increase of the learning quality and reduction of the time for studying activities. The student is getting the advanced professional knowledge regarding the technological equipment operation by graduating the session. This e-learning system is designed and used to keep and develop, in time, deep in knowledge, about Fuelling Machine Head construction and working, for F/M Test Rig operators and technicians, from INR Piteşti. The e-lessons for F/M snout clamp, magazine and separators have been already implemented, the rest of materials in data base is following.

Key words: eLearning, SIPEC, CANDU, fuel handling, knowledge

Introduction

Education using the computers, named eLearning, has had an explosive development during the last years, and has become one of the integrated components in the educational strategies in most of companies and universities. The term eLearning is associated to all the programs and materials, distributed to beneficiaries in electronically form, that are used in a learning process, instruction or education.

Was initiated and developed an Informatics System for Perfecting and Evaluating the Professional Knowledge - SIPEC Pro (fig. 1), in order to consolidate the professional knowledge of the implied personnel in operation activities and exploitation of the test rig, so they can be able to test the F/M Head (fig. 2) for the CANDU reactor. The SIPEC Pro is currently in the development stage.
The courses in electronic format (eBook) that are found in the SIPEC, contain information and knowledge about the construction and function of the Fuel Handling System’s equipment, the adjustment, calibration and maintenance operations, all collected in a dedicated data base [1], [2].

Until now the data base was completed with information regarding the F/M Head, its hydraulic driving system, as well as the new and spent fuel transfer equipment. In the future the data base is going to be completed with the information regarding the testing of the F/M Head, and of the spare Rams.

There were implemented electronic courses for the coupling device, the F/M Head storage and separators; in the future it will be realized for the rest of the materials existing in the data base.

The F/M Heads are mechanical robotics, with hydraulic driving, from the Fuel Handling System, which has the following functions in the nuclear plant:

- Takes the nuclear fuel in its own magazine and transports the fuel between the new fuel port and reactor, in order to supply the reactor;
- After positioning in front of the reactor, at the upstream end-fitting of the fuel channel, it checks the correct alignment and the channel sealed coupling;
- Opens the fuel channel, which has a 23 kg/s heavy water flow, a 110 bars pressure and 307°C temperature and transfers the new fuel to the channel;
- Closes the fuel channel and leaves the reactor room towards the maintenance room, where it stays until the next fuel charging operation.

Simultaneous with this operation, at the downstream end-fitting of the same fuel channel, another F/M Head takes over the spent fuel, according to the following operations:
  - After positioning and moving in front of the reactor, it checks the correct alignment at the downstream end-fitting of the fuel channel, subject to the charging / discharging operation, and executes the sealed coupling with the fuel channel;
  - Opens the fuel channel and takes over the spent fuel inside its own storage;
  - Closes the fuel channel and leaves the reactor room towards the spent fuel port where it will transfeere the whole fuel charging;
  - It moves toward the maintenance room, where it stays until the next fuel discharging operation.

Next it will be presented examples of the electronic courses for different F/M Head subassemblies.

**DESCRIPTION**

F/M Head includes the mechanisms necessary (fig. 3) for fuel bundle manipulation, as it follows:

- The coupling device assembly is a mechanism that blocks the fuel channel terminal fitting and seals at pressure the F/M Head on terminal fitting;
- The F/M Separators assembly realizes the fuel bundle separation from the bundles column, and so facilitates their stocking in the F/M Head storage;
- The F/M Magazine is an assembly that shelters the fuel bundles that are to be inserted in the channel or for the ones removed from the channel, as well as for the protection plugs, channel shutting plugs, bundles guiding sleeve, Ram adaptor and FARE tool;
- Ram pushers, or the telescopic cylinders, is an assembly used to remove or to put in place the plugs, installs or retreats the bundles guiding sleeve and manipulates the fuel bundles.

![Fig.3 F/M Head subassemblies](image)

The fuel swooping operations must be possible when the CANDU reactor is at nominal power or when is shut down, in any possible temperature or pressure range. The fuel is always charged in the reactor in the coolant flow direction. There is also a possibility that the column bundles can be moved with 2 to 4 cm between the protection plugs and the two heads of the pressure tube, in order to compensate the column thermic dilatation.
During the reactor operation, the bundles are moved from the core of the reactor only by the fluid flow action, under the action of the hydrodynamics forces exerted above it. Because the flow through the fuel channel varies according to its position, for the channels in the peripheral zone on which the flow is reduced, it is necessary using of the FARE device (Flow Assist Ram Extension).

This device isn’t necessary for the tubes in the central zone of the reactor core, here being enough the FAF method (Flow Assist Fuelling). The FARE device is charged at the end of the bundle column, after the fresh fuel bundle had been inserted in the upstream machine, and is removed after the bundle column had had reached on position.

**F/M Snout Clamp** - it is positioned in the front of the F/M Head storage carcass, forming an extension of the pressure zone between the storage carcass and the terminal fitting of the fuel channel, also containing the mechanisms that allows the F/M Head to couple on the terminal fitting, to unlock the channel closure and to assure the air-tight coupling. Its functions are: the aliening of the F/M Head with the terminal fitting and maintaining an air-tight without losses between the machine head and the terminal fitting.

![Fig.4 – F/M Head Coupling device](image-url)
**Fig. 5** – *F/M Head Coupling. Components*

**F/M Magazine** – is an equipment central positioned, between the snout clamp and the pushers, used for new or spent fuel bundle deposition or transportation toward or from the reactor, and the storage of different tools and devices necessary in the fuel changing operations.

**Fig. 6** – *F/M Head storage*

The main magazine subassemblies (fig. 6 & 7) are:
- Rotor assembly, that has 12 tubes with different destinations;
- Action—indexing mechanism, that assures the correct positioning of the tubes;
- Breakdown action mechanism, for manual auctioning of the F/M Head Magazine.

![Diagram of F/M Head storage](image)

**Fig. 7 – F/M Head storage. Constructive description**

**F/M Separators** – two assemblies of the F/M Head (fig. 8), mounted behind the coupling device, that:
- Feels the position of the gap between two fuel bundles, between a fuel bundle and the biological protection plug or between the fuel bundle and the Ram C adaptor;
- Pushes the bundles in the storage and prevents the axial movement of the fuel column.

![Diagram of F/M Head separators assembly](image)

**Fig. 8 – F/M Head separators assembly**

The main subassemblies of the separators (fig. 9) are:
- Sliding assembly – constituted by a palpator and a pusher (or a retractor) and two sides stoppers, each one being put in function by a piston;
- Potentiometers and magnetic microswitchers assembly that seizes the separators position.
The separators are in rest in normal function position (fig. 10).

It is mentionable that at the end of each electronic course a questionnaire appears from the presented material, named “quiz” (fig. 11) its structure has been conceived so the complexity grade can determine different grades of difficulties.
Results

In this paper it is presented the execution mode for the courses in electronic format, parts of the Informatics System for Perfecting and Evaluating the Professional Knowledge (SIPEC Pro) of the FHS personnel, with the help of the Toolbook Instructor program, using a data base with information and knowledge about the FHS function.

Conclusion

Next to the practical activities conducted in the technological installations, this courses system contributes to the maintaining and developing of the human resources used in testing and operating of the F/M Heads at the RATEN/ICN.

References

[2] F/M Head. Test requirements – code 2-3-35000-460A
RADIOACTIVE WASTE RESULTED IN $^{99}$MO FISSION PRODUCTION

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ABSTRACT

Management of radioactive waste is an integral and very important part of $^{99}$Mo production, especially for fission $^{99}$Mo, and should be given high priority during production process development and operation. To ensure a continued reliable and cost-effective supply of $^{99}$Mo, the management of associated waste must be carried out in accordance with internationally accepted safety criteria and related national regulations.

The waste products form and characteristics will be dependent on the process design and chemistry. Typically, vital aspect of the Fission Production Mo processing is the holding of waste streams at the site of production for initial decay of the shorter lived fission products. This usually follows a sequence of in-cell storage, storage adjacent to the hot cell operations, and transfer to longer term storage or processing. Adequate capacity must be allowed for this initial waste storage at the hot cell area.

Key words: molybdenum-99, fission molybdenum, molybdenum waste

Introduction

Molybdenum-99 ($^{99}$Mo) and its daughter product $^{99m}$Tc are the most important radionuclides used in nuclear medicine practice. The importance of these radionuclides stems from the fact that $^{99m}$Tc is used in about 80% of all diagnostic nuclear medicine procedures. $^{99}$Mo production for $^{99m}$Tc generators gives rise to a number of waste streams, some of which need special care in handling. Radioactive waste from the production of $^{99}$Mo must be managed in such a way that the protection of people and the environment is ensured, both now and in the future.

$^{99}$Mo is produced mainly by nuclear fission of $^{235}$U. Processing of irradiated uranium targets for $^{99}$Mo production generates considerable amounts of radioactive wastes containing fission products as well as small quantities of transuranium elements. Small amounts of Mo are also produced by neutron activation of $^{98}$Mo targets. However, this method is not yet fully developed for medium or large scale production.

The nuclear and radioactive waste management is based on well-established safety standards for the management of radioactive waste. Romanian legislation and regulations are all developed from internationally agreed standards, guidelines and recommendations in order to ensure the protection of public and the environment.
Molybdenum production by separating out of the fission products resulted from the fission of uranium involves four stages [1].

First stage is the *manufacturing of the target* to be irradiated. The target used in the acid dissolution method is a uranium foil, and for the alkaline process is a uranium plate.

The second stage is the *irradiation of the target* in the reactor; after a cooling period the irradiated target is transferred to a post irradiation facility for further processing.

The third stage involves the *chemical processing of the target*. This stage implies the dissolution of the target in acid or alkaline solution. The waste resulted depends on the process.

Lastly, the fourth stage deals with the *management of the radioactive waste*. All the materials used for the separation of $^{99}$Mo become radioactive waste. Although some of these materials are high level waste (HLW), most of them are intermediate level waste (ILW).

**Waste characteristics**

Production of $^{99}$Mo by means of neutron induced fission of $^{235}$U (as HEU or LEU) is operated in some countries on an industrial scale. Although this production method supplies nearly the whole worldwide demand for $^{99}$Mo, it also generates the most radioactive waste of all production methods [2]. This arises from the fact that a wide range of nuclides are formed during $^{235}$U fission.

The waste products form and characteristics will be dependent on the process design and chemistry. The waste streams to be handled are summarized below:

- **Solids** from processing, such as precipitates or cans that held targets.
- **Liquids** from processing are common to all Mo-99 production processes. These are usually intermediate level waste at time of production and often remain so for long time periods due to the fission product content. There is more than one stream of liquid waste from target processing and Mo-99 purification and these may be of similar or differing chemistry. These streams will contain different major nuclides or different concentrations of the same nuclides.
- **General cell solid** waste must be considered and incorporated into the “existing” ILSW (Intermediate Level Solid Waste) handling system. In-cell consumables such as tubing, columns and wipes, fittings, etc, are regularly replaced and are accumulated for batch-wise removal and shielded transfer to a shielded storage facility.
- **Gaseous** waste arises from all forms of processing and must be handled effectively for safety and regulatory reasons. For FPMo (Fission Product Mo-99) the iodine and the noble gases in the streams must be held for sufficient decay or treated before gases can be released to the extract system and stack.
- **Low level solid** waste will be produced by hot cell processing (at the rear of cell operations and transfers, and for any decontamination activities). This must be incorporated in the existing site low level operating system.
- **Low level liquid** waste is produced from decontamination and other activities at rear of cells and also must be planned for, and incorporated into the existing site system by either a batch transfer system or a reticulation system.

Vital aspect of the FPMo processing is the holding of waste streams at the site of production for initial decay of the shorter lived fission products. This usually follows a sequence of in-cell storage, storage adjacent to the hot cell operations, and transfer to longer term storage or processing. Adequate capacity must be allowed for this initial waste storage at the hot cell area.
Other aspects to consider for the FPMo route are Safeguards and the tracking and security of the uranium throughout the process. The uranium deports to different waste product streams depending on whether acid processing or alkaline processing is used for target dissolution.

**FPMo acid processing route**

The acid route in obtaining FPMo uses a LEU foil target. The irradiated target is dissolved in nitric acid. During the dissolution of the LEU foil fission gasses the rest of the gasses resulted (NOx) are drained from the dissolver using a cold finger gas trap. By this method the gasses are treated as solid waste.

On further are presented the wastes resulted during the acid processing route in obtaining FPMo:

- **Liquid waste.** A primary liquid waste results from the dissolution of the targets in nitric acid (or other acid mixtures) after the extraction of the Mo-99. Due to the total dissolution of the target this results in the major proportion of the mixed fission products reporting to the liquid waste. This is a longer lived waste, dominated over the long term by Cs-137 and Sr-90. In the medium term Ce-144, Ru-106/Rh-106, Zr-95, Nb-95 also contribute significantly to the activity. This is accumulated in cell on a filling and hold for decay basis before removal from the cell. This waste will required further storage, either at location or in a separate area and may be combined in larger tanks, subject to criticality assessment and certification.
- **Liquid waste.** A secondary liquid waste results from the purification steps on the original Mo-99 adsorption column. This may be similar in chemistry to the primary solution but more dilute in most species. This is similarly dominated over the longer term by the Cs-137 and Sr-90.
- **Solid waste** from target cladding or enclosure. This may be aluminum or stainless steel and of high initial activity and are accumulated in cell until a combined batch is removed as intermediate level solid waste.
- **Solid waste** from hot cell operations. This includes other cell waste, or consumables, such as adsorption columns, tubing and other redundant equipment. This is also accumulated in cell and removed on a campaign basis with the other ILSW. Both these intermediate level solid wastes are contaminated with mixed fission products and must be transferred in shielded containers to shielded storage.
- **Other in cell liquid wastes** that are produced in smaller volume. These have been accumulated at one operation, transferred separately in a shielded flask and accumulated in a storage facility. Due to the lower activity and shorter lived nuclide content this waste can be processed and disposed of through the low level liquid waste treatment system after a moderate storage time.
- **Gaseous waste** is generated during the target dissolution and must be treated and/or stored before HEPA and charcoal filtration before passing to the stack within the regulatory requirements. This does lead to other wastes which periodically have to be removed, stored and processed. The fission gas traps (treated charcoal) the HEPA filters and charcoal filters all contribute to waste.
- **Low level solid waste** from rear of cells activities (decontamination from items or transfer operations).
- **Low level liquid waste** from rear of cells decontamination activities. Since this liquid waste will contain mixed fission products it may or may not be suitable for direct treatment in the effluent plant. A pretreatment may be required.

Waste Facilities required at FPMo Processing Plant that uses the acid processing route.

- Heavily shielded Hot Cells for processing targets and initial storage of solid and liquid wastes. Hot cells fitted with master-slave manipulators and with extract ventilation through HEPA and charcoal filtration. Cells to have means of transferring waste material to shielded flasks or containers, or shielded transfer mechanism to storage facilities adjacent or near to the hot cells.
- Overhead crane and/or other mechanism to move and accurately locate heavily-shielded flasks to allow transfers of waste.
- In cell storage tanks for accumulation of liquid wastes (possibly with segregation of types of liquid waste). To have sufficient capacity to hold waste for sufficient time to allow significant decay of shorter lived nuclides.
- In cell storage ability to hold solid waste for removal on a (full) batch basis.
• Additional storage and/or processing capacity to transform liquid waste into a solid product in shielded facilities suitable for long term storage or disposal.
• Safeguards monitoring of this waste since this contains the uranium from the targets.

**FPMo Alkaline Processing Route**

The alkaline processing route uses a LEUAlx plate target. The general waste regimes are as follows:
• **Uranium precipitate.** For alkaline dissolution the uranium remains as a precipitate in a filter at the base of the dissolver. This dissolution technique results in a distribution of the fission products between the solution and the precipitate. The filters containing the filter cake are removed from the dissolver. There are some variations on locally dealing with this material from capping and interim storage in cell, from removal and transfer to a different vessel, to redissolving and reprecipitating in a different filter for subsequent encapsulation. No producer worldwide is currently reprocessing this precipitate for uranium recovery whereas several producers are multiple encapsulating (welded sealing) for longer interim storage in shielded facilities. The longer lived isotope Sr-90 reports to the uranium precipitate.
• **Primary intermediate level liquid waste.** This waste results from the dissolution liquid after removal of the Mo-99. The liquid contains the Cs-137 and therefore remains as intermediate level. Due to the short lived nuclides also present it is necessary to store this in-cell or beneath cell (underground) to allow sufficient decay before shielded transfer (or transport) to further consolidated storage or processing to a solid product.
• **Secondary level liquid waste.** This may be alkaline or can be acid depending on the Mo-99 purification flowsheet. Often it may be a number of streams which are combined for storage. This liquid waste will initially be classified as intermediate level but has potential to decay to the low level category in a reasonable period (years). Similarly, this waste is accumulated at the site of production to allow initial decay followed by transfer to accumulated storage for further decay and processing to solid.
• **Solid cell waste** which is collected in cell(s) and removed by the shielded ILSW transfer system to the shielded ILSW storage system. This waste includes the columns, tubing, fittings and other general cell waste.
• **Gaseous waste.** There is a requirement to retain the off gas from the dissolution process to allow for sufficient decay of the noble gases before release through the cell extraction HEPA filtration and carbon adsorption systems.
• **Low level solid waste** from rear of cells activities (as acid route)
• **Low level liquid waste** from rear of cells decontamination activities. Since this liquid waste will contain mixed fission products it may or may not be suitable for direct treatment in the effluent plant. A pretreatment may be required.

**Waste Facilities required at FPMo Processing Plant that uses the alkaline processing route.**
• Heavily shielded Hot cells for processing targets and initial storage of solid and liquid wastes. Hot cells fitted with master-slave manipulators and with extract ventilation through HEPA and charcoal filtration. Cells have to have means of transferring waste material to shielded flasks or containers adjacent or near to the hot cells.
• Overhead crane and/or other mechanism to move and accurately locate heavily-shielded flasks to allow transfers of waste.
• In cell storage tanks for liquid wastes (with segregation of types of liquid waste). To have sufficient capacity to hold waste for sufficient time to allow significant decay of shorter lived nuclides.
• In cell storage to hold uranium precipitate in suitable containment to allow decay of shorter lived nuclides and ability to transfer material to further processing (encapsulation) without spread of contamination. Safeguards monitoring of this waste since this contains the uranium from the targets.
• In cell storage ability to hold general solid waste for removal on a (full) batch basis.
• Additional storage and/or processing capacity to transform liquid waste into a solid product in shielded facilities suitable for long term storage or disposal.
Conclusion

Radioactive waste from the production of Mo, which is the most important radionuclide for medical application, must be managed in such a way that the protection of people and the environment is ensured, now and in the future.

The volumes and types of radioactive wastes generated by $^{99}$Mo production are largely dependent on the production method. Most methods used for $^{99}$Mo production are based on nuclear fission of targets containing $^{235}$U. Processing of these targets generates considerable amounts of radioactive wastes containing long lived fission products and $\alpha$-emitters. The presence of fission products and actinides in solid and liquid radioactive wastes complicates all stages of waste management. The treatment and conditioning of fission $^{99}$Mo production wastes require more progressive technology and equipment, hot cells remote handling, shielded waste packages, and capacity for interim storage. The waste will be generated as solids, liquids, and/or gases, and will include material in the low, intermediate and even high level radioactive categories. Initial treatment of waste streams is usually required at the production site, prior to short or long term storage. The treatment required is dictated by both the form of the waste and its activity level. Final disposal of waste should be considered right from the beginning of process development.

References


INVESTIGATION OF DELAYED HYDRIDE CRACKING MECHANISM IN THE CANDU PRESSURE TUBE

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ABSTRACT

The paper presents the testing methodology on the CANDU pressure tube specimens (Zr-2.5% Nb alloy) to determine the fracture mechanics parameters. The crack initiation and subsequent DHC propagation tests were performed on compact-tension (CT) specimens prepared from pressure tube off-cuts. Previously, the CT specimens were subjected to the specific method to increase the hydrogen concentration up to 50 ppm. This process consists of two stages: the first one is the electrolytic disposal of a uniform zirconium hydride layer and, the second one is to apply a homogenization thermal treatment causing the diffusion of hydrogen into the zirconium matrix from the hydride surface layer. The testing temperature was around 280°C, which is the temperature of interest for the pressure tubes operating in CANDU 6 reactor.

Key words: fracture mechanics parameters, KIH, CT specimens, DHC mechanism

Introduction

Zirconium alloys are used in nuclear reactors because of their combination of high strength, high corrosion resistance, and low neutron absorption cross-section. Their most demanding applications in nuclear reactors are as fuel cladding and in CANDU, RBMK, and other Pressurized Heavy Water (PHW) reactors as pressure tubes containing the fuel bundles. It is important for the safe and economic operation of these reactors that these components maintain their integrity throughout their design life. However, during their residence in the reactor these components are subject to aging mechanisms resulting from thermal - and pressure driven changes, fast neutron bombardment, and corrosion at the water/metal interface, the latter resulting in a small fraction of the released hydrogen produced during the corrosion reaction being absorbed in the zirconium alloy. When the hydrogen concentration in the material exceeds the Zr–H solvus composition, zirconium hydrides are formed. These hydrides, which are less ductile than the surrounding metal matrix, can have deleterious effects on the mechanical properties of these components when present at sufficiently high volume fraction. Their deleterious effects are enhanced by increases in yield strength and decreases in fracture toughness of the zirconium material. These changes are produced as a consequence of the production of dislocation loops and other microstructural changes during fast neutron bombardment in the reactor core [1].
The paper presents the crack initiation and subsequent DHC propagation tests performed on Zr-2.5% Nb alloys from CANDU pressure tubes. Its objective is focused on the experimental DHC tests and to develop the fracture mechanical testing methodology on the specimens cut from the back and front end of the pressure tubes from Cernavoda NPP. This methodology will be used for implementing the fracture mechanics tests in the Post-Irradiation Examination Laboratory on irradiated material cut from removed pressure tubes in Cernavoda NPP. Also, the experimental results will be used also to build-up a database in order to perform some structural integrity assessment activities based on the in-service inspection results.

**Overview on the Delayed Hydride Cracking mechanism in the CANDU pressure tubes**

Delayed Hydride Cracking (DHC) involved the accumulation of hydrides to some critical size at a region of elevated tensile stress, the fracture of this hydrided region up to its leading edge and the repetition of this process. In pressure tube material, DHC is largely limited to the radial-axial plane of the tube. Optical micrographs of arrested DHC cracks show that hydrided regions formed at a crack can be idealized as plate shaped, extending in the crack growth directions with thickness much smaller than their in-plane lengths. The hydrides platelets have an irregular aspect, darker in contrast with zirconium matrix (Fig. 1). Lying on the pressure tube’s radial-axial plane these plate shaped (often tapered) hydrided regions are also referred to as radial hydrides. Observations of fracture surfaces at low magnifications often showed periodic rows of ridges (striations) extending in rows parallel to the crack front. The fracture surfaces between these striations were of brittle appearance, being flat with cleavage-like river patterns. The lengths between these striations were strongly temperature dependent.

![Fig 1. Micrograph showing hydrides in Zr-2.5%Nb pressure tube](image)

The interpretation of the experimental observations is that the hydrided region grows to some critical length from the crack tip, fracturing along its length up to its leading edge (Fig. 2). The fracture of the hydrided region causes an abrupt increase in length of the macroscopic crack, which is arrested at the leading edge of the hydrided region by the ductile matrix. The striations are the physical evidence of this arrest and the distance between each row of striations, called the striation spacing, represents the critical fracture length of the hydrided region while the striation length itself is associated with the plastic zone of the macrocrack at its point of arrest.

![Fig 2. Crack growth by DHC mechanism in Zr-2.5%Nb](image)
The rate of growth of the DHC crack suggests that the process is diffusion driven. The presence of a ductile stretch zone (the striation) after each growth event suggests that the first step for continuation of the process is the nucleation of new radial hydrides. Although there may be, by chance, transverse hydrides located at the new location of the crack tip, these hydrides are not in favorable orientations to cause embrittlement of the crack front. The requirement for the nucleation of new, reoriented (radial) hydrides after each microgrowth step then means that the concentration of hydrogen at the flaw must be at a concentration that is at least as great as the solvus composition for nucleation of new radial hydrides [2].

Experimental Activity

Sample preparation. The fracture mechanics tests were performed on CT (Compact Tension) samples cut from Zr-2.5% Nb pressure tubes. The geometry of the samples used in fracture mechanics tests, carried out to determine the parameter $K_{IH}$, must meet specific standards for fracture mechanics tests: ASTM E 399, ASTM E 647, ASTM E 1820. The samples are obtained from the axial direction of the pressure tube. The sketch of the CT sample is shown in Figure 2.

![Fig. 2 The CT specimen for DHC test](image)

To obtain $K_{IH}$ parameter on the CT specimen, one of the requirements is to have a small fatigue crack at the CT tip flaw. Thus the tip crack play the role of stress concentration factor and further will promote the hydride reorientation process during the test. Also, the fatigue crack constitutes the start point from which the cracking initiates through DHC mechanism, according to E-399 ASTM.

The pre-cracking process consists of cycling test on which the load has been gradually reduced while maintaining the stress ration is kept constant, $R \geq 0.28$. The fatigue pre-cracking process should be done carefully to diminish the probability of development plasticity at the crack tip.
Fig. 3 Stress versus time during pre-cracking fatigue process

The testing and analyses facilities. Figure 4 shows a photo with tensile creep machine where the experimental fracture tests were performed, in order to obtain the values of threshold stress intensity factor, $K_{IH}$, which characterize the initiation of DHC.

Fig. 4 DHC testing facility

Fig. 5 Optical microscop
The fracture surface and crack propagation measurements were performed by using metallographic examination with the optical microscopy (Fig. 5).

**The experimental methodology used for obtaining $K_{IH}$ parameter**

The mechanical test to obtain the threshold stress intensity $K_{IH}$ requires a constant load applied to the specimen simultaneous with a thermal cycling in a specific range. The load monitoring is realized by means of dedicated software developed at RATEN ICN Pitesti. This software is able to visualize and to adjust the testing parameters while the test is going on. The initial applied load, before test starting, is to develop a stress intensity value of $K_I=18.5$ MPa√m. At each step of 50 µm crack advance, the software automatically decrease the load with 3% from previous load. This process is repeated during 24 hours, until the cracking is stopped. In this last stage, the test is stopped and the specimen is removed from the furnace in order to perform its examination. During the test the crack propagation is monitored by the potential drop (PD) method. A specimen ready for test is shown in Figure 6.

![Fig. 6 The CT specimen wired for PD method](image)

A window with parameters monitored during the test are displayed in Figure 7.

![Fig. 7 Windows with evolution of main parameters $K_{IH}$ test](image)
To infer the $K_{ih}$ value, the last load step is considered, which the step is when the crack does not propagate for 24 hours. On the metallographic capture (Fig. 8) the crack length has been evaluated by means of the “nine segments method”. This allows obtaining the $K_{ih}$ value according the E-399 ASTM.

![Fig. 8 Măsurarea propagării fisurii prin metoda „celor 9 segmente” pe proba B55: a) fața A a probei, b) fața B a probei](image)

The stress intensity factor, used for KIH assessment is given by the following relationship:

$$K_I = \frac{P_0}{B \sqrt{W}} f\left(\frac{a}{W}\right)$$

(1)

With

$$f\left(\frac{a}{W}\right) = \frac{2 + \frac{a}{W}}{\left(1 - \frac{a}{W}\right)^{\frac{3}{2}}} \left[0.866 + 4.64 \cdot \left(\frac{a}{W}\right) - 13.32 \cdot \left(\frac{a}{W}\right)^2 + 14.72 \cdot \left(\frac{a}{W}\right)^3 - 5.60 \cdot \left(\frac{a}{W}\right)^4\right]$$

(2)

Here the meanings of involved parameters are:

- $P_0$ = applied load (N);
- $B$ = specimen thickness (m);
- $W$ = specimen width (m);
- $a$ = crack length (m)

By using the above equations in present work two values of threshold stress intensity factor $K_{ih}$ were obtained at $280\, ^\circ C$: $K_{ih} = 13.1\, MPa\cdot\sqrt{m}$ and $K_{ih} = 19.6\, MPa\cdot\sqrt{m}$. For a temperature range between $180-250\, ^\circ C$, the reference [2] declared values displayed in Figure 9.

To obtain the J integral, the following relation is used:

$$J = \frac{K_I^2}{E}$$

(3)

with $K_I$ given by equation (1), and $E$ is Young modulus.
Fig. 9 Values of $K_{IH}$ versus temperature Zr-2.5%Nb [2]

One can see that the results from the present paper are in a rather good agreement with those mentioned in cited reference.

Conclusions

The main paper outlines are:

- The short overview of Delayed Hydride Cracking (DHC) mechanism is given; DHC is the main damaging mechanism for the CANDU pressure tubes made from Zr-2.5%Nb;

- The experimental methodology to obtain the threshold stress intensity factor $K_{IH}$, which characterize the crack initiation by DHC in Zr-2.5%Nb alloy. In the present work two values of threshold stress intensity factor $K_{IH}$ were obtained at 280 °C: $K_{IH} = 13.1$ MPa·m and $K_{IH} = 19.6$ MPa·m. The results are in a good agreement with those mentioned in the scientific literature;

- This methodology will be used for implementing the fracture mechanics tests in the Post-Irradiation Examination Laboratory on irradiated material cut from removed pressure tubes in Cernavoda NPP. Also, the experimental results will be used also to build-up a database in order to perform some structural integrity assessment activities based on the in-service inspection results.

References

INFLUENCE OF SOME PARAMETERS ON THE VOID REACTIVITY AND PIN POWER CALCULATION FOR TYPICAL CANDU CELLS

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ana.segarceanu@nuclear.ro

ABSTRACT

The void coefficient of the reactivity in a typical CANDU cell is positive and it represents a sensitive issue for the regulatory discussions. The paper investigates the influence of some parameters (as geometry, composition, temperatures) on the void reactivity and power distributions. The methodology is based on integral transport calculation by the first flight collision probability approach (WIMS and CP_2D codes). The results offer a general view regarding the importance of different parameters and physical phenomena on the magnitude of the void reactivity, useful for educational & training purpose, and for the future development of new fuel bundles. The influence of each parameter is separately treated in order to understand the individual importance in the global effects.

Key words: void reactivity, pin power, CANDU

Introduction

One of the most difficult task of the CANDU (CANada Deuterium Uranium) Reactor Physics Analysis is related to the correct treatment of the deviations from the reference coolant properties. The most significant problem is the reactivity induced by a given coolant density variation. The void reactivity is calculated by the aid of the eigenvalue $k_{\text{eff}}$ taking into consideration the normal operation state ($k_{\text{ref}}$) and the perturbed state ($k_{\text{pert}}$) as:

$$\rho = \frac{1}{k_{\text{ref}}} - \frac{1}{k_{\text{pert}}}$$

In CANDU type reactors the void reactivity effect is positive ($\rho > 0$) which means that a decrease in coolant density will result in an increase of the reactivity. Consequently an increased heating of the fuel will be produced which further contributes to the decrease in the coolant density.

The effect is usually calculated by the direct method that means two successive $k_{\text{eff}}$ calculations by solving the transport equation. Generally a homogeneous model is used [1], but also stratification of the coolant including two phases model was developed [2] in order to perform detailed calculation at the level of pin powers.
From the point of view of the associated experimental data only fresh, clean and cold fuel obtained by AECL (Atomic Energy of Canada Ltd.) in the ZED-2 reactor at Chalk River Laboratory, many years ago, may be taken into consideration. There are not any specific data for burned fuel. On the other hand the validation of the theoretical models and computer codes requires an estimation of the errors associated with the void reactivity effect. Also the understanding of the influences of different parameters on the magnitude of the void reactivity is important from the point of view of the improvement of the current models.

**The model and the methodology**

The magnitude of the void reactivity is influenced by some parameters such as the burnup, the composition (including the enrichment), the temperature, the geometrical structure. In order to investigate these influences as separate effects some cases are defined as follows:

- **(Case 1)** CANDU standard cell - based on a cluster with 37 fuel elements (Figure 1), with fresh fuel,
- **(Case 2)** CANDU standard cell - based on a cluster with 37 fuel elements, at a burnup of 4000 MWd/tU,
- **(Case 3)** CANDU SEU43 cell – slow enriched uranium with 43 elements (Figure 2), with fresh fuel,
- **(Case 4)** CANDU SEU43 cell – slow enriched uranium with 43 elements, at a burnup of 4000 MWd/tU.

Both for CANDU standard cell and SEU43 cell the geometry shows a horizontal channel topology consisting of a pressure tube filled with heavy water coolant, a helium filled gap, and a Calandria tube surrounded by heavy water moderator. The fuel elements are made from UO$_2$, with natural proportions of U$^{235}$ and U$^{238}$ for standard cell, and an enrichment of 1.10% in U$^{235}$ for SEU43. A fuel gap and a Zircaloy cladding surrounds each rod. For the normal operation (reference state) a coolant density of 0.80623 g/cm$^3$ and hot operating conditions were used. For void case (perturbed state) a decreasing in the coolant
densities from 100% to 0% was used. A homogeneous distribution of the coolant into the channel was used.

The evaluation of void reactivity effect was performed by direct calculation of $k_{eff}$ in the perturbed state (voided) and reference state. Two codes are used (WIMS [3] and CP_2D [4]) for these calculations. Both of them solve the transport equation in the first flight collision probability (FFCP) approach. The difference between the two codes consists of the more detailed treatment of CP_2D compared with the simplified scheme of WIMS.

CP_2D is a two dimensional transport FFCP code for detailed fuel assembly hyperfine flux distribution calculation. The flexibility of the geometry allows treating circular or rectangular fuel pin clusters and also a relatively large local zone with homogeneous fuel assemblies. A detailed pin structure and individual fuel elements description is permitted. Also the boundary of the assembly can be exactly treated. The first version CP_2D1.0 was released in 1998. The second, CP_2D2.0, was released in 1999 and uses a multi-stratified coolant model for CANDU loss of coolant accident analysis. The third version, CP_2D3.0, incorporated a generalized burning scheme and it was released in 2000.

The integral transport equation is treated by FFCP method:

$$
\Psi(\vec{r},\hat{Q}) = \Psi(\vec{r}_0,\hat{Q}_0) \exp[-\tau(\vec{r},\vec{r}_0)] + \int dR' Q(\vec{r}',\hat{Q}) \exp[-\tau(\vec{r},\vec{r}')] 
$$

(1)

under the following approximations:

(A1) - the isotropy of the boundary in-coming flux:

$$
\Psi(\vec{r}_0,\hat{Q}_p) \approx \Psi(\vec{r}_0) = 4J^{-}(\vec{r}_0)
$$

(A2) - the isotropy of the local sources and scatterings:

$$
Q(\vec{r},\hat{Q}_p) \approx \tilde{Q}(\vec{r})
$$

(A3) - the flat scalar fluxes on each volumetric regions:

$$
\Phi(\vec{r}_i \in V_i) = \Phi_i
$$

(A4) - the flat partial currents on each boundary surface subdivision:

$$
J^\pm(\vec{r}_0 \in A_m) = J^\pm_m
$$

is equivalent with the linear equation system:

$$
\begin{cases}
\Phi^g_i A^g_i \sum^g_i = \sum_j Q^g_j A^g_j P^g_{j \rightarrow i} + \sum_m J^-_{m,g} L_{m \rightarrow i} P^g_{m \rightarrow i} \\
J^+_{m,g} L_m = \sum_j Q^g_j A^g_j P^g_{j \rightarrow m} + \sum_m J^-_{m',g} L_{m' \rightarrow m} P^g_{m' \rightarrow m}
\end{cases}
$$

(2)

where $P^g_{j \rightarrow i}$, $P^g_{m \rightarrow i}$, $P^g_{m' \rightarrow m}$ are the first flight collision probabilities (FFCPs), the transfer probabilities (TPs), and, respectively, the escape probabilities (EPs), $i$ and $j$ are generic discrete region indexes, $m$ and $m'$ are generic segment subdivisions indices.

For the boundary condition the following relation is used:

$$
J^-_{m,g} = \tilde{J}^-_{m,g} + a_{g-g'} J^+_{m,g} + \sum_{g'} a_{g'-g} J^+_{m,g'}
$$

(3)

where $a_{g-g'}$ are the albedo coefficients and $\tilde{J}^-_{m,g}$ is the given in-current on the boundary. The angular dependence is assumed isotropic.

Results and discussions

Table 1 Void reactivity for the investigated cases
<table>
<thead>
<tr>
<th>Case</th>
<th>Fuel type</th>
<th>Void fraction [%]</th>
<th>(\rho_{\text{mk}})</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>CP-2D</td>
</tr>
<tr>
<td>1</td>
<td>CANDU standard cell, with fresh fuel</td>
<td>10</td>
<td>1.68</td>
</tr>
<tr>
<td></td>
<td></td>
<td>20</td>
<td>3.26</td>
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<tr>
<td></td>
<td></td>
<td>30</td>
<td>4.95</td>
</tr>
<tr>
<td></td>
<td></td>
<td>40</td>
<td>6.56</td>
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<tr>
<td></td>
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<td>50</td>
<td>8.27</td>
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<td>10.00</td>
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<tr>
<td></td>
<td></td>
<td>70</td>
<td>11.70</td>
</tr>
<tr>
<td></td>
<td></td>
<td>80</td>
<td>13.48</td>
</tr>
<tr>
<td></td>
<td></td>
<td>90</td>
<td>15.30</td>
</tr>
<tr>
<td></td>
<td></td>
<td>100</td>
<td>17.11</td>
</tr>
<tr>
<td>2</td>
<td>CANDU standard cell, at a burnup of 4000 MWd/tU</td>
<td>10</td>
<td>1.32</td>
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<tr>
<td></td>
<td></td>
<td>20</td>
<td>2.61</td>
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<td></td>
<td></td>
<td>30</td>
<td>3.97</td>
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<td></td>
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<td>40</td>
<td>5.33</td>
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<tr>
<td></td>
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<td>50</td>
<td>6.69</td>
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<td></td>
<td></td>
<td>60</td>
<td>8.03</td>
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<td></td>
<td></td>
<td>70</td>
<td>9.43</td>
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<td></td>
<td></td>
<td>80</td>
<td>10.81</td>
</tr>
<tr>
<td></td>
<td></td>
<td>90</td>
<td>12.25</td>
</tr>
<tr>
<td></td>
<td></td>
<td>100</td>
<td>14.25</td>
</tr>
<tr>
<td>3</td>
<td>CANDU SEU43 cell, with fresh fuel</td>
<td>10</td>
<td>1.31</td>
</tr>
<tr>
<td></td>
<td></td>
<td>20</td>
<td>2.63</td>
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<tr>
<td></td>
<td></td>
<td>30</td>
<td>3.93</td>
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<td>5.27</td>
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<td></td>
<td></td>
<td>50</td>
<td>6.58</td>
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<td></td>
<td></td>
<td>60</td>
<td>7.95</td>
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<tr>
<td></td>
<td></td>
<td>70</td>
<td>9.30</td>
</tr>
<tr>
<td></td>
<td></td>
<td>80</td>
<td>10.71</td>
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<tr>
<td></td>
<td></td>
<td>90</td>
<td>12.11</td>
</tr>
<tr>
<td></td>
<td></td>
<td>100</td>
<td>13.58</td>
</tr>
<tr>
<td>4</td>
<td>CANDU SEU43 cell, at a burnup of 4000 MWd/tU</td>
<td>10</td>
<td>1.34</td>
</tr>
<tr>
<td></td>
<td></td>
<td>20</td>
<td>2.76</td>
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<tr>
<td></td>
<td></td>
<td>30</td>
<td>4.13</td>
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<td></td>
<td></td>
<td>40</td>
<td>5.53</td>
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<td>12.75</td>
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<tr>
<td></td>
<td></td>
<td>100</td>
<td>14.63</td>
</tr>
</tbody>
</table>

In Table 1 the main results for the above defined cases are presented. The void reactivity values are obtained both by WIMS-D4 and CP_2D in homogeneous model. Void fractions from 10% to 100% were used.

A relative good agreement between the results of the two codes is observed taking into consideration the differences in the geometrical treatment. The maximum void effect is obtained...
for the Case 1 (standard cell) around 17 mk for the fully voided cell. In Figure 1 the comparison between CP_2D and WIMS results is presented. A better agreement is observed in Figure 2 for the SEU43 cell (Case 4).

![Fig.1 Void reactivity effect, CP_2D to WIMS comparison for Case 2](image1)

The introducing of the slow enriched uranium (Case 3 and 4) leads to a slow decreasing of the void effect in comparison with natural Uranium (Case 1 and 2) (see Table 1). In Figure 3 the variation of the void reactivity with the enrichment is presented by using standard cell geometry configuration and a variation of the enrichment between 0% and 5%. Fresh fuel and void

![Fig.2 Void reactivity effect – comparison for CP_2D and WIMS (Case 4)](image2)
fractions between 10% and 100% are used. A decreasing tendency is observed with the increase of the enrichment.

![Variation of void reactivity with the enrichment](image)

**Fig.3 Variation of void reactivity with the enrichment**

A comparison of the variations of void reactivity versus void fractions for Case 1 (CANDU standard, fresh fuel) and Case 3 (CANDU standard, 4000 MWd/tU) is presented in Figure 4. In case of averaged burnt fuel a reduction of the void reactivity in comparison with the fresh fuel case is observed (from 17.11 mk to 14.25 mk, for CP_2D results).

The tendency is also observed for SEU43 configuration (Fig. 5), but the effect is quite diminished. A direct comparison of the fresh fuel cases (SEU43 versus standard configuration) is presented in Figure 6. For fully voided channel (100% void fraction) a reduction of void reactivity from 17.11 (standard cell) to 13.58 (SEU43) is observed. A combined effect of the enrichment’s increase and of the geometry leads to this decrease of the void reactivity.

In Figure 7 the values of the pin powers obtained by CP_2D calculations are presented. The values of the pin power are normalised to the average pin power of the bundle, therefore the average is represented by 1.00 value. A comparison between reference state (0% void) and fully voided (100%) (Case 4) for SEU43 configuration is represented. The results show a decrease of the pin powers for the outer ring of fuel elements and an increase of the others in voided configuration in comparison with the normal state.
Fig. 4 Void reactivity – comparison between fresh and 4000 MWd/tU fuel (standard configuration)

Fig. 5 Void reactivity – comparison between fresh and 4000 MWd/tU fuel (SEU43 configuration)
Fig. 6 Void reactivity effect – comparison between standard and SEU43 configuration (fresh fuel)

Fig. 7 Void reactivity effect in the pin powers – comparison between normal state (bellow) and full voided (up) SEU43 configuration (4000 MWd/tU)
The values of the pin powers for SEU43 configuration (Case 4) for fully voided cell is presented in Table 2. For standard cell (Case 2) the pin powers are presented in Table 3. In order to compare the void effect for standard and SEU43 configurations the averaged pin powers per fuel ring are presented in Table 4. In both cases 100% void fraction was used.

**Table 2** Pin power values for SEU43 fully voided (Case 4)

<table>
<thead>
<tr>
<th>Fuel el.</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
<th>11</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pin powers</td>
<td>1.098</td>
<td>1.122</td>
<td>1.130</td>
<td>1.145</td>
<td>1.175</td>
<td>1.175</td>
<td>1.145</td>
<td>1.130</td>
<td>0.853</td>
<td>0.850</td>
<td>0.863</td>
</tr>
<tr>
<td>Fuel el.</td>
<td>12</td>
<td>13</td>
<td>14</td>
<td>15</td>
<td>16</td>
<td>17</td>
<td>18</td>
<td>19</td>
<td>20</td>
<td>21</td>
<td>22</td>
</tr>
<tr>
<td>Pin powers</td>
<td>0.865</td>
<td>0.873</td>
<td>0.883</td>
<td>0.884</td>
<td>0.891</td>
<td>0.884</td>
<td>0.883</td>
<td>0.873</td>
<td>0.865</td>
<td>0.862</td>
<td>0.850</td>
</tr>
<tr>
<td>Fuel el.</td>
<td>23</td>
<td>24</td>
<td>25</td>
<td>26</td>
<td>27</td>
<td>28</td>
<td>29</td>
<td>30</td>
<td>31</td>
<td>32</td>
<td>33</td>
</tr>
<tr>
<td>Pin powers</td>
<td>1.027</td>
<td>1.028</td>
<td>1.036</td>
<td>1.034</td>
<td>1.030</td>
<td>1.033</td>
<td>1.035</td>
<td>1.031</td>
<td>1.031</td>
<td>1.033</td>
<td>1.044</td>
</tr>
<tr>
<td>Fuel el.</td>
<td>34</td>
<td>35</td>
<td>36</td>
<td>37</td>
<td>38</td>
<td>39</td>
<td>40</td>
<td>41</td>
<td>42</td>
<td>43</td>
<td></td>
</tr>
<tr>
<td>Pin powers</td>
<td>1.045</td>
<td>1.033</td>
<td>1.031</td>
<td>1.031</td>
<td>1.035</td>
<td>1.033</td>
<td>1.030</td>
<td>1.034</td>
<td>1.036</td>
<td>1.028</td>
<td></td>
</tr>
</tbody>
</table>

**Table 3** Pin power values for standard cell fully voided (Case 2)

<table>
<thead>
<tr>
<th>Fuel el.</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
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<tr>
<td>Pin power</td>
<td>0.833</td>
<td>0.854</td>
<td>0.867</td>
<td>0.853</td>
<td>0.854</td>
<td>0.866</td>
<td>0.852</td>
<td>0.939</td>
<td>0.935</td>
<td>0.913</td>
<td>0.910</td>
</tr>
<tr>
<td>Fuel el.</td>
<td>12</td>
<td>13</td>
<td>14</td>
<td>15</td>
<td>16</td>
<td>17</td>
<td>18</td>
<td>19</td>
<td>20</td>
<td>21</td>
<td>22</td>
</tr>
<tr>
<td>Pin power</td>
<td>0.930</td>
<td>0.938</td>
<td>0.939</td>
<td>0.935</td>
<td>0.912</td>
<td>0.910</td>
<td>0.930</td>
<td>0.938</td>
<td>1.111</td>
<td>1.105</td>
<td>1.105</td>
</tr>
<tr>
<td>Fuel el.</td>
<td>23</td>
<td>24</td>
<td>25</td>
<td>26</td>
<td>27</td>
<td>28</td>
<td>29</td>
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<td>31</td>
<td>32</td>
<td>33</td>
</tr>
<tr>
<td>Pin power</td>
<td>1.105</td>
<td>1.103</td>
<td>1.103</td>
<td>1.105</td>
<td>1.105</td>
<td>1.105</td>
<td>1.111</td>
<td>1.105</td>
<td>1.105</td>
<td>1.105</td>
<td>1.103</td>
</tr>
<tr>
<td>Fuel el.</td>
<td>34</td>
<td>35</td>
<td>36</td>
<td>37</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pin power</td>
<td>1.103</td>
<td>1.105</td>
<td>1.105</td>
<td>1.105</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Table 4** Averaged pin power values per each fuel ring – comparison between Case 2 and Case 4, for 100% void fraction

<table>
<thead>
<tr>
<th>Fuel ring</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
</tr>
</thead>
<tbody>
<tr>
<td>CANDU standard</td>
<td>0.833</td>
<td>0.858</td>
<td>0.927</td>
<td>1.105</td>
</tr>
<tr>
<td>SEU</td>
<td>1.098</td>
<td>1.146</td>
<td>0.870</td>
<td>1.033</td>
</tr>
</tbody>
</table>
Conclusions

(C1) CP_2D and WIMS results for the void effect are in a good agreement. Both the models use the first-flight collision probability method to solve the transport equation. Additionally CP_2D is able to simulate a two-phase flow (liquid and vapors) of the partially voided cell compared with WIMS that use only the homogeneous model.

(C2) The paper presents the dependence of the void reactivity on the: geometry configuration, enrichment, and burnup. Generally the increase of the enrichment and also of the burnup leads to the decreasing of the positivity of the void reactivity.

(C3) The pin powers were calculated by CP_2D simulation and compared for the voided cell and normal state operation. In case of SEU43 a major redistribution of the powers is present mainly due to the effect of the geometry (two-dimensional types of fuel rods).

References

METALLOGRAPHIC ANALYSIS OF AS RECEIVED AND AUTOCLAVING INCONEL 617 ALLOY

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*University from Piteşti, **RATEN - INR Piteşti, România

ABSTRACT

This paper make a comparative metallographic analysis between “as received” and steam-autoclaving samples of Inconel 617 alloy (UNS N06617). The samples were tested by steam-autoclaving. Autoclaving test parameters: time 16 days, 550°C temperature and 250 atmosphere pressures. The analysis of samples consisted in metallographic examination: thickness layer measuring, microstructure modification, and Vickers microhardness (MHV0.1) determinations. The tested material was compared with the as received material. For metallographic analysis are used the Olympus GX 71 optical microscope and OPL tester by automatic cycle. Average grain size were automatic determined by linear interception method and reported as ASTM Micro Size number (G). Material microhardness was calculated with relationship of technical book of device. Average grain size for tested and “as received” samples is finer than the specification requirements (G max. 3.5). The results showed a good metallographic behavior of Inconel 617 at this steam-autoclaving test.

Key words: Inconel 617, steam-autoclaving, microstructure, microhardness

Introduction

INCONEL® alloy 617 (UNS N06617) is a solid-solution, strengthened, nickel-chromium-cobalt-molybdenum alloy with an exceptional combination of high-temperature strength and oxidation resistance. The alloy also has excellent resistance to a wide range of corrosive environments, and it is readily formed and welded by conventional techniques. The high nickel and chromium contents make the alloy resistant to a variety of both reducing and oxidizing media. The aluminum, in conjunction with the chromium, provides oxidation resistance at high temperatures. Solid-solution strengthening is imparted by the cobalt and molybdenum. The combination of high strength and oxidation resistance at temperatures over 1800°F (980°C) makes Inconel alloy 617 an attractive material for such components as ducting, combustion cans, and transition liners in both aircraft and land-based gas turbines. Inconel alloy 617 also offers attractive properties for components of power-generating plants, both fossil-fueled and nuclear, [1].

Experimental Methods

The experiment consisted in a steam-autoclaving test, in a Baskerville Autoclave, with further parameters: 16 days time, 550°C temperature and 250 atmosphere pressure. Metallographic analysis of samples consisted in the thickness oxide layer determination and the microstructure examination (average grain size, Heyn method), [2].
Vickers microhardness were calculate with relationship of technical book of device (micro-load 0.1 kgf): 
\[ MHV = \frac{1854.4 \cdot F}{d^2}, [\text{kgf/mm}^2] \]
*Where: F -force/charge [gf]; d -average diagonal of indentation [μm]; 1854.4 -device coefficient for Vickers microhardness, [3].*

In Table no.1: The average grain size and hardness for cold-rolled sheet of Inconel 617 (solution treated), in conformity with prescript requirements in Table 4 of ASME SB-168, [4].

<table>
<thead>
<tr>
<th>Table 1</th>
<th>Inconel 617 (UNS N06617) Average grain size and hardness, [4]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cold-Rolled Sheet, 1.42 m Wide and Under</td>
<td>Calculated Diameter of Average Grain Section, Max., mm</td>
</tr>
<tr>
<td>Over 1.3 to 6.4, incl.</td>
<td>0.110</td>
</tr>
</tbody>
</table>

**Results and Discussion**

*Thickness oxide layer*

The thickness oxide were automatic measured with “analySIS” soft of Olympus GX 71 metallographic microscope (x1000). In general, after the steam-autoclaving test the thickness oxide layer was very thin, below 0.5 μm. Figure 1: oxide layer on autoclaving Inconel 617, [5].

![Figure 1](image1.png)

**Fig. 1.** Tested Inconel 617: oxide layer <0.5µm, [6]

*Microstructure*

Structure were revealed by electrolytic etching: 10% oxalic acid solution, 6V, 20-50 seconds. Average grain size were automatic determined, in cross-section (TS). 
Microstructure of as-received Inconel 617, Figure 2 (TS): Austenite (A) with twined grains and carbide particles (dark) at grain boundaries and intragranular. Average grain size correspond of ASTM No. 5.5, [6]

Microstructure of autoclaving Inconel 617, Figure 3: Austenite (A) with partial recrystallized grains and uniform dispersed carbide particles (dark). Average grain size correspond of ASTM No. 7.0, [5]

![Figure 2](image2.png) ![Figure 3](image3.png)

**Fig. 2.** As-received Inconel 617, A+carbides, TS  **Fig. 3.** Tested Inconel 617: A+carbides, TS
The microstructure of autoclaving samples is finer than the as-received samples, average diameter of grain decreasing with 1/3 approximate (see Figure 4).

![Inconel alloy 617: Average grain size](image)

**Fig. 4:** Comparative diagram of average grain size

In general, average grain size of as-received and tested samples is finer than the ASTM requirements [4]: Max. ASTM No.3.5 for Alloy 617 (UNS N06617).

**Vickers microhardness**
In Figures 5-6: Vickers microhardness imprints (MHV0.1) on as-received and autoclaving samples, in cross-section (TS), [5]

![Fig.5. As-received Inconel 617: MHV imprints](image) ![Fig.6. Tested Inconel 617: MHV imprints](image)

After steam-autoclaving test the microhardness increase with 10 units comparative with as received samples (see Figure 7)
Conclusions

✓ **Thickness layer.** The surface samples after steam-autoclaving covered with a very thin oxide layer.
✓ **Microstructure.** The alloy after steam-autoclaving undergo some microstructure changes comparative with as-received alloy: partial recrystallization and finer grains.
✓ **Vickers microhardness.** A little growth of microhardness have place after steam-autoclaving (light hardening).

The results show that Inconel 617 alloy have a good behaviour at this steam-autoclaving test.

References

[1]. Special Metals – www.specialmetals.com
FRESH FUEL INVESTIGATION USING THE DRY NEUTRON RADIOGRAPHY FACILITY

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ABSTRACT

The non-destructive imaging examination by thermal neutrons and gamma radiations is a testing method for nuclear fuel, used in INR, useful for the improvement of the CANDU nuclear fuel fabrication. For the investigation of fresh nuclear fuel, the dry neutron radiography facility placed at the tangential channel of the Annular Core Pulsing Reactor (ACPR), can be used. INUS uses a modern digital detector with scintillators for neutrons and gamma radiations and CCD cameras for the presented experiments with experimental CANDU nuclear fuel. Some of the images obtained are shown and discussed from the point of relevance for geometrical resolution, internal structure, contrast etc. The tomography reconstruction for a nuclear fuel element will be presented. The paper will outcome some conclusions on the possibility to offer meaningful information about the internal structure of the fuel, grains in the structure of the pellets, dishes between pellets, eventual defects etc..

Key words: Imaging facility, tomography of nuclear fuel, experimental nuclear fuel

Introduction

The main components of the INUS facility (detector with scintillators and CCD cameras, object holder, gear for remote control of the lead cork of the tangential channel, two fast neutron shutters of cadmium and Boral and the thick neutron and gamma shutter) are placed in a biological protection [1,2,3,4].

The detector (Photo 1) is a key component of the experimental neutron radiography setup, which allows image acquisition regarding the internal structure of the tested object. In the upper side of the photo, the two cameras can be seen, and the mirror in the lower side. The cameras are moving together, horizontally in order to position on the axis represented by the centre of the scintillator and the mirror and vertically, to ensure the optimum working distance.

Two interchangeable scintillators for neutron and gamma radiation detection, with 300 mm x 300 mm size are used:
- For neutrons (6LiF-ZnS, 0.3 mm thick);
- For gamma radiation (Gd2O2S, commercial, LANEX type).

The CCD cameras of the detector are:
• CCD STARLIGHT XPRESS SXV-H9 camera with ICX285AL Exview HAD sensor from Sony (1392 pixels (H) x 1040 pixels (V) and 8.98 mm (H) x 6.7 mm (V) dimensions) + XD-4 type image intensifier + lenses. This CCD camera operates with exposure times from μs to minutes.
• EM-CCD Hamamatsu C9100-02 camera with 1000 pixels x 1000 pixels and 8.0 mm x 8.0 mm dimensions of the sensor. This camera operates with exposure time from 100 μs to 10 s.

These cameras of the detector are connected at PC by cables with repeaters (additional 12 m cable for USB 2.0 connection of the CCD STARLIGHT camera and additional 10 m CameraLink cable for EM-CCD Hamamatsu).

![Photo 1. The neutron and gamma radiation detector fully equipped with two cameras, Starlight (left) and Hamamatsu (right), mirror and step by step TRINAMIC motors.](image)

The main parameters of INUS are:

• Thermal neutron intensity measured with gold foils through neutron activation analysis is $1.22 \times 10^5$ n/cm$^2$/s without Bi filter in the neutron beam;
• Inlet aperture is of 45 mm and is placed at 4178 mm from the middle of the rotary table (L/D = 92.84 for this position but ~ 95 for distance inlet aperture-detector);
• Effective beam size is ~288.3 mm in diameter at the middle of the rotary table;
• Acquisition time/position with Starlight camera is 35-50 s and the sensor reading is 3-5 s, thus it is not usable for real time imaging;
• Fixed working distances with Starlight camera are used, one at 40 cm for 100 mm x 75 mm field of view and the other one at 75 cm for 286.5 mm x 214 mm field of view (this camera has two interchangeable folding lenses);
• Divergent angle of the collimator is 3.34;
• Gamma dose with Bi filter raised is 9.58 mSv/h;

Operational desk (Photo 2) is at approximately 7 m away from the additional wall of the biological protection, fact that ensures the safety of the personal during operation. Surveillance system consisting of three fixed cameras, one of them attached at the front side of the detector (Photo 3) is used to observe the investigated object and different movements carried out by step by step motors.
Performed work

In the current experiment, a series of CANDU experimental nuclear fuel elements were investigated in order to observe the fuel characteristics such as the internal structure of the fuel, grains in the structure of the pellets, gaps between pellets and fuel cladding, dishes between pellets, welding seams appearance or eventual defects.

For the purpose of the experiment, taking into account the different dimensions of the experimental fuel elements and the need for them to spin around their axes, an existing special fuel holder was adapted in order to hold the fuel elements vertically (Photo 4) not horizontally as in initial design in order to have more precise rotation axis. End adapters were added to the modified holder, according to the length, diameter and the shape of the end plugs of the fuel elements.

The specified seven CANDU experimental fuel elements (Table 1) with different constructive characteristics, due to the specific irradiation tests to which they will be subjected, have been investigated.
Table 1. Experimental nuclear fuel characteristics

<table>
<thead>
<tr>
<th>No.</th>
<th>Name</th>
<th>Total length [mm]</th>
<th>Pellet column length [mm]</th>
<th>Pellet density [g/cm³]</th>
<th>Pellet enrichment [%U²³⁵]</th>
<th>Pellet diameter [mm]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>B27/B29</td>
<td>~283,6</td>
<td>245,2</td>
<td>10,70</td>
<td>5+/−0,1</td>
<td>12,22</td>
</tr>
<tr>
<td>2</td>
<td>B63</td>
<td>~321</td>
<td>287,1±0,2</td>
<td>10,6±0,15</td>
<td>10 (standard pellets) 0,71(natural UO₂, end pellets)</td>
<td>12,22±0,014</td>
</tr>
<tr>
<td>3</td>
<td>B53/ B54/ B55/ B56</td>
<td>~213</td>
<td>150,4</td>
<td>10,70</td>
<td>10</td>
<td>12,22</td>
</tr>
</tbody>
</table>

For the tomography investigation, the Starlight camera was used in order to capture 115 images with an exposure time of 50 seconds. 105 images were projections with an increment of 1.8°, 5 images as dark images (images without neutron beam) and 5 images as flat field (images without object in the neutron beam). The fuel holder rotates all seven nuclear fuel elements simultaneously with an electric motor remote controlled from PC. On the scintillator, a geometrical resolution indicator was placed to be used for camera focalization and to indicate the level of geometrical resolution of the obtained images (seen in Photo 4).

With Octopus 8.1 software in manual mode, the proper parameters were set and the volume of the B27 experimental CANDU fuel element was reconstructed.

In order to determine the degree of activation of the fuel elements after the exposure, a radiological characterization has been made before and at 30 minutes after investigation. The results of the measurements are presented in Table 2.

Table 2. The radiological characterization of the experimental fuel elements

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Maximum value (α+β+γ) Bq/cm²</td>
<td>Gamma (γ) µSv/h</td>
</tr>
<tr>
<td>1</td>
<td>B27</td>
<td>27,5</td>
<td>7</td>
</tr>
<tr>
<td>2</td>
<td>B29</td>
<td>25,6</td>
<td>6,3</td>
</tr>
<tr>
<td>3</td>
<td>B63</td>
<td>27,6</td>
<td>5,2</td>
</tr>
<tr>
<td>4</td>
<td>B53</td>
<td>28,4</td>
<td>4,5</td>
</tr>
<tr>
<td>5</td>
<td>B54</td>
<td>29,4</td>
<td>10,6</td>
</tr>
<tr>
<td>6</td>
<td>B55</td>
<td>32,4</td>
<td>6</td>
</tr>
<tr>
<td>7</td>
<td>B56</td>
<td>28,7</td>
<td>5,5</td>
</tr>
</tbody>
</table>

As shown in the table, an expected increase of the radioactivity of the fuel elements has occurred due to exposure to thermal neutrons.

Results

For the experiment, the seven nuclear fuel elements which will be used for testing in a number of irradiation devices have been investigated. This investigation with neutron and gamma radiation allows the fabrication evaluation of the fuel before the beginning of the irradiation tests.
Photo 5. Neutron exposure image with all 7 experimental nuclear fuel elements

Photo 6. Neutron exposure image for the B27, B63 and B29 fuel elements

Photo 7. Gamma exposure image for the B27, B63 and B29 fuel elements

Photo 5 presents the image obtained with thermal neutrons with the entire visual field (286.5 mm x 214 mm) of the CCD Starlight camera with all 7 experimental nuclear fuel elements and the geometric resolution indicator whose image puts in evidence the narrowest gap of 0.1 mm between gadolinium stripes. Here, the internal structure of all the fuel elements can be distinguished as was described in technical specifications. For B53, B54, B27, B29 and B55 the axial gap of the pellets column is about 1 mm, for B56 is about 0.5 mm but at B63 there is not any evidence of the gap. The difference between pellet enrichment of the experimental element is observable on image. The elements B27 and B29 have 5% enrichment in U-235 and the other ones have 10% enrichment in U-235. Thermal neutrons penetrate in a much number through less enriched elements, this fact conducting at a lighter image for B27 and B29.
nuclear fuel pellets. There are visible the central cylindrical holes for five pellets in the top of the column for B27 and B29 nuclear fuel elements.

Neutron obtained images (Photo 6) and gamma radiation obtained images (Photo 7), with 100 mm x 75 mm field of view for CCD Starlight camera shows central parts of the B27, B63 and B29 fuel elements. The gap between two pellets is about 0.36 mm according to fabrication specifications. Comparing the width of the gaps on image with the gaps revealed by the image of the geometric resolution indicator these are in the range of 0.32 mm – 0.42 mm known on initial measurements on indicator. In Photo 7, the skids used to fit the fuel element into the irradiation device fuel holder can be seen. Also, from Photo 7, using γ radiations, can be seen the clad of the nuclear fuel elements that is not visible with neutrons. The difference between diameters put in evidence with neutrons and gamma is about 11-12 pixels, that correspond to 0.74 -0.81mm, approximately two times the thickness of the experimental nuclear fuel clad.

For the tomographic reconstruction, the B27 experimental fuel element has been picked (Photo 8). Its relevant characteristics are shown in Table 1.

![Photo 8. B27 experimental fuel element](image)

In the left side of the image, the channel that will be used for instrumentation can be seen going through the end plug and all the way into five fuel pellets. In Photo 9, relevant slices obtained with tomographic reconstruction are shown. Important areas such as the instrumentation channel (a) with the fuel pellet dish (b), a mid-section pellet (c) with a dish (d) and the skids (e) can be observed.

![Photo 9. 5 slices obtained with tomographic reconstruction of the B27 fuel element](image)

**Conclusions**

INUS operated with CCD STARLIGHT XPRESS SXV-H9 camera proved to be proper for imaging investigations, primarily with neutrons to reveal structure of the pellets of the nuclear fuels and secondly with gamma radiations to reveal the structure of the other components of the nuclear fuel elements (clad, skids).

In the captured images, the pallets, the gap and the general internal structure of the fuel element can clearly be seen. From these images, can be concluded that there are no evident defects for six experimental nuclear fuel elements (B53, B54, B55, B56, B27, B29) and these fuel elements can safely be used for further experiments. A question has raised for B63 whose axial gap of 1.3-1.5 mm, according specifications, is not seen on image.
The tomography reconstruction with Octopus 8.1 software have shown the potential of good revelations even with 105 projections, instead of at least 200 projections/180°. The reconstructed images indicate that the fuel element pellets have a homogenous structure and does not present internal macroscopic defects.

INUS can be used for imaging applications on nuclear fuel elements after fabrication process for fast investigations to control the internal dimensional structure to certify its correspondence with fabrication specifications.

For images with higher geometrical resolution could be used the transfer method with dysprosium and indium converter neutron screens and radiographic films. These images are the witness images for those that will be obtained with underwater neutron radiography facility after irradiation activities with transfer method.

References

VACUUM BRAZING TECHNIQUES FOR IRRADIATION DEVICES AT TRIGA RESEARCH REACTOR

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ABSTRACT

Metallic thin-walled thermocouples are required for monitoring the temperature value for experiments that are conducted in a nuclear research reactor. The different location wall crossing is made by instrumented passage. Such a passage produced by vacuum brazing using a BNi-7 alloy, represents the proper way to obtain a sealed joint, which can withstand corrosion and high temperatures, having in the same time a small neutron cross section.

This paper presents the brazing experiments of K-type thermocouples with stainless steel and Inconel 600 sheath. The sheaths brittleness, hardness changing in joint’s vicinity and structural modification emphasized by metallographic analysis are aspects treated by comparing different samples obtained in brazing laboratory.

For finding the correct answer regarding the attenuation of negative effects which are occurring during brazing procedure using Inconel 600 – BNi-7 combination, one can assess both the adopted solution used in designing instrumented passage and thermal regime parameters and its precisely control.

Key words: thin-walled thermocouples, vacuum induction brazing, BNi-7, instrumented passage, Inconel 600

Introduction

The experimental installations designed for irradiation testing require specific instrumentation techniques. When is needed an operation such as passing one pressurized enclosure walls by a thermocouple which has a sheath thickness under 0.15 mm, made from steel or Inconel 600 and a diameter φ 1 mm, it is strongly required finding some unconventional solution for obtaining the joint. These must be sealed and must have mechanical resistance having in mind the work environment in which they are settled. Moreover, it must take into account the restrictions related to the usage of this type of materials in nuclear installations, copper being one of the forbidden material for these locations.

In order to achieve the necessary sealed passages in instrumentation of devices used in irradiation process, a solder nickel-based alloy, BNi-7, is used. It has a low section of the absorption of neutrons, and is suitable for brazing stainless steels and providing mechanical strength at high temperatures. It looks like metallic powder of different grain sizes, embedded in a gel that evaporates when heating without leaving any residue.

Brazing technique used in our laboratory is based on inductive heating in a vacuum chamber. All the elements to be joined are positioned in the center of a coil made of copper pipe. The temperature is
monitored both by direct observation using a disappearing filament pyrometer and through the acquisition from a piece-fixed thermocouple, in an interval of one second, using a software application. Brazing installation is shown in Figure 1.

The main components of the brazing installation are:
1. Brazing installation Control-Command panel
2. Multichannel Controller -ALCATEL ACM 1000- for monitoring the vacuum value
3. Glass vacuum chamber
4. Optic pyrometer with disappearing filament for measuring the temperature
5. F.C.A Panel – feeding-control-acquisition
6. Medium frequency generator 10 kHz, SIEMENS
7. Computer for running the control application
8. Pressure transducer, ALCATEL ACC 1009 connected to the multichannel controller
9. Diffusion pump for obtaining high vacuum \(\sim 10^{-6}\) torr
10. Water cooling loop for coil, generator, transformer and diffusion pump.
11. Pump for obtaining preliminary vacuum \(\sim 10^{-2}\) torr - AGILENT/220 V

BNi-7 alloy used for the experiments has the following characteristics which are presented in table 1.

<table>
<thead>
<tr>
<th>Chemical composition</th>
<th>Melting temperature (^{\circ}\text{C})</th>
<th>Brazing temperature (^{\circ}\text{C})</th>
<th>Brazing atmosphere</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ni76% Cr14% P10%</td>
<td>888 888</td>
<td>927÷1093</td>
<td>Vacuum</td>
</tr>
</tbody>
</table>

Between base material and the filler material must be at least a partial solubility for the brazing process to take place. The process that takes place at the contact surface between the base material and the filler material is based on the alloying capacity between these two materials or at least of some of their components.

The filler materials used for brazed joints must meet the following general technical requirements:
- melting point is lower than that of the base material;
- melting range to be as small as possible, otherwise the components with lower melting temperature tend to separate from the ones which fuses harder which hampers the normal process;
- must have good wetting ability, fluidity and capillarity;
- components of filler and base material must be mutually soluble and must diffuse between them without giving rise to fragile intermediate stages;
- molten alloy must have low surface tension and low viscosity;
- the components of the alloy must have while in contact with the base material, as small as possible potential differences for obtaining a joint to be more resistant to corrosion;

One of the basic conditions of a brazing operation is cleanness of surfaces to be joined in order to wetting the base material by molten brazing alloy and its expansion in the joint’s gap.

A prerequisite condition from a constructive point of view, is the usage and the proper placement of the alloy chosen for the joint implementation. Geometric shape of the joint must provide a convenient and effective addition of filler, therefore when melting this must penetrate by capillary action and by its own weight directly into the joint.

**Experiments**

In the experiments were tested two different ways of placing the alloy. In the first case we used an alloy reservoir practiced right around the element to be brazed, in this case the thermocouple. In the second case we designed a piece on which the alloy reservoir is on the lateral side, from where a channel was practiced through which the molten alloy will fill the technological space resulting the final joint.

We can say that both methods rely on capillary phenomena, but in the second case we have a great advantage.

It is represented by the opportunity to observe the direct method, when the alloy is passing in liquid state and fills the interstice between the two pieces, forming a meniscus in the visible area. High fluidity of the liquid alloy makes this phenomenon to occur quite rapidly.

Once the meniscus is formed, we can take the decision of turning the power off by stopping the generator. In this way we can reduce some of the negative effects that could occur during maintaining the molten alloy into contact with the thermocouple sheath.

In the brazing process the following phenomena can occur:
1. diffusion in the base material;
2. dilution with the base material;
3. erosion of base material;
4. forming of fragile compounds.

The effects (1), (2) and (3) depends on the mutual solubility of the brazing filler metal and the base metal, the amount of brazing filler present, the temperature and the brazing cycle duration.

Some fillers materials excessive diffuse, changing base metal properties. To control the diffusion, one selects the suitable filler material, applies a minimal amount of it and seeking appropriate brazing cycle. In long capillaries, mutual solubility can change the composition of alloying filling metal. This will increase normally the liquid state temperature causing the solidification before complete filling of the joint. The most important factor affecting the degree of erosion for a given combination filler – brazing base metal is the brazing technique that includes techniques for assembly and achievement, application of brazing alloy and the brazing cycle itself.

In conclusion, in order to be efficient, a filler material used in brazing must be alloyed with the base material surface without undesirable level of diffusion, without dilution in base material, without the erosion of the base material or without forming fragile compounds on interface. These effects are dependent on the mutual solubility of the filler material and the base materials, the amount of brazing filler material present and the temperature and time profile of the brazing cycle.
The fundamental problem that arises is that of finding the optimum combination between materials used, design the joint area and the thermal profile used in brazing, so as to avoid weakening and rupture in the sheath and breaking in the instrumented passage entry area.

To prepare a comparative study on the behavior of thermocouples with sheath of stainless steel versus thermocouples with Inconel 600 sheath when brazing with BNi-7 alloy, we constructed an assembly consisting of a transition piece from one irradiation device and a total of four thermocouple segments of which two of steel sheath and two of Inconel 600 sheath. Figure no. 2 shows the assembly prepared for brazing.

![Fig. 2 – Pieces positioning for brazing process](image)

The elements preparation for being joined by brazing was started by checking the dimensions and tolerances so that they meet all the technical requirements. Choosing the filler material determines the joint gap and the work environment temperature. For BNi-7 brazing alloy, the recommended values taken from literature are: maximum gap = 0.002 inch = 0.055 mm, and brazing temperature, $T = 927-1093 \, ^\circ C$.

In the temperature / time graph in Figure no. 3, besides the temperature plateau set at 500 °C, we have set another one at 620 °C, which lasts for 10 minutes. This plateau was aimed to maintain a vacuum value between the established limits because when the temperature increases, a damage of the vacuum (pressure increase) was observed. This phenomenon occurs due to release of volatile elements in particular from brazing alloy composition. Therefore, it must be kept a relatively low temperature compared to the temperature at which the melt solder occurs, until the evacuation from the vacuum chamber of components that may damage the prescribed value of pressure needed for conducting the brazing process.
Fig. 3 Thermal profile on simultaneous Inconel 600-steel brazing

Results

Figure no.4 shows the comparative cross sections through the brazed area of a steel sheath thermocouple and an Inconel 600 sheath thermocouple. It can be observed that on simultaneous brazing of the same basic material, with the same brazing alloy and the same thermal conditions, the Inconel 600 sheath undergoes radical transformations.

![Cross section of a TC brazed joint; a) – steel; b) – In600](image)

1. Brazing layer
2. Piece
3. TC sheath
4. TC conductors
5. TC insulator
6. Base piece

In Figure no. 5 are shown the same brazed areas from Figure no. 4 but in longitudinal section. It is confirming the same changes of the Inconel 600 sheath compared with the steel sheath.
The cross section and longitudinal section for both stainless steel and In600 sheaths TC were analyzed using a microscope; differences were observed between brazed areas for the two types of sheaths (steel and Inconel 600). In the case of Inconel sheaths, it was observed the phenomenon of conversion of a portion of the sheath, in a metallic compound, in the contact area with the brazing alloy.

After performing metallographic analysis, we considered necessary to find out the hardness values in the surrounding area of the brazed joint. It was determined hardness of the brazing metal compound and the surrounding areas with the help of micro-hardness measuring equipment. In general, micro-hardness was determined in four (4) distinct areas: the sheath, alloy, alloy particles and metallic compound (alloy + sheath). Hardness values obtained fall (roughly) between 200-800 kgf / mm².

Figure no. 6 shows a hardness measuring of In600 sheath on a witness thermocouple segment not subjected to brazing process. Major differences can be seen between the hardness of the witness thermocouple sheath and the hardness of the brazed thermocouple.

In Figures 7 and 8 it can be seen precisely every point where Vickers hardness was measured with micro-hardness-meter and its value.
Fig. 7 - Vickers hardness calculated in alloy particles a) inside sheath b) at brazed In600 TC

Fig. 8 – Vickers hardness determination inside In600 sheath

The observations made after experiments allowed us to conclude that whenever is possible, brazing alloy must be settled directly on piece inside a small reservoir situated on the solid part of base material (BM), and is connected with joint zone through a capillary channel therefore to be in contact with the joint zone as little time as possible. Adding material (AM) must reach joint area only when its fluidity is sufficient enough for penetrating small interstices and the temperature inside that is sufficient for a reasonable wetting. In some cases, piece’s configuration and its dimensions disallow the usage of this method, thus it is necessary to operate in the inferior zone of the permitted temperature range of 930 - 940 °C.
Conclusions

1. Simultaneous BNi-7 brazing behavior of the two types of materials from which the thermocouple sheath can be made – steel or Inconel 600 – shows major differences, these being emphasized by metallographic analysis.

2. Inconel 600 sheath shows obvious changes of structure at contact with BNi-7 alloy. These changes occur starting the temperature of 935 °C, and are materialized by diffusion and erosion at sheath-BNi-7 alloy interface.

3. At temperatures near 1000 °C, steel sheath shows structural changes only in the area of alloy reservoir, where the quantity of alloy is much bigger than the one from the length of the channel. Instead of this, at lower temperatures liquid state alloy does not properly wetting the steel and thus it occurs the possibility of not penetrating through capillarity in interstices and the joint cannot follow the imposed conditions.

4. At temperatures going towards the upper end of the range recommended for brazing BNi-7, close to 1000 °C or above this temperature Inconel sheath 600 suffers major changes. These changes are highlighted by both metallographic analysis and hardness values determined in those areas. In comparison, stainless steel sheath does not suffer these changes.

5. The method of placing the alloy in a tank laterally from the area to be assembled proves to have the most advantages and good results. In this way it can be determined by direct observation when the liquid alloy fills the interstice and forms the meniscus. Thus, the time of contact between the molten alloy and thermocouple sheath can be minimized. In this way, the negative impacts like erosion and sheath weakening can be reduced.

References

FATIGUE TESTING OF ZIRCALOY-4 CLADDING BY THERMO-MECHANICAL CYCLING

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ABSTRACT

This paper aims the fatigue testing by thermo-mechanical cycling of zircaloy-4 material used for CANDU nuclear fuel claddings. Testing was done on "C-ring" samples, sampled on the transversal direction of non-irradiated tubes.

In this paper is presented an experimental testing facility for investigations of fatigue cycling tests, designed and constructed in RATEN ICN.

The experimental facility is able to achieve mechanical cycling tests at various temperatures of interest (up to 400°C), with one cycle/second frequency. These parameters are registered and shown on the PC display. The piston movement can be controlled using a displacement transducer.

For each strain (1, 2 and 3%) of the sample corresponds a displacement of the piston.

Calibration of the transducer depending on the deformation is performed either by means of high temperatures strain gauges stamps, or by means of ANSYS Calculation Code.

To obtain calibration curves by means of ANSYS Calculation Code are required tests traction (ring-test) data.

Are achieved “durability curves” at different temperatures, up to 400°C. These highlight the fatigue behaviour of the fuel cladding material for CANDU bundle types (classic and SEU-43).

It can highlight the evolution of durability dependent on the number of cycles, temperature and deformation.

Key words: zircaloy-4, cycles fatigue testing

Introduction

CANDU type reactors produce economic electrical power by utilizing natural uranium fuel and structural materials having a low cross section for thermal neutrons, like zirconium alloys.

The UO₂ fuel is clad in Zircaloy-4 tubes sealed by welding of the end caps to form a fuel element. The claddings, having Zircaloy wear and spacer pads brazed one, are held together by Zircaloy-4 end plates to form a fuel bundle.

Though CANDU reactor is generally operated at steady high power, there is a trend towards periodic power reactor variations, load following, due to the grid requirements. During a power increase the cladding is plastically strained by the fuel. Under a subsequent power decreasing, the contact between the fuel and cladding is lost, the cladding can became oval by creep collapse, bending being the dominant state of stress. After a second power increasing the cladding is strained by the fuel and the plastic bending
cycle is completed. As a consequence Zircaloy–4 cladding may be subjected to plastic strain cycles that after a sufficient number of power cycles are responsible of low-cycle fatigue fracture.

The objectives of the paper were to simulate the load following operation conditions of Zircaloy-4 CANDU cladding using the low cycles fatigue tests [1,2,3].

**Experimental method**

The method consists of the symmetrical loading of the C-shaped samples under a cycling axial force inducing a maximum localized strain on the middle point of the inner surface of the tested ring. The cycling force shall be applied through the grips rod device coupled to an eccentric oscillating system. The total strain amplitude induced into the sample will be estimated by means of the calibration curve “Total Strain Amplitude – Rod Displacement”,

The testing conditions were [1,2]:
- Temperature: room temperature, 300°C, 350°C;
- Total Strain Amplitude: 1%, 2% si 3%;
- Cycle rate: 60 cycles / min;
- Test environment: air.

Test Ring, simulating the load following Zircaloy–4 sheaths conditions, was carried out at the Institute for Nuclear Research, Pitesti so as the mentioned testing conditions are satisfied.

The rings are cut from Zircaloy-4 CANDU tube types, by turning; a gap was split along the generatrix by milling and dimensional measured, (Figure 1).

![C-ring sample](image1)

**Fig.1. C-ring sample**

The experimental facility for fatigue testing under conditions of power cycling (Figure 2) of the Zircaloy-4 case was designed and developed in in 2002 at ICN.

![Experimental facility](image2)

**Fig. 2. The experimental facility for simulating the behaviour in power cycling conditions.**
The sample holders device (Figure 3) are two “yoke”-like pieces, one of which is fixed to the support beam of the testing chamber, the other being coupled with a crank gear system, necessary for cyclic movement.

![Sample holders](image)

**Fig. 3. Sample holders**

The calibration curves represent the dependency between the central deformation of the sample and the displacement of the piston, variables which can be measured with a strain gage and a displacement transducer. Every deformation (1%, 2% and 3%) is equivalent to a certain displacement of the piston which is measured by the transducer in millimetres (Figure 4.).

![Calibration curve](image)

**Fig. 4. Type of calibration curve**

For calibration curves determination, an alternative method is used: the finite element analyses in the ANSYS computer code [1,2,3].

Using this method, the calibration curves for the 1% ÷ 3% deformation domain were traced for “C-ring” samples. For this, a transversal cross-section of the specimen was modelled (Figure 5).

![Modelling of “C-ring” samples using ANSYS](image)

**Fig. 5. Modelling of “C-ring” samples using ANSYS**
The software code uses material parameters (yield strength, tensile strength and elastic deformation) determined through calculation using formulas:

\[
\varepsilon_{\text{real}} = \ln \left( 1 + \frac{\varepsilon_{\text{exp}}}{\sigma_{\text{exp}}} \right)
\]

\[
\sigma_{\text{real}} = \sigma_{\text{exp}} \cdot \left( 1 + \frac{\varepsilon_{\text{exp}}}{\sigma_{\text{exp}}} \right)
\]

where: \( \varepsilon_{\text{real}} \) is calculated deformation

\( \sigma_{\text{actual}} \) is calculated yield strength

To characterize the mechanical properties "ring tensile test" type samples are used (Figure 6).

![Fig. 6. "Ring tensile test" type samples](image)

The experimental values are determined from the traction curves (Figure 7) obtained after testing at room temperature.

![Fig. 7. Type of traction experimental curves for a Zircaloy-4 sample](image)

The mechanical parameters resulted from the calculated “real” curves are used for the ANSYS modelling.
Results and Comments

The fatigue lifetime of the tested sheaths, measured as number of cycles to rupture, are correlated as function of the total strain amplitude, Figure 8.

The dependence $N - \Delta \varepsilon$ follows the Coffin-Manson relationship:

$$N^{\alpha} \Delta \varepsilon = C$$  \hspace{1cm} (1)

where $N =$ number of cycles to rupture; $\Delta \varepsilon =$ total strain amplitude; $\alpha =$ material constant; $C =$ parameter measuring the material ductility to fracture.

The values of $\alpha$ and $C$ parameters of the tested tubes are determined from Figure 8.

Fig. 8. Type of Low cycle fatigue lifetime: $N - \Delta \varepsilon$

The low cycles fatigue data can be correlated with the micro- and macro results. For microstructural analysis of break surface, SEM (Figure 9) and metallographic (Figure 10) analysis can be performed.

Fig. 9. Type of SEM analysis of break surface
Conclusions

1. Low cycle fatigue tests are performed in order to simulate the load following conditions of the CANDU fuel using out – of – pile testing;
2. The experimental results are assessed by means of the Coffin–Manson law, dependence usually used to characterize the low cycle fatigue behavior of the metallic materials;

References

THE UPGRADED CONTROL AND INSTRUMENTATION SYSTEM OF C9 IRRADIATION DEVICE

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ABSTRACT

C9 Capsule is an irradiation device of TRIGA SSR, which was designed for nuclear fuel cycling testing. It simulates the load follow-up by the power reactor, i.e. CANDU 700MW operating with variable load. The irradiation tests in the power cycling conditions were conceived for complete characterization of the fuel behaviour. The irradiation conditions are similar to those found in nuclear power plant when it is operated in load following mode. The power cycling tests are performed by using an under flux moving system. The paper presents the upgraded control and instrumentation system of the C9 irradiation device, designed and manufactured in order to enhance the performance of this system for better surveillance and processing of the acquired experimental data.

Key words: C9 irradiation device, control and instrumentation system, TRIGA Reactor, nuclear fuel cycling

Introduction

C9 irradiation device consists of three systems [1]:

- The in-pile section with sample holder and adequate instrumentation in which the test samples are exposed to the radiation field;
- The primary and auxiliary circuits maintaining key parameters under control;
- The instrumentation system (C9 rack and control room PC) allowing critical parameters measurement and regulation.

In the C9 irradiation device the generated heat is transferred through the in pile section walls to the reactor cooling water.

Some important values for interesting parameters [2] are given in the following table:
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operation pressure (Mpa)</td>
<td>10.7</td>
</tr>
<tr>
<td>Coolant max. temperature (°C)</td>
<td>150-300</td>
</tr>
<tr>
<td>Fuel cladding max. temperature</td>
<td>325</td>
</tr>
<tr>
<td>Heat removal capacity (KW)</td>
<td>30</td>
</tr>
<tr>
<td>Water flow inside the pressure tube (l/h)</td>
<td>2</td>
</tr>
<tr>
<td>Water flow in calorimeter (m³/h)</td>
<td>3</td>
</tr>
<tr>
<td>Oxigen content in pressure tube(μg/kgH₂O)</td>
<td>10-50</td>
</tr>
<tr>
<td>Impurities content in pressure tube (mg/kgH₂O)</td>
<td>0.1-100</td>
</tr>
<tr>
<td>Fuel rod length (mm)</td>
<td>500</td>
</tr>
<tr>
<td>Maximum no. of fuel rods</td>
<td>1</td>
</tr>
<tr>
<td>Maximum fuel rod linear power with enriched uranium (W/cm)</td>
<td>600</td>
</tr>
</tbody>
</table>

In order to be able to cycle the fuel rod power an angular motion of the in pile section of the capsule located in a peripheral reactor channel is possible allowing position changing of the test section. The change of position can be done with a servomotor or manually [2].

**The upgraded control and instrumentation system**

The manufacture [3] of new control and instrumentation system of the C9 irradiation device is based on design theme prepared by TRIGA Reactor Department-Irradiation Devices in accordance with the rules and specific requirements for quality management, applied to the design of nuclear devices.

Main requirements for the system are:

1. Energy supply (220 / 380 Vac) for C9 devices: P2 pump (secondary cooling circuit of C9), heat resistances (from pressurizer and test section), electro-valves EV1 and EV2 (for test section separation from the rest of the circuit);
2. Optically and acoustically signaling of threshold overreach from C9 installation parameters (established in “Diagrama de Securitate”- Security Diagram);
3. Data transmission through RS-485 standard (between ADAM modules and PCs from reactor hall and reactor control room) and through Ethernet standard (between the master PC located in the control room and Reactor Server located in another room);
4. SCRAM signal provided for Reactor Protection System racks (RPS);
5. Measurement and conversion of analogue signals provided by device instrumentation into digital signals;
6. Automatic control of the servomotor from SDF (“Sistem Deplasare sub Flux”- under Flux Movement System) versus neutron flux from Reactor active zone. This kind of control must follow the cycling schema;
7. Monitoring, recording and displaying all the important signals of C9 Capsule on the PC monitor.

Bellow (see Figure 1) the C9 irradiation device bloc diagram is presented. It contains the two hydraulic racks, test section, SDF, calorimeter, P2 pump, power supplying rack, control computer and C9 rack. All signals are acquired with the ADAM modules (some of them are located in the ADAM distribution box, some inside of the C9 rack) and sent through RS-485 cable to main PC. The output signals are sent back in order to control the desired parameters (pumps, heat resistance, servo-motor, electro-valves).
The measurement and security actions system is an analogue-digital system which performs data acquisition and then processes and generates the security actions.

The system is composed of several process data acquisition modules (ADAM - Advantech Data Acquisition Modules) placed in two locations (in the distribution box near the hydraulic rack and in the C9 instrumentation rack). These perform the measurement of the analogue signals from the transducers of the installation, the conversion of the process values to physical units (e.g. temperature) and serial transmission to the central station through an RS-485 isolated serial numeric interface. The input signals come from thermocouples, neutron flux detectors, pressure and level gauge (analogue signals and low-high digital levels), gamma radiation detectors and limit-switches for the SDF safety stop. It contains the following modules:

- Two Advantech ADAM 4018+ modules for thermocouple measurement (analogue input);
- Three Advantech ADAM 4017 modules for pressure and level measurement (analogue input);
- One Advantech ADAM 4051 module for digital input measurement (low, high limits);
- Three Advantech ADAM 4021 module for analogue outputs required by heaters;
- Two Advantech ADAM 4060 module for digital outputs (security diagram implementation).
Front and rear view of measurement and security actions system are presented in Figure 2.

![Figure 2](image)

**Fig. 1. Front and rear view of C9 instrumentation and control rack**

The central station equipment is based on industrial processor technology which is able to monitor, control and coordinate the operations of selection, conversion and transmission for the whole system and is located in the Reactor control room. The central station ensures the interface between operator and process when the TRIGA Reactor is operated (also the control of the system can be made locally-inside the reactor hall- but only when the reactor is in shutdown mode).

Operator interface is provided by a main screen (see Figure 3). In the upper part the security diagram outputs (optically and acoustic signaling, SCRAM, servomotor stop, pressurizer heaters turn-off, test section heaters turn-off and P1 pump turning on or off) are displayed. The “Rearmare” button is used by the operator to acknowledge the actions taken by the software system. Below are displayed the actual values of some important parameters and their low and high thresholds. If any threshold is over-reach then the threshold-box (H1, H2, L1, and L2) will be displayed in red and it will flash and appropriated
actions required by the security diagram will be taken. At the bottom of the page, the controls (software buttons) for another supervising pages are located. Using them we can see:

- The signals (“Semnale”) coming from transducers, exactly how they are acquired;
- The trends for the meaningful signals (“Grafice 1” and “Grafice 2”);
- The temperature regulator (“Regulator temp.”) used for heat resistance of the pressurizer;
- The servomotor regulator (“Regulator motor”) which is used to move the test section (and SDF) according to the cycling schema of the irradiation campaign, watching the neutron flux as input and then moving the SDF to reach the desired power- the closer it gets to the reactor core the more power flux is generated on the probe mounted in the testing section.

An important feature of the C9 irradiation device is the fact that it is a moving device so it allows power cycling experiments to be accomplished. This feature is made possible with the help of the servomotor which moves the test section and is controlled by the software on the central PC. Here (see Figure 4) is the part of LabVIEW software which is called “Regulator motor” (Motor regulator). The operator can see:

- The current position of the SDF (“Pozitia curenta”);
- The on/off state of the two mechanically limit-switches (Lim1, Lim2);
- The neutron flux;
- The speed of the SDF (“Viteza curenta”).

The device can be driven versus flux or versus position- see the two boxes (“Regulator flux”-flux regulator and “Regulator pozitie”-position regulator). For example, the SDF can stay still for four hours (at 100% neutron power), then go down in three hours at 50% power, stay there for another two hours, go
up at 100% power in three hours and then (after a twelve hours cycle) start again the entire process for a number of predefined cycles required by the experiment or beneficiary.

Also, using the controls in the bottom box of this C9 control page, the operator can move the test section in the required position with different motor speed (see “Pozitionare SDF”-SDF positioning box).

![C9 Control diagram]

**Fig. 4. C9 control tab for motor regulator**

**Conclusions**

The upgraded control and instrumentation system of the C9 irradiation device enhances the performance of this system for better surveillance and processing the acquired experimental data.

Main improvements consist of:

1. Significant enhancements in human interface which is more intuitive, consistent and easy to use;
2. The addition of trend display (graphics) for main parameters allows better surveillance of experimental conditions;
3. Additional features for servo-motor controlling part of the software- multiple possibilities for operators to choose from;
4. Fast acknowledge of important events-especially regarding the security diagram-and access (only with password) to all the signals coming from the transducers;
5. On-line connection with central data acquisition system allows access to experimental data from other locations outside reactor control room in real-time;

References

[2] Marian Ionescu, Mirela Ancuta, Dan Georgescu, Mirel Mladin, Alex. Pulpă, Stefan Stoenescu, Power cycling test for B63 experimental fuel rod RI 10860/2015
APPLICATION OF NEUTRON ACTIVATION ANALYSIS METHOD ON BIOMONITORS FOR ASSESSING ENVIRONMENT QUALITY

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ABSTRACT

The purpose of this paper is to determine the chemical elements concentrations which affect both the population health and environment, such as toxic agents which are contained in air and are retained by precipitation inside the biomonitors vegetal tissues. The Neutron Activation Analysis is an analytic technique based on measuring the numbers and the energy of gamma radiation emitted by the radioactive isotopes produced in the sample matrix by irradiation with thermal neutrons in a nuclear reactor [1]. Usually, the samples are irradiated together with specific neutron flux monitors, duplicates and interest elements standards for a prior selected period of time inside the core of nuclear reactor. After the irradiation experiment and the specific radioactive decaying, can proceed measuring the gamma energies spectrum by using a high resolution detection system (HPGe – High Purity Germanium crystal) for gamma spectrometry and then assess the impact of the traced elements on population and environment.

Key words: Neutron Activation Analysis, biomonitor, HPGe

Introduction

In urban areas, air quality is strongly influenced by the many human activities. High population density, heavy traffic, domestic heating in winter and other industrial activities, are influencing concentrations of trace chemical dangerous elements in the atmosphere. Consequently, the population is exposed to potential adverse effects arising from changes in the composition of urban air. Thus, monitoring air quality has become one of the standard quality control procedures for assessing urban environment. The environment must be protected from pollutants to avoid its destruction and hence affecting the population in terms of health. What must be kept under control, especially nowadays, are massive heavy metal pollution from various industrial activity and beyond. For this reason we must have a fair and accurate assessment of these quantities of metals (how much is allowed to exist in the environment without affecting our health) and also qualitative assessment of these pollutants (what contains the air we breathe every day). There are many studies about atmospheric contamination but most of these have been limited because of the financial problems, i.e. high costs and the difficulty of carrying out extensive studies in terms of both time and space. To determine these dangerous chemical elements that adversely affect human health, we used an indicator namely moss plant as explained in references [2] and [3]. This plant has the ability to retain in the vegetative tissues precipitated chemical elements from the atmosphere because it lacks the cuticle that would stop the entry of elements within cells. Mosses were sampled and processed and the final form of dry residue obtained was introduced into polypropylene cartridges and then they were irradiated in the TRIGA ACPR rabbit. After the samples were activated in thermal neutron flux for a certain period of time, they were measured using a high resolution gamma-ray detection system.
with crystal detector and an electronic system (preamplifier, pulsating unit, polarization module, amplifier) and a multichannel analyzer provided with a genuine specific software. These devices have led to the highest achievement of qualitative analysis of the sample (which chemical elements are contained in the sample) and quantitative analysis (concentration of the traced elements) using neutron activation analysis k0 standardization. The results can be used to achieve a graphical representation of areas with a high degree of pollution and based on it important decisions can be made to reduce the pollution.

**Experimental**

Analytical steps of Neutron Activation Analysis are as follows:

- Phase I of analysis: sample preparation means in most cases just grinding (pulverization), homogenization, mass determination, roasting, packaging and also selecting the best analytical process and preparation standards if any.
- Phase II of analysis: irradiating the samples. For irradiating the samples we used TRIGA ACPR facility which is a high efficiency neutron source for applying neutron activation analysis (a neutron flux in the range from $10^{12}$ to $10^{14}$ neutron · cm$^{-2}$ · s$^{-1}$).
- Phase III of analysis: measurement, evaluation and calculation considering gamma spectrum and the concentrations level of the trace elements. The most widely used gamma spectrometer consist of semiconductor detectors based on germanium connected to a computer using as a multi-channel analyzer for assessing and calculating the spectrum.

**Phase I of analysis**

For the experimental part of the research work we have taken a number of 4 samples from different locations of Pitesti city as shown in the Table 1:

**Table 1. Samples identification for irradiation experiment conducted in TRIGA ACPR Reactor**

<table>
<thead>
<tr>
<th>Sample no.</th>
<th>Sampling location</th>
<th>Sample mass (g)</th>
<th>Sample ID</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>TRIGA Reactor stack</td>
<td>4470E-5</td>
<td>P1</td>
</tr>
<tr>
<td>2</td>
<td>Pitesti South Railway</td>
<td>6715E-5</td>
<td>P2</td>
</tr>
<tr>
<td>3</td>
<td>ARPECHIM Refinery</td>
<td>10884E-5</td>
<td>P3</td>
</tr>
<tr>
<td>4</td>
<td>ROLAST S.A. („GAVANA” area)</td>
<td>12113E-5</td>
<td>P4</td>
</tr>
</tbody>
</table>

After sampling, the mosses were stored in polyethylene bags at room temperature for 10 days after which they were weighed and then were placed in an oven at a temperature of 40°C. Measurements were performed regularly after which the mass was further dried in an oven until the weight of the mosses remained constant. The samples were then burned out in CTD 2-type oven at the temperature of 450°C until there was obtained a white ash. The reason of burning is to remove the water content from the vegetative tissue and because we need to have a homogenous composition of elements in the sample during irradiation experiment. Figure 1 indicates the variation of masses for all samples from initial state to the ash state. For measuring the value of thermal neutron flux inside the irradiation location gold thin foil was used (aluminum diluted gold 0.1%). This monitor foil was washed with alcohol and then dried at 40°C.
Phase II of analysis
The samples were inserted in polyethylene cartridges after being weighted and then they were placed inside the TRIGA ACPR rabbit together with the gold foil monitor. Table 2 shows the values for the reaction rates of the flux monitor reaction, $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ and thermal neutron flux values indicated by the monitor (obtained from the calculations).

Table 2. Thermal neutron flux measurements indicated by gold monitor inside the TRIGA ACPR rabbit (D-10 location of the core grid).

<table>
<thead>
<tr>
<th>No.</th>
<th>Tr(h)</th>
<th>$R$ (dis/(nucl*s))</th>
<th>Rcd</th>
<th>Thermal flux (n/cm$^2\cdot$s)</th>
<th>R(average) (dis/nuc/*s)</th>
<th>Dev std (n-1) (rate)</th>
<th>Average flux (n/cm$^2\cdot$s)</th>
<th>Devstd (n-1) (flux)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>23.55</td>
<td>1.7790E-10</td>
<td>2.10</td>
<td>9.432E+11</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>28.58</td>
<td>1.8020E-10</td>
<td>2.10</td>
<td>9.554E+11</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>53.58</td>
<td>1.7653E-10</td>
<td>2.10</td>
<td>9.359E+11</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>67.66</td>
<td>1.7679E-10</td>
<td>2.10</td>
<td>9.373E+11</td>
<td>1.7862E-10</td>
<td>1.4677E-12</td>
<td>9.470E+11</td>
<td>7.78E+09</td>
</tr>
<tr>
<td>5</td>
<td>145.83</td>
<td>1.7930E-10</td>
<td>2.10</td>
<td>9.506E+11</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>151.07</td>
<td>1.8030E-10</td>
<td>2.10</td>
<td>9.559E+11</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>163.7</td>
<td>1.7830E-10</td>
<td>2.10</td>
<td>9.453E+11</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>8</td>
<td>168.78</td>
<td>1.7960E-10</td>
<td>2.10</td>
<td>9.522E+11</td>
<td></td>
<td></td>
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</tbody>
</table>

Calculations resulting from irradiation showed an average thermal neutron flux of $9.47\times10^{11}$ neutrons*cm$^{-2}\cdot$s$^{-1}$).

**Fig. 1** Mosses mass trend between sampling and burning (ready to be irradiated)
Phase III of analysis
It should be noted that the irradiation time was too short and so it did not permit to activate any of heavy metal elements. So in 4142 seconds irradiation it were successfully activated isotopes with short lifetime like: $^{55}$Mn, $^{41}$K, $^{45}$Sc, $^{121}$Sb, $^{139}$La, $^{151}$Eu, $^{75}$As, $^{81}$Br, $^{58}$Fe. For every irradiated sample we conducted a set of 8 measurements for each.

Qualitative - quantitative measurements of the elements retained in the moss were done by using k0 standardization method of NAA. Accordingly to NAA k0-standardization method, one trace element concentration in a sample is obtained from the following equation detailed explained by reference [4] and [5]:

$$\rho_a (\mu g/g) = \left( \frac{N_p / t_c}{SDCW} \right)_{sp,m} \cdot \frac{1}{k_0,m} \cdot \left( \frac{G_{th,m} \cdot f + G_{e,m} \cdot Q_{0,m} (\alpha)}{G_{th,a} \cdot f + G_{e,a} \cdot Q_{0,a} (\alpha)} \right) \cdot \frac{1}{\varepsilon_{p,m}} \cdot 10^6$$

Where:

- $\rho_a$ - The concentration of the traced element "a" ( $\mu g/g$);
- $m$ - Co-irradiated monitor for neutron fluency monitoring;
- $N_p$ - Net peak area measured and corrected by pulse losses (dead time, real coincidence);
- $t_c$ - measuring time [s];
- $S$ - Saturation factor: $= 1 - \exp(-\lambda t_{irr})$, with $t_{irr}$ – irradiation time and $\lambda = (\ln 2)/T_{1/2}$ with $T_{1/2}$ - halftime;
- $D$ - Decay factor: $= 1 - \exp(-\lambda t_d)$, where $t_d$ – decay time (from the end of irradiation to the start of measurement);
- $C$ - Counting factor: $= [1 - \exp(-\lambda t_c)]/\lambda t_c$, correction for decay during measurement;
- $W$ - Sample mass [g];
- $A_{sp}$ - $= (N_p/t_c)/SDCw$, specific counting rate, with $w$ – monitor element mass [g];
- $k_{0,m}(a)$ - $k_0$ factor experimental calculated of the traced element "a" relative to “m” monitor, defined as:

$$k_{0,m}(a) = \left( \frac{M_a \theta_a \sigma_{0,a} \gamma_a}{M_m \theta_m \sigma_{0,m} \gamma_m} \right)$$

with $M$ – molar mass,

- $\theta$ – isotopic abundance, $\sigma_0$ – cross-section of (n,γ) reaction at 2200 m/s$^{-1}$ and $\gamma$ – absolute gamma-ray intensity;

- $G_{th}$ - thermal neutron self-shielding correction factor;
- $G_{e}$ - epithermal neutron self-shielding correction factor;
- $f$ - $= \Phi_{th}/\Phi_e$, thermal to epithermal neutron ratio; (for ACPR reactor f factor value at the time of the experiment was 17.12);

$$Q_0(\alpha) = 0.429\overline{E}_r^{\alpha} + 0.429/0.55^{2\alpha + 1}(2\alpha + 1)^{\alpha}$$

where $Q_0 = I_0 / \sigma_0$, and $I_0$ represents the resonance integral defined as:

$$I_0 = \int_{0.55eV}^{\infty} \sigma(E)dE / E$$

- $\overline{E}_r$ - Effective energy of the resonance expressed in eV;
- $\alpha$ - value for the distribution deviantion of the epithermal neutron compared with the ideal shape 1/E, approximated by a 1/E$^{1+\alpha}$ dependency; (for ACPR reactor f factor value at the time of the experiment was 0.01159);
- $\varepsilon_p$ - Detection efficiency for full energy peak;

**Results**
The results of the Neutron Activation Analysis k0-standardization obtained after measuring all the samples on HPGe detector are presented in the Figures 2 – 10.

**Bromine**

Long-term use of potassium bromide (or any of the salts of bromine) may lead to bromism. This state of central nervous system depression is caused by bromide moderate doses on the order of several grams to humans or other mammals. Ingesting bromine can also cause a rash on the skin similar to acne.

![Bromine concentration](image)

**Fig. 2. Bromine concentration in the samples**

**Antimony**

Antimony and many of its compounds are toxic and effects of antimony poisoning are similar to arsenic poisoning. Inhalation of antimony dust is very harmful and in some cases can be fatal; in small doses, antimony causes headache, dizziness and depression. Higher doses and prolonged skin contact may cause dermatitis; it can also cause liver and kidney damage, causing violent vomiting and lead to death within days.

![Antimony concentration](image)

**Fig. 3. Antimony concentration in the samples**

**Arsenic**

Arsenic and many of its compounds are generally very strong poisons. Many water sources near mines are contaminated with these poisons. European Union under Directive 67/548/EEC of International Agency for Research on Cancer recognizes arsenic and its compounds as being in the first group of carcinogens. Arsenic is responsible for the most cases of poisoning of all heavy metal. Arsenic is released into the environment through the process of smelting copper, zinc and lead as well as by producing chemicals or bottles. “Arsine” gas is a by-product that occurs following the occurrence of pesticides that contain arsenic. Arsenic can be found in water sources all over the world, involving crustaceans and fishes exposure. Other sources are paints, poisons for rats, fungicides and inhibitory substances of wood aging. Target organs are the blood, kidneys, central nervous system, digestive tract and skin.
Fig. 4. Arsenic concentration in the samples

**Europium**
There are clear indications that europium is in particular less toxic compared to other heavy metals. Europium dust presents a danger of fire (flammable) and explosion. Variation is represented in the following figure.

Fig. 5. Europium concentration in the samples

**Scandium**
Elemental scandium is considered non-toxic element. Average lethal dose for scandium chloride (III) for guinea pigs was determined at a value of 4 mg/kg for the inter-stomach area and 755 mg/kg for oral administration. In the light of these results, scandium compounds must be treated as having moderate toxicity.

Fig. 6. Scandium concentration in the samples

**Potassium**
Because of the powerful reactive nature of potassium, it must be handled with care, with skin and eye protection and preferably an explosion-resistant barrier between the user and metal. Ingesting large quantities of potassium can lead to hyperkalemia strongly influencing the cardiovascular system.
Iron

This heavy metal is of major importance, especially because of the iron based supplements in the diet that can poison young children acutely. Ingestion is the main way that the iron enters the body exposing the toxic effects because iron is rapidly absorbed in the gastrointestinal tract. Corrosive nature seems to increase the absorption of iron. Most overdoses are the result of wrong ferrous sulphate tablets ingestion by children because they are covered in red capsule or by adults on basic multivitamin preparations. Other sources of iron include drinking water, iron pipes and cooking vessels. Target organs are the liver, kidneys and cardiovascular system. Large amounts of ingested iron can cause excessive increase iron levels in the blood. High levels of free iron in the blood reacts with peroxides and produce free radicals, which are highly reactive and can attack DNA, proteins, lipids and other cellular compounds. Iron usually affects heart cells, liver and is causing adverse effects including coma, metabolic acidosis, shock, impaired liver functions, coagulopathy, long-term organ damage and even death. A dose of 20 mg and 60 mg of iron per kg mass is considered to be lethal.

Manganese

Manganese compounds are less toxic than other metal such as nickel and copper. However, manganese dust exposure limit value should not exceed 5 mg/m³ even for small periods of time because of its toxicity level. Manganese poisoning was coupled with disturbances in the body and locomotor function and with cognitive disorders. Manganese exposure among miners was associated with a form of neurodegeneration similar to Parkinson’s disease called “manganism”. High levels of exposure to manganese in drinking water have been associated with impaired intellectual and reducing the coefficient of intelligence at school children. Manganese occurs in people engaged in the production or processing of manganese alloys, in workers exposed to manganese and fungicides. In general, concentrations of manganese in the environment more that 5 mg Mn/m³ can cause specific symptoms.
**Lanthanum**

Lanthanum has a moderate level of toxicity and should be handled with care. Lanthanum produce hyperglycemia, lowers blood pressure, degeneration of the spleen and liver functions alteration.

**Conclusion**

The NAA-k0 method allows to trace a wide range of chemical elements which can be activated in a nuclear reactor. Solid and liquid samples can be studied with this method. The characterization of the biomonitors samples made in this paper represents a detailed work which can help to draw a map of the most polluted area from the Pitesti city. These results can also help in taking actions of prevent future development of industrial activities in these areas to avoid the increasing of pollution and by this affecting the humans and environment as well.

**Acknowledgements**

Special thanks to NAA personnel from TRIGA Reactor from ICN PITESTI for all the support brought in this experimental work.

**References**

[2] D. Popovic – “Trace elements and radionuclides in urban air monitored by moss and tree leaves”.
MECHANICAL PROPERTIES OF STRUCTURAL MATERIALS IN HLM

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ABSTRACT

The Generation IV nuclear systems are nowadays in the design stage, and this is one of the reasons of testing stage for candidate materials. The purpose of this paper is to present the tensile tests, for candidate materials. The studied tests are: on temperature of 500 °C in air, on mechanical testing machine Walter + Bie by using the furnace of the testing machine, and environmental molten lead using testing machine Instron, equipped with a lead testing device attached to it. Also the mechanical parameters will be determined on tensile strength and yield strength for steel 316L material to be used as candidate in achieving LFR reactor vessel type, and the microstructural analysis of surface breaking will be performed by electronic microscopy. The paper will present the main components, the operating procedure of the testing system, and the results of tensile tests in molten lead.

Introduction

Generation IV reactors are truly innovative technology. They will performed maximum security with economic competitiveness. The technology is based on passive safety aspects to ensure a "zero impact" outside the directly affected area, even for the most severe accidents scenarios. Some types of reactors use closed fuel cycles to 'burn' waste in the interior of the reactor, thereby reducing final quantities of radioactive waste and increasing resistance to nuclear proliferation. [1]

LFR pathway for fast reactors cooling (Cooled Lead Fast Reactor) is done through heavy liquid metals, such as lead (Pb) and alloys (LBE - Lead Bismuth Eutectic).

The favorable characteristics of heavy metals fluid used as a coolant for the reactor quick are reactive lower bound of a hypothetical loss of coolant, better barrier against gamma radiation and neutrons active, high solubility of the actinides in the coolant leading to critically minimize the potential for the molten core events and also not react with air and water, thus eliminating the possibility of a fire, does not require a moderator. [7]

It also allows operation at higher temperatures, improving the efficiency and feasibility of the process; relatively low melting point enables use of lower temperatures and reduced risk of uncontrollable freezing. In this propose the liquid metals are the best choice, and liquid metals of lead-based alloys are preferred in high temperature operating conditions. [1]
The objective of this paper is to present the steels candidate characteristics that can be used as structural materials for plants type LFR, and they will be submitted testing methods, with the results from this testing.

2. Steels candidate selection criteria and their properties.

The development of new materials, that can cover attacks of rapid corrosion / erosion on the operating conditions of the LFR and shown that austenitic stainless steels are candidates to be used for the reactor vessel, which is a low carbon content and operating temperature conditions relative low radiation and low fluence, while martensitic steels represent the feritico-martensitic steels candidate materials for fuel shield, which must withstand high radiation flows.

Among the criteria for selection of structural materials necessary for building the reactor mention increased operating temperature and corrosion resistance.

To obtain conclusive results of corrosion tests conducted for austenitic and feritic steels LFR on, they must be tested at operating temperatures of the primary system below 500 ° C, and for the cladding of the fuel to temperature less than 600 ° C.

Research and Development needs qualification consist of the following materials:
- Austenitic steels welded joints operating in pure lead at 400-450 ° C (low flow speed – under 1m / s);
- Martensitic steels and the welded joints feritico-operating in the pure lead (flowing at a rate of – under 2m / s) and under irradiation rapidly flow at a temperature of 400-500 ° C;
- Protective coating for the cladding of the fuel, operational molten lead (the flow - under 2m / s) and a stream of low irradiation at 400 ° C to 550-650 ° C;

As a candidate material to be used in reactors Gen. IV, mention steel structure type 316L (used as material for the basic structure of the reactor). [2]

3. Study on thermo-mechanical testing techniques applied to steels candidate for the construction and operation of reactors Gen. IV

Investigation of thermo-mechanical behavior of materials especially austenitic steel and martensitic - feritico steels, candidate to be used in reactors LFR, is one of the directions of utmost importance.

In special literature are presents also the conditions of testing. Prior to use materials for the construction of reactors, they have suppose to a lot of testing methods in order to increase the advantages and disadvantages. The tests are classified according to the test as follows:

3.1 The testing technique in dynamic mood
3.1.1 Mechanics tests fracture

The investigations of mechanics tests fracture were performed by three-point bending tests according to E1820-99 ASTM. For the determination of the resistance curves (J-R), the J- integral technique uses a single sample by the method of resistance to bending, charging and uncharging.

The difficulty in applying the technique to a single sample, is to determine the crack length of the sample, which is normally accessible by the technique of multiple samples. Through the technique of multi-samples may be determined and the relationship between the crack length and bending movement. The samples used for this type of test are taken from the table to the size 20x4x2mm and executed a "notch", root radius is about 1 mm deep. The outer surfaces of the samples are finished with sandpaper
grit 4000. The samples were subjected to fatigue by thermique cycle a rate crack length / width crack (A0/W = 0.5). [3]

Fig. 3.1.1 presents the force-displacement diagram of plastic deformation crack for a test run in LBE compared with air.

![Force-displacement diagram of plastic deformation crack for a test run in LBE compared with air](image)

**Fig. 3.1.1** Force-displacement diagram of plastic deformation crack for a test run in LBE compared with air [4]

### 3.1.2 Fatigue tests by thermique cycle

These tests were performed on 316L steel, frequency 20Hz standard samples and small samples, results are similar for both types of samples, such practice can be used to save small samples of the material.

Also samples were executed micro-hardness tests in the weld area and its immediate area. The temperature test was between 300 °C and T Tocam. There were also performed on this material dynamic stress tests using Charpy hammer.

Fatigue limit for these samples depends on the strength of the material, the direction of sampling, temperature stress and welding processes.

### 3.2 Testing Techniques in static mood

#### 3.2.1 Creep tests

This is the test for creep in the air and molten lead with oxygen to 10-6 (wt%) at a temperature of 650 °C, the mechanical pressure between 100 and 200 MPa, and time of 200h.

Tests were conducted in a test system designed to break through molten lead creep in static condition (CRISLA). The thermocouples are NiCr-Ni type, the inner protection layer is FeAl test capsule, the capsule volume is 900ml. The samples are measured before and after testing using a laser scanner the spatial resolution of 0.1μm and 0.1 mm along the vertical axis of the sample vertical sections of the samples are measured at four points at 90 degrees, sample dimensions are in accordance with DIN EN ISO 204; 2009 after removal of lead are cleaned with a solution of CH₃COOH - H₂O₂ to study fracture surfaces by scanning electron microscopy. [4]

In Fig. 3.2.1.1 are presented the creep speed variations in air and LBE.
3.2.2 Specific test of tensile

Mechanical behavior of structural materials in contact with molten lead alloys must take account of the mechanism of penetration of liquid along the grain boundary by perfect wetting or thickened liquid absorption to limit grain. During penetration of liquid into solid, liquid component changes limit grain surface appearance.

Tensile tests were performed on steel AISI 316L SCK-CEN, after exposure of the LBE loop LECOR the flow conditions of the test are the following: temperature 400 °C, $V_{\text{LBE}} = 1 \text{ m/s}$, the concentration of O$_2$ = 10/8 - 10 to 10 wt%. [6]

The test results are shown in the table below.

**Tabel 3.2.2.1 The result of specific test of tensile** [6]

<table>
<thead>
<tr>
<th>AISI 316L</th>
<th>UTS(Mpa)</th>
<th>R(0.2%)(Mpa)</th>
<th>A%</th>
<th>Z%</th>
</tr>
</thead>
<tbody>
<tr>
<td>AS(Val. Medie)</td>
<td>457±10</td>
<td>165±35</td>
<td>69±3</td>
<td>73±3</td>
</tr>
<tr>
<td>1500h(expunere)</td>
<td>477±9</td>
<td>183±26</td>
<td>68±3</td>
<td>72±5</td>
</tr>
<tr>
<td>4500h</td>
<td>455±5</td>
<td>185±19</td>
<td>57±14</td>
<td>66±1</td>
</tr>
</tbody>
</table>

It is observed significant changes in tensile properties for 316 by hardening its steel - feature as evidenced by increased mechanical strength and elongation decreased.

To test steel 316L was used; tensile tests were performed at a speed of deformation 10-3 to 10-7 s-1 and the maximum force of 20KN, according to ASTM E8-01 and E21-92. The sample is placed in a test device which sets introduced into the molten lead from the autoclave and the beam speed was 0.2 mm / min corresponding to a deformation speed of about 2.8x10-4sec-1. No used extensometer.

3.3 Microstructural analysis

Microstructural properties of candidate materials for LFR tested in molten lead alloy or pure molten lead can be determined by transmission electron microscopy investigations (TEM) and scanning electron microscopy (SEM).
These analyzes are obtained information for understanding the mechanisms of weakening and cleavage, state metallurgical materials, highlighting the ductile-brittle transition, breaking analysis of surfaces and how breaking tensile tests, creep and fracture mechanics.

4. Analysis of mechanical behavior of steels candidate after testing
4.1 Thermo-mechanical test facility in molten lead

Test facility in molten lead was designed and built at ICN Pitesti RATEN following subassemblies and major landmarks:
An overview of the installation is shown in Fig. 4.1.1:

![Fig. 4.1.1 Testing installation](image1)

The test system is the equipment with which the tensile tests were conducted as well as the creep. The equipment consists of the following components:
- ✓ Furnace;
- ✓ Gearmotor;
- ✓ Support crucible;
- ✓ Actuators, parts that are used for fixing the sample and its scope.

An overview of the testing system is shown in Fig. 4.1.2 (1. Gearmotor; 2. Actuators; 3. Support crucible; 4. Furnace metals)

![Fig. 4.1.2 Testing sistem](image2)

Furnace
The Furnace is designed to melt the lead and maintained at the maximum temperature of 600ºC. It will be purchased by the customer and will respect the dimensions and geometry of the sealing surface of the "Cover".
**Gearmotor**
The gearmotor is meant to provide the necessary mechanical strength to tensile test process. It consists of a stepper motor and a reduction gear 12 µm.

**Support crucible**
The support crucible for gearmotor is designed to support and immobilization of the Furnace cover thus possible to achieve tensile test sample.

**The actuators**
The actuators are designed to secure evidence and using gearmotor and support crucible to apply tension.

An overview of how fixing and stretching the sample is shown in Fig.4.1.3 (1.Rod; 2. Mechanical support mounting sample; 3.Cover; 4.Furnace ; 5.Sample)

**4.2 Tensile tests on steel AISI 316L**

To test the molten lead in the environment were used 3 sample made by lathing of AISI 316 steel rod as shown in Fig.4.2.1

![Fig. 4.2.1 Cubic test sample for traction, creep and LCD](image)

Test parameters were in accordance with the requirements of ASTM E8, draw rate of 0.5% / min without using extensometer for strain determination. Tensile tests on steel AISI 316L were conducted in static in molten lead at 500 °C, the test results are shown in the diagram sample tested for traction in Fig. 4.2.2 and Table 4.2.1, where: Rm = resistance fracture; R (0.2%) = yield strength; A = strict reduction in the area.

![Fig. 4.2.2 The diagram of the 3 sample tested for traction in molten Pb -T=500°C](image)

<table>
<thead>
<tr>
<th>Table 4.2.1 Test results of tensile in molten Pb, T=500°C</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Otel 316L</strong></td>
</tr>
<tr>
<td>Proba1</td>
</tr>
<tr>
<td>Proba2</td>
</tr>
<tr>
<td>Proba3</td>
</tr>
</tbody>
</table>
The tensile tests similar to those tested in the molten lead, were tested in the air Furnace at 500 °C, and the mechanical properties of the test chart determined are shown in Fig. 4.2.3.

![Tensile Test Diagram](image)

**Fig. 4.2.3** The diagram of tensile test in air - T=500°C

From the analysis of mechanical parameters determined following extensive testing at the same temperature (500 °C) don’t result significant differences in parameter values for tensile strength and yield strength.

### 4.3 Review of breaking surface

To analyze the fracture surface, the samples were cleaned of molten lead traces using a solution of acetic acid and hydrogen peroxide. Breaking surface was investigated by SEM electron microscope provided.

The appearance of the fracture surface is shown in Fig. 4.3.1, which is observed brittle-type fracture (red arrow) towards the edges of the sample type and ductile (blue arrow), cup-con way toward the center of the sample.

![Breaking Surface SEM](image)

**Fig. 4.3.1** Aspect of breaking surface using SEM technical at T=500°C in molten lead
5. Conclusions

This paper presents the tensile tests on temperature of 500 °C in air, on mechanical testing machine Walter + Bie by using the Furnace of the testing machine, and environmental molten lead using testing machine Instron, equipped with a lead testing device attached to it.

Also mechanical parameters were determined on tensile strength and yield strength for steel 316L material to be use as candidate in achieving LFR reactor vessel type.

After testing in average air temperature and molten lead at 500 °C, there was no major differences in parameter values for tensile strength and yield strength.

The values obtained for the resulting mechanical parameters the results are similar to the special literature and confirm the usefulness of this type of steel for the intended purpose.

Microstructural analysis of surface breaking was performed by electronic microscopy, the tearing-type fracture being performed at the interface between the contact surface between the sample and molten lead, ductile type in the center of the sample analyzed.

6. References

[1] www.iaea.com
[7] Iacopo Buongiorno, "Conceptual Design of a Lead Bismut Cooled Fast Reactor with In-Vessel Direct-Contact Steam Generation", ASSACHUSETTS INSTITUTE OF TECHNOLOGY (February 2001) (online);
SIMULATOR FOR CANDU600 FUEL HANDLING SYSTEM.
ENVIRONMENTAL IMPLICATIONS

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ABSTRACT

Personnel training are a main topic in the security and reliability of several industrial processes. The simulator is a physical device that reproduces real operation of a device used in a production process technology. Typically, a simulator is intended to train the operators to work properly with the real device in the production process, but simulators can be involved frequently in the research and evaluation of performance of human operators. Process simulation has a significant role in the training of operators of nuclear plants. To ensure the safe operation, protection of workers and the environment, of any nuclear power plant, the training of its operators in all operating modes of the plant is essential. A trained operator who can handle any emergency in a controlled manner, without panic, protecting equipment and personnel is an asset for a nuclear power plant.

Key words: Simulator, CANDU Fuel Handling, Environment

Introduction

Personnel training are a main topic in the security and reliability of several industrial processes. In order to fulfil this objective, it is necessary to develop distributed applications that allow simulation, interactive, with an interface similar to the real one. The simulator is a physical device that reproduces real operation of a device used in a production process technology. Typically, a simulator is intended to train the operators to work properly with the real device in the production process, but simulators can be involved frequently in the research and evaluation of performance of human operators. There are several approaches to accomplish this task:
- simulators which reproduce accurately elements from the control room;
- virtual simulators that use displays;
- hybrid simulators that use a combination of the two.
Simulators used to train nuclear power plant operators must conform to the domain normative - in detail - such as ANSI 3.5.
Process simulation has significant role in the training of operators of nuclear plants. To ensure the safe operation, protection of workers and the environment, of any nuclear power plant, the training of its operators in all operating modes of the plant is essential. A trained operator who can handle any emergency in a controlled manner, without panic, protecting equipment and personnel is an asset for a nuclear power plant.
Simulators in real time, i.e. “Full Scope”, play an essential role in extensive training of operators and staffs from a central analytical and functional simulator are no longer considered as suitable for training.
This paper presents case study of an experimental model for FHS independent simulator, "Full Scope" type, intended to train Fuel Handling System (FHS) operators in the control room of CANDU nuclear power plant 600 for Ram cylinders of Fuel Handling Machine Head.

The process computer. Development system.

The process computer is an OS9 machine, produced by the PEP Modular Computers, Germany, company, with VM42 motherboard equipped with CPU Motorola MC68040. The process interface from the structure of OS9-VME computer ensures the type and number of the signals of simulation for experimental model, being equipped with I/O required modules:

- interface of analogue outputs (A/O)
- interface of digital outputs (D/O)
- interface of analogue inputs (A/I)
- interface of digital inputs (D/I)

The interface of digital inputs (D/I) acquisition is constituted from two VMOD-2D modules (I/O industrial interfaces for VME bus) equipped with PB-DIN3 piggybacks which have 20 optoisolated channels (16 for entries of logical level and 4 for state and interruptions entries). The way of functioning of this piggyback is set in the phase of initialization by registration in general control register at required command word. The interface of digital inputs acquisition (D/I) allows realization of manual commands to the experimental simulator desk level.

The interface of digital outputs (D/O) is constituted from three VMOD-2D modules (I/O industrial interfaces for VME bus) equipped with PB-DOUT2 piggybacks which allow the process computer to signal at A and B desks, the status of experimental simulator.

The interface of analogue inputs (A/I) acquisition is constituted from a VMOD-2D module (I/O industrial interface for VME bus) equipped with PB-ADC3 piggybacks which allows realization of manual commands to the experimental simulator desks level.

The interface of analogue outputs (A/O) is constituted from two VMOD-2D modules (I/O industrial interfaces for VME bus) equipped with PB-DAC3 piggybacks which allows the process computer to produce analogical signals which shows information of simulated equipment (positions, pressures, temperatures etc.) to A and B desks.

The main programming environments used to develop the simulator software application of combustible manipulation system parts included in experimental model, are installed on calculation systems of type MS-Windows/ix86, the chosen topology implementing a system of open network type, concept which allows interconnection of different software/hardware platforms, between different development systems and VME system on which runs a system of operation OS9-68k (operation system like UNIX, with real time facilities, multi-user and multitasking):

ISaGRAF allows implementation of logical part of programs package, facilities designated to visualization / registration (Historian) of analogical and digital sizes involved in simulator process, administration and acquisition of data; the operator from the command desk having possibility to select the technological sizes group wanted in some moment of simulation process to be viewed under bargraphs or trend forms.

Acquisition tasks and system facilities of the simulator are developed using cross-computer Microware HAWK IDE (for developing C++ applications used on platforms based on MOTOROLA processors. Developing system provides visualization of technological sizes evolution under graphic form (engineering units), important part of experimental model, is hosted also of Development System.

Instructor interface include management mechanisms and fault scenarios development, interface which will use facilities offered by the graphic ISaGRAF environment.

Developing simulation algorithms

Simulation on computer of an existing physic system involves creation of his mathematical model. This kind of model is often obtained as a result of identification of the system or less frequent based on
structural analyze, if is possible. Anyway, not all models are obtained in this way. From this reason is important to notice that from a large set of mathematical models of various physic systems, a small subset can be separated. This subset is characterized through the fact that for his models exist values of parameters legal and official tested, often defined by its excellent data. This form results from an objective function which is an abstract mathematical formula, which must be met by the system to be moulded. This is the reason why this kind of models is named standard models. Being a reference, standard models play an important role in errors determination, especially in automatic control systems, dynamic metrology etc.

We used important resources for modelling the fragmented technological process in independent functional modules, projecting and testing instruments dedicated to some functions, as [1]:

- Simulation in real time of the machine behaviour when responds at human operator commands or at state changes of others machine parts as a result of other commands, mechanical connections or technological interlocks; presentation of appeared results after the functioning.
- The creation of databases, their population with information, administration, maintenance instruments and procedures of including obtained data at technological parameters calibrators and information exposure to human operator.
- Safety communication mechanisms between developed concurrent tasks under different programming environments facilitated on a side by the real time controller from software core of the system of operation OS9/68k and on the other side by ISaGRAF kernel, which has provided synchronization mechanisms between concurrent tasks and possibility of administration data under OS9 data models form.
- I/O administration for data acquisition functions in real time and verification instruments of data acquisition.
- Visualization/Registration, HMI, with local access or form distance for on-line programs and off-line.
- Procedures and rules of administration integrated for users and users groups, using OS9/68k operation system functions and application mechanisms.
- Manoeuvrings errors administration and software application maintenance.
- Defining technological JOBs and administration of their execution, for work mode with automated commands of experimental model.

**Projecting and implementing simulation algorithms for Ram cylinders B and Latch of Fuel Handling (F/H) Machine Head.**

Implicated Devices in simulation of Ram cylinders B and Latch functioning presents 2 types of signals:

- Analogue output with proportional evolution
- Analogue output with "saw tooth" evolution with pass through 0

**B Ram simulation.**

RAM B subroutine must ensure generating of analogical signals that simulates the B cylinder position when it moves or stay, in contact with different elements from the inside of the F/H Machine Head Magazine or inside the combustible channel [2]. Generating of analogical/digital signals respects algorithm described in figure 1.
Automation devices, at Latch RAM, that realize the leading, control of its position are simulated through AO signals and DI, and signing on operator panel is made through some DO signals (see figure 2).

<table>
<thead>
<tr>
<th>Device</th>
<th>SIM F/H</th>
<th>Indicator</th>
</tr>
</thead>
<tbody>
<tr>
<td>The coarse trimmer potentiometer RAM B main</td>
<td>AO_01</td>
<td>Voltmeter</td>
</tr>
<tr>
<td>The coarse trimmer potentiometer RAM B reserve</td>
<td>AO_02</td>
<td>Voltmeter</td>
</tr>
<tr>
<td>The fine trimmer potentiometer RAM B main</td>
<td>DI_02</td>
<td>DO_01</td>
</tr>
<tr>
<td>The fine trimmer potentiometer RAM B reserve</td>
<td>DI_03</td>
<td>DO_02</td>
</tr>
<tr>
<td>Advance RAM B</td>
<td>DI_04</td>
<td>DO_03</td>
</tr>
<tr>
<td>Retract RAM B</td>
<td>DI_05</td>
<td>DO_04</td>
</tr>
<tr>
<td>RAM B at high speed</td>
<td>RESET</td>
<td>DO_05</td>
</tr>
<tr>
<td>RAM B at medium speed</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RAM B at low speed</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

RAM B simulation subroutine needs some parameters: \( C_{\text{HMPOZ}}(0.3366V) \), \( F_{\text{HMPOZ}}(2.45V) \) = values in volt, read on the coarse potentiometer, and fine one, when RAM B can be found at Z position (HOME).

RAM B speed domains:

\[
\begin{align*}
\text{MINSPD} &= \text{minimum speed:} \\
\text{SLOW} &= 1.8 \text{ mm/s} \\
\text{MEDIUM} &= 10.9 \text{ mm/s} \\
\text{HIGH} &= 29.1 \text{ mm/s} \\
\text{MAXSPD} &= \text{maximum speed:} \\
\text{SLOW} &= 6.3 \text{ mm/s} \\
\text{MEDIUM} &= 18.2 \text{ mm/s} \\
\text{HIGH} &= 75.0 \text{ mm/s} \\
\text{SPDTM} &= \text{time to reach some speed} = 1.0 \text{ s} \\
\text{MAXRD} &= \text{maximum value read on the fine trimmer potentiometer} = 4.9 \text{ V} / 26 \text{ mm} \\
\text{Coarse potentiometer range} &= 0-4.9V(0-3620mm) \\
\text{Fine potentiometer range} &= 0-4.9V(0-26mm) \\
\text{B Ram Nominal Speed:} \\
\text{Slow} &= 5.2 \text{ mm/s} \\
\text{Medium} &= 15.2 \text{ mm/s} \\
\text{High} &= 51.7 \text{ mm/s}
\end{align*}
\]
This information must be found in the application database.
By positioning we understand moving (as a result of actuating some outputs) to a point (defined by a Set Point value).

The moving speed can be modified with the selection switch of Ram B moving speeds.
In case of Ram B simulation we will have electrical signals of ramp type (AI’s) in domain 0-5 V, with signal saturation at positioning on setpoint or at mechanical stop in the described situations and digital signals at levels 0 and 24 V [2].

**Fig. 2. Desk drawer – Ram B**

**Simulation operating under semiautomatic mode - RAM B**

a) Positioning control RAM B with coarse potentiometer to a specified position (A … Z) at low speed [3].
Insert the appropriate mnemonic from the alphanumeric keyboard [3].

It creates the initial conditions for:
- RAM B position > FINLIM = limit control with fine potentiometer
- RAM B position <corresponding setpoint value (advance)
- Verify program loop:
  Use coarse potentiometer for position indication;
  Set the digital outputs accordingly:
  - DO_01 = SET / RESET DO_02 =, i.e. the direction of movement of the RAM B is advance,
  - DO_03 and DO_04 = RESET / DO_05 = SET speed of the RAM B is slow selected (see figure 3).

**Fig. 3. Ram B advance at low speed**
b) Positioning control RAM B with coarse potentiometer to a specified position (A ... Z) with medium speed [3]:

- Is introduced mnemonic CRBM.
- Insert using the alphanumeric keyboard corresponding mnemonic (CRB *).
- It creates the initial conditions for:
  - the position of RAM B > FINLIM = limit control on fine potentiometer
  - RAM B position <setpoint value corresponding to the position (advance)

Verify loop program:
- Use coarse potentiometer for position indication;
- Set the digital outputs accordingly:
  DO_01 = SET / RESET DO_02 =, i.e. the direction of movement of the RAM B is advancing;
  DO_03 and DO_05 = RESET / DO_04 = SET speed of the RAM B is selected at MEDIUM value (see figure 4).

![Fig. 4. Ram B advance at medium speed](image)

c) Positioning control RAM B on coarse potentiometer to a specified position (A ... Z) at high speed [3].

Is introduced the appropriate positioning mnemonic from the alphanumeric keyboard (CRB *).

It creates the initial conditions for:
- the position of RAM B _> setpoint value corresponding to the position (retract)
- RAM B position > FINLIM = limit control on fine potentiometer.

Verify program loop:
- Use coarse potentiometer for position indication;
- Set the digital outputs accordingly:
  DO_01 = RESET / DO_02 = SET, that is the direction of movement of the RAM B is retract;
  DO_04 and DO_05 = RESET / DO_03 = SET, speed of RAM B HIGH value is selected (see figure 5).
Latch Ram simulation

Simulation subroutine of Ram Latch running generates electrical signals which characterizes positioning move of bolt cylinder - LATCH RAM to a certain position. Automation devices, Latch RAM afferent, which achieve the lead and its positioning control, are simulated through AO signals and DI, and flagging on operator panel is made through some DO signals [2].

<table>
<thead>
<tr>
<th>Device</th>
<th>SIM_SMC</th>
<th>Indicator</th>
</tr>
</thead>
<tbody>
<tr>
<td>The Latch Ram trimmer potentiometer main</td>
<td>AO_04</td>
<td>Voltmeter</td>
</tr>
<tr>
<td>The Latch Ram trimmer potentiometer reserve</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Advance Latch Ram</td>
<td>DI_12</td>
<td>DO_14</td>
</tr>
<tr>
<td>Retract Latch Ram</td>
<td>DI_13</td>
<td>DO_15</td>
</tr>
</tbody>
</table>

LATCH RAM subroutine use technological set of data memorized in database specific modules:
- CKTIME = completing time LATCH RAM=60 s;
- MINSPD = minimum speed = 0.89 mm/s;
- MAXSPD = maximum speed = 20.9 mm/s;
- SPDTM = time to reach some speed = 1.6 s;
- Max travel = maximum race drive = 34 mm;
- Nominal speed = nominal speed = 1.8 mm/s;
- TLL = tolerance to setpoints SPLL= 558 counts.

The domain of measuring of potentiometer for Z cylinder position measure is 0...4,9V in 10 rounds of potentiometer, same as 0...43, 7 mm in 10 rounds.

In case of simulation Ram Latch we will have electrical signals of ramp type (AI’s) in domain 0-5 V, with signal saturation at setpoint positioning or at mechanical stop in described situations and digital signals at 0 and 24 V levels [2]. Generating of analogical/digital signals respects algorithm described in figure 6.
Fig. 6. Simulation algorithm for Latch Ram signal

Simulation operating under semiautomatic mode - RAM LATCH

a) Check Latch movement in a given position [3]:
Is introduced using the alphanumeric keyboard corresponding mnemonic (CLL *)
Simulating that:
if Latch Ram position < corresponding setpoint (advance).
Check:
- DO_14 - set, the direction of travel is advance;
- Loop program is running;
if Latch Ram position > corresponding setpoint (retract).
Check:
- DO_15 - set, the direction of travel is retract;
- Loop program is running (see figure 7).

Fig. 7. Ram Latch advance
Conclusions

This paper can be considered as part of a series of activities intended for knowledge significant role of process simulators in training nuclear plant operators to ensure safe operation by training its operators in all operating modes of the plant.

Activities within a nuclear power plant must be correlated with protection of the environment so that the negative impact on his reduced to the minimum rates.

Operating a nuclear power creates risks. CANDU 600 type plants present additional risks arising from the use of natural uranium as fuel and heavy water as coolant. These features create a risk like that a CANDU 600 reactor could feel a violent deviation of power, which may cause a failure of the safety and release of radioactive substances into the environment. Therefore, to ensure the safe operation, protection of workers and the environment, training operators in all operating modes of the plant is essential.

Therefore, operators of plant should be well trained and able to handle any emergency in a controlled manner, without panic. This involves participation of operators in a variety of abnormal situations operating the plant in order to develop the skills necessary to operate the plant. These abnormal situations can be reproduced with simulators.

Additionally, by using simulators can analyze hypothetical events or occurred from other nuclear power plants in order to prepare personnel to promptly respond to that situation and to identify technical measures to be taken to prevent unwanted events.

References


CHARACTERIZATION TECHNIQUES OF ELECTRODES DEPOSITION MATERIALS FOR IONIZATION CHAMBER DETECTORS

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**University of Piteşti, Romania

ABSTRACT

The paper focuses on evaluating the deposits and support materials used in radiation detectors, by microstructure characterization techniques. Boron depositions on aluminium surface and colloidal graphite deposition on polyethylene support were experimentally obtained. Advanced examination techniques, such as Scanning Electron Microscopy (SEM) and Energy Dispersive X-Ray Spectrometry (EDS) were used for investigations. Because boron cannot be detected by EDS, its deposition was investigated using backscattered electron signals in SEM. The adherence, compactness, chemical composition and content of impurities are the main characteristics of graphite deposition used in radiation detectors. In order to obtain an accurate detection for all deposition types, the impurities content from the deposits has to be small. As result, the examined boron deposition, which was not dense, presented un-connected, non-adherent crystals with different shapes and sizes. The graphite deposition has proven to be adherent, continuous, compact, but not uniform in terms of thickness.

Key words: radiation detectors, boron, graphite, deposition

Introduction

Development of nuclear power imposed the development and improvement of detection systems and the Instrumentation and Control field had also to be adapted to the new generations of reactors. To improve the effectiveness of charges collection, which is one of the main tasks in designing an ionization chamber detector, theoretical studies were conducted, with the aim to determine the formula of the current collected by an ionization chamber, having a cylindrical-hemispherical geometry [1], [3].

Radiation detectors

The reactor instrumentation can be divided into two categories: systems within the core and systems outside the core [1], [2], [3].

The first category includes small detectors that provide detailed information on the neutron spectrum; they can be placed on a mobile system or at the centre of the core. This type of detectors can provide information continuously or at regular time intervals. From this category, the paper uses detectors having components with depositions in their composition and working in intense neutron fields - the "Boron Detector" (with boron deposition) and the "Fission chamber detector" (with uranium deposition).
In the second category, where the environmental conditions are less severe, detectors are larger and they answer to the neutron flux properties integrated throughout the core. From this category, the "Ionization chamber detector from intelligent detection assemblies used for surrounding monitoring" (with graphite deposition) is used. Materials such as aluminium and polyethylene were used as support material for boron and, respectively, for graphite deposition.

**Examination of boron deposition on aluminium layer**

To examine the boron depositions, both physical phenomena and problems induced by neutron irradiation of boron depositions on different layers need to be taken into account. One of the critical issues for the deposited boron in the case of neutron detectors is the minimization of the amount of impurities which interfere with the neutron detection by spreading the resultant reaction products of the nuclear reaction, produced at the neutron irradiation of boron depositions inside the detector [4], [6].

To make an accurate detection, the deposited boron amount must be maximized and the amount of impurities in the deposition must be as small as possible. Therefore the boron depositions will be examined in order to determine the impurity type chemical elements and their distribution in the deposition. Also, the submitted films must have a good adherence to the layer, although the design characteristics have not set special conditions for the films crystallography [5], [6].

The results consisted in obtaining an experimental colloidal graphite deposition on polyethylene support and in analyzing, together with the support materials, of the boron depositions on aluminium and graphite polyethylene support. Advanced analysis techniques, as Scanning Electron Microscopy (SEM) and Energy Dispersive X-ray Spectroscopy (EDS) were used [5], [6].

Because boron is not detectable by the EDS method, the electron backscatter signal (instead of using secondary electrons) was used for analysis. The chemical composition and the level of impurities from depositions and from support material were determined; an evaluation of the chemical element distributions on the deposition surface, using cartography maps, was performed.

The boron depositions were visually examined to observe the deposition color. The electronic microscopy examination were conducted in order to observe both the deposition and particles adherence and an energy dispersive X-ray spectroscopy was conducted to observe the impurities distribution on the surface with deposition.

Even if EDS analysis cannot reveal the boron, it can be used for observing the locations where boron is located. Because these regions are covered and do not provide a strong signal, boron may be detected in regions where other elements have minimum signal. After an EDS analysis a correlation can be made between the regions with minimal EDS signal and the regions with low gray level from the backscattered electron images.

The backscattered electron (BSE) images can achieve an atomic number contrast, so that elements with small atomic number, such as boron, will produce a very low gray level in the BSE image. Secondary electron images will provide information about adherence, shape and distribution of the crystallites present on the surface with boron deposition.

Figure 1 shows the aluminum cylinder used in the neutron detector. On the inner surface, the matted, dull, gray color boron deposition, can be observed.
Figure 2 presents a topographic SEM image of the deposition surface, showing that the deposition is not dense. The un-connected crystals can be observed, and the deposition consists of crystals which are probably non-adherent, with different shapes and sizes. This suggests that these crystals are composed by different chemical elements, which would mean a high impurities content.

Therefore, the visual field in Figure 3 was examined by X-ray spectroscopy, to determine their possible chemical nature. Since boron cannot be detected, all others chemical elements presented in the investigated field were identified.

Figure 4 shows the EDS spectrum obtained after the field investigation illustrated in Figure 3.

Table 1 shows the chemical elements which were present on the mentioned surface, except boron.
As seen from the spectrum in Figure 4 and Table 1, the following chemical elements were present on the surface: Aluminium, Oxygen, Silicon, Copper. The presence of aluminium in high volume shows that not all surface is covered with boron deposition, and there are regions not covered by the deposition. Aluminium can also occur from the sub-layer in the EDS analysis. The analysis shows that boron was not identified on certain areas, and silicon, in percentage of 17.7 %, is in the form of impurity deposit. Therefore, a mapping of the chemical elements present on the surface was performed.

**Table 1. Impurities present in the deposition [6]**

<table>
<thead>
<tr>
<th>No.</th>
<th>Chemical element</th>
<th>Atomic no.</th>
<th>Spectral line</th>
<th>Relative quant. (% mass)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Aluminium</td>
<td>13</td>
<td>K series</td>
<td>54.8</td>
</tr>
<tr>
<td>2</td>
<td>Silicium</td>
<td>14</td>
<td>K series</td>
<td>17.7</td>
</tr>
<tr>
<td>3</td>
<td>Copper</td>
<td>29</td>
<td>K series</td>
<td>0.4</td>
</tr>
<tr>
<td>4</td>
<td>Sulphur</td>
<td>16</td>
<td>K series</td>
<td>0.3</td>
</tr>
<tr>
<td>5</td>
<td>Magnesium</td>
<td>12</td>
<td>K series</td>
<td>0.1</td>
</tr>
<tr>
<td>6</td>
<td>Oxygen</td>
<td>8</td>
<td>K series</td>
<td>26.7</td>
</tr>
</tbody>
</table>

Figures 5, 6 and 7 show: secondary electron image of the investigated field, global EDS mapping for Al, Si, O and Mg [6].
Examination of graphite deposition on nonconductive polyethylene layer

The important characteristics of the boron depositions are the following: adherence, compactness, chemical composition and impurities content. In order to obtain an accurate detection, the impurities level for all depositions must be as small as possible. Therefore, the deposited graphite will be examined to determine the impurity type chemical elements of and their distribution in the deposition.[5], [6].

Figure 8 shows the graphite deposition on polyethylene layer. The polyethylene tubing was cut into two components on which the adherent, black, uniform color graphite deposition can be observed. The film adherence and compactness will be highlighted by electron microscopy.

Figure 9 shows the secondary electron image, at a relatively low magnification rate (x200), of the cylinder outer surface with graphite deposition. Note that on surface are present ditches in two parallel lines, which were originally produced during the manufacture of polyethylene support. The fact that they are visible shows that the deposition is not very thick and the ditches can help at a better adherence of the deposited graphite film.

Figure 10 shows the image of surface with high magnification rate: regions where the deposition thickness is high can be seen and rarely in some regions pores may be present in the deposition, in the initial ditches on the polyethylene. Still, they do not have any effect on the detection, due to their small number and size. High magnification images confirm that the layer is adherent, continuous and compact.

The image in Figure 11 shows the graphite layer thickness measurement, the measured value being 50.73 microns. The deposition was measured in four different points, and the measured thickness value had a variability between 8 and 100 microns (8.6μm, 10.56μm, 71.75μm, 50.73μm).
After SEM examination, the chemical composition of the deposition was determined and a mapping of the present chemical elements was made, the aim being to find out the nature and distribution of impurities in the graphite deposition.

Figure 12 shows the EDS spectrum obtained after investigating the analysis field, with a magnification rate of 500.

Table 2 shows the chemical elements which were present on the scanned surface.

<table>
<thead>
<tr>
<th>No.</th>
<th>Chemical element</th>
<th>Atomic no.</th>
<th>Spectral line</th>
<th>Rel. quantity (% mass)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Carbon</td>
<td>6</td>
<td>K series</td>
<td>25.1</td>
</tr>
<tr>
<td>2</td>
<td>Calcium</td>
<td>20</td>
<td>K series</td>
<td>0.3</td>
</tr>
<tr>
<td>3</td>
<td>Aluminium</td>
<td>13</td>
<td>K series</td>
<td>0.3</td>
</tr>
<tr>
<td>4</td>
<td>Silicium</td>
<td>14</td>
<td>K series</td>
<td>0.3</td>
</tr>
<tr>
<td>5</td>
<td>Iron</td>
<td>26</td>
<td>K series</td>
<td>0.2</td>
</tr>
<tr>
<td>6</td>
<td>Sulphur</td>
<td>16</td>
<td>K series</td>
<td>0.2</td>
</tr>
<tr>
<td>7</td>
<td>Sodium</td>
<td>12</td>
<td>K series</td>
<td>0.4</td>
</tr>
<tr>
<td>8</td>
<td>Oxygen</td>
<td>8</td>
<td>K series</td>
<td>73.2</td>
</tr>
</tbody>
</table>

Figure 13 shows the SEM image of the marked area and the mapping for the following chemical elements: C, Si, O, Ca, Na and Al.
From the two types of mappings, it can be seen that the chemical elements presented as impurities in small amounts are evenly distributed in the deposition and very likely they do not have a significant influence on the detection properties of the detector.

5. Conclusions

The paper has made an assessment of the capability to use potential microstructural characterization techniques for the materials or depositions on various support materials which are present in the composition of radiation detectors. These techniques have shown a good characterization of materials or depositions specific to radiation detectors.

The requirements of radiation detectors manufacturers specify that the support materials must have a high level of purity and their impurities must present a small neutron capture section, in order to maximize the Signal to Noise Ratio.

- It was found that boron deposition is not dense, is formed of non-adherent crystals which are not interconnected, having different shapes and sizes.
- The deposition contains large amounts of impurities of Silicon, Copper, Magnesium.
- Although the polyethylene could not be examined by electron microscopy because it is not conductive, there were no examination problems for the graphite deposition on polyethylene support.
- The graphite deposition was found to be adherent, continuous and compact, but with variable thickness.
- Very few pores were detected in the initial polyethylene ditches; they do not have any effect on the detection due to their small numbers and sizes.
- The deposition has a very small amount of impurities: Calcium, Aluminium, Silicon, Iron and Sodium.
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COMPUTERIZED TECHNIQUES FOR COLLECTING THE RADIOPROTECTION DATA

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ABSTRACT

An important component of a computerized radioprotection system is the module for the collection of the radioprotection data. The data collection can be made automatically from the measurement equipment or manually by the operators after they read the values measured by the mobile devices. Database systems are used for storing the data, they offer higher performances, more efficient data organization, ensure data integrity and controlled access to the data into a multiuser environment.

The experimental program for the automatic collection of the remote data transfers periodically, at programmable time intervals, data files from the fixed radiation monitoring stations to a centralized system for radioprotection data. For this is used the File Transfer Protocol (FTP). A Radiation Monitoring Equipment designed and assembled in the Electronics Department of ICN Pitesti was used as a data source for the testing of the experimental programs.

Key words: radioprotection, data collection

Introduction

In the nuclear field there are necessary complex measurements of the specific parameters. The results of the measurements are used for the control and the security of the nuclear facilities and also for the personnel and the environment protection.

The evolution of the systems for data measuring, monitoring and centralization is in accordance with the technology development in the electronics and computing areas.

The usage of computerized systems for acquisition, processing and storage offers essential advantages as: high capacity databases, flexibility, high reliability, high acquisition speed and processing power. Beside these performances, the computing systems offer the possibility of communication, which allows the connection to the local area or wide area computer networks, inside of national or international networks. This integration property constitutes an (essential) advantage of the modern systems.

The databases constitute an essential component of a modern radioprotection system. Their main functions are:

- data storage,
• data protection and security,
• data sorting and presentation.

The real-time data acquisition and processing systems are known mainly for their capacity of reaction to external requests as fast as possible. The main task of a real-time system is to survey a number of input signals and to produce a response or actions depending on the external conditions inside a predefined time interval.

**Techniques for the collection of the data**

*Techniques for the automatic collection of the data*

The techniques for the automatic collection of the data involve/imply the periodic transfer, at programmable time intervals, of the data files from the fixed radiation monitoring stations to a system for the centralization radioprotection data where the database for the data storage is installed.

For the data files transfer it is used the FTP (File Transfer Protocol) which allows the communication between two computers located in two different places/networks. The application takes the file containing the data recorded by the monitoring stations and convert it into its own table used by the database. This table is update each time the file is transferred to the central system, this way it contains the last data recorded by the monitoring stations [3].

The add or the removal of the monitoring station from the configuration is done dynamically. For this it is used a configuration file which contains the ID of the station from which the data is collected, the IP address of it, the information for the authentication of the user which is authorized to access the data the transfer type (binary or alphanumeric), the time interval at which the collection of data is made, the source and the destination paths of each file which is transferred. The configuration file may be manually edited with the parameters that are specific to each of the monitoring station [3], [1].

After the data files are transferred to the system for the centralization of the radioprotection data, the data inside them is saved into a database which allows their storage into a structured way and the query of them based on different criteria.

*Techniques for the manual collection of the data*

The radioprotection data obtained by reading of the indication the mobile measurement equipment and the data connected that are accessories to the radioprotection area are inserted manually by the operators in the system for the centralization of the radioprotection data.

The interaction of the operators with the system for the collection and the centralization of the radioprotection data is done through some forms inside Web pages. The main advantage of such approach is the flexibility of the usage – there is no need for the installation of a software component on the user’s computer and the software used for accessing the system is a general purpose one, respectively a Web browser [1].

The user interfaces were created in order to permit the access to the resources of the application. The authentication of the user is required for the usage of the application. After the successful authentication, the user is automatically redirected to the main page of the application which contains a multiple menu. Every article in the multiple menu link to a dedicated Web page. Depending on the category the user belongs, he has access to the forms for manual insert of the data or only the right to view the data from the system.
The architecture of the system for the collection and the centralization of the radioprotection data

The software architecture of the system (Figure 1) was developed starting from the usage of the modern operating systems, completed with the utility programs used for the development of the modules specific for each system.

The following characteristics which conferee the system capabilities for processing and presentation of the information are withdrawn [3]:

- configuration of the system during the runtime, which offers flexibility for the applications (passwords for the ensurance of the access control and the security of the information, the number of the inputs, the adresses of the monitoring stations);
- real-time collection of the data from the radiation monitoring systems;
- portability – the developed programs run on windows family platforms;
- off-line collection of the data through forms;
- the possibility to generate reports and statistics;
- real-time update of the data;
- security of the information is ensured by the avoidance of the unauthorized access, the update of the access rights, the integrity of the transmitted data;
- extract of the data depending on the authorization level;
- functioning of the system based on clearand intuitive menus;
- integration in computer networks and the internet.

The concept of the system for the collection and the centralization of the radioprotection data is based on the modular functioning of the specific programs and on the information exchange between them [2].
Fig. 1. Software architecture of the system for the collection and the centralization of the radioprotection data

Experimental programs for the collection of the data

Experimental programs for the automatic collection of the data and the storage into the database

The experimental program for the collection of the remote data was developed in Visual C++.net and uses the FTP (File Transfer Protocol) to transfer periodically, at programmable time intervals, the data files from the radiation monitoring stations to a system for the centralization of the radioprotection data where the database for storing of data is installed [3].

For a flexible utilization the program for the automatic collection uses a configuration file which contains the address of the computer from which the data is collected, the information for the authentication of the
authorized user, the type of the transfer the frequency of the data transfer, the source and the destination of each file that is transferred.

At the moment of the start of the start of the execution, the application reads from the configuration file the parameters of the communication, opens a connection to the monitoring station from which will collect the data, transfers all the files included in the configuration file and saves them then closes the connection; after the defined time interval it restarts the procedure for the transfer of the files [3].

The storage of the data from the data files into the database is done through some experimental programs for writing in the database that were developed in Perl language.

The programs are made of some program modules with different functions which are then called successively to insert the data contained in the transferred files into the database. The functions of the program modules are as follow [3]:
- Read of the configuration file
- Display the information messages
- Display the error messages
- Display the debug messages
- Check the status of the database
- Deletion of the files that were imported into the database

**Experimental programs for the manual collection of the data**

The user interaction with the System for the collection and centralization of the radioprotection data is made through a graphical interface in the form of Web pages [2].

For the control of the access to different modules of the application the following categories of application users were defined:
- regular user – is the user who has the right for accessing (visualization) the application data but not being allowed to make changes on them. He can view only the predefined reports inside the application;
- advanced user – is the user with extended rights who, beside the right to access the application has the right to record/modify/delete the data in the application and has also the right to view the reports;
- administrator of the application is the user with full right who, beside the rights grated to the advanced users, has the right to create other application users and to access the module for the configuration of the application.

For security reasons and in order to separate the roles of the users inside the application, the access to different modules of the application was restricted based on the category of the user.

The module for the administration of the users is accessible only to the application administrator and permits him to create new users, to change the category of the existing users or to delete user accounts. For this purpose, it was created a dedicated application page (named “Users Administration”) [3].

The module for the configuration of the System for the collection and the centralization of the radioprotection data is used for adding the monitoring stations, for changing the configuration parameters for the stations that are already registered with the system or for deletion of some monitoring stations from the system. The module for the configuration is accessible to the administrator of the application only and is accessed from the dedicated page (named “Configuration of the Application”) [3].
The module for the access to the radioprotection data allows different access types to the application data depending on the category the user belongs. The regular users (without special rights) have only access (visualization) rights to the application data without being able to make changes on them; the advanced users and the administrators beside the right to access the application data have the right to record/modify/delete the data in the application [3].

As a data source for the test of the experimental programs was used a Radiation Monitoring Equipment which was designed and assembled by the SXI Electronics Department of ICN Pitesti.

The experimental programs for the collection and the centralization of the radioprotection data were verified by running them on an experimental platform connected into the same local network with the Radiation Monitoring Equipment.

**Conclusions**

The need to implement computerized radioprotection systems is given by the need to ensure the control and the security of the nuclear facilities and also for the personnel and the environment protection.

An essential component of a modern computerized radioprotection system represents the centralized collection of the radioprotection data. It is also necessary to ensure the integrity and the consistence of the data and also the control of the access to data. To achieve this the use of the databases is the most indicated option.

There were identified two techniques for the collection of the data in the radioprotection systems the automatic collection of the data and the manual collection through the forms. The two techniques for the collection of the data were validated by the mean of the experimental programs that were developed.

**References**

INFLUENCE OF TEMPERATURE ON THE MECHANICAL PROPERTIES OF INCOLOY 800

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ABSTRACT

The purpose of this paper is to determine the mechanical properties of Incoloy 800 by tensile tests performed with the SSRT autoclave at operating temperatures and pressures of the steam generator of a CANDU nuclear power plants. The tests were performed in air, at room temperature and atmospheric pressure, and in demineralized water at 3100C and 100bar pressure, respectively 2600C and 50bar pressure. The strain rate was 1.67x10^{-4}s^{-1}.

The main mechanical properties of Incoloy 800 under these conditions were determined, i.e.: elastic strength, yield strength, ultimate tensile strength, necking and elongation. In the range 25-3100C, the alloy breaking is preponderant ductile: cut-off section is oriented at about 45° to the direction of applied load, the necking is very strong, the presence of the slide which is formed during the very strong plastic deformation. Increasing the temperature conducts to the diminishing of elastic strength and to the growth of the ultimate tensile strength and elongation.

Key words: Incoloy 800, mechanical properties, high temperature

Introduction

The steam generators represent the functional connection between primary and secondary circuit of a NPP, being very important in the operation of nuclear power plants [1]. The role of steam generator is to transform the cooling water from the secondary circuit in saturated steam using the heat received from the coolant of the primary circuit (heavy water). The heat transfer is achieved through heat exchange surface of the steam generator tubing. Because the walls of steam generator tubes represent the boundary between the circuit of heavy water (contaminated with radioactive corrosion products) and light water circuit (non-radioactive), the integrity of steam generator tubes in CANDU power plants is very important: any loss of the heavy water from the primary circuit involves contamination of the secondary side. This fact has economic damages due to the NPP stops operation while the tubes are changed. However, due to the severe operating conditions, degradation of the steam generator tubes has been reported in the case on many nuclear plants [1].

Thus, for the manufacture of steam generator tubes of CANDU nuclear power plant was chosen Incoloy 800 alloy, which has a very good thermal conductivity and a high corrosion resistance in high temperature corrosive environments. On the other hand, for a better management of coolant from primary circuit (heavy water) steam generator tubes have small diameters (15.9mm outside diameter, wall thickness1.13mm), being manufactured in accordance with ASME SB-407/2001 [3].
In this context, the knowledge of the mechanical properties of Incoloy 800 alloy under specific temperature and pressure allows the assessment of CANDU time life of steam generators in safe conditions.

**Experimental**

**Sample preparation**

The composition of Incoloy 800 is shown in Table 1 [5]. Tensile samples were prepared from Incoloy 800 bar, according to ASTM E8M-96 (Fig. 1).

<table>
<thead>
<tr>
<th>Elements</th>
<th>Ni weight%</th>
<th>Cr</th>
<th>Fe min.39.5</th>
<th>C max</th>
<th>Mn max</th>
<th>S max.</th>
<th>Si max.</th>
<th>Cu Max.</th>
<th>Al 0.15-</th>
<th>Ti 0.15-</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>30.0-35.0</td>
<td>19.0-23.0</td>
<td></td>
<td>0.10</td>
<td>1.50</td>
<td>0.015</td>
<td>1.0</td>
<td>0.75</td>
<td>0.60</td>
<td>0.60</td>
</tr>
</tbody>
</table>

![Table 1 The chemical composition of the Incoloy 800 alloy (%) [5]](image)

**Fig. 1 Example of a test specimen used for testing in the autoclave SSRT**

**Tensile tests**

The tensile samples were tested in air, at room temperature, respectively in aqueous environment (demineralized water vapor) at 260°C or 310°C (the working temperatures of the steam generator in the secondary and primary sides). The strain rate was 1.67x10^-4 s^-1 (displacement rate of pull rod of 250μm/min). Table 2 systematizes the testing conditions used in our tensile tests.

<table>
<thead>
<tr>
<th>Temperature (°C)</th>
<th>Pressure (bar)</th>
<th>Testing media</th>
<th>Strain rate (s^-1)</th>
</tr>
</thead>
<tbody>
<tr>
<td>25</td>
<td>Atmospheric pressure</td>
<td>Demineralized water</td>
<td>1.67x10^-4</td>
</tr>
<tr>
<td>260</td>
<td>50</td>
<td>Demineralized water</td>
<td>1.67x10^-4</td>
</tr>
<tr>
<td>310</td>
<td>100</td>
<td>Demineralized water</td>
<td>1.67x10^-4</td>
</tr>
</tbody>
</table>

The specimens were tested using an autoclave type SSRT 50KN, manufactured by CORMET Finland. This equipment gives the possibility to carry out the traction test in corrosive environments and high temperature, to measure the rate of cracking in corrosive environments and high temperature and the study of cyclic fatigue behaviour of metals. The SSRT autoclave is presented in Figure 2. Figure 3 shows a detail with clamping device corresponding to the tensile sample.
Results and Discussions

Tested specimens were analyzed macroscopically (Fig. 4). It was observed that the elongation of the specimens increases with the test temperature; the break section is oriented at approximately 45° of direction of applied stress; the necking is very pronounced; the slip lines formed during the strong plastic deformation are present. All of these observations indicate a ductile fracture.

The stress-elongation curves corresponding to the tested specimens are presented in Figure 5. The mechanical properties of Incoloy 800 obtained by tensile tests are given in Table 3. As expected, the temperature increasing conducts to a decrease of the yield strength ($\sigma_{0.2\%}$), the ultimate tensile strength (UTS) and, respectively, at an increasing of the elongation. This fact is in agreement with the mechanical behaviour of Incoloy 800 hot-rolled rod in air, at elevated temperature [4].
Table 3  
Tensile properties of Incoloy 800 tested in demineralised water function of testing temperature

<table>
<thead>
<tr>
<th>Testing Temperature</th>
<th>Yield Strength – $\sigma_{0.2%}$ (MPa)</th>
<th>Ultimate Tensile Strength – UTS (MPa)</th>
<th>Elongation (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Room – 25°C</td>
<td>345</td>
<td>590</td>
<td>11.7</td>
</tr>
<tr>
<td>260°C</td>
<td>285</td>
<td>518</td>
<td>12.3</td>
</tr>
<tr>
<td>310°C</td>
<td>260</td>
<td>512</td>
<td>12.5</td>
</tr>
</tbody>
</table>

Conclusions

In the temperature range of 25-310°C, the breaking of Incoloy 800 is preponderate ductile. The raising of temperature conducts to the decreasing of yield strength, ultimate tensile strength and necking, respectively the increasing of elongation. At room temperature, 260°C and 310°C the values of yield strength obtained in our testing conditions are: 345 MPa, 285 MPa and 260MPa, respectively. At the same temperatures, corresponding values to the ultimate tensile strength are: 590MPa, 518 MPa, respectively 512MPa

References

[3] *** – „Assessment and management of ageing of major nuclear power plant components important to safety: Steam generators”, IAEA-TECODC 981, IAEA, VIENNA, noiembrie 1997, ISSN 1011-4289
METHOD AND EQUIPMENT FOR TREATING WASTE WATER RESULTING FROM THE TECHNOLOGICAL TESTING PROCESSES OF NPP EQUIPMENT

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ABSTRACT

Modern methods and technologies coupled together with advanced equipment for treating residual substances resulted from technological processes are mandatory measures for all industrial facilities. The correct management of the used working agents and of the all wastes resulted from the different technological process (preparation, use, collection, neutralization, discharge) is intended to reduce up to removal of their potential negative impact on the environment. The high pressure and temperature testing stands from INR intended for functional testing of nuclear components (fuel bundles, fuelling machines, etc.) were included in these measures since the use of oils, demineralized water chemically treated, greases, etc. This paper is focused on the method and equipment used at INR Pitesti in the chemical treatment of demineralized waters, as well as the equipment for collecting, neutralizing and discharging them after use.

Key words: demineralised water, environment management, waste waters

Introduction

The development, into an accelerated rhythm, of the industry, demographic explosion followed by the urbanization, destruction of some large forest area, degradation of some great rivers, release in the atmosphere of huge amount of foul gases, determined, in the last dozens of years, a significant disturbance of the environment. Lack of the flop-out measures, until deletion, of the itemized causes, will determine an irreversible disturbance of the environment balance, with inestimable consequences, on long and average term.

At this stage of the human civilization, we still assist at situation as the safety regulations and laws of the environment are flagrant broken, as the technical – economic interests are coming first to the detriment of „pure life conditions” [1].

RATEN-INR Pitești is developing the complex research, design, fabrication, control and functional testing activities of the equipment for the nuclear area, operation of the nuclear plant and the equipment and material testing, activities that can result in waste with a possible impact on environment – at small scale.
The Out of Reactor Testing Department (TAR) from RATEN - Institute for Nuclear Research, has as endowment a series of installations to control and functional testing of NPP equipment for CANDU 600 type reactors.

The testing installations are operating at high pressure and temperature conditions, using as working agents: oils, demineralised chemically treated water, vaseline, etc.

Application of some modern methods and technologies, doubling by using of some efficient equipment for treating waste resulted from technological processes in installations are mandatory measures for the RATEN - INR Pitesti, also.

Because of the Environment Management is one of the major priority, and the environment performance has an increased meaning for all interested internal and external parties, the Board of Institute decided development and implementation of an Environment Management System, according to the SR EN ISO 14001/2005 requests. Complying with these requests will result in decreasing, until deletion, of the negative impact of the waste on the environment.

**The preparation installation of water as working agent for testing rigs**

The NPP components testing rigs (nuclear fuel, Fuelling Machines, etc.), represent an installations and equipment assembly capable to provide operation parameters (pressures, temperatures, flow rates) similar to those from reactor. The testing rigs are using demineralized, outgassed and chemically treated water as working agent, and complying with the testing request with: pH of 9.5 -10.5; a maximum conductivity of 30 µS/cm; the Oxygen remised in water = max. 200 ppb; content of un-dissolved solid slurry = max.0.2 mg/l [3].

The water preparation is carried out in the Water Preparation Station of the Out of Reactor Department (Fig 1).

To obtain the physical-chemical parameters of water (pH, Oxygen concentration, particulates, etc.) the substances used are:
- Lithium hydroxide;
- Hydrazine.

![Diagram of the outgassed and chemically treated demineralised System](image)

**Fig. 1 The outgassed and chemically treated demineralised System**
Where:
RAD - Demineralised water tank;
P - Pump;
CRR - Oxidation-reduction column;
CRI - Ion-exchange resin column;
RADD - Demineralised and outgassed water tank;
PD - Metering pump;
PR - Recirculating pump;
RH - Hydrazine collector.

Before chemical treatment, the water passing through complex ion-exchange filters. Then, complying with the procedure instructions, the water – poisonous substance mixture is carried out to obtain the specified parameters.

The thermo-hydraulic circuits of the Out of Reactor testing stands are closed, with round circulation, carrying-out a continuous transfer between water treatment station and thermo-hydraulic loops during testing, the water coming back through the return pipeline to the RADD [4].

After use, the industrial effluents are resulted. These are collected in the headers placed close by the testing installation. There are taking place the treating process in order to its neutralizations (pH < 8.5, hydrazine = 0). In each stage of the technological process, the water samples are taken off and Analysis Bulletins are issued.

Thus, the water is analysed for the following conditions:
- at the acceptance time;
- after treating in special vessels from outgassed station;
- during testing process, at time period well-established;
- before and after neutralization, in the collecting headers.

After neutralization, the water is circulated to the INR Water-purification Plant, where the new Analysis Bulletins are issued after water is checked-out. The neutral water is finally discharged into environment, without any detrimental impact on it [2].

The component parts of the waste waters collecting and neutralization plant
The effluents resulted from water supply system of the Out of Reactor Department testing rigs contain Lithium hydride with concentration appropriate to a pH of 10.3 ± 0.2 and Hydrazine.

The Safety Regulations specific for the effluents discharge resulted from technological process impose a series of quality requests for the water following discharging to environment. From this point of view, the residual water from testing rigs exploitation have to comply with a Hydrazine concentration according to Standards and a pH between ranges 6.5 – 8.5. Because of this reason, these waters have to be chemically treated in order to complete Hydrazine deletion and to bring the pH between acceptance limits.

For the residual waters treating resulted in the technological NPP components testing process, have been required fabrication of a taking-over and circulating system to their collecting into drainage basin.

The components of the collecting and neutralization plant have been placed close by the testing rigs.

Have carried out:
- Two collecting and treating drainage basins for residual water;
- Separation of the meteoric water and industrial lines;
- Lines and component parts for effluents.

![Fig.2. The air discharge and effluents discharge/collecting pipes from testing rigs – partial view](image)

Have also provided the required equipment for barbotage, to decant and circulate the water to the INR Water-purification Plant [2].

**The water treating method having hydrazine content**

The chosen method is consisting in Hydrazine oxidation using an iodine solution into a slightly alkaline or neutral agent, at the room temperature. In this condition, are taking place the following chemical reactions:

\[
2I_2 + N_2H_4 \rightarrow 4HI + N_2
\]

\[
HI + LiOH \rightarrow LiI + H_2O
\]

The materials required for neutralization process are:
- The iodine solution of 0.5 N;
- Reacting substances and laboratory equipment for Hydrazine proportioning and the pH measuring.
The plant is operating as follows:

- The water is removed from installation into the treating tank, by gravity circulation through the pipelines;
- After about 15 minutes of bubbling the sample (1 l) is taken off to determine the Hydrazine content and the pH.
- After analysing the sample in the laboratory, is determined the iodine solution volume required to destruction the Hydrazine, taking into account the water volume being treated.
- Is bubbling with compressed air, adding, under stirring, the specified iodine solution volume.
- The stirring is going on during 30 minutes.
- The stirring stops, is taking off the sample for pH determination.
- In case of the solution pH decreased to the values less than 6, is adding under stirring, Sodium hydrate solution 1M, bringing the pH into range of 7.5 – 8.5.
- The stirring stops and is taken off sample for Hydrazine content determination.
- In case of the complete Hydrazine destruction did not took place, the above operations are iterating, adding an iodine solution volume established until getting the desired pH [5].

**The water volumes balance-sheet from technological processes of NPP components testing**

The amount of demineralized and chemically treated water volume supplied in the thermo-hydraulic circuits before starting a technological process, have to be equal to total of volumes collecting as effluents in the tanks, after testing process.

At the preparation stages of the testing installation and during tests, are necessary the water controlled discharges, for air discharge thermo hydraulic pipelines.

At the same time appear the losses due to leakiness, also. These are not significant quantitative. The building floor where the testing equipment are installed is all concrete and shows geometries (slopes, angles of surfaces) allowing the entire their collection.

In this case, the relation between the water volumes is as follows:

\[(1) \quad \Sigma v_i = \Sigma v_c \]

where:
\( \Sigma v_i = \) total of the inlet water volumes;
\( \Sigma v_c = \) total of the water volumes from collecting tanks after testing process.

\[\Sigma v_c = V_p + V_{\text{leak}}\]

where:
\( V_p = \) the water volume used in the technological process;
\( V_{\text{leak}} = \) the water volume resulted after air discharge pipelines plus the loss volume due to leakiness.

After measurements, if the relation (1) is complying with there, is not uncontrolled loss of the residual water.

**Conclusions**

- The discharging, admission and transfer pipelines, of the residual waters as well as the collecting tank of them provide the tightness required through entire thermo hydraulic circuit lay-out, thus deleting the water loss;
- The new collecting system of the waste waters existing in the Out of Reactor Department provided and still does the entire collection of them.
- The solution chosen for the residual water treating / neutralization is simple and efficient;
- The impact over environment, in this case, is vanishing.

References

[3] “The F/M Head Testing Rig”- Project no. 3-627
C-14 ANALYSIS IN RADIOACTIVE WASTE BY COMBUSTION AND DIGESTION TECHNIQUES

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ABSTRACT

Carbon-14 is a long lived radionuclide (half life of 5730 years) present in almost all radioactive waste streams generated by a CANDU nuclear power plant. It is a pure beta emitter that decays to $^{14}$N by emitting low energy beta-radiation with an average energy of 49.5keV and a maximum energy of 156keV. Before the beta radiation of $^{14}$C can be measured from radioactive waste liquid scintillation counting (LSC), the samples must be transformed in a stable, clear and homogeneous solution. Two methods were tested for carbon-14 recovery and analysis in radioactive wastes from nuclear power plants. The combustion process is a simple automatic method of sample preparation, in which all carbon isotopes, including $^{14}$C are oxidized to gaseous carbon dioxide that is subsequently trapped in form of carbonate in a column filled with a carbon dioxide absorbent. The microwave digestion is the method wherein the samples are transformed totally or partially in liquid phase depending on the sample matrix using adequate digestion reagents. The samples were counted with a normal and low level count mode liquid scintillation counter Tri-Carb3110TR. The tests performed on the simulated radwaste showed a $^{14}$C recovery of 90% by combustion and higher than 75% by microwave digestion method.

Introduction

In general, a 600 MW CANDU reactor in normal operation conditions produces the following C$^{14}$ and tritium containing radwaste types:
- spent ion exchange resins;
- spent filters;
- solid waste in the form of paper, protective clothing, rags;
- gaseous radioactive waste discharged by general ventilation;
- reactor’s condensation ventilation chambers;
- molecular sieves.
Table 1 shows the concentrations of C-14 in different types of reactors and standardized on GWe and per operating year.
Table 1. C-14 production in different types of nuclear reactors

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>C-14 production speed (TBq/GWe · year)</th>
<th>Heat transport system</th>
<th>Moderator</th>
<th>Fuel</th>
<th>Reactor’s structural materials</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>CANDU</td>
<td>Less than 1%</td>
<td>20</td>
<td>1,1</td>
<td>1,9</td>
<td>23</td>
<td></td>
</tr>
<tr>
<td>PWR</td>
<td>Less than 1%</td>
<td>0,4</td>
<td>0,6</td>
<td>1,4</td>
<td>2,4</td>
<td></td>
</tr>
<tr>
<td>BWR</td>
<td>Less than 1%</td>
<td>0,4</td>
<td>0,6</td>
<td>2,3</td>
<td>3,3</td>
<td></td>
</tr>
<tr>
<td>GRAFIT</td>
<td>0,3</td>
<td>8,0</td>
<td>2,2</td>
<td>1,8</td>
<td>12,3</td>
<td></td>
</tr>
</tbody>
</table>

Quantities of C-14 produced in reactor systems will be found distributed in solid, liquid and gaseous radioactive waste, as shown in Table 2.

Table 2. Distribution of C-14 on the types of waste by the type of reactor

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>C-14 production speed (TBq/GWe · year)</th>
<th>Accumulating speed of C-14 inside gaseous radwaste (TBq/GWe · year)</th>
<th>Accumulating speed of C-14 inside solid radwaste (TBq/GWe · year)</th>
<th>Accumulating speed of C-14 inside fuel (TBq/GWe · year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CANDU</td>
<td>23</td>
<td>13,1</td>
<td>8,8</td>
<td>1,1</td>
</tr>
<tr>
<td>PWR</td>
<td>2,4</td>
<td>0,4</td>
<td>1,4</td>
<td>0,6</td>
</tr>
<tr>
<td>BWR</td>
<td>3,3</td>
<td>0,4</td>
<td>2,3</td>
<td>0,6</td>
</tr>
<tr>
<td>GRAFIT</td>
<td>12,3</td>
<td>0,3</td>
<td>9,8</td>
<td>2,2</td>
</tr>
</tbody>
</table>

As shown, the production of C-14 in CANDU reactors in much higher compared to other types of reactors, as the main production reaction is the thermal neutron capture of O-17 and O-17 amount in heavy water moderator is much greater than in light water moderator of PWR and BWR reactors (O-17 in heavy water has a concentration of 0.058% compared to 0.037% from light water). Studies carried out in different types of reactors have led to the evaluation of C-14 concentration in radioactive waste. Exact distribution of C-14 in various radwaste is unknown [1]. However, it is estimated that the greater part of the C-14 production is discharged in the atmosphere, because this isotope is found in the form of CO₂ and only a small part of C-14 production remains in the fuel covering gas cleaning system.

Analysis methods of soft-beta emmitter radionuclides determination from radwaste

1. “307 PerkinElmer Sample Oxidizer” combustion facility description for solid radwaste sample calcination

“307 PerkinElmer Sample Oxidizer” combustion facility [1] is used in the radiochemistry laboratory for calcinating radwaste samples, to be subsequently measured using 3110TR liquid scintillation Tricarb analyzer. The sample is burned in an oxygen stream resulting CO₂ which is collected in a bottle used to measure the C-14 concentration using liquid scintillation analyzer. The following requirements regarding the preparation of samples for carrying the combustion operations and spectral analysis must be accomplished:

- the sample must be representative in terms of soft-beta emmitters radionuclides content;
- the sample weight must be correlated with the technical operating parameters of the combustion facility (calcination time, the volumes of chemicals used in the combustion process etc.);
- physical state of samples to be analyzed: in case of wet samples or samples which requires a slow calcination it must be increased the burning level by using ignition substances.

Combustion facility consists of:
1. Combustion system;
2. The system of tritium;
3. The system for collecting C-14;
4. The system of nitrogen and oxygen;
5. The programming facility.

Figure 1 is a schematic diagram used for combustion facility used to calcinate the radwaste samples to determine the C-14 content with liquid scintillation.

![Model 307 PerkinElmer Sample Oxidizer combustion facility scheme](image)

where:
- **M** – power supply switch;
- **N** – switch on/off the facility;
- **P** – combustion regulator: fixes the period of ignition for a complete combustion cycle;
- **S** – pressure regulators: pressure system setting for providing oxygen, air/nitrogen and water installation;
- **T** – toggle switch: enables tightness testing of facility circuits;
- **U** – toggle switch: indicates the system water pressure and allows testing on leakage;
- **V** – Switch: allows manual adjustment of combustion time, reducing the time of combustion in cases when the sample is completely ignited before the combustion reset time has expired;
- **X** – reagent metering pumps located on the three chemicals reservoirs of the facility;
- **Z** – reservoirs filling reactive holes;
- **W** – switch: controls the sample calcination time reset, allowing the combustion time to be prolonged in situation when the initial reset time is not enough for sample calcination.

2. **TRICARB3110TR liquid scintillation analyzer description to determine the soft beta emitting radionuclides from radwaste.**

Liquid scintillation measurement of radioactivity is based on the interaction of beta emitting radionuclides and the scintillator which is the component of the scintillator cocktail. Scintillator converts the ionizing radiation emitted by radionuclides into light quantum (scintillation) [2]. By placing the vials with samples analyzed in the closed detection chamber, the device measures the intensity of the photon. Emitted photons can be detected by a photomultiplier which generates a pulse signal and amplifies the voltage proportional to the number of incident photons. Liquid scintillators are organic chemicals that allow the spectral beta radiation energy to be converted in scintillation light, which can further detected by the analyzer. Due to the sample preparation, a part of the energy of disintegration can be absorbed into the
sample. Also a part of the emitted photons can be absorbed into the sample so they can no longer be detected. This phenomenon is called extinction (quench). Quench can be defined as the phenomenon of mitigation of sample photon emission, both due to chemical agents (chemical quench) [3], which reduces the efficiency of beta energy transfer from the solvent to the scintillator and due to the presence of color agents (color quench) [4], which absorb the photons emitted by the scintillator inside the sample. Figure 2 shows a block diagram of a Tricarb 3110TR liquid scintillation analyzer.

Fig. 2. Block diagram of TriCarb 3110TR liquid scintillator analyser

The description of BERGHOF SPEEDWAVE 4 digestion system for solid radwaste samples mineralization

Microwave field acid digestion involves decomposition of a solid material in the presence of a digestion appropriate reagent, in a microwave permeable container and resistant to high temperatures. The process involves the heating of the sample through direct absorption of microwaves by digestion reagents, solutions which normally contain ionic components. The reagents used in the digestion process can contain nitric acid, hydrochloric acid, hydrofluoric acid, phosphoric acid, sulfuric acid and other mixtures of acids, in the presence of hydrogen peroxide and water. Efficiency of digestion depends directly on the temperature at which decomposition takes place in all amount of the sample to be analyzed, on the reagent mixture and the period of time necessary to finish the digestion. Dak-model 100/4 digestion facility [5] (Figure 3) has the caps and pressure vessels made of teflon, being protected by a protective high pressure resistant ceramic layer. Magnetron frequency is 2450MHz and the microwave power can reach up to 1450W.

Fig. 3. BERGHOF SpeedWave4 digestion system
Experimental

For determining the content of C-14 in the liquid samples resulted from the mineralization process, analyzes were performed by liquid scintillation spectrometry in Low Activity counting mode using Tri-Carb 3110 / TR analyzer at a counting rate between 50 and 500 CPM and measurement time of minimum 1000 minutes.

Experimental results on samples of solid waste paper type

Table 4. The results obtained on radwaste solid paper type

<table>
<thead>
<tr>
<th></th>
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<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>H1(1)</td>
<td>1000</td>
<td>302.48</td>
<td>185</td>
<td>5.04</td>
<td>3.08</td>
<td>61.1</td>
<td>0.63</td>
</tr>
<tr>
<td>H2(1)</td>
<td>1000</td>
<td>302.48</td>
<td>186</td>
<td>5.04</td>
<td>3.10</td>
<td>61.4</td>
<td>0.63</td>
</tr>
<tr>
<td>H3(4)</td>
<td>1000</td>
<td>302.48</td>
<td>206</td>
<td>5.04</td>
<td>3.43</td>
<td>68.0</td>
<td>0.60</td>
</tr>
<tr>
<td>H4(4)</td>
<td>1000</td>
<td>302.48</td>
<td>206</td>
<td>5.04</td>
<td>3.43</td>
<td>68.0</td>
<td>0.60</td>
</tr>
<tr>
<td>H5(7)</td>
<td>1000</td>
<td>302.48</td>
<td>210</td>
<td>5.04</td>
<td>3.50</td>
<td>69.3</td>
<td>0.60</td>
</tr>
<tr>
<td>H6(7)</td>
<td>1000</td>
<td>302.48</td>
<td>209</td>
<td>5.04</td>
<td>3.48</td>
<td>69.0</td>
<td>0.60</td>
</tr>
</tbody>
</table>

where:
*%2s is the percentage of uncertainty (with confidence limits 95%) for gross counting value. As can be seen in Table 4, the average recovery of C-14 was approximately 66.13%. Table 5 presents the results obtained after measuring samples taken from the second bubble flask by liquid scintillation spectrometry.

Table 5. The results of the samples taken from the two bubble flusks

<table>
<thead>
<tr>
<th>Index probă</th>
<th>Activitate măsurată [DPM/proba]</th>
<th>Activitate măsurată [Bq/ml]</th>
</tr>
</thead>
<tbody>
<tr>
<td>B1</td>
<td>18</td>
<td>0.03</td>
</tr>
<tr>
<td>B2</td>
<td>25</td>
<td>0.401</td>
</tr>
<tr>
<td>B(fond)</td>
<td>17</td>
<td>0.028</td>
</tr>
</tbody>
</table>

For an accurate assessment of C-14 experimentally determined activity besides the radiochemical analysis of the solutions from the two bubble flusks, was carried out and a decontamination of the teflon tubes by washing with a solution of citric acid, so that at the end of the process, in each tube to be samples (D), ready to be measured by the liquid scintillation spectrometry, to detect a possible contamination of digestion tube with C-14. Table 6 presents the samples preparation mode using scintillation liquids and the measurement results made by spectral analysis. Those samples were taken from the washing solution.

Table 6. C-14 activity in samples taken from the washing solutions

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>D1(H)</td>
<td>1</td>
<td>19</td>
<td>20</td>
<td>0.33</td>
</tr>
<tr>
<td>D4(H)</td>
<td>1</td>
<td>19</td>
<td>24</td>
<td>0.4</td>
</tr>
<tr>
<td>D7(H)</td>
<td>1</td>
<td>19</td>
<td>23</td>
<td>0.38</td>
</tr>
</tbody>
</table>

As can be seen from Tables 5 and 6, a part of C-14 with which the samples of paper were marked on, remained on the digestion tube’s walls (about 0.4 Bq / ml) and another part was released in the Off
digester gas system, explaining the low recovery efficiency of C-14 in the experiments performed with such a solid matrix type. When taking into account the C-14 activity values measured in the solutions taken from the bubble flusk vessels solutions and the one from the decontamination of digestion vessels, the recovery efficiency increases to about 81.43% (Table 7).

Table 7. The recovery efficiency of C-14 after correction with the measured activity inside the bubble flusks and inside the decontamination solution of digestion vessels

<table>
<thead>
<tr>
<th>Sample ID</th>
<th>Reference activity [Bq/ml]</th>
<th>Measured activity* [Bq/ml]</th>
<th>Recovery efficiency %</th>
</tr>
</thead>
<tbody>
<tr>
<td>H1(1)</td>
<td>5.04</td>
<td>3.81</td>
<td>75.5</td>
</tr>
<tr>
<td>H2(1)</td>
<td>5.04</td>
<td>3.83</td>
<td>75.9</td>
</tr>
<tr>
<td>H3(4)</td>
<td>5.04</td>
<td>4.23</td>
<td>83.9</td>
</tr>
<tr>
<td>H4(4)</td>
<td>5.04</td>
<td>4.23</td>
<td>83.9</td>
</tr>
<tr>
<td>H5(7)</td>
<td>5.04</td>
<td>4.28</td>
<td>84.9</td>
</tr>
<tr>
<td>H6(7)</td>
<td>5.04</td>
<td>4.26</td>
<td>84.5</td>
</tr>
</tbody>
</table>

* activity measured in solutions taken from the digestion tubes corrected with losses inside the bubble flusks and on the digestion tube’s walls.

Experimental results obtained on solid radwaste textile type

Table 8. Results of samples of solid radwaste textile type

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>T1(1)</td>
<td>1000</td>
<td>261.82</td>
<td>212</td>
<td>4.36</td>
<td>3.53</td>
<td>80.9</td>
<td>0.60</td>
</tr>
<tr>
<td>T2(1)</td>
<td>1000</td>
<td>261.82</td>
<td>214</td>
<td>4.36</td>
<td>3.56</td>
<td>81.7</td>
<td>0.59</td>
</tr>
<tr>
<td>T3(4)</td>
<td>1000</td>
<td>261.82</td>
<td>195</td>
<td>4.36</td>
<td>3.25</td>
<td>74.4</td>
<td>0.62</td>
</tr>
<tr>
<td>T4(4)</td>
<td>1000</td>
<td>261.82</td>
<td>212</td>
<td>4.36</td>
<td>3.53</td>
<td>80.9</td>
<td>0.59</td>
</tr>
<tr>
<td>T5(7)</td>
<td>1000</td>
<td>261.82</td>
<td>212</td>
<td>4.36</td>
<td>3.53</td>
<td>80.9</td>
<td>0.60</td>
</tr>
<tr>
<td>T6(7)</td>
<td>1000</td>
<td>261.82</td>
<td>208</td>
<td>4.36</td>
<td>3.46</td>
<td>79.4</td>
<td>0.60</td>
</tr>
</tbody>
</table>

where:

*% 2s is the percentage of uncertainty (with confidence limits 95%) for gross counting value.

Table 9. The recovery efficiency of C-14 after correction with the activity measured in bubble flusks and inside the digestion vessels decontamination solution

<table>
<thead>
<tr>
<th>Sample ID</th>
<th>Reference activity [Bq/ml]</th>
<th>Measured activity* [Bq/ml]</th>
<th>Recovery efficiency %</th>
</tr>
</thead>
<tbody>
<tr>
<td>T1(1)</td>
<td>5.04</td>
<td>4.34</td>
<td>86.1</td>
</tr>
<tr>
<td>T2(1)</td>
<td>5.04</td>
<td>4.37</td>
<td>86.7</td>
</tr>
<tr>
<td>T3(4)</td>
<td>5.04</td>
<td>4.05</td>
<td>80.3</td>
</tr>
<tr>
<td>T4(4)</td>
<td>5.04</td>
<td>4.33</td>
<td>85.9</td>
</tr>
<tr>
<td>T5(7)</td>
<td>5.04</td>
<td>4.35</td>
<td>86.3</td>
</tr>
<tr>
<td>T6(7)</td>
<td>5.04</td>
<td>4.28</td>
<td>84.9</td>
</tr>
</tbody>
</table>

* activity measured in solutions taken from the digestion tubes corrected with the losses from bubble flusks and from the digestion tubes
C-14 Determination from solid radwaste using liquid scintillation spectrometry

After experiments conducted on different types of solid radwaste and presented in the previous section, the samples resulted from the combustion process were analyzed by liquid scintillation spectrometry in order to determine the C-14 content. The samples obtained are measured by spectrometry using the TriCarb3110TR liquid scintillation analyzer.

Results obtained on paper type solid radwaste

In Table 10 are shown the results obtained on paper type solid radwaste, labelled with A3, A4, A5 and in Table 11 are shown the results obtained on samples of paper type solid radwaste, labelled with A6, A7 and A8.

Table 10. Results obtained on paper type solid radwaste

<table>
<thead>
<tr>
<th>ID</th>
<th>Reference activity (DPM)</th>
<th>tSIE</th>
<th>Calculated activity (DPM)</th>
<th>Activity* (DPM)</th>
<th>Recovery efficiency (%)</th>
<th>Relative deviation of activity value (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>A3</td>
<td>2444.5</td>
<td>251.29</td>
<td>2407</td>
<td>2354</td>
<td>96.2</td>
<td>3.7</td>
</tr>
<tr>
<td>A4</td>
<td>2444.5</td>
<td>176.76</td>
<td>2354</td>
<td>2302.2</td>
<td>94.1</td>
<td>3.7</td>
</tr>
<tr>
<td>A5</td>
<td>2444.5</td>
<td>130.28</td>
<td>2410</td>
<td>2356</td>
<td>96.3</td>
<td>3.5</td>
</tr>
</tbody>
</table>

Table 11. Results obtained on paper type solid radwaste

<table>
<thead>
<tr>
<th>ID</th>
<th>Reference activity (DPM)</th>
<th>tSIE</th>
<th>Calculated activity (DPM)</th>
<th>Activity* (DPM)</th>
<th>Recovery efficiency (%)</th>
<th>Relative deviation of activity value (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>A6</td>
<td>2444.5</td>
<td>176.56</td>
<td>2407</td>
<td>2354</td>
<td>96.2</td>
<td>3.7</td>
</tr>
<tr>
<td>A7</td>
<td>2444.5</td>
<td>186.64</td>
<td>2436</td>
<td>2382.4</td>
<td>97.4</td>
<td>2.54</td>
</tr>
<tr>
<td>A8</td>
<td>2444.5</td>
<td>204.39</td>
<td>2424</td>
<td>2370.6</td>
<td>97</td>
<td>3.02</td>
</tr>
</tbody>
</table>

Analyzing the results shown in Table 10 and Table 11 it is noted that for all the measured samples the outcome is a high recovery rate, ranging between 94.1% and 97.4%, measurement errors being between 2.54% and 3.7%, very low values for liquid scintillation spectrometry. After the experiments and measurements carried out it was found that for a amount of fabric between 0.2-0.3g, the optimum complete combustion time is 1 minute and the optimal volumes of scintillators CarboSorb-E and PermaFluorE are 8 ml respectively, 12 ml per sample.

Table 12. Results obtained on textile type solid radwaste

<table>
<thead>
<tr>
<th>ID</th>
<th>Reference activity (DPM)</th>
<th>Calculated activity (DPM)</th>
<th>Activity* (DPM)</th>
<th>Recovery efficiency (%)</th>
<th>Relative deviation of activity value (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>A9</td>
<td>977.8</td>
<td>1088</td>
<td>1064</td>
<td>99.9</td>
<td>8.8</td>
</tr>
<tr>
<td>A10</td>
<td>1955.6</td>
<td>2090</td>
<td>2044</td>
<td>99.9</td>
<td>4.5</td>
</tr>
<tr>
<td>A11</td>
<td>2933.4</td>
<td>3024</td>
<td>2957</td>
<td>99.9</td>
<td>0.8</td>
</tr>
<tr>
<td>A12</td>
<td>3911.2</td>
<td>3809</td>
<td>3724</td>
<td>95.2</td>
<td>4.7</td>
</tr>
</tbody>
</table>

By analyzing the results from the table 12 it is noted that for all the measured samples the outcome is a very high recovery rate (99.9%), the calculate measurement errors being between 0.8% and 8.8%, very low values for liquid scintillation spectrometry.
Experimental results obtained on oil type samples

Table 13. Results obtained on oil type liquid radwaste

<table>
<thead>
<tr>
<th>Index probă</th>
<th>Timp de măsurare [min.]</th>
<th>Activitate referință [DPM/ml]</th>
<th>Activitate măsurată [DPM/ml]</th>
<th>Randament recuperare [%]</th>
<th>*% 2s</th>
</tr>
</thead>
<tbody>
<tr>
<td>U1(1)</td>
<td>1000</td>
<td>179.87</td>
<td>114</td>
<td>63.2</td>
<td>0.86</td>
</tr>
<tr>
<td>U2(1)</td>
<td>1000</td>
<td>179.87</td>
<td>115</td>
<td>63.9</td>
<td>0.86</td>
</tr>
<tr>
<td>U3(1)</td>
<td>1000</td>
<td>179.87</td>
<td>114</td>
<td>63.3</td>
<td>0.85</td>
</tr>
<tr>
<td>U4(4)</td>
<td>1000</td>
<td>179.79</td>
<td>144</td>
<td>80.0</td>
<td>0.76</td>
</tr>
</tbody>
</table>

where: *% 2s is the percentage of uncertainty (for a time confidence 95%) for gross counting value. In order to assess the efficiency of the method used in the experiments for determining the activity of radionuclide C-14, the results obtained from the analysis with liquid scintillation of samples taken from digestion solution, from the solution from bubble flask no. 2 and from the washing solution used inside the digestion vessels, it were reported relative to the mean activity value from all three samples of radioactive oil processed inside the acid digestion facility in a microwave field. Thus, following the calculations made and shown in Table 14, the recovery efficiency of C-14 activity from radioactively contaminated oil is 82.9%.

Table 14. The recovery efficiency of C-14 from oil samples by acid digestion

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>179.88</td>
<td>128.88</td>
<td>24</td>
<td>0.37</td>
<td>153.25</td>
<td>85.19</td>
</tr>
</tbody>
</table>

Conclusion

The proposed method for the characterization of solid radwaste in terms of C-14 content by acid digestion comprises the following steps:

✓ mineralization of the radwaste samples in order to recover the radionuclide of interest in a solution which can be characterized by liquid scintillation spectral analysis;
✓ C-14 radionuclide activity measurement using TriCarb3110TR liquid scintillation analyser;

The proposed method for the characterization of radioactive waste in terms of solid content C-14 by combustion comprises the following steps:

✓ Radwaste sample calcination and the recovery of C-14 in the form of carbon dioxide in the presence of liquid scintillator;
✓ Measurement of C-14 radionuclide concentrations using the Tricarb 3110TR liquid scintillation analyzer

References

THE BURNUP INFLUENCE ON THE INTEREST PARAMETERS OF A CANDU LATTICE WITH NATURAL AND SLIGHTLY ENRICHED URANIUM

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ABSTRACT

The paper goal is to find out the burnup influence on lattice parameters of interest when nuclear fuels with Natural Uranium (NU) and Slightly Enriched Uranium (SEU) are used. The considered lattice parameters consist of the infinite multiplication factor and the isotopic contribution to the fission power for some of the major actinides such as Uranium and Plutonium. The widely spread transport equation solver computer code WIMS (Winfrith Improved Multigroup Scheme) was used. The working configurations correspond both to fresh and to irradiated nuclear fuel up to the discharge burnup from a generic CANDU power reactor. Three Uranium enrichments were used: 0.72 w% U235 (corresponding to the Natural Uranium), 1%w U235 and 2%w U235 corresponding to the SEU fuel. The unique critical (discharge) burnup values are also presented for every of the three enrichments. The results showed that almost one and a half discharge burnup can be obtained using a nuclear fuel with a quite light enrichment of 1%w U235, while 2%U235 configuration is able to supply almost three times larger discharge burnup. The peculiar individual fissile actinides’ contributions to the total fission power with respect to enrichments and burnups are also discussed.

Key words: WIMS, Natural Uranium, Slightly Enriched Uranium, fission power

Introduction

CANDU (CANada Deuterium Uranium) is the widest pressure tubes reactor in operation around the world. CANDU is heavy water moderated and cooled reactor using Natural Uranium fuel in a simple and flexible fuel bundle design. With two nuclear units in operation at Cernavoda NPP, Romania should be interested in the use of advanced fuel cycles in actual CANDU power reactors. The goal of our work is to find out the burnup influence on lattice parameters of interest when nuclear fuels with Slightly Enriched Uranium (SEU) are used, compared to the case of Natural Uranium (NU). The lattice parameters pursued in calculations were the infinite multiplication constant and the contribution to fission power of some major actinides such as: Uranium-235, Uranium-238 along with Plutonium series (Pu-239, Pu-240, Pu-241 and Pu-242).

Methodology Outlines

The methodology used in the paper is based on performing neutron lattice calculations to solve the transport equation using the WIMS computer program [1] and its updated public library, [2] supplied by
the International Atomic Energy Agency (IAEA). The transport equation models the behavior of neutron population inside of nuclear system. The WIMS program allows to calculate a lot of lattice interest parameters. We pursued the infinite multiplication constant and the contribution to the total fission power brought by the major actinides alluded in introduction.

The fuel composition, the bundle design, the working configurations and the main WIMS region radii are synthesized in Table 1. Three enrichments for nuclear fuel placed in the standard 37-rods CANDU bundle were used: 0.71, 1.0 and 2% mass percent of U235. Fig. 1 presents the CANDU 6 elementary lattice [3].

### Table 1. The working configurations

<table>
<thead>
<tr>
<th>Configuration Name</th>
<th>NU</th>
<th>SEU1%</th>
<th>SEU2%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Config. #</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Enrichment %U235</td>
<td>0.71</td>
<td>1.0</td>
<td>2.0</td>
</tr>
<tr>
<td>Number of rods per fuel ring</td>
<td>CE=1, R1=6, R2=12, R3=18</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bundle Geometry</td>
<td>CANDU 37-rods (CANDU standard)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total Uranium mass (kg)</td>
<td>19.9</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total fuel mass (kg)</td>
<td>22.6</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total Zircaloy mass (kg)</td>
<td>2.15</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total bundle mass (kg)</td>
<td>24.74</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure Tube Internal Radius, [4]</td>
<td>5.17 cm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure Tube External Radius (ANNULUS 5), [4]</td>
<td>5.6 cm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Annular Gas Thickness, [4]</td>
<td>0.85</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Calandria Tube Internal Radius (ANNULUS 6), [4]</td>
<td>6.45 cm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Calandria Tube External Radius (ANNULUS 7), [4]</td>
<td>6.59 cm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Lattice Pitch (ANNULUS 8), [4]</td>
<td>28.575</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*CE = Central Element; R1,R2,R3=inner rings from CE to outside, see Fig. 1

**Fig. 1. The CANDU elementary lattice, [3]**

The data regarding CANDU lattice dimensions was taken from public domain [4], while the fuel and zircaloy masses were calculated supposing a density of 10.5 g/cm³ for fuel and an equivalent density of 7.5 g/cm³ for zircaloy alloy. In the WIMS model we supposed that the entire mass of zircaloy is distributed around the fuel rods in the clad. As result, the zircaloy density should be increased from 6.5 g/cm³ to 7.5 g/cm³ in order to preserve its total bundle mass. Fig. 2 illustrates the WIMS model built with the main 8 regions (identified in the input data file by the keyword "ANNULUS").
Instantaneous burnup calculations have been performed in order to find out the exit discharge (critical) burnups i.e. those burnup values for which the lattice is still critical. The major actinides contribution to fission power with respect to the fuel burnup were also pursued. The "REACTION" keyword was used to specify all the actinide isotopes possible to have contribution to fission power. They are denoted firstly by the order number, then the periodic table symbol and lastly, by their atomic mass as following: $^{92}\text{U-233}$, $^{92}\text{U-235}$, $^{92}\text{U-236}$, $^{92}\text{U-238}$, $^{94}\text{Pu-238}$, $^{94}\text{Pu-239}$, $^{94}\text{Pu-240}$, $^{94}\text{Pu-241}$, $^{94}\text{Pu-242}$. Despite the fact that the initial fuel composition only included $^{92}\text{U-235}$ and $^{92}\text{U-238}$ isotopes, the final output files reveal the presence of all above mentioned isotopes.

The fission power is that power released only from fissions underwent by fissile isotopes. It is generally known that a fission reactor works using Uranium as fuel, but it seems usually to be less known the contribution to fission power of other heavy isotopes generated by nuclear reaction during the fuel burning process. In order to capture the contributions and their share in the total fission power we took these values from WIMS output file sections' entitled "0element isotope# reactions", where the "isotope#" key word denotes the isotope identification number in the WIMS library.

The infinite multiplication constant $k_{\text{inf}}$ is plotted under "diagonal transport corrected flux solution" WIMS output file section for every set of 45 MW/tU x 11.11 days burnup interval. The paper's results are presented and discussed below.

**Results and discussions**

In Table 2 the discharge burnup values are presented for every of the three configurations.

<table>
<thead>
<tr>
<th>Configuration Name</th>
<th>NU</th>
<th>SEU1%</th>
<th>SEU2%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Config.#</td>
<td>1</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td>Discharge Burnup (MWd/kgU)</td>
<td>6.5</td>
<td>10.5</td>
<td>21.5</td>
</tr>
</tbody>
</table>

The discharge burnup can rapidly be discovered by plotting the infinite multiplication factor (sometimes referred as the infinite reactor multiplication constant, [3]) with respect to the burnup, as in Fig. 3. It is a measure of the total fission energy released by the nuclear fuel during its residence in the reactor core.
Fig. 3. The infinite multiplication factor by fuel burnup

The discharge burnup value corresponds to an infinite multiplication reactor constant of unity, when the initial reactivity excess has been exhausted and the fuel should be removed from the core. As it can be observed, an increase by 0.3% in the fuel enrichment (from 0.7 to 1%) is able to produce 61% more energy from every kilogram of Uranium. Moreover, adding another one percent to the enrichment supplied 21.5 MWd/kgU as discharge burnup, almost three time more than in the case of Natural Uranium. Knowing that the radioactive waste amount is inverse proportional to the discharge burnup, we expect to have benefits regarding radioactive waste amount reducing by the same order along with a lower usage of the refueling machine.

The fissile isotope share into the total fission power is represented with respect to the fuel burnup in Figs. 4, 5 and 6.

Fig. 4. NU Contribution to the fission power  
Fig. 5. SEU1% Contribution to the fission power

Despite the fact that the fission power share evolutions in Figs. 4, 5 and 6 seems to be very close each other, some differences can be disclosed representing the contribution of a single interesting fissile isotope for every of the three configurations in the same picture, as in Figs. 7 to 10.
Fig. 6. The SEU2% contribution to the fission power

Fig. 7. The U-235 contribution to the fission power

Fig. 8. The U-238 contribution to the fission power

Fig. 9. The Pu-239 contribution to the fission power

Fig. 10. The Pu-241 contribution to the fission power

In Figs. 7, 8, 9 and 10 the contribution to the fission power of the main contributors is illustrated. The main contributors are: 92-U-235, 92-U-238 94-Pu-239 and 94-Pu-241. The rest of contributions (supplied by Pu-240 and Pu-242) are insignificant and therefore have not been represented.

Figures 5, 6 and 7 reveal that the most important contribution to the fission power is brought by the two fissile isotopes, namely U235 and Pu239, the first one being found "as is" in the natural Uranium ore while the second one being generated in the reactor during the burning process through nuclear reactions. While the U235 contribution decreases by burnup, the Pu239 contribution increases with respect to the fuel burnup. The equilibrium state, when the two contributions are equal takes place at a farther and farther time from the middle burnup moment, as follow: for the NU configuration the moment is 88.8 days (when the corresponding burnup is 4 MWd/kgU = 0.045 MW/kg*88.8 days, see Fig. 4), for the
SEU1% configuration the moment is 155.6 days (when the corresponding burnup is about 7 MWd/kgU = 0.045 MW/kg*155.6 days, see Fig. 5) and for the SEU2% configuration the moment is 377.8 days (when the corresponding burnup is about 17 MWd/kgU = 0.045 MW/kg*377.8 days, see Fig. 6).

In Fig. 7 the U235 contribution to the fission power with respect to the burnup is presented for every of the three configurations considered in the study. As it was expected, the U235 contribution decreases by the fuel burnup and it is greater at higher initial enrichments. Near the discharge burnup, the U235 isotope only contributes by 30% to the fission power.

Of interest is the contribution brought by U238 isotope which is usually less investigated. We can notice that 3 to 6% of the fission energy is released by fast fissions in U238 whose contribution slowly increases by burnup and as U235 is being consumed.

The Plutonium 239 contribution is revealed in Fig. 9 while that of Plutonium 241 is shown in Fig. 10. Being generated by two subsequent beta decays of U-238 isotope, the Pu239 contribution to fission power is direct proportional to the instantaneous concentration of its precursors, U238 and Neptunium-239. The Pu-239 contribution overrides fifty percents near the discharge burnup, see Fig. 9, being therefore, after U235, the most important contributor to the fission energy generation. Also, the odd atomic mass number Pu241 isotope reveals its significant contribution to the fission power in Fig. 10. This contribution rises up to 6, 7 and 10.5% with respect to the corresponding discharge burnups of the working configuration.

Conclusions / Remarks

The fissile and fissionable isotopic share to the total fission power has been revealed for a CANDU lattice fuelled with Natural and Slightly Enriched Uranium by using the WIMS burnup calculations.

Tracking the infinite multiplication factor variation with respect to the burnup, the unique discharge burnup values for every analyzed fuel configuration were found out.

Peculiar and less frequently investigated aspects regarding fissionable isotope contributions to the fission power have also been revealed.

References

MANAGEMENT OF THE USED ION-EXCHANGERS CONTAMINATED WITH C-14 GENERATED BY NPP CERNAVODA

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ABSTRACT

For the conditioning of ion-exchangers generated from operation of Cernavoda NPP Unit 1, techniques of direct immobilization in cement, bitumen and organic polymers have been experimented. The selected process for conditioning of spent resins is the bituminization. The bituminization process consists of the incorporation in bitumen of the spent resin, at temperatures between 110 and 120°C, and the solidification of the mixture by cooling. The percentage of incorporated spent resin in bitumen is ranged from 40 to 50% dry resin. The advantages of bituminization are: the bitumen is insoluble in water, the bituminization installation is simple, the matrix is not cost expensive, the temperature process is low, bitumen is compatible with wastes having various compositions, and the volume of the final product is smaller. The main disadvantage is that bitumen is combustible, although not easily flammable, and, at high temperature, interaction between bitumen matrix and chemical components, might occur.

Key words: C-14, spent resins, bituminization

Introduction

$^{14}\text{C}$ is contained in important amounts in the ion exchangers used at the purification of the moderator and of the primary transport systems. Because of the unusually long life (5730 years) and the facility with which $^{14}\text{C}$ can be assimilated in the biological cycles, it is necessary the immobilization of ion exchangers contaminated with $^{14}\text{C}$ to be made in a matrix, as a monolithic block, so that the safety of the environment be ensured, during final disposal, according to international standards throughout the radioactive decay, until an acceptable level is attained.

The researches in field have made applicable more techniques of treatment and conditioning of the ion exchangers contaminated with $^{14}\text{C}$ [1]. In spite of all these it might not be admitted that the perfect matrix for immobilization has been found. Every technique has pro and contra arguments, it has advantages and disadvantages and, perhaps, different concepts for application. The methods which turned out to be the nearest to imposed conditions (direct immobilization or pre-treatment and volume reduction followed by immobilization of residues in a suitable matrix) were materialized in technologies, they have been applied today, on an worldwide scale, at an industrial scale.

The study consisted in immobilization experiments in bitumen of spent resins and leaching studies on the bitumen products to determine the leaching rates of $^{14}\text{C}$. 
Characteristics of ion-exchange resins used in experiments

In the experiments there were used cationite C100H and anionite A600, similar to IRN-77 and IRN-78 (used at Cernavoda NPP). C 100H resin is a strong-acid cationite in the –SO_3H form, based on styrene-divinylbenzene, contains minimum 99.9% ion exchangers in the R-H form and it is similar to IRN-77. A600 resin is a strong-base anionite, based on same support and it have linked –N(CH_3)_3^+ functional groups; it contains minimum 95% ion exchangers in the R-OH form, max.0.1% in R-Cl form, max. 5% R_2CO_3, max. 0.3% in R_2SO_4 form and is similar to IRN-78 [2].

Table 1. Formulas of embedding resins in bitumen [3, 4, 5]

<table>
<thead>
<tr>
<th>Sample no.</th>
<th>Composition</th>
<th>Dry resin/Bitumen embedding rate</th>
<th>Resin type</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Dry resin (g)</td>
<td>Bitumen (g)</td>
<td></td>
</tr>
<tr>
<td>0.</td>
<td>240</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>1</td>
<td>266.6</td>
<td>80</td>
<td>0.667 cationite (w=45.2%)</td>
</tr>
<tr>
<td>2</td>
<td>266.6</td>
<td>400</td>
<td>0.667 anionite (w=51.45%)</td>
</tr>
<tr>
<td>3</td>
<td>400</td>
<td>400</td>
<td>1 anionite (w=51.45%)</td>
</tr>
<tr>
<td>4</td>
<td>266</td>
<td>400</td>
<td>0.664 carbonated anionite (w=50.97%)</td>
</tr>
<tr>
<td>5</td>
<td>267</td>
<td>400</td>
<td>0.667 carbonated anionite (w=49.81%)</td>
</tr>
<tr>
<td>6</td>
<td>400</td>
<td>400</td>
<td>1 carbonated anionite (w=51%)</td>
</tr>
<tr>
<td>7</td>
<td>400</td>
<td>400</td>
<td>1 carbonated anionite (w=49.81%)</td>
</tr>
</tbody>
</table>

Tests of spent resin-bitumen products

Visual characterization of the products

Under macroscopic examination it has been found that the changing of bitumen-ion-exchange resin in the form of solid, monolite, cylindrical blocks, with glossy bases and surfaces, without pores or cracks. In section, does not present gas bubbles.

Leaching tests of samples of bituminized resins

The inactive bitumen-resin samples, within which the anionite was loaded with known quantities of inactive C (under the form of CO_3^2-) were subjected to the leaching test.

The mixture was poured into cylindrical metal molds with the diameter of 28 mm and the height of 40 mm.

After the solidification of the bitumen-resin mixture, for about 24 hours, the samples are removed from molds and are subjected to the leaching tests. The leaching test consisted in the immersion of the bitumen-resin products in deionized water such as the whole surface comes in contact with the water. Because the carbon release is small, for many times under the detection limit of the method, the water change frequency was not respected for all samples, according to IAEA and ISO recommendations. For the samples 4 and 6 the leaching water was changed daily in the first 7 days, once a week in the next three weeks, and remain unchanged until the end of the experiment, and in the case of the other samples the leaching water was never changed. In table 2 the characteristics of the samples tested for leaching are presented.
Table 2. Characteristics of the inactive probes of bitumen-resin tested for leaching

<table>
<thead>
<tr>
<th>Sample no</th>
<th>Weight sample (g)</th>
<th>Volume sample (cm³)</th>
<th>Dry resin in sample (g)</th>
<th>Density (g/cm³)</th>
<th>Surface exposed (cm²)</th>
<th>Volume lechant (cm³)</th>
<th>¹²C incorporated* (mg/sample)</th>
<th>Leaching medium</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>28.5</td>
<td>24</td>
<td>9.41</td>
<td>33.02</td>
<td>1.190</td>
<td>54.96</td>
<td>500</td>
<td>-</td>
</tr>
<tr>
<td>2</td>
<td>28.0</td>
<td>26</td>
<td>10.86</td>
<td>38.78</td>
<td>1.080</td>
<td>53.99</td>
<td>500</td>
<td>-</td>
</tr>
<tr>
<td>3</td>
<td>27.5</td>
<td>26</td>
<td>14.10</td>
<td>51.27</td>
<td>1.060</td>
<td>53.03</td>
<td>500</td>
<td>-</td>
</tr>
<tr>
<td>4</td>
<td>27.9</td>
<td>26</td>
<td>9.91</td>
<td>35.90</td>
<td>1.073</td>
<td>50.14</td>
<td>500</td>
<td>109.99</td>
</tr>
<tr>
<td>5</td>
<td>28.0</td>
<td>24</td>
<td>10.83</td>
<td>38.68</td>
<td>1.170</td>
<td>53.99</td>
<td>500</td>
<td>120.02</td>
</tr>
<tr>
<td>6</td>
<td>28.0</td>
<td>26</td>
<td>12.11</td>
<td>43.25</td>
<td>1.077</td>
<td>50.14</td>
<td>500</td>
<td>108.82</td>
</tr>
<tr>
<td>7</td>
<td>28.0</td>
<td>28</td>
<td>14.05</td>
<td>50.18</td>
<td>1.000</td>
<td>53.99</td>
<td>500</td>
<td>147.42</td>
</tr>
</tbody>
</table>

* the amount of incorporated carbon was calculated by the difference between the initial amount of carbon which has been charged on the anionite and the amount of carbon that was lost during the bituminization

Characteristics of deionized water (DW) used as leaching medium:
- pH=6.85
- conductivity=4.68 µS/cm
- content of carbon=0.8 mg/l

Leaching Rate of ¹²C (assimilated with ¹⁴C)

The carbon releases were evaluated by titration of the leaching water samples with 0.1 mol/L HCl. In the table 3 there are presented the carbon content in leaching water and the leaching rates of ¹⁴C, and in figure 1, the leaching rates of ¹⁴C for the samples 4 and 6 which have carbonated anionite.

Table 3. ¹⁴C releases (Bq) from resin-bitumen samples in leach medium. The leach rates of ¹⁴C

<table>
<thead>
<tr>
<th>Sample no</th>
<th>¹⁴C content (mg/sample)</th>
<th>C in leaching medium (mg)</th>
<th>¹⁴C leached a (mg)</th>
<th>Leaching rate ¹⁴C (g/cm²/day)</th>
<th>Total days of leaching</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>-</td>
<td>0.40*</td>
<td>-</td>
<td>-</td>
<td>84</td>
</tr>
<tr>
<td>2</td>
<td>-</td>
<td>0.40*</td>
<td>-</td>
<td>-</td>
<td>84</td>
</tr>
<tr>
<td>3</td>
<td>-</td>
<td>0.40*</td>
<td>-</td>
<td>-</td>
<td>84</td>
</tr>
<tr>
<td>4</td>
<td>109.99</td>
<td>1.34</td>
<td>1.34</td>
<td>1.3012 b1</td>
<td>2.10×10⁻⁵</td>
</tr>
<tr>
<td>5</td>
<td>120.02</td>
<td>0.41</td>
<td>0.01</td>
<td>0.0093 b2</td>
<td>5.14×10⁻⁷</td>
</tr>
<tr>
<td>6</td>
<td>108.82</td>
<td>2.41</td>
<td>2.41</td>
<td>2.3029 b3</td>
<td>6.10×10⁻⁵</td>
</tr>
<tr>
<td>7</td>
<td>147.42</td>
<td>0.50</td>
<td>0.10</td>
<td>0.0760 b4</td>
<td>4.19×10⁻⁶</td>
</tr>
</tbody>
</table>

a The carbon found in leaching medium is in fact the carbon of the deionised water used as the leaching medium

b the amount of leached C-14 was calculated from the difference between the amount of carbon in the blank sample and the samples containing carbonated anionite: b1= C exp.16 - C exp.9; b2= C exp.17 - C exp.9; b3= C exp.18 - C exp.11; b4= C exp.19 - C exp.11
Figure 1. Leaching rates of $^{14}$C for samples 4 and 6

The leach rates were calculated using the follow equation:

$$R = \frac{a_s}{a_0} x \frac{a_x}{S \cdot t} \quad \text{(Eq.1)}$$

where,

- $R$ - leach rate (g/cm$^2$·day)
- $a_x$ - amount leached during the leaching period (g)
- $a_0$ - amount initially present in sample (g)
- $a_s$ - sample weight (g)
- $S$ - exposed surface area of sample (cm$^2$)
- $t$ - leaching duration (days)

It was found that for the samples with 50% carbonated anionite the leaching rates are higher than for the samples containing 40% resin. Therefore, a higher percentage of embedding resin and a higher $^{14}$C amount incorporated causes a slightly higher rate of leaching.

**Conductivity and pH**

In parallel with the above analyses, for the same samples of bitumen-resin the conductivity and pH of the leaching water have been measured.

The conductivity is a parameter that gives information with regards at the concentration of ions in solution.

The conductivity and pH of the leaching water were measured with a standard laboratory pH conductometer. The values are shown in Tables 4 and 5.
Table 4. Conductivity (μS / cm) of water leaching of samples of bitumen-resin

<table>
<thead>
<tr>
<th>Sample no</th>
<th>day 1</th>
<th>day 2</th>
<th>day 3</th>
<th>day 4</th>
<th>day 5</th>
<th>day 6</th>
<th>day 7</th>
<th>day 14</th>
<th>Day 21</th>
<th>Day 28</th>
<th>After 84 days&lt;sup&gt;a&lt;/sup&gt;</th>
<th>169 days&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>DW</td>
<td>1.80</td>
<td>1.80</td>
<td>1.70</td>
<td>1.50</td>
<td>1.50</td>
<td>2.40</td>
<td>2.10</td>
<td>5.90</td>
<td>3.61</td>
<td></td>
<td>0.98</td>
<td></td>
</tr>
<tr>
<td>DW'</td>
<td>1.68</td>
<td>1.68</td>
<td>1.80</td>
<td>1.50</td>
<td>1.50</td>
<td>2.90</td>
<td>2.90</td>
<td>6.80</td>
<td>1.87</td>
<td></td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>1.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>4.68</td>
<td></td>
<td></td>
<td></td>
<td>22.30</td>
<td></td>
</tr>
<tr>
<td>2.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>4.68</td>
<td></td>
<td></td>
<td></td>
<td>16.64</td>
<td></td>
</tr>
<tr>
<td>3.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>4.68</td>
<td></td>
<td></td>
<td></td>
<td>58.90</td>
<td></td>
</tr>
<tr>
<td>4.</td>
<td>8.28</td>
<td>3.69</td>
<td>3.44</td>
<td>3.60</td>
<td>4.34</td>
<td>4.34</td>
<td>6.80</td>
<td>10.25</td>
<td>5.98</td>
<td></td>
<td>86.10</td>
<td></td>
</tr>
<tr>
<td>5.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>4.68</td>
<td></td>
<td></td>
<td></td>
<td>37.60</td>
<td></td>
</tr>
<tr>
<td>6.</td>
<td>13.5</td>
<td>5.74</td>
<td>5.08</td>
<td>4.34</td>
<td>9.26</td>
<td>16.23</td>
<td>17.63</td>
<td>12.63</td>
<td></td>
<td></td>
<td>199.26</td>
<td></td>
</tr>
<tr>
<td>7.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>4.68</td>
<td></td>
<td></td>
<td></td>
<td>54.60</td>
<td></td>
</tr>
</tbody>
</table>

DW - deionized water (initial) (leaching medium)
DW' - deionized water (final) (conductivity and pH of were measured at the period of changing the leaching medium for the samples)
(a) leaching medium was not changed until the end of tests
(b) leaching medium was not changed after 28 days until the end of tests

First, it is noticeable that the conductivity values are very low which indicates the low number of ions in solution and the value range is relatively short.

There is an increasing of the conductivity with the amount of resin embedded, such as for the samples with 50% resin (samples 3, 6, 7) the value is higher than for the samples with 40% resin. There are higher conductivities for bitumen-cationite (sample 1) si bitumen-anionite (sample 3) than the samples containing carbonated anionite (samples 6, 7). This is explained by the higher mobility of ions H<sup>+</sup> than OH<sup>-</sup> and especially much higher than CO<sub>3</sub><sup>2-</sup> and HCO<sub>3</sub><sup>-</sup>. 
Table 5. pH of the of water leaching of samples of inactive bitumen-resin

<table>
<thead>
<tr>
<th>Sample no</th>
<th>day 1</th>
<th>day 2</th>
<th>day 3</th>
<th>day 4</th>
<th>day 5</th>
<th>day 6</th>
<th>day 7</th>
<th>day14</th>
<th>Day 21</th>
<th>Day 28</th>
<th>After 84 days&lt;sup&gt;a&lt;/sup&gt;</th>
<th>169 days&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>DW</td>
<td>6.3</td>
<td>6.3</td>
<td>6.1</td>
<td>6.1</td>
<td>6</td>
<td>6</td>
<td>6.1</td>
<td>5.7</td>
<td>6.2</td>
<td>5.5</td>
<td>6.2</td>
<td>5.5</td>
</tr>
<tr>
<td>DW'</td>
<td>6.3</td>
<td>6.2</td>
<td>6.1</td>
<td>6.3</td>
<td>6.6</td>
<td>6.1</td>
<td>6.7</td>
<td>5.5</td>
<td>-</td>
<td>-</td>
<td>5.5</td>
<td>5.5</td>
</tr>
<tr>
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<td>6.6</td>
<td>6.4</td>
<td>6.3</td>
<td>6.4</td>
<td>6.6</td>
<td>5.9</td>
<td>6.4</td>
<td>6.7</td>
<td>-</td>
<td>-</td>
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Mass variation of the bitumen-resin samples

Although the bitumen is impermeable, however, in contact with water for a long time, there is an increasing of the mass of bitumen-resin blocks due to water absorption.

Mass increases were determined on samples kept in water for 84 or 169 days and are presented in Table 6 and Figure 2.

Table 6. Mass variation of the bitumen-resin samples in leaching water

<table>
<thead>
<tr>
<th>Pr. nr.</th>
<th>Dry resin (%)</th>
<th>Sample' s weight before leaching (g)</th>
<th>Sample' s weight after leaching (g)</th>
<th>Gain in weight (%)</th>
<th>Number of days of immersion in water of leaching</th>
<th>Environment of leaching</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>40.00</td>
<td>28.50</td>
<td>29.00</td>
<td>1.75</td>
<td>84</td>
<td>A.D.</td>
</tr>
<tr>
<td>2</td>
<td>40.00</td>
<td>28.00</td>
<td>28.70</td>
<td>2.50</td>
<td>84</td>
<td>A.D.</td>
</tr>
<tr>
<td>3</td>
<td>50.00</td>
<td>27.50</td>
<td>27.60</td>
<td>0.36</td>
<td>84</td>
<td>A.D.</td>
</tr>
<tr>
<td>4</td>
<td>39.92</td>
<td>27.90</td>
<td>30.70</td>
<td>10.03</td>
<td>169</td>
<td>A.D.</td>
</tr>
<tr>
<td>5</td>
<td>40.00</td>
<td>28.00</td>
<td>28.90</td>
<td>3.21</td>
<td>84</td>
<td>A.D.</td>
</tr>
<tr>
<td>6</td>
<td>49.98</td>
<td>28.00</td>
<td>32.50</td>
<td>16.07</td>
<td>169</td>
<td>A.D.</td>
</tr>
<tr>
<td>7</td>
<td>50.00</td>
<td>28.00</td>
<td>28.90</td>
<td>3.21</td>
<td>84</td>
<td>A.D.</td>
</tr>
</tbody>
</table>
Figure 2. Mass variation of the bitumen-resin samples in leaching medium

It is found that the mass of all samples increase, the increases varie between 0.36 and 16.07%, higher increases being in case of samples that were kept a longer period in contact with water. It also notes that the carbonated anionite samples have a higher increase. Mass variations are very close and they are directly proportional with the amount of resin incorporated. Also the mass increases with the amount of resin embedded.

Conclusions

The immobilising matrix, bitumen, has many advantages such as insolubility in water and high resistance against diffusion that results in final products with low leaching rates. In addition, the bitumen has a high capacity for resin incorporation and therefore high volumetric efficiency. There are also some disadvantages that might be important, for example the bitumen is flammable and also there is the possibility to chemical interact with components of the radioactive waste and has limited stability to radiation. Inconveniences of the bitumen can be offset by carefully selecting the type of bitumen and operating conditions.

As a result of the bituminization tests and as the result of the obtained data can be mentioned the following:

- During the bituminization process it also takes place the evaporation of water contained in the wet resin, this being collected as condensation product. Therefore the volume of conditioned waste is significantly reduced. Thereby, there is no longer necessary a preliminary drying of resin, which would generate new waste due to the retention of $^{14}$C in the absorbant solution;
- Leaching rates of carbon are lower, of about $10^{-5} – 10^{-6}$ g/cm$^2$day;
- Mass variation of the resin-bitumen products during the contact with water, is low;
- The conductivity values are low during the leaching tests and pH values are ranged from neuter to slightly acid;
- a disadvantage of bituminization is that the bitumen is burning at a temeparture $> 250^0$C
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TECHNIQUE FOR DETACHMENT OF THE ROLLED JOINT BETWEEN END FITTING AND PRESSURE TUBE. TECHNOLOGICAL CONDITIONS

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ABSTRACT

This study aims to develop a technology that allows the reuse of the end fittings by non-destructive detachment of the rolled joint between the end fitting and the pressure tube. Therefore, this technique would provide the detachment and removal of the pressure tube from the end fitting faster, reducing the reactor shutdown time, and eliminates the need of transport, storage and production of new fittings. The system works on the principle of inductive heating, in which the rolled joint area is rapidly heated up to $T_3 [^\circ C]$ followed by a quick axial load applied in order to detach the end fitting from the rolled joint. Certain information about the rolled joint main features was analyzed in order to determine the characteristics of the equipment such as the axial load required for the detachment considering different conditions. The paper is dedicated to specialists working in research and technological engineering.

Key words: end fitting, pressure tube, joint, rolled joint, drag

Introduction

Characteristic to the CANDU reactor compact structure is the replacement of the high-pressure reactor vessel with a low pressure cylindrical vessel (calandria) placed in bearings, horizontally, on the generators. The calandria vessel is crossed by a number of horizontal fuel channel assemblies integrated into the primary heat transport circuit.

Given the fact that the integrity maintenance for the CANDU nuclear reactor is an essential condition, the main material of the vessel and for the associated joints (welding techniques, mechanical joining method) should assure the mechanical resistance and adequate tenacity throughout the entire lifetime [1].

A fuel channel assembly is mainly composed of two end fittings, one pressure tube, the calandria tube, bearings, fasteners, expansion takeovers and biological protection. The whole assembly is placed between the end plates of the calandria vessel.

The end fittings ensure a tight inclusion of the fuel channel assembly in the primary heat transport system through the attached feeders and the possibility for the loading-unloading machine to load the fresh fuel and unload the used fuel.

The pressure tube is rolled on both ends in end-fittings; it is the fuel channel assembly’s limit for pressure and temperature in the fuel channel assembly inside the calandria vessel and also contains the nuclear fuel, cooled by the turbulent flowing of the cooling agent, D$_2$O.
The active zone designer used rolling as a method of pressure-tight sealing tubes (alloy Zr-2.5% Nb) with end fittings (stainless steel) which is a classic process of plastic deformation. Thus the ends of the pressure tube fittings are rigidly assembled by applying a roll joining process using a procedure which diminishes the size of the residual mechanical stresses Fig. 1, [2].

The calandria tube, the pressure tube envelope, limits the low pressure in the calandria vessel in the fuel channel assembly. The space between the calandria tubes is forcefully flowed by the moderator D₂O, which also ensures the compliance with the low temperature limit.

The annular space between the pressure tube and calandria tube can be "open" or "closed" and flowed by CO₂. Monitoring and continuous analysis of the isolation environment provides useful data to characterize the integrity of the pressure tube wall.

Direct contact between the pressure tube and the calandria tube is achieved by using four garter spring spacers.

![Diagram of the rolled joint between the end fitting and the pressure tube](image)

*Fig. 1 The rolled joint between the end fitting and the pressure tube*

The pressure tube, which was considered for a long time as the main component in plant operation, is exposed to environmental conditions (temperature, pressure, radiation flux, mechanical vibrations induced by the flow of the primary cooling agent), conditions that reduce its designed lifespan requiring its replacement. [4]

The pressure tube replacement technology involves disassembling the fuel channel assembly, a process which has a high difficulty degree. It includes cutting off the ends of the pressure tube near the end fittings and in the middle which involves containerizing, transporting and storing the active components.

By applying this method for the disassembly of the rolled joint between the pressure tube and the end fitting without damaging the end fitting, there is no need for containerizing the end fittings, transporting
and especially storing them, because they can be re-enter the primary circuit by rolled joining with the ends of another pressure tube.

After a brief general characterization of the rolled joint pressure tube end fitting, the article shows the technological conditions of the rolled joint disassembly, the results and, finally, some conclusions.

**General characterization of a rolled joint between the end fitting and the pressure tube**

Dimensions for the end fitting:

- exterior diameter: \( D_{F_i} \) [mm];
- total length: \( L_{F_i} \) [mm];
- outer diameter of the end fitting: \( D_{Fi} \) [mm].

Characteristics of the pressure tube-end fitting assembly, Figure 2:

- inner diameter: \( d_{Fi} \) [mm];
- usable length for rolling: \( l_{mF} \) [mm];
- minimum fitting displacement (practically „zero” possibly a slight grip fit);
- the length from the border of the rolled joint to the end fitting: \( L_{mF} \) [mm];
- the inner diameter of the pressure tube in the rolled joint area: \( d_{ti} \) [mm].

Functional characteristics generated by the geometry of the end fitting and its joint with the pressure tube which are useful for configuring the device that snatches the end of the pressure tube from the rolled joint.

- distances: between the free end of the end fitting and the jaws locking the slot closing the channel, between the free end of the end fitting and the tube inner slot and between the free end of the end fitting and the end of the pressure tube with the rolled joint;
- the joint between the inner tube ending and the rolled joint pressure tube ending;
- the pressure tube free end’s inner diameters (next to the sealing ring) and the inner liner tube diameter.

Details of the rolled joint, figure 2, ensure the necessary data for the configuration of some basic elements of the device (super heater and the dragging tool’s entering end)

The three realms are characterized by: height \( h \) [mm]; width \( s \) [mm]; the distance between the grooves end \( l_1 \) [mm]; the inner diameter of the pressure tube in the rolled joint area: \( d_{t1} \) [mm]; the wall thickness of the tube with the rolled joint in front of the groove \( h_{t1} \) [mm] and between the grooves \( h_{t2} \) [mm].

*Fig.2 Details of the rolled joint between the end fitting and the pressure tube*
The rolled joint is a sealed and rigid radial clamping that was obtained cold by using a rolling tool on a grooved surface. By rolling of the inner pressure tube in the groove area a plastic deformation to the alloy of Zirconium in two directions: radial and longitudinal directions, thereby reducing the internal volume of the fittings groove, by filling them in a certain proportion. The distribution of the contact pressure between the grooves is shown in the Fig. 3

![Fig.3 The distribution of the contact pressure between the grooves](image)

The joining degree mandrel in this case determined by the equation [2]:

\[ H = 100 \frac{d_{Fi}}{d_{ti}} \% \]

Determining the indicative actual axial load:
Assuming that we are working with a stress environment without the risk of losing stability with a rolled joint on a smooth surface, without grooves, in which case, the maximum axial load for tensile / compression that the pressure tube joint could meet is given in the relationship, [3]:

\[ Q_{max} = \Pi \cdot h_2 \cdot d_{Fi} \cdot h_2 \cdot \sigma_{0.2} \cdot C \cdot [\text{KN}] \]

Since there is a risk of losing the stability the same maximum axial load becomes a safety factor, [2]:

\[ Q_{cap} = \frac{Q_{max}}{C_s} \cdot [\text{KN}] \]

On the other hand, a rolled joint with three grooves on a surface involves an increased bonding strength of the resulting rolled joint [2]. Thus an axial load three times larger is obtained at about maximum \( 3 x Q_{cap} \).

**Technological conditions for the detachment technique**

Disassembling the rolled joint requires designing and constructing a complex equipment with the following functions:
- rapid end fitting volume heating in order to obtain an increase in the outer pressure tube diameter which remains at a constant temperature around the initial value;
- applying a fast axial load in order to remove the pressure tube from the rolled joint.

A rapid volume heating for the end fitting provides a diametrical expansion that based on the level of temperature would weaken the rolled joint [4]. For a local temperature level of $T_2 = 100^\circ$C on the end of the end fitting ($T_1 = 20^\circ$C) and a coefficient of expansion: $\alpha_{100^\circ}$C, [K$^{-1}$] we achieve, [4]:

- an outline expansion:
  \[ \Delta L = \Pi \cdot d_{Fi} \cdot \left( T_2 - T_1 + \alpha_{100^\circ}C \right), [mm] \]

- an diameter increase that at first glance would lead to a separation between the two surfaces in contact (between grooves). The detachment cannot be total because after the rolling there is an uneven residual stress distribution in the pressure tube material that will cause a significant camber, Figure 3.

For this level of temperature, we do not expect a significant separating force reduction [4].

For a temperature level of $T_3 = 200^\circ$C locally applied on the end of the end fitting ($T_1 = 20^\circ$C) and a coefficient of expansion: $\alpha_{200^\circ}$C, [K$^{-1}$] we obtain, [4]:

- an outline expansion:
  \[ \Delta L = \Pi \cdot d_{Fi} \cdot \left( T_3 - T_1 + \alpha_{200^\circ}C \right), [mm] \]

- an increase in the end fitting inside diameter of $\delta \sim$ five times larger than what previously obtained.

The axial load for the wrench in this case can be obtained with the equation (for $\delta = \frac{\Delta L}{2}$):

\[
Q_{max}^{200^\circ}$C = $\Pi \cdot \phi - \delta \left[ D_r - \phi - \delta \right] \sigma_{0,2}^{20}
\]

where: $D_r$, [cm], is the diameter of the pressure tube distorted in the groove.

The requested axial load would correspond to the technological possibilities of the equipment design [4].

For a temperature level of $T_4 = 300^\circ$C locally applied on the end of the end fitting ($T_1 = 20^\circ$C) and a coefficient of expansion: $\alpha_{300^\circ}$C, [K$^{-1}$] we obtain, [4]:

- an outline expansion:
  \[ \Delta L = \Pi \cdot d_{Fi} \cdot \left( T_4 - T_1 + \alpha_{300^\circ}C \right), [mm] \]

- an increase in the inside diameter of the end fitting of $\delta \sim$ ten times higher than what previously obtained.

The axial load for the wrench in this case can be obtained with the equation:

\[
Q_{max}^{300^\circ}$C = $\Pi \cdot \phi - \delta \left[ D_r - \phi - \delta \right] \sigma_{0,2}^{20}
\]
The axial plucking force value obtained does not represent a technological problem and can be accomplished by attaching a hydraulic cylinder (pneumatic) in the longitudinal area of the fitting, either by embedding a hydraulic cylinder (pneumatic) specially shaped in the linner tube. [4].

The chamfered shape of the groove at $\alpha$ (towards the inside of the channel) causes a slight movement of the pressure tube in the direction of the plucking force.

The temperature required for the expansion of the end fitting in the rolled joint area can be achieved by inducing turbulent currents for short time intervals so that the pressure tube temperature in this area is kept in acceptable limits (+10°C) [4]. After increasing the pressure tube temperature by 10°C we obtain a required axial load that would be technologically possible. [4].

Based on calculations, the equipment functions are [4]:

- rapid volume heating of the end of the fitting that contains the rolled joint at a constant power by induced currents.
- applying a compressive axial force to the contour of the rolled joint at the end of pressure tube until the separation of the fitting from the joint.

The application of the second functions can be:

- immediately after reaching the specified temperature on the inner surface of the fitting in the rolled joint.
- throughout the heating

To reduce the danger of damaging the surface involved in the rolled joint with the pressure tube, which would lead to the impossibility of reusing the fitting, we choose the first mode of application [4].

**Environmental impacts**

After applying this technique for the detachment of the rolled joint, the end fitting must remain intact in the groove area. The absence of Ni from the stainless steel used for the execution of the terminal fitting, leads us to sustain that the fitting decontamination would substantially reduce its activity making it reusable. [5]. This reduces the time of intervention and the dose of radiation exposure received by the personnel, and massively reduces the active material inventory that needs to be containerized, transported and stored.

**The results**

The technology for replacing the pressure tube involves removing the fuel channel assembly, which has a high degree of difficulty. Following the cuttings in the subassembly (pressure tube-end fittings) a number of active components are obtained in the form of end fittings with pressure tube fragments and the two sections of cut pipe, which need to be containerized, transported and stored.

The technique for detaching the rolled joint between the pressure tube and the end fitting, without damaging the fitting, removes the need to containerize the end fittings with pressure tube fragments, transporting and storing them so they can be reintroduced into the primary circuit by joining them with another pressure tube. Besides eliminating the transport and storage containers for end fittings resulted from the fuel channel disassembly and the opportunity to reuse them, this method shortens the intervention on fuel channel and also reduces the radiation exposure of the personnel involved.
In order to disassemble the rolled joint, we apply heating by induction to the rolled joint followed by a quick application of an axial force that will detach the pressure tube from the rolled joint without affecting the integrity of the fitting [3].

A general geometrical and physical characterization of the rolled joint between the pressure tube and the end fitting was made considering both the rolled joint geometry and some functional characteristics of the device involved in dismantling of this junction.

We established by known data calculus the rolled joining degree applied, the axial plucking forces and the maximum axial forces for cold plucking with the risk of losing the stability $Q_{\text{max}}$ and without this risk $Q_{\text{cap}}$. For the rolled joints made between the pressure tube and the end fitting, for the three grooves, the plucking resistance increased three times $Q_{\text{cap}}$, [3].

A quick volume heating applied to the end fitting in order to obtain a diameter increase compared to the outer diameter of the pressure tube remained at a constant temperature around the initial value, causes a rolled joint weakening that leads to a much reduced force for the plucking required in order to disassembly the junction.

The local diametric expansions of the end fitting were calculated for T1, T2 and T3, and the corresponding plucking forces, representing their variations with temperature in Fig 4 and 5.

![The variation of diametric expansion of the end fittings](image1)

**Fig.4** The variation of diametric expansion of the end fittings

![The variation of the plucking force of the end tube](image2)

**Fig.5** The variation of the plucking force of the end tube

**Conclusions**

- The technique for detachment of the pressure tube from the rolled joint in the technology of replacement for the pressure tube shortens the time for the intervention, reduces the radiation exposure of the personnel involved and minimizes the environmental impact by reusing the end fittings in the primary heat transmission circuit;
- estimations for the plucking forces in the cold junction disassembly were made with and without the risk of losing the stability;
- the variation of the local expansion of the end fitting was presented as a function of temperature and also the wrench force required for the disassembly of the junction with temperature;
- based on calculated estimations, the functions and sequences of application for the equipment were set.

Bibliography

ASPECTS RELATED TO THE TESTING OF SEALED RADIOACTIVE SOURCES

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ABSTRACT

Sealed radioactive sources are commonly used in a wide range of applications, such as: medical, industrial, agricultural and scientific research. The radioactive material is contained within the sealed source and the device allows the radiation to be used in a controlled way. Accidents can result if the control over a small fraction of those sources is lost. Sealed nuclear sources fall under the category of special form radioactive material, therefore they must meet safety requirements during transport according to regulations. Testing sealed radioactive sources is an important step in the conformity assessment process in order to obtain the design approval. In ICN Pitesti, the Reliability and Testing Laboratory is notified by CNCAN to perform tests on sealed radioactive sources. This paper wants to present aspects of the verifying tests on sealed capsules for Iridium-192 sources in order to demonstrate the compliance with the regulatory requirements and the program of quality assurance of the tests performed.

Key words: sealed source, test, transport, quality assurance

Introduction

Radioactive sealed sources have been used since several years for a wide range of applications in a variety of shapes, sizes and radioactivity levels. In industry, they are widely used for non-destructive testing, radiation processing, “on-line” process control systems, online elemental analysis for raw materials, semi-formed and final product, mineral resources evaluation, food irradiation and smoke detection. One well-known example of such sources is $^{192}$Ir sources for industrial radiography.

The miniature size of these sources and the high degree of uniformity in activity distribution needed technical challenges in the process production and development of quality control methodologies. Same aspect related to necessity of specially designed and remotely operated positioning systems for source assembling and sealing (welding) in a hot cell, offers another area of interest.

A sealed radioactive source is classified as “Law dispersible radioactive material“ in accordance with Specific Safety Requirements –SSR no.6:2012[1]. For use of radioactive sources is necessary to require the approval of National Competent Authority. An application for approval include: a description of the radioactive material (capsule and content), a statement of the design of capsule, a statement of the tests that have been carried out and their results, a specification of applicable management system, any proposed pre-shipment actions, etc.
This paper presents some aspects related to quality assurance in the production process of $^{192}$Ir sealed sources for industrial radiography, as well as in the testing process of the welding sealed capsules in INR Pitesti.

**The $^{192}$Ir sealed sources for industrial radiography**

$^{192}$Ir is one of the most widely used sealed sources for radiography applications. Taking into account the relatively short half-life of the isotope (74 days), it is essential to organize the production of sources as close as possible to the client to avoid logistic problems.

The assembling and welding techniques of $^{192}$Ir sealed sources for industrial applications were optimized. An active core of this kind of source is a set of irradiated iridium disks with a diameter of 0.5 to 3.5 mm and thickness of 0.2 to 0.5 mm (depending on required source activity and dimensions). The sources of radiotherapy made in INR are disks of 3 mm with a maximum activity of 4,440 TBq (120 Ci). The capsules made from stainless steel are sealed by TIG welding. This design provides a source classification C 43515 according to ISO 2919: 2012 [2].

Radioactive sources are manufactured in accordance with strict control methods. Stringent tests for leakage are an essential feature of radioactive source production. The methods adopted depend on the design and intended application source, and also on statutory requirements, and the standard method is ISO 2919:2012. A test report is delivered for each source or batch of sources.

**The Quality Assurance System**

In order to guarantee sustainable assurance of test results delivered to the approving authority as well as to applicants or customers, Reliability and Testing Laboratory, operates and maintains an effective quality management system designed to ensure that the applied test methods meet international accepted quality requirements.

This complete quality assurance plan encompasses both, quality assurance and quality control functions. Quality Assurance involves meeting programmatic requirements but on occasion requires the implementation of external checks on testing quality. These external checks include independent system audits, third party sample and analysis for accuracy and precision or comparison to calibration standards. Quality Assurance audits confirmed that operational and maintenance procedure and quality control are a series of frequent routine internal checks, such as system inspections, periodic calibrations, and routine maintenance.

The Quality Management System developed and implemented by INR’s Reliability and Testing Laboratory is based on the international standards-ISO 9001:2008 [3] and ISO CEI 17025:2005 [4] and includes the processes necessary to achieve the organization's overall objectives that consist of:

- The description of the management system in the Quality Management Manual of Reliability and Testing Laboratory - MC LIF [5];
- Documents that describe the management system processes-system procedures;
- Detailed work control documents - technical procedures, instructions, checklists, process control cards and forms.

A significant part of Quality Management System involves documentation and the scheme of a documentation structure for a Quality Management System of INR’s Reliability and Testing Laboratory is given in Figure 1.
Following the implementation of Quality Management System the INR’s Reliability and Testing Laboratory was notified by the National Commission for Nuclear Activities Control (CNCAN) through the NOTIFICATION no. LI 02/2015.

The Quality Management System implemented by the Reliability and Testing Laboratory is intended to assure that the sealed source testing are accomplished in a quality assured way, using appropriate equipment and calibrated instruments working within their recognized capabilities and within the limits of accuracy. Only by controlling all design-related activities in such a way, a manufacturer, user or certifying body have a reasonable assurance that the tested sealed sources complies with the designer's requirements.

The quality control programme is implemented for sealed sources manufacturing in INR. For manufacturing sources the supplies and materials have to pass inspection tests. They should have certificate for their identification. Sources must be hermetically sealed. Leakage tests should be provided in accordance with immersion method 5.1.1. (SR ISO 9978:1996)[6]. The measured activity of liquid where sources have been treated must not exceed 0,2 kBq.

Estimation of source (or simulated sealed source) leak tight and strength must be carried out according to the result of examination before and after carrying out tests. Not less than two sources of the type specified are to be subjected to each test. The programme of tests is developed according to ISO 2919:2012 [2]. When changing the design and the technology for manufacturing the source of the type specified which influence on its safe application as to purpose, new sources must be tested.

**Experimental tests for welding capsules**

The sealed radioactive sources must comply with the performance tests and general requirements in accordance with classification system established in International Standard ISO 2919:2012[2]. This standard provide a set of tests by which manufacturers of sealed radioactive sources can evaluate the safety of their products in use, and users of such sources can select types which are suitable for required application.
The tests fall into several groups including exposure to abnormally high and low temperatures and a variety of mechanical tests. Each test can be applied in several degrees of severity, the criteria of pass or fail depends on leaking contents of the sealed radioactive source.

The radioactive sources for industrial radiography must comply the following tests:
- temperature test;
- external pressure test;
- impact test;
- puncture test.

**General requirements**

All tests, except the temperature tests, were carried out at ambient temperature. For each test, at least two test sources of the model type shall be subjected to the test, and shall pass the criteria as defined in ISO 2919:2012[2] point 7.1.5. “Compliance with the tests was determined by the ability of the sealed source to maintain its leak tightness after each test has been performed. After each test the source shall be examined visually for loss of integrity and it shall also pass an appropriate leakage test carried out in accordance with SR ISO 9978[6]”.

The tests, for welding capsules for $^{192}$Ir sealed source for industrial radiography, were performed on three capsules marked P1, P2 and P3 as shown in Fig.2:

![Fig.2. The sealed welding sources](image)

**External examination**

External examination should be fulfilled by visual inspection giving particular attention to quality of welding joints (the plug and the capsule). Correspondence of dimensions of the assembled source with requirements to specifications is checked with the certified gauges ensuring measurement accuracy by drawings.

**Puncture test**

Puncture test was performed in accordance with ISO 2919[2] point 7.6. The 300g steel hammer (Figure no.3), with the pin rigidly fixed at the lower part, was dropped onto sealed source positioned on the hardened steel anvil. The hammer is dropped from the height of 1 m onto the top of sealed source by the
means of the smooth vertical tube (Figure no.4). The pin is positioned to drop as close as possible to the welding joint.

**Fig.3.** The steel hammer  **Fig.4.** The device for the puncture test

After the puncture test each source (Figure no 5) was examined visually for loss of integrity and a leakage test with helium was conducted to verify its leak tightness. All sources fulfil the criteria.

**Fig.5.** The sources after the puncture test

**Impact test**

Impact test was performed in accordance with ISO 2919[2] point 7.4. The impact hammer (Figure no.6) with the mass of 5 Kg was dropped onto sealed source positioned on the hardened steel anvil with the diameter of section of the flat striking surface of 25 mm (with its outer edge rounded to a radius of 3mm). The drop height, measured between the top of the sealed source, positioned on the anvil, and the face of the hammer in its position prior to release, was 1 m. The hammer is dropped from the height of 1 m onto the top of sealed source by the means of the smooth vertical tube (Figure no.7):
After the impact test, each source (Figure no 8) was examined visually for loss of integrity and a leakage test with helium was conducted to verify its leak tightness. All sources fulfil the criteria.

**Temperature test and thermal shock**

The heating or cooling test of sealed capsules, are made in the climatic chambers with a test zone volume of at least five times the volume of the specimen.

The sealed capsules were cooled at -40°C for 20 min., then were exposed to temperature below the ambient. After this test, the sealed sources were tested at thermal shock as follow: Sealed sources at ambient temperature were heated at +400°C and then were kept inside at that temperature for at least 1 h. After this time, the sources were subject to thermal shock by transferring them, within 15 s, to water at ambient temperature (about 20°C).

Details during temperature and thermal shock tests are shown in the figure no 9 and 10:
After the temperature test and thermal shock, each source (Figure no 10) was examined visually for loss of integrity and a leakage test with helium was conducted to verify its leak tightness. All sources fulfil the criteria.

**External pressure test**

The test was carried out in a special chamber at 25 kPa pressure for two periods of 5 min each. Between these periods the pressure was returned at atmospheric pressure.

After the external pressure test, each source was examined visually for loss of integrity and a leakage test with helium was conducted to verify its leak tightness. All sources fulfil the criteria.

**Conclusions**

The adequate implementation of the Quality Management System applicable to testing activities will assure a greater safety for the testing of sealed radioactive sources. When a task is carried out in a quality assured manner the whole process employed by the company or organization involved will evolve systematically. Through the application of an appropriate QA data base record, safety and compliance of the sealed radioactive sources can be easily demonstrated to the third parties.

A consistently high quality and reliability of welding capsules tests, as an integral part of design approval procedure for radioactive sealed $^{192}$Ir sources, can only be achieved if the quality requirements for manufacture and testing of prototypes and test models are clearly defined, if the requirements for the quality management system are specified and the quality control methods are documented accurately. Based on a quality assurance system, the test results can provide objective evidence for tested specimens or prototypes within the design source approval in order to demonstrate the compliance with safety requirements.

**References**

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DESIGNING ADVANCED MATERIALS BY ENVIRONMENTAL FRIENDLY PLASMA ELECTROLYTIC OXIDATION

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ABSTRACT

In the CANDU-PHWR nuclear reactors, Zr-2.5Nb coated with a black adherent oxide film of $\approx 1$ to 2 $\mu$m in thickness is currently used for the manufacture of pressure tubes. The black oxide thin film has corrosion protective properties. However, it can be damaged during the regular refueling process, thus causing hydrogen/oxygen ingestion. Therefore, an enhanced wear and corrosion resistance coating is needed. Plasma electrolytic oxidation (PEO) is an anodic electrochemical treatment, both cost-effective and environmentally friendly, widely used in the formation of a protective oxide film on the metal surface to enhance wear and corrosion resistance as well as prolonging component lifetime. The state of the art reveals that PEO method is suitable for improving the wear resistance of Zr-2.5Nb alloy. Few studies are performed in this field and thus, it is necessary to conduct a more detailed insight study on the processing parameters for PEO treatment. By understanding the influence of process parameters, such as electrolyte temperature and electrolyte composition, we can find the way to obtain a coating with improved mechanical and corrosion properties on zirconium alloys.

Key words: zirconium alloys, black oxide film, plasma electrolytic oxidation (PEO), wear, corrosion resistance

Introduction

Manufactured for the first time in 1957, the pressure tube is a major component of the CANDU reactor. Each pressure tube in a CANDU contains 12 fuel bundles (500 mm long) and heavy water coolant, which is thermally insulated from a low-pressure cooling moderator by a gas annulus formed between the pressure tube and the calandria tube surrounding it [1].
Originally, the reactors used Zircaloy-2. Later, the Zr-2.5Nb alloy was selected to be used because this alloy had a better creep resistance and its higher strength permitted the use of a thinner wall tube with a resultant advantage in neutron economy over Zircaloy-2 in the reactor [3].

The Zr-2.5Nb pressure tubes are heated for 24hr at 400°C in an autoclave to form, on the both inner and outer ring surfaces, a black adherent oxide film of ~1 to 2 μm in thickness [4]. This commercially thin black oxide film has corrosion protective properties. During the refueling process, the oxide film could be damaged, thus causing hydrogen/oxygen ingress. Therefore, an enhanced wear and corrosion resistance coating is needed.

When an abrupt change it happens in the corrosion resistance, the oxide film, in a relatively short time, changes its color to gray or white and becomes friable and spalling occurs. This change in corrosion resistance is called “transition”. This phenomenon (the “transition”) is characteristic of zirconium and its alloys and limits the life time of the metal. In the nuclear industry, the pressure tubes in CANDU reactors have the life time operation for about 25 years.

Recently, different studies on aluminum, magnesium, titanium and zirconium have been employed in order to increase wear and corrosion resistance.

**Plasma Electrolytic Oxidation**

Plasma electrolytic oxidation (PEO) is a relatively recent surface modification technique providing ceramic coatings with improved corrosion and wear resistance on the metals surface such as aluminum, magnesium, titanium, zirconium and their alloys. This process implies plasma generation in spark discharge channels on the electrode of a metal–electrolyte system under the action of external electric field, chemical reactions between the metal and plasma components, and deposition of the reaction products on the electrode. An additional advantage of the PEO technology is its environmental friendliness and relative low costs [5,6].
**The PEO plant**

![PEO plant diagram](image_url)

**Fig. 2.** Schematic description of the PEO chamber and micro-discharges photo. C – cathode, A – anode, PP – peristaltic pump, HE – heat exchanger [7].

Metal alloys such as aluminum, magnesium, titanium, zirconium can be used as anode material. Rectangular shaped anodes will be placed in electrolytic chamber, see Figure 2, leaving an active surface area. In this example, two platinum wires are used as cathodes. During PEO, the electrolyte is circulated through the chamber-reservoir system (peristaltic pump and heat exchanger) and its temperature is measured in the close vicinity of the anode [7].

PEO is applied above the breakdown voltages of the original oxide films, typically in the range of 400-800 V. For the PEO process can be used various types of power sources including direct current (DC), pulsed DC and alternating current (AC) [5].

**The PEO Mechanism**

In the coating growth mechanism, discharges play a very important role. In PEO process the discharge events are very short and this makes very difficult to catch them instantaneously to analyze the physical and chemical processes occurring in the discharge channels. Thereby, an existing controversy over the growth mechanism of PEO coatings occurs [9].

The studies show that the PEO coating grows simultaneously above and below the original substrate surface by the combination of two growth mechanisms:

- an outer growth, from the substrate towards the electrolyte, by the melting, oxidation and solidification of ejected species;
- an inner growth into the substrate by an oxygen transport due to the high electric field.
During the inner growth, oxygen anions were transferred into the coating and react with metal cations from the metal substrate to form an oxide ceramic coating. Due to the high cooling rate enforced by the cold substrate, the molten oxide at the coating/substrate interface rapidly solidifies, creating a thin crystalline layer with small uniform nano-sized grains. The nano-crystalline layer is constantly formed during PEO and moves inwards by ‘eating’ the substrate and is considered as the main inner growth mechanism, see Figure 3 [9,10].

State of the art concerning PEO of Zr-2.5Nb alloy

Few studies were performed for improving the wear resistance of Zr-2.5Nb alloy. Chen [1] has studied the process of plasma electrolytic oxidation of zirconium alloy Zr-2.5Nb and he has compared the ceramic oxides coatings obtained with the black oxide coating produced commercially. After the rotating mode wear tests, the SEM analysis shows that the PEO coating obtained at small current densities has excellent wear resistance (Fig 4 b). The wear trace on black oxide coating was much wider and deeper and it is clearly seen that all coating material was removed from the substrate (Fig 4 a). Abrasive wear is clearly observed on black oxide coating with wear debris presents in the wear track. For the PEO coating, no obvious abrasive wear and material removal was observed. Thus, the PEO coating had much better wear resistance than the black oxide coating and less wear debris was produced which could lead to contamination of the service environment.

Fig. 4 SEM showing wear traces of (a) the commercially black oxide coating and (b) PEO coating [1].
The PEO coating obtained at high current densities showed an improvement of the corrosion resistance of 130 times compared to the uncoated substrate (Zr-2.5Nb), while the commercially black oxide coating is 293 times higher than the substrate. The conclusion was that the commercially black oxide has the best corrosion resistance compared with the PEO coatings obtained until now.

Some prospects for further research work

The factors that affect the PEO processes are: the substrate material, the electrolyte, the temperature, the oxidation time, the electrical parameters, the additives etc. The electrolyte temperature can greatly affect the PEO process. If the temperature is too low, the oxidation process becomes weak, resulting in less thickness and lowers hardness of the PEO coatings. If the temperature is too high, the dissolution of oxide film will be enhanced, and thus cause the coating thickness and hardness to decrease significantly [11]. Therefore, the processing temperature should be studied. Generally, in the PEO process the electrolyte temperature is controlled in the range of 20 - 40°C [11]. But, this is not a rule; this domain depends on the electrolyte - substrate system used first. (K₂Al₂O₄ – Na₃PO₄ – NaOH) – Ti alloy system showed that coatings with more improved wear resistance were obtained at low temperatures (5°C) [12].

The studies of Chen [1] were performed below 60°C. In this context, we consider very important to study the influence of the processing temperature on the coatings properties obtained on Zr-2.5Nb using different electrolytes.

Conclusions

- PEO technique is suitable for improving the wear and corrosion resistances of the zirconium alloys;
- Low current densities increase the wear resistance, while high current densities enhance the corrosion resistance; the black oxide coating has the best corrosion resistance, compared with the coatings obtained until now on Zr-2.5Nb;
- The electrolyte temperature is a critical factor influencing the PEO coating properties; a study about the influence of electrolyte temperature on the formation of PEO coatings on Zr-2.5Nb does not exist.

References


III.2 INTERNATIONAL COOPERATION
ASSESSMENT OF NUCLEAR ENERGY COST COMPETITIVENESS AGAINST ALTERNATIVE ENERGY SOURCES IN ROMANIA ENVISAGING THE LONG-TERM NATIONAL ENERGY SUSTAINABILITY

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ABSTRACT

The paper includes some of the results obtained by RATEN ICN Pitesti experts in the IAEA’s Collaborative Project INPRO-SYNERGIES. The case study proposed to evaluate and analyze the nuclear capacity development and increasing of its share in the national energy sector, envisaging the long term national and regional energy sustainability by keeping collaboration options open for the future while bringing solutions to short/medium-term challenges. The following technologies, considered as future competing technologies for electric energy generation in Romania, were selected: nuclear technology (represented by PHWR CANDU Units 3 and 4 – CANDU new, advanced HWR – Adv. HWR, and advanced PWR – Adv. PWR) and, as alternative energy sources, classical technology (represented by Coal-fired power plant using lignite fossil fuel, with carbon capture – Coal_new, and Gas-fired power plant operating on combined cycle, with carbon capture – Gas_new). The study included assessment of specific economic indicators, sensitivity analyses being performed on Levelised Unit Energy Cost (LUEC) variation due to different perturbations (e.g. discount rate, overnight costs, etc). Robustness indices (RI) of LUEC were also calculated by considering simultaneous variation of input parameters for the considered power plants. The economic analyses have been performed by using the IAEA’s NEST program. The study results confirmed that in Romania, under the national specific conditions defined, electricity produced by nuclear power plants is cost competitive against coal and gas fired power plants electricity. The highest impact of considered perturbations on LUEC has been observed for capital intensive technologies (nuclear technologies) comparatively with the classic power plants, especially for discount rate changes.

Key words: INPRO, nuclear energy system, cost competitiveness, LUEC, long term sustainable development

Introduction

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of the International Atomic Energy Agency (IAEA) was established in 2000 to help ensure that nuclear energy is available to contribute to meeting the energy needs of the 21st century in a sustainable manner. INPRO’s activities are organized in four major tasks (Global Scenarios; Innovations; Strategies; Policy and Dialogue) consisting in collaborative projects, dialogue forums and other activities, [1]. Romania participated in INPRÓ activities starting from 2007, as observer in the Steering Committee meetings (2007-2011), after 2012 getting involved as a member in Collaborative Projects, Dialogue Forums and Steering Committee
activities. Raten ICN Pitesti is involved in four INPRO Collaborative Projects (SYNERGIES completed by the end of 2015, KIND, ROADMAPS, FANES) and the Dialogue Forum activities.

The main objective of SYNERGIES CP was to identify and evaluate mutually beneficial collaborative architectures and the driving forces and impediments for achieving globally sustainable Nuclear Energy Systems (NES), being focused on the synergies related to the nuclear fuel cycle (NFC) and specifically regional and/or global collaborative approaches providing Member States the opportunity to keep options open to the future while bringing solutions to short/medium-term challenges, [2].

Based on the case studies performed by project participants, SYNERGIES project modelled and examined various synergies among nuclear technologies and forms of collaboration among nuclear technology suppliers and users in order to identify mutually beneficial strategies for working together to promote the sustainable expansion of nuclear energy worldwide and to identify the corresponding driving forces and possible impediments involved in achieving globally sustainable NES.

The Romania’s case study proposed to analyze the development of nuclear capacity and increasing of its share in the national energy mix in order to assure the sustainability, by considering regional collaborative architectures both in the front-end and back-end of the nuclear fuel cycle. The assessment of several potential national nuclear energy development scenarios took into consideration both the collaboration in NFC based on “win-win” approach and the current existing and near-term projected technologies and infrastructure. [3].

Romanian Nuclear Programme started in 1950. Two research reactors were commissioned, VVR-S in 1957 (decommissioning started in 1997) and TRIGA 14MW in 1979. Romania’s current policy is for a Once-Through Nuclear Fuel Cycle based on indigenous facilities, without enrichment or reprocessing (which are prohibited by national laws). The front-end activities are carried on in the U ore mines, Feldioara UO2 Powder Plant, Nuclear Fuel Plant from Pitesti and Heavy Water Plant from Drobeta Turnu-Severin. The nuclear electricity generation is assured by the operation of the Cernavoda Nuclear Power Plant (NPP) with two PWHR reactors, CANDU 6 type (700 MWe each, Unit1 since December 1996 and Unit2 since 2007, respectively). The management of Spent Fuel (SF) at Cernavoda NPP is assured by the interim wet storage in SF Bay (at least for 6 years), interim dry storage, Canadian MACSTORE type (for 30 years; first module became operational in 2003) and interim storage for solid Radioactive Waste. Final disposal of LILW from Cernavoda NPP (currently in stage of site authorization) is based on a Near-Surface Repository with multiple barriers (Saligny site, inside the NPP’s exclusion zone). Research is carried on the geological environment for SF and HLW Deep Geological Repository (very preliminary stage). The National Repository for LILW at Baita operates since 1985, [4].

Romania has a balanced portfolio of electric energy generation capacity comprising hydro, nuclear, coal and gas-fired power plants, with renewable (other than hydropower) representing a small but rapidly growing subsector of the generation market. The electricity generated by Cernavoda NPP represents about 20% from the national electricity production. Romanian electricity generation sector is facing major challenges as a significant percentage of the generation assets are already past their useful technical life (30% are ~ 40 years old). Taking into account that ~ 28% (5.5 GW) of the total installed capacity must be replaced by 2020, and ~ 55% (11 GW) by 2035, Romanian Government considers nuclear power as a stable component of the national energy-mix taking into consideration security of supply, reliability, economic efficiency, and GHG low emissions, [5].

In order to assure a useful technical support for the nuclear energy long-term sustainable development strategic approach, Romania’s case study performed under IAEA SYNERGIES CP framework included an economic analysis focused on specific economic parameters calculation, such as: Levelised Unit Energy Cost - LUEC, Internal Rate of Return - IRR, Return on Investment - ROI, Net Present Value – NPV, and Total investment costs/ Investment limit.
The main objective of the economic analysis consisted in the assessment of nuclear energy cost competitiveness comparatively with other competing technologies for electric energy generation in Romania, namely classical technology represented by coal and gas – fired power plants. Sensitivity analyses have been performed highlighting the effect of various perturbations on LUEC (e.g. discount rate, fixed O&M costs, overnight costs, etc). To confirm the validity of the economic analysis, Robustness indices of LUEC were calculated by considering simultaneous variation of several input parameters for the nuclear and alternative source (coal and gas) power plant.

Specific economic parameters calculation

The economic analysis has been performed by using the IAEA’s NEST (NESA economic support tool), available on the IAEA official website, IAEA/INPRO section, [6]. For the above mentioned specific economic parameters (so called financial figures of merit) calculation, the formula and support provided in [7, 8] have been used.

INPRO methodology recommends using of the levelised discounted costs, also called Levelised Unit Energy Costs (LUEC), as input for comparing electricity production costs of different plants. LUEC is equivalent to the average price that would have to be paid by consumers for electricity delivered at the plant "gate" to repay exactly all costs incurred by the owner/operator of a plant at the selected discount rate in a defined time frame (lifetime of the plant) and without profits. LUEC includes three factors, namely: the capital costs, the operation and maintenance costs and the fuel costs. LUEC, [mill$/kWh] or [10^{-3} $/kWh], is equivalent to the average price that would have to be paid by consumers to repay exactly for capital, O&M and fuel costs, with a proper discount rate (and without profits), [7, 8].

\[
\text{LUEC} = \text{LUAC} + \text{LUOM} + \text{LUFC}
\]

Discount rate is a very important economic term and is used as input parameter to calculate levelised costs. This rate takes the time value of money into account, i.e. money earned in the future has less value than received today. The value of the discount rate is linked to the interest that an investor has to pay for long term bonds.

Levelised unit lifecycle amortization cost, LUAC, consists of "specific" overnight costs, "specific" interest during construction (specific costs are obtained by the division to the electric power of the plant), levelised back fitting (exist only in the case of plant design envisages life time extension by the replacement of parts of main equipment) and levelised decommissioning costs.

Levelised unit lifecycle operation and maintenance cost (including refurbishment cost), LUOM, includes all costs save power plant construction/decommissioning and fuel front-end/back-end costs (e.g. staff salaries, auxiliary equipment and materials purchasing, refurbishment of buildings and equipment, non-fuel waste management etc).

Levelised unit lifecycle fuel cost, LUFC, represents the levelised cost of the fuel including both front-end and back-end per unit of electric energy received from this fuel.

There are two aspects of the investment, somewhat related one to the other, namely, the attractiveness of the investment in terms of the financial return to be expected and the size of the investment that is required. The total investment comprises the costs to adapt a given design to a given site, and then to construct and commission the power plant, including the interest during construction that depends both on the construction time and the time to commission. The attractiveness of an investment is usually quantified by determining IRR, ROI and NPV of cash flows. Private sector investors will be attracted by a competitive IRR, corresponding to associated risks. However, NPV of cash flows may be more suitable
for government investors because takes into account other benefits, such as security of energy supply and technology development.

**Internal rate of return, IRR**, represents the discount rate that makes the net present value of all cash flows from a particular project equal to zero. The higher a project's internal rate of return, the more attractive is to undertake the project, [7, 8].

\[
\begin{align*}
\text{IRR} &= r \\
\text{LUEC}(r) &= \text{PUES}
\end{align*}
\]

where \( r \) = the real discount rate and \( \text{PUES} \) = the reference price per unit of electricity sold or the selling price of electricity, [mill$/kWh] or \( 10^{-3} \$/kWh].

**Return on investment, ROI**, is frequently derived as the "return" (incremental gain) from an action divided by the cost of that action (really one divides it by "total overnight cost"); this parameter is not levelised, [7, 8]. The higher a project’s ROI, the more attractive is to undertake the project.

\[
\text{ROI} = \frac{\text{PUES} - \text{OM} - \text{FC}}{\left( \frac{\text{CI}}{\text{P}} \right)_{\text{ON}}} \times 8760 \times L_f
\]

where: \( \text{OM} \) = unit lifecycle operation and maintenance cost - includes simple (not levelised) sum of all costs save power plant construction/decommissioning and fuel front-end/back-end costs (e.g. power plant staff salaries, auxiliary equipment and materials purchasing, non-fuel waste management etc.), \( \text{FC} \) = unit lifecycle fuel cost - represents the cost simple (not levelised) of the fuel including both front-end and back-end per unit of electric energy received from this fuel; \( (\text{CI}/\text{P})_{\text{ON}} \) = Total Overnight cost (per unit of installed capacity), including contingency and owner costs; \( L_f \) = average load factor.

**Net Present Value, NPV**, can be used as an evaluation parameter for measuring the net financial benefit of a project investment, [7, 8]. Since it represents the total net value of the investment, discounted to time 0; its absolute value will depend on the size of the investment.

\[
\text{NPV} = \text{PUES} - \text{LUEC} \times L_{fp}, \quad L_{fp} = 8760 \times L_f \times \left( \frac{1 - \left( \frac{1}{1 + r} \right)^{\gamma_{inf}}}{1 - \left( \frac{1}{1 + r} \right)} \right)
\]

**Total investment costs** are given by, [7, 8]: Investment costs x Net Capacity of the power plant. Investment costs include Overnight costs (pre-construction/owner’s costs, construction - engineering, procurement and construction) - costs and contingency costs) and Interest during construction.

\[
\text{INV}_{\text{N}} = \left( \frac{\text{CI}}{\text{P}} \right)_{\text{ON}} + \left( \frac{\text{CI}}{\text{P}} \right)_{\text{IDC}} \times P
\]

where \( (\text{CI}/\text{P})_{\text{ON}} \) = Total Overnight cost (per unit of installed capacity), including contingency and owner costs; \( (\text{CI}/\text{P})_{\text{IDC}} \) = Interest during construction cost (per unit of installed capacity); \( P \) = installed net electric capacity of the plant unit, [kW(e)].

**Investment limit** is the maximum investment a private company can afford, taking into account the (private) market conditions the company is working in, [7, 8].

\[
\text{INV}_{\text{LIMIT}} = M \times Sh \times Pm \times t_{\text{growth}} \times \alpha
\]
with \( M = \) total income from the considered market, [\$]; \( Sh = \) market share of the tentative plant owner; \( Pm = \) profit margin of the tentative plant owner; \( t_{\text{growth}} = \) time till the electricity consumption growth is equal to plant installed capacity; \( \alpha = \) adjusting coefficient.

In order to validate the economic analysis performed for the selected technologies, Robustness indices, \( RI \), of LUEC have been calculated by simultaneous variation of input parameters of the nuclear and alternative source (coal and gas) power plant, [7].

\[
RI = \min \left( \frac{LUEC_{\text{Alternative}}}{LUEC^*_{\text{Nuclear}}}, \frac{LUEC^*_{\text{Alternative}}}{LUEC_{\text{Nuclear}}} \right); \quad \text{for perturbed value of LUEC parameter} \quad (7)
\]

**Input data used for figures of merit calculation**

The proposed economic analysis has been performed for five types of power plants competing in Romania’s national energy system for electricity generation, representing: nuclear technology (CANDU_new, Adv. PWR and Adv. HWR) and classical technology on fossil fuels, but using advanced technologies for \( \text{CO}_2 \) capture (Coal_new and Gas_new).

The electricity generation calculated costs are plant-level costs, at the station, and do not include transmission and distribution costs. In the comparative study, for the initial capital investment the uniform investment schedule has been used for all considered technologies.

In Tables 1-4, the basic assumptions for the comparative economic study are given. The values presented in the tables have been collected from [9-23] and were used as initial input values for the NEST calculations.

**Table 1 - Country specific economic input parameters, [9, 20, 21]**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Discount rate</td>
<td>1/year</td>
<td>0.08</td>
</tr>
<tr>
<td>Price of unit electricity sold</td>
<td>mills/kWh</td>
<td>112</td>
</tr>
<tr>
<td>Tax rate</td>
<td>%/100</td>
<td>0.5</td>
</tr>
<tr>
<td>Market income *</td>
<td>MS/year</td>
<td>3800</td>
</tr>
<tr>
<td>Market share *</td>
<td>%/100</td>
<td>0.5</td>
</tr>
<tr>
<td>Profit margin *</td>
<td>%/100</td>
<td>0.2</td>
</tr>
<tr>
<td>Time of growth *</td>
<td>years</td>
<td>6</td>
</tr>
<tr>
<td>Adjusting coefficient *</td>
<td>%/100</td>
<td>2</td>
</tr>
</tbody>
</table>

* Above parameters are used only for investment limit calculation, according to country specifics

By taking into consideration the parameters presented above, the calculated *Investment limit* for the case study economic analysis was \( 4560 \cdot 10^9 \) \$. 

Taking into account that LUEC has a central role in the energy cost competitiveness assessment, its sensitivity to different parameters variation was studied. Figure 1 (a-f) illustrates the impact on LUEC in the conditions of perturbation on annual discount rate, total construction costs, fuel costs, load factor, power plant life time and construction schedule compliance.
Table 5 – Calculated values for economic parameters of interest

<table>
<thead>
<tr>
<th>Technology</th>
<th>Net Capacity [GW(e)]</th>
<th>Overnight Costs [$/kWe]</th>
<th>Investment Costs ** [$/kWe]</th>
<th>Total Investment *** [10^9 $]</th>
<th>LUEC [10^3 $/kWh]</th>
<th>IRR</th>
<th>ROI</th>
<th>NPV [$/kWe]</th>
</tr>
</thead>
<tbody>
<tr>
<td>CANDU new</td>
<td>0.720</td>
<td>5820</td>
<td>7375</td>
<td>5310</td>
<td>64.11</td>
<td>0.129</td>
<td>0.286</td>
<td>5024</td>
</tr>
<tr>
<td>Adv. PWR</td>
<td>1.000</td>
<td>3400</td>
<td>4653</td>
<td>4503</td>
<td>47.71</td>
<td>0.164</td>
<td>0.460</td>
<td>6775</td>
</tr>
<tr>
<td>Adv. HWR</td>
<td>0.720</td>
<td>3000</td>
<td>3973</td>
<td>2861</td>
<td>39.58</td>
<td>0.184</td>
<td>0.537</td>
<td>7632</td>
</tr>
<tr>
<td>Coal_new</td>
<td>0.400</td>
<td>1520</td>
<td>1849</td>
<td>740</td>
<td>77.25</td>
<td>0.232</td>
<td>0.436</td>
<td>2874</td>
</tr>
<tr>
<td>Gas_new</td>
<td>0.400</td>
<td>1099</td>
<td>1234</td>
<td>494</td>
<td>92.04</td>
<td>0.313</td>
<td>0.414</td>
<td>1512</td>
</tr>
</tbody>
</table>

* Include pre-construction/owner’s, construction and contingency costs; ** Include Overnight Costs and Interest During Construction; *** Is given by Investment costs multiplied by power plant Net Capacity

Fig. 1. Impact on LUEC due to variation in: a) annual discount rate; b) overnight costs; c) fuel costs; d) load factor; e) plant lifetime; f) construction schedule compliance
Robustness indices of Levelised Unit Electricity Cost were calculated by simultaneous variation of several input parameters for the nuclear and alternative source (coal and gas) power plants, such as: plant lifetime, average load factor, overnight capital costs, delay in construction schedule, fuel costs, nuclear back-end cost together with fossil fuel price escalation rate, and thermal efficiency of the power plant (see Table 6).

### Table 6 Robustness indices of Levelised Unit Electricity Cost

<table>
<thead>
<tr>
<th>Perturbed parameter/ NPP type</th>
<th>Perturbation of NPP data</th>
<th>Perturbation of Classic PP data</th>
<th>Robustness index against Coal_new</th>
<th>Robustness index against Gas_new</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Power plant lifetime</strong></td>
<td>- 5%</td>
<td>+ 5%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CANDU new</td>
<td></td>
<td></td>
<td>1.20</td>
<td>1.43</td>
</tr>
<tr>
<td>Adv. PWR</td>
<td></td>
<td></td>
<td>1.62</td>
<td>1.93</td>
</tr>
<tr>
<td>Adv. HWR</td>
<td></td>
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<td>1.95</td>
<td>2.32</td>
</tr>
<tr>
<td><strong>Average load factor</strong></td>
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<td>+ 5%</td>
<td></td>
<td></td>
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<tr>
<td>CANDU new</td>
<td></td>
<td></td>
<td>1.14</td>
<td>1.36</td>
</tr>
<tr>
<td>Adv. PWR</td>
<td></td>
<td></td>
<td>1.56</td>
<td>1.86</td>
</tr>
<tr>
<td>Adv. HWR</td>
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<td>1.87</td>
<td>2.23</td>
</tr>
<tr>
<td><strong>Overnight costs</strong></td>
<td>+ 5%</td>
<td>- 5%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CANDU new</td>
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<td></td>
<td>1.16</td>
<td>1.38</td>
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<tr>
<td>Adv. HWR</td>
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<td>1.88</td>
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</tr>
<tr>
<td><strong>Construction schedule</strong></td>
<td>+ 12 month</td>
<td>- 12 month</td>
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<tr>
<td>Adv. HWR</td>
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</tr>
<tr>
<td><strong>Fuel costs</strong></td>
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<td>- 5%</td>
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<td></td>
</tr>
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<td></td>
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<td>1.37</td>
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<td></td>
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</tr>
<tr>
<td>Adv. HWR</td>
<td></td>
<td></td>
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<td>1.95</td>
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<td><strong>Nuclear back-end costs</strong></td>
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<td>- 10%</td>
<td></td>
<td></td>
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<td></td>
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<td>CANDU new</td>
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<td>Adv. HWR</td>
<td></td>
<td></td>
<td>1.88</td>
<td>2.23</td>
</tr>
</tbody>
</table>

* For nuclear technologies, perturbation was applied to the natural Uranium purchase cost

** For classical technologies, perturbation was applied to fuel price escalation

According to the INPRO methodology, [7, 8], a robustness index greater than 1.0 indicates sufficient robustness of the economic analysis results, which means that all perturbations studied in Table 6 above show acceptable results.
Regarding the calculated economic parameters presented in Table 5, the following observations have to be mentioned:

• For the selected nuclear technologies LUEC values are less than the ones calculated for the considered classical technologies. The lowest LUEC value was obtained for Adv. HWR technology, 39.58 [10^3 $/kWh], while the highest LUEC value, 92.04 [10^3 $/kWh], is associated with Gas_new technology. Among the selected nuclear technologies, CANDU new has the highest LUEC value (35% and 60% higher comparatively with Adv. PWR and Adv. HWR, respectively).

• IRR values calculated for nuclear technologies (less than 0.2) are lower than IRR values for alternative technologies (0.23 for Coal_new and 0.31 for Gas_new, respectively), due to the significantly higher capital investment required for a NPP. The lowest IRR value was obtained for CANDU new technology, 0.13. It means that, for a private investor, the considered projects of fossil fuel power plants, improved by using modern technologies, are more attractive than the NPP ones.

• ROI values calculated for nuclear technologies (about 0.5) are higher than ROI values for the considered classical technologies (about 0.4), except for CANDU new (ROI = 0.286) whose capital investment is significantly higher than those considered for the advanced NPPs.

• NPV calculated for nuclear technologies are higher than those associated with the competing classical technologies considered. The highest NPV value is obtained for Adv. HWR, 7632 [$/kWe], the lowest NPV value being calculated for Gas_new, 1512 [$/kWe].

• The value of needed capital (Total Investment costs) for the planned nuclear projects in Romania, as shown in Table 5, is lower than the calculated Investment limit, 4560 [10^9 $], except for CANDU new, 5310 [10^9 $]. It means that for CANDU new the national utility on its own would not be capable of raising the needed capital by itself, and would need a Government support. The capital investment needed for the selected classic power plants (740 [10^9 $] for Coal_new and 494 [10^9 $] for Gas_new, respectively), is much lower than those needed for nuclear projects.

The sensitivity analyses performed in the study illustrate the impact of several interest parameters variation on LUEC values (see Figure 1 a-f). The impact on LUEC due to the considered perturbations is higher for capital intensive technologies (nuclear technologies) comparatively with the classic power plants.

The variation of annual discount rate has the highest impact on nuclear technologies, calculated LUEC values being about 50% lower/higher than the reference value (LUEC for annual discount rate of 8%) for annual discount rates of 5% and 12%, respectively. Meanwhile, variation of annual discount rate doesn’t affect significantly LUEC calculated for classical technologies (less than 0.5%).

The lowest impact on calculated LUEC values corresponding to the considered competing technologies for electricity generation was registered for the variation in the power plants lifetime> Considering variations of ± 5 years and ± 10 years for the power plants lifetime, the maximum LUEC values variation was the following: less than1% for the advanced nuclear projects, about 4% for CANDU new, about 1% for Coal_new and about 3% for Gas_new.

The variation in the fuel costs have a low impact on LUEC values calculated for selected nuclear technologies (less than 0.5%), but the situation changes for the selected classic technologies where the impact of fuel costs variation on corresponding calculated LUEC was rather high, namely 8% for Coal_new and 10% for Gas_new.
Conclusions

The study performed by Romania in the framework of IAEA INPRO-SYNERGIES collaborative project proved that Nuclear Energy is an important candidate for the national production of electricity, in conditions of cost competitiveness, safety and security of supply. In order to assure the projected national electricity demand, the Nuclear Energy share in the National Energy Mix can be increased from the present value (about 20% from the total production of electric energy) according to the strategic documents in force.

Specific economic parameters have been calculated, as follows: Levelised Unit Energy Cost - LUEC, Internal Rate of Return - IRR, Return on Investment - ROI, Net Present Value – NPV, and Total investment costs/Investment limit.

The following competing technologies for electric energy generation in Romania have been considered in the economic analysis: nuclear technology (represented by PHWR CANDU Units 3 and 4 – CANDU new, advanced HWR – Adv. HWR, and advanced PWR – Adv. PWR) and, as alternative energy sources, classical technology (represented by Coal-fired power plant using lignite fossil fuel, with carbon capture – Coal_new, and Gas-fired Power Plant operating on combined cycle, with carbon capture – Gas_new).

For the selected nuclear technologies LUEC values are less than the ones calculated for the selected classical technologies.

CANDU new has a significantly higher capital investment comparatively with the other selected nuclear technologies, aspect that can be explained by the multiple delays and financing challenges registered in the CANDU Units 3 and 4 project, including the investors withdrawal from the consortium in 2011-2013.

The selected nuclear technologies were better in the comparison against the considered fossil fuel power plants, only by ROI and NPV parameters. Based on IRR parameter comparison, nuclear technologies are less attractive than the coal and gas classic technologies. However, IRR = 0.164 (Adv. PWR) and IRR = 0.184 (Adv. HWR) are higher enough to consider that the nuclear technology can become attractive for long term development of Romania’s national energy system. Nevertheless, the value of IRR for the selected nuclear technologies is high enough for the Government to accept the nuclear project, taking into account strategic considerations such as increased security of supply by diversification of energy sources.

The total investment needed for selected nuclear technologies was lower than Investment limit calculated taking into consideration specific national financial environment for Romania, except for CANDU new. It should be noticed that the capital investment needed for considered classic power plants is much lower than the one needed for nuclear projects. Still, it must take into account the Government long term commitment for nuclear energy and the strategic considerations such as increased security of supply by diversification of energy sources, reducing the greenhouse gases emission, the link to the European Energy Policy, and the actions for minimization of the global climate changes impact.

Sensitivity analyses have been performed highlighting the effect of various perturbations on LUEC (e.g. discount rate, fixed O&M costs, overnight costs, etc). To confirm the validity of the economic analysis, Robustness indices of LUEC were calculated by considering simultaneous variation of several input parameters for the nuclear and alternative source (coal and gas) power plant.

The highest impact on LUEC due to the considered perturbations has been observed for capital intensive technologies (nuclear technologies) comparatively with the classic power plants, especially for the annual discount rate changes. The variation of the power plants lifetime registered the lowest impact on calculated LUEC values for considered competing technologies for electricity generation in Romania’s
national energy system. The fuel costs changes led to a low impact on LUEC calculated for considered nuclear technologies, but for the selected classic technologies the impact of fuel costs variation on corresponding LUEC was rather high.

The economic analysis performed in order to offer a useful technical support for the nuclear energy long-term sustainable development strategic approach, highlighted the nuclear energy cost competitiveness comparatively with other competing technologies for electric energy generation in Romania, namely classical technology represented by coal and gas – fired power plants.

References

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MODELLING HUMAN INTERACTIONS IN THE ASSESSMENT OF MAN-MADE HAZARDS

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ABSTRACT

The human reliability assessment tools are not currently capable to model adequately the human ability to adapt, to innovate and to manage under extreme situations. The paper presents the results obtained by ICN PSA team in the frame of FP7 Advanced Safety Assessment Methodologies: extended PSA (ASAMPSA_E) project regarding the investigation of conducting HRA in human-made hazards. The paper proposes to use a 4-steps methodology for the assessment of human interactions in the external events (Definition and modelling of human interactions; Quantification of human failure events; Recovery analysis; Review). The most relevant factors with respect to HRA for man-made hazards (response execution complexity; existence of procedures with respect to the scenario in question; time available for action; timing of cues; accessibility of equipment; harsh environmental conditions) are presented and discussed thoroughly. The challenges identified in relation to man-made hazards HRA are highlighted.

Key words: external hazards, human interactions, PSA

Introduction

The operating experience acknowledges that the effects generated by external hazards could have the potential to adversely impact the plant safety and the response of plant personnel. External events may lead to harsh personnel working conditions, problems in getting external aid and increases in emotional stress (site isolation induced as a consequence of the occurrence of a severe external event, adverse conditions for countermeasures requiring working outdoors, affecting the possibility of implementing emergency procedures).

The human reliability assessment (HRA) tools are currently not capable to model adequately the human ability to adapt, to innovate and to manage under extreme situations, even if the reliability of operator actions in mitigation of the external initiating event consequences represents a key issue for the evolution of an event. Not all the external hazards have a well-developed HRA methodology, only in case of seismic events or induced internal fire events, detailed information and HRA models are available. [1] The international nuclear community is making serious efforts trying to develop and reach consensus on the modalities to perform a HRA in the occurrences of external events context.
**Investigation of human interactions**

More systematic developments after post-Fukushima lessons learned have emphasized the importance to assess the human performances in the context of external hazards occurrences. The reconsiderations for human performance analyses include the treatment of:

- the psychological impact on operators and on persons taking decisions (uncertainties, stress, loss of trust or hope);
- the feasibility of recovery actions and time delays (some actions require long time to be performed);
- the effects of long-term scenarios (including fatigue, stress and cumulative dose);
- prioritization of work by the people who make decisions (possibly due to incorrect information concerning the status of the plant or system and information received from external organizations);
- high potential for commitment errors.

The Fukushima event showed that the events post core-damage can become much complex because of the potential loss of information and unexpected instrumentation failures or due to lack of personnel trained for accident specific conditions. The severe accident management guides (SAMG) cannot always cover the situations faced by operators. Sometimes, even supplementary resources (from off-site) do not timely secured the control of the situation, so there is a need to realistically analyze the potential time delays.

The current human reliability analysis (HRA) methods are limited in their ability to represent all important aspects of human performances due to lack of data, methodological limitations related to the limitations processing of time scale and the uncertainties regarding people behavior in accident conditions.

Among the HRA methods used in human factor analysis that may be used also to model the interventions in external events conditions, the followings can be mentioned:

- THERP (Technique for Human Error Rate Prediction); [2]
- SPAR-H (Standardized Plant Analysis Risk Human Reliability Analysis); [3]
- ATHEANA (A Technique for Human Error Analysis); [4]

THERP is an evaluation method of man-machine system degradation using: individual human error; human error in interaction with operating equipment, practices and operating procedures, and other system or human characteristics which influence the system objectives and functions. [2] This technique is used to generate quantitative estimations for actions reliability and recovery factors and represents a human reliability model defined as a set of operation relationships and principles.

The aim of SPAR-H [3] is human error probability estimation in case of actions and decisions realized as a response of operator/ intervention team to initiating events occurred in NPP. This method is based on a cognitive response and is related to a behavior model.

ATHEANA method [4] is based on a multidisciplinary framework which considers both human centered factors (man-machine interfaces, procedures content and form, training) and NPP conditions which increase the necessity for actions and create the operational causes for man-machine interactions (wrong indications, equipment unavailabilities or other configurations or abnormal operation circumstances). This HRA method is especially designed to identify, model and quantify the commission or omission human errors, in a specific complex scenario (it involve diagnosis or cognitive complexities resulting in many credible judgments from which the operator has to make a choice). A key observation representing the basis for application of this method is that the real human failure event (HFE) cannot usually randomly appear or as a result of a simple spontaneous behavior.

After an external initiator two contributions should be considered in HRA: the success of operators to follow related emergency procedures, as the success of improvised recovery actions for human and equipment failures, in opposition with inadvertent and erroneous actions having the potential to worsen the situation.
New factors should be considered in the assessment and in the estimation of HEP, different from those taken into account in internal events PSA level 1. For example, the operator actions taken outside of the control room should be evaluated for their feasibility including the necessary time, location availability, availability of the right personnel and environmental hazards (after-shocks, radiation).

The “new” factors (redefined by the context) that influence the performance are relevant to events extended in time, such as fatigue (the Fukushima operators have spent much time on-the-job) and level of stress (Fukushima operators were clearly anxious because of their personal safety).

No specific methods have been proposed up to now for modeling the impact of external hazards on the quantification of human factor in the external event PSA. The impact of external events on the quantification of human factor in the external events is in general based on the “extension” of the existing HRA methods, with the idea that the assessment of human error probabilities for external hazards should follow the basic assumptions from PSA for internal events that will be tailored on external hazard conditions. As results, more pessimistic factors in the HEP quantification, or rough modification of the quantified HEP is used.

**ASAMPSA_E project**

The ASAMPSA_E (Advanced Safety Assessment Methodologies: extended PSA) project has started on July 2013. The project is coordinated by the French Institute for Radiological Protection and Nuclear Safety (IRSN), and is based on the contributions from 28 project partners (from 19 European countries), representing well-known organizations with activities in the development and application of PSA. US-NRC, TEPCO and Japan Nuclear Safety Institute (JANSI) have also joined the project.

The project offers a new framework to discuss, at a technical level, how good is the existing PSA methodology in identification of any major risk induced by the interaction between a NPP and its environment, and to derive some technical recommendations for PSA developers and users. [7]

The ASAMPSA_E partners have examined how they can collect relevant experience or formulation of needs through an external survey for PSA stake-holders (utilities, vendors, safety authorities, research and TSO). The developed survey intends to foster useful exchanges in the PSA community, helping to specify the needs for guidance in performance and application of extended PSA.

Among further actions related to common issues for HRA (for external hazards) the following might be mentioned: [9]

- Analyse HRA modelling demands for extreme external events
- Analyse HRA modelling demands for multi-unit PSA (sharing the team/ resources between units, site management complexity - capabilities and limitations, equipment restoration possibilities, inter-reactor positive or negative effects)
- Examine how to improve HRA modelling for multi-unit external hazards conditions to tackle specific issues, such as high stress of NPP staff, number of tasks to be performed, potential lack of written operating procedures, accident management and possible site contamination

As potential challenges, the following were identified: [9]

- Human performances in the context of combinations of external hazards with other anticipated events or hazards
- Better understand, and predict human performance at all levels
- Systematic multi-unit analysis covering in particular those issues with operating and non-operating units
- Better analyses tools, easy to use, support for enhanced and more cost-effective analyses

The obtained results were presented at the ASAMPSA_E end-user workshop held in Uppsala, and it was recommended that the project shall examine how to improve HRA modelling for external hazards conditions to tackle the following issues: [8]
- the high stress of NPP staffs,
- the number of tasks to be done by the NPP staffs,
- the impossibility, for rare events, to generate experience or training for operators actions (no observation of success/failure probability (e.g. simulator),
- the possible lack of written operating procedures,
- the possible wrong information in the MCR or maybe the destruction of the MCR,
- the methodologies applicable to model mobile barrier installation (for slow developing event),
- the methodologies available to model use of mobile equipment (pumps, DGs) and conditional failure probability (human and equipment),
- the methodologies applicable to model equipment restoration (long term accident sequences, specific case of multi-units accidents)

Considerations on the Performance Shaping Factors

Any factor influencing the human performance, through mechanisms which may have heterogeneous, endogenous or exogenous nature, is named performance shape factor (PSF). The accurate identification of these PSFs and of their influence mechanisms on the human factor require an interdisciplinary approach that combines sociological, psychological and technical expertise. These factors are evaluated in modeling of the human actions, and provide real opportunities to reduce the likelihood for human errors.

The factors considered to have the greatest effect on the performance are discussed below:

Available time
The available time refers to the amount of time that an operator or a crew has to diagnose and act upon an abnormal event [10]. A lack of time can affect the operator's ability to think clearly, to consider and analyze the alternatives, as well as the ability to perform the required tasks. Time pressure leads to complex situations and high and extremely high stress levels. It is important that the time available and the time required carrying out the action to be considered along with many other factors and requirements of the sequence. The following special considerations can be identified in the external hazard occurrence context: [7]
  o use of less familiar or even different procedure steps and sequencing could change the anticipated timing of actions in response to a man-made hazard;
  o interfacing with other organizations (e.g. fire brigade) working in the vicinity or on the site may delay performing some actions;
  o accessibility issues, harsher environments, and/ or the need for other special tools may impact the overall timeline of how quickly actions normally addressed in response to internal events can be performed under the conditions imposed by man-made hazards;
  o for ex-control room local actions, the available resources, the number and locations of the necessary actions and the overall complexity of the actions that must be taken may have a most significant impact on the time required to perform the actions.

Stressors and stress
The stress is defined as bodily or mental tension and stressor as any external or internal force that causes stress. [2] The environmental factors are often referenced as stress factors (heat, noise, humidity, smoke, radiation or poor ventilation) that can impair thinking or operators physical performances. The stress elements can be classified as elements of psychological stress (load speed, monotonous work) and physiological elements (fatigue, discomfort). The stress can include mental stress, excessive workloads or physical stress (difficult environmental factors). It includes aspects of limited attention, and may induce general fear or panic associated with an event.

The relationship between stress and human performance is curvilinear (figure 1). Both too little and too much stress can lead to a poor performance.
A routine task has usually assigned three stress levels (up to moderately high level). The psychological tests for prediction of performance of the individual in a stress situation have not been sufficiently analyzed and evaluated to be widely used. The highest stress level is dangerous and involves emotional reactions to the situation, even the possibility of losing control. In modeling human performance considering extreme external events, only two levels of stress should be used: high and extremely high, allocation depending on the location and on the dimensions of impact.

**Task complexity**
Most man-made hazard related scenarios may be considered as complex tasks due to multiple induced transients, unavailability of multiple equipment, large number of actions required, misleading or absence of indications, transitioning between multiple procedures and large amount of communication required. Moreover, for local and MCR abandonment actions, the crew may be required to visit various locations that may increase the complexity of the situation.

**Experience and training**
Training or experience should be relevant to the situation of external hazard. Another important aspect to be considered is whether the script is new (whether or not the team or the individual has been involved in a similar scenario, in training or operational situation).

**Procedure**
Specific operational procedures must be set to monitoring real time and human action that follows an external event. [10] This PSF could have negative influence if it has features such as those below:
- Situations where operators are likely to have trouble identifying a way forward transmitted by procedure (ambiguous, unclear or non-detailed steps)
- Requirements to rely on considerable memory
- Situations for which there is no procedure or the procedure is likely to be unavailable, confusing situations, interpretable

**Alarms and indications**
The instrumentation is an important factor in human performance, both for the control room actions and for local actions (indications should be clear so that operators know the status of the plant and when to carry out the required actions). This factor may have a negative influence in the following cases: [10]
- loss of indications, inaccuracies or defects in the instrumentation could lead to wrong actions
- too many changes of indications and alarms or indication in short period (difficult to notice the important indications and alarms)
In case of external hazards, possible erroneous information in MCR could be considered as having a high probability. The human-machine interfaces differently impact the performance of operators, depending on the location of the action [5].

**Communication**
For local actions, this factor may be more important due to possible difficult situations. Communication issues that may impact negatively the response implementation might be: [10].
- Mistakes in communication. The data communicated in verbal instructions are revoked incorrectly by people who implement the instructions
- Inadequate hearing. Information or data sent by an individual are not heard as it was intended (e.g. environmental noise, inadequate channels (radio, mobile)).

**Accessibility**
Especially for local actions, the availability and operability of the equipment needed to be manipulated cannot be always ensured. After a human induced external event, there is the potential that the personnel path to the component location will be blocked and this fact will lead to a delay or inability to reach the action location. Where alternative routes are possible, the demands associated with identifying such routes and the extra time associated with the use of the alternative routes should be factored into the analysis. Sometimes, unique fitness needs are needed to access the important components: [7]
- having to climb up or over equipment to reach a device because the external event has caused blockage of the initial access path;
- need to move and connect hoses, especially if using a heavy or awkward tool;
- using SCBA, which can be physically demanding and hinder communication.
The mobile equipment should be available at demand and should be located in a safe environment.
The habitability within the building in toxic gases, smoke or extreme heat conditions are to be analyzed and evaluated. [7]

**Results**
To define the human interactions, similar stages as those used in SHARP-1 methodology [6] were proposed:

- Definition and modelling of human interaction events
- Quantification of human failure events (HFEs)
- Recovery analysis
- Review

Consistent with PSA tasks, the HRA stages are intended to emphasize the integration of the HRA into PSA model, with a special focus on the dependencies that exist between human interactions and other events. The four stages should be performed iterative, rather than in a stepwise manner.

**Definition and modelling of human interaction events**
The most important objectives of this stage are the followings:
- To provide an understanding of the context of human interactions
- To understand the impact of the human interactions on accident sequence development
- To incorporate the human interaction events into the plant logic models

Post-initiator operator response can be divided into four stages: detection of a critical situation, diagnosis of the situation, deciding on the necessary actions, and implementation of these actions. The human interactions could be very scenario-dependent, related to actions dictated by plant operating procedure or related to recovery of failed equipment, establishing cross-connection within units, repairing equipment, etc.
The human interactions could be incorporated in the PSA model in the definition of initiating event and in accident sequence development. The interaction ways will be a function of the various conditions that can occur, as defined by the development of the PSA accident sequences and associated equipment
unavailabilities and failure modes. Some of the operator actions may be performed immediately and without regard to the specific situation, while others will be dependent on the plant status and cues. Recovery actions that cannot be performed due to the impact of external hazards of certain magnitude should be removed from the Level 1 PSA model or probabilities of failure whilst performing the action should be increased.

**Quantification of human failure events**

This stage provides as output the probabilities of human interaction basic events (HEPs) for each of HFE, the uncertainties of estimations and whatever revisions to the models are needed to properly account for the final human interaction definitions. The probabilities may be quantitative screening values, or the results of a detailed evaluation.

In conditions of external hazards occurrence, a thorough check and associated adjustment should be performed in relation to recovery actions and probabilities of human errors. All post-initiator human errors that could occur in response to the initiating event, as modelled in the Level 1 PSA for internal initiating events, should be revised and adjusted for the specific external hazard conditions. As a minimum, the following induced effects on the operators performance shaping factors should be taken into account:

- Availability of pathways to specific structures, systems and components after an external hazard occurrence;
- Increased stress levels; Compared to accident scenarios caused by internal initiating events, the operators stress levels and conditions in the plant may differ considerably after an external initiating event.
- Failures of indication or false indication;
- Failure of communication systems.

There are likely to be interdependencies between the individual human failures events included in the logic model. Such interdependencies could arise from the use of a common cue or procedural step, incorrect procedures, an incorrect diagnosis or a plan of action in carrying out response actions, etc. Dependencies among human failure events in the same sequence, if any, can significantly increase the human error probability, and they should be identified and quantified in the analysis. Proper consideration of the dependencies among the human actions in the model is necessary to reach the best possible evaluation of both the relative and absolute importance of the human events and related accident sequence equipment failures.

Whether it use conservative or detailed estimation of the post-initiator HEPs, the evaluation should include both diagnosis and execution failures. Diagnosis tasks consist of reliance on knowledge and experience to understand existing conditions, planning and prioritizing activities, and determining appropriate courses of action.

Criteria for selecting or modifying the HRA models include availability of data, experience of the user with the model, importance of the action being modelled and the correspondence between the key influence factors identified for the human interaction and parameters used as input to the quantification model (e.g. such as the time available to complete the action).

Some performance factors may affect the decisions taken, while other influence factors will affect only the value of the human interaction probabilities. The plant model should be revised to account for additional scenario dependencies on human interactions which were not considered previously.

**Recovery analysis**

The recovery actions are identified for the scenarios, judged as feasible, explicitly defined and quantified, and are referring to other reasonable actions the operators might take to avoid severe core damage and/or a large early release (they are not specifically modeled already). The failure to successfully perform such actions would subsequently be added to the accident sequence model thereby crediting the actions and further lowering the overall accident sequence frequency because it takes additional failures of these actions before the core is actually damaged.
Usually, the possibilities to worsen an accident by the operators, as the possibilities to perform recovery actions unplanned are omitted from the model.

The following issues should be considered in defining appropriate recovery actions:
- whether the cues will be clear and provided in time to indicate the need for a recovery action
- whether the recovery is a repair action of a failed equipment
- whether sufficient time is available
- whether sufficient crew resources exist to perform the action
- whether there is procedure guidance to perform the action
- whether the crew has trained on the recovery action including the quality and frequency of the training
- whether the equipment needed to perform the action is still accessible and in a non-threatening environment/ location

The influence factors may not only increase the time to complete the tasks but also cause unsuccessful recoveries.

The possibility to use mobile equipment (pumps, DGs) should be considered.

Another important point in modeling equipment restoration is the consideration of shared resources in case of multi units (e.g. management difficulties, sharing of human resources and equipment).

**Revision**

This step includes the revision of validity and completeness of the results obtained in the first stages of the procedure.

**Conclusions**

✓ The success of operator actions in mitigation of the external event consequences is a subject that has grown in importance after the Fukushima accident. Many international organizations have begun to develop approaches to analyze the reliability of operator actions after the occurrence of an extreme external event.

✓ The general procedure and the major analysis steps in HRA within a PSA for man-made hazards are actually in good agreement with the ones of HRA in general. However, some specific analysis tasks need particular attention or even further developmental efforts, especially regarding the identification of external performance shaping factors.

✓ The complex environment created by the occurrence of an external event can be modelled successfully, if there is an appropriate knowledge of the main factors that have an impact on the human factor, leading directly or indirectly to human performance.

✓ The estimated probability of human errors, are scenario specific and it reflects factors that may influence the performance of operators, including stress level, time available to accomplish the necessary tasks, availability of operating procedures, the training provided, equipment availability, communication, etc.

**Acknowledgement**

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BUILDING LOCAL PARTICIPATION AND SUPPORT FOR ALFRED PROJECT

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ABSTRACT

At the beginning of 2015, a Local Dialog Group (LDG) was set-up under the FP7 ARCADIA project, as an interface between the implementer of ALFRED demonstrator (Fostering ALfred CONstruction - FALCON consortium) and local community from Mioveni town. The paper discusses the current situation and the possible evolution of the group beyond the ARCADIA project (November 2016). The possible evolution could be the transformation in a Local Committee (LC), this fact leading to the extension of LDG’s actual role, towards a larger participation in decision making process (DMP). Aspects regarding LC construction elements as local context, mission and role, legitimacy, funding and resources, organizing mode, products and outputs, the interaction with other stakeholders, are presented in the paper. In order to describe the context, some general elements concerning the Local Committees (LCs) as tools for the communities’ involvement in the DMP are also introduced.

Key words: ALFRED, participation, Local Dialog Group, Local Committee, Decision Making Process

Introduction

During the last decades the approach of public involvement in the decisions related to nuclear investment has progressively shifted from public acceptance approach to a fair and real participation. Nowadays the efforts of the nuclear industry and RDI organizations are not oriented to convince the public and other stakeholders (like non-governmental organizations - NGOs, local authorities, politicians, etc.) to accept a nuclear investment, but rather to involve them in DMP. It is a smart way to share the responsibilities and to build sustainable decisions.

If years ago the NIMBY (not in my back yard) syndrome was the main difficulty of the interaction between the implementers and potential hosting communities, today the participation approach introduces more complexity and higher efforts. Due to this fact some of the implementers insist yet to build a reasonable acceptance level by using the old classical methods and tools.

In Romania the re-construction of the democracy after the communist regime has introduced some additional difficulties, mainly related to the effects of the inheritance of the organizational culture and due to the lack of the practice of the dialogue between citizens and policy-makers. However, during the last decade, stimulated by the integration process of the country into the European Union, both the public and nuclear organizations have progressively understood the important role of the construction of a continuous dialogue aimed to create the conditions of a real participation.

Romanian communities hosting the nuclear power plant (NPP) and the future low and intermediate level waste (LILW) repository (Cernavoda, Saligny respectively) were involved in the activities of some
international projects (COWAM2 [1], CIP [2], IPPA [3]) and succeed to learn about the experience of other countries and about the methods and tools to be used. Institute for Nuclear Research (ICN in Romanian) acted as a pioneer in this field and as a national facilitator. Based on this experience the involvement of the public in ALFRED project was started in the early phase (at the level of strategic planning).

ALFRED is a demonstrator for Lead Fast Reactor (LFR) technology supported by the European vision of Sustainable Nuclear Energy Technology Platform (SNETP) [4] and by Generation IV International Forum (GIF) [5]. In 2013 FALCON consortium aimed to prepare the future implementation was set-up by the Ansaldo Nucleare (Italy), ICN, and ENEA (Italy). In 2014 CVR (Czech Republic) joined the consortium. According to the Consortium Agreement the reference site is considered at Mioveni nuclear platform. Therefore, Mioveni town is the hosting community and a process to involve the representatives of the community in the implementation decision was started.

The Local Committee – a tool for community to participate in DMP

Usually, the local communities are solicited and contacted by national authorities, waste management or other important institutions involved in nuclear field, when a new facility is going to be host on their territory (new reactors, waste repositories, etc.). This hosting can affect the life of community for years to come. In the last time, authorities and other stakeholders have understood that the public participation is requested by national legislative framework and international conventions (like Aarhus and Espoo). But, in many cases, the involvement of public is strictly kept at the level mentioned in the national laws. The public information and formal public hearings during Environmental Impact Assessment (EIA) and Strategic Evaluation Assessment (SEA) are seen and considered as being enough, due to the cost and the risk to delay the nuclear facility implementation process. On the other hand, many stakeholders simulate the agreement of public participation.

However, a positive new trend concerning communities involving exists, the Local Committees. They are aimed to stimulate the DMP participation as early as possible and to ensure the voice of community will become important, active one and heard by the other stakeholders. The LCs can provide a forum for community discussion concerning nuclear related issues (new facilities implementation, how waste is going to be managed, etc.) “gather and disseminate information, follow scientific research performed by other players, develop and deliberate on solutions to address community impact of an installation, give recommendations to other players, monitor other players’ performance” [6] and can make some part of the complex socio-technical decisions involving nuclear related issues. The LCs can be organized spontaneously (by residents) or can be created officially (by national law or arrangements). They can be set-up to have one, two or all the following competences [6], depending on the potential to influence the decision making process:

- gather information from the implementer or central authorities and disseminate it to the community;
- give advice to the implementer and/or to other decision makers;
- has decision making authority;
- has the ability to grant or refuse legal authorizations;
- has the capacity to repeat some of the technical studies through commissioning independent studies or reports.

An important experience concerning LCs in different European context exists and examples of best practices in applying local democracy to nuclear related issues can be found in [6]. Many LCs are information committees. For example, in France, there are Local Information Commissions (LIC), foreseen by national legislation, around all nuclear sites which enable local communities to discuss nuclear issues together with the site operators and official inspectorates. The Commissions do not have the power of decision, but the power of discussion. They disseminate information from different sources in order to facilitate public debate, may give their own opinion and make recommendations to assist site
operators and public authorities in their decisions, by representing the concerns of the local area [7]. Also, they have the ability to assure a continuous monitoring of nuclear sites and of other nuclear facilities from their area.

A successful model of LCs is Belgian Local Partnerships: STOLA in the municipality of Dessel and MONA in the municipality of Mol. The partnerships are independent decision making bodies. Each organization received federal funds of a 250,000 EUR yearly budget through ONDRAF/NIRAS (national radioactive waste management agency) which they use according to their convenience. Mol Consultation on Nuclear Waste (MONA) is a partnership between Mol municipality and ONDRAF/NIRAS, and was created in February 2000, in order to consider whether the Mol municipality can accept a low and intermediate radioactive waste repository, and works out technical and social aspects [6]. The partnership consists of a general assembly (36 members) with a broad representation of the local community (political actors, delegates from socio-economic, environmental, cultural and other locally based organizations) and a representative of ONDRAF/NIRAS; an executive committee (10 members and a representative of ONDRAF/NIRAS); two full-time project coordinators; and four working groups (10 - 15 members each): siting and design, environment and health, safety assessment and local development. Each working group comprises representatives of the political, economic and social organizations that founded the partnership and individual citizens who took an interest in the debate. All participate on a voluntary basis.

Taking into account the local context and the potential to influence the DMP, in the ladder of citizen participation (figure 1), the LCs can be located on level 3 (information committees) or level 4 (local partnerships).

![Fig. 1. Levels in the ladder of citizen participation](image)

A description of the methods and tools used for each of the 5 levels can be found in [8].

**The Cernavoda Local Committee**

The Cernavoda LC was formally created on October 21, 2004 [6] based on a protocol between the mayor of Cernavoda, local councilors (local authority) and 8 local non-governmental organizations. The intention was to inform the public from this community about the impact of nuclear power plant and intermediate dry storage for spent fuel presence and to act as an important player in DMP concerning RWM issue. As an interface between the community and the Romanian nuclear authorities, another role of Cernavoda LC was to promote the local development (economic and social), to discuss on the health of the people and environmental protection in connection with the nuclear issues. At the beginning, this group was spontaneously organized, by the concerned people who were willing to participate in decision making concerning NPP and RWM issue (especially that of hosting the LILW repository). The LC did not benefit by funds; only free rooms for meetings were provided by the local authority. Even if the LC did not have a legal statute to empower the community to take part in DMP, its legitimacy was gained by the fact that it has represented community diversity and has acted for the community’s interests. The
communication with people from Cernavoda was made by newspapers, radio, TV, public meetings and local information sheets. An important role of Cernovoda LC consisted of the avoiding to remain in isolation. Some actions were organized, highlighting the international conference “Local Competence Building and Public Information in European Nuclear Territories” from April 2006, having the support of Group of European Municipalities with Nuclear Facilities (GMF) and European Commission. Representatives from France, Slovenia, Sweden, Spain, Hungary and Romania shared their experience concerning the impact on the neighboring areas and the demands from the local authorities.

Efforts were made by the National Stakeholders Group (NSG) in FP6 CIP project (2007 – 2010), in order to support the foundation of a LC for information and appropriate decision making regarding the planned LILW repository. In the first meeting of NSG (NSG1), all participants agreed on the interest for discussion about the LC for Cernavoda-Saligny area. In NSG2, a first draft of the LC for Cernavoda area was presented. It included a list of stakeholders, the role of the LC, the major mission, the functioning and funding aspects. The proposed role of the LC was [9]: the increase of the local voice; continuous dialogue between the actors involved in the radioactive waste management; creation of an integrated vision of the local perspective; debate of the technical program of LILW disposal based on presentations easy to be understood by common people; decrease of the stressed situations and relationships. The proposed mission of the LC consisted in information – debates – influence of the decision making process. Concerning the functioning, a Steering Committee with 11 members led by a President elected for three months was proposed. Working Groups centered on environment and local development, legislation and public information was also proposed to carry out the LC activities. The planned LC was not designed as a legal entity, therefore the solution was that owner of Cernavoda NPP (SNN) and National Agency for Radioactive Wastes (ANDRAD) to give money for LC functioning. Due to some insurmountable difficulties, political aspects, different interests of the two communities, etc., the creation of LC was delayed.

In 2011, Cernavoda NPP decided to complete its communication and community consultation programme by setting-up the Council for Information and Community Consultation (CICC). This Council was created in order to identify the problems, concerns and community interests and to provide for Cernavoda NPP consultations, advices, opinions on community expectations in all areas of interest in order to continuously improve the activities from the site and to contribute to community welfare. A statute, functioning rules, different commissions and activities plan exist [10]. The CICC is indirectly funded by Cernavoda NPP based on a protocol of activities. From its setting-up until now, 17 meetings took place, 5 of them only in 2015. Members of CICC, representatives of Cernavoda NPP, a facilitator and different guests participate at each meeting.

**ALFRED Local Dialog Group – a step in involvement of local community in DMP**

As a result of discussions between RATEN ICN representatives, the Local Council (LoC) of Mioveni and the mayor of the city, a LDG was set-up at the beginning of 2015 under FP7 ARCADIA project. Its aim is to create an interface between implementer (responsible entity for ALFRED implementation) and local community and to discuss the sustainability of the development from the local community point of view, the benefits and potential concerns of the public.

Regarding the structure, legitimacy and representativeness of ALFRED LDG, the group consists of 5 persons elected as members of the LoC, 5 representatives of citizens and 3 representatives from RATEN ICN as permanent invited persons.

The activity performed last year is in close relation with the accomplishment of the main objectives of the group which refer to [11]: continuous information of public on the project’s evolution; public participation in decision making process concerning ALFRED; stimulation of a real and prompt feedback from the local community in order to be used by the implementer to follow the local needs.

The necessity of this dialog group creation derived from the following reasons [11]:
- legal context considerations derived from Aarhus convention, especially from the second pillar „participation” involves public „as early as possible”;
- building a sustainable relation between implementer and local community, for identification of the benefits and the best solutions for both parts;
- creation of a starting point for a safe space of expressing opinions, ideas, concerns in relation with a nuclear research infrastructure investment;
- stimulation of the continuous dialog construction between the stakeholders.

ALFRED LDG functions under rules established during the first meeting. A leader and a secretary were also elected during first LDG meeting. At least two meetings with pre-established agenda (including debates on the issues raised by the community or the implementer) are planned during ARCADIA project. Experts from RATEN ICN and from ALFRED working group can be invited, to clarify the problems raised by the group. According to the group evolution, the possibilities to invite other representatives of public and policy makers exist. Also, the group could construct commissions (e.g. environment aspects, community development, risks). Between meetings, discussions with citizens by means of various channels (media, meetings with citizens, local events) are proposed. After meetings, important activities will be focused on the capture of feedback from the citizens, the potential transformation in recommendations to the implementer and the construction of an actions plan (if it is necessary), together with the implementer.

Two meetings took place at the Mioveni LoC headquarter in 2015. First meeting (5th of March) consisted also of 3 presentations and a debate on the following items [12]: existing knowledge about ALFRED project; concerns in relations with investments and perceived benefits for the local community. The presentations approached socio-economic, investment and management aspects of ALFRED project, the most important items which have to be solved in ARCADIA project in order to support ALFRED implementation, and the role, objectives and functionality of ALFRED local group.

Regarding the existing knowledge (“Before the local group construction, did you hear about ALFRED project? From where?”), the members of LDG received information on the project from local and national newspapers and from some public meetings of the Local Council. They also raised some questions of interest for the local community members, in relation to the possible technological transfer to small and medium enterprises from Mioveni, the replacement of existing TRIGA reactor with ALFRED demonstrator and the possible risks of the operation of ALFRED for the inhabitants of Mioveni.

Regarding the concerns in relation with the demonstrator’s construction (“What kind of any concerns do you have in relation with ALFRED construction?”) the answers have shown a low level of concern, especially due to the low risk demonstrated during the 30 years of experience of TRIGA research reactor with no major impact on the population and environment and due to the fact that ALFRED is a new and improved technology with a good level of safety. Other questions referred to the exclusion zone for ALFRED, the design status and the planning for its commissioning.

The final question was in relation to the benefits of this infrastructure from the perspective of the local community (“Do you consider this investment important from the point of view of the benefits for the local community?”). The members of LDG considered ALFRED would be a real chance for new jobs, but young people could be interested if the remuneration level is attractive. Also, the local services could be improved by ALFRED construction on Mioveni nuclear platform.

Some interesting conclusions were drawn at the end of the meeting: a real interest on RATEN ICN activities was observed; a preliminary knowledge about ALFRED project exists; there is a trust in RATEN ICN based on the accumulated experience; there is a perception of a positive impact of ALFRED investment; there is a confidence in Generation IV as a new and improved technology; ALFRED is seen as a chance for new jobs and for improvement of local services.

The second meeting was held on 2015, 26th of August. The initiated debate was based on the following presentations, according to the request formulated by LDG members [12]: Generation IV characteristics; the role of ALFRED and implementation perspective; Information and public participation in
implementation process of a nuclear project; ALFRED – the social-economic impact at local and regional level.

A short presentation of nuclear reactors starting from generation I to generation IV (Gen IV), the main objectives of Gen IV, Gen IV and sustainable development, the perspective of ALFRED implementation and demonstrator’s roadmap were the most important aspects outlined in the first presentation.

Concerning the second presentation, national legal framework regarding public consultation and participation in decision making process, implication steps of local authorities/local community in ALFRED project, aspects of environmental impact assessment and elements for public participation in decision related to ALFRED demonstrator were key aspects.

The third presentation introduced the main benefits of ALFRED implementation and social-economic impact at regional and local level. Stimulation of local development by job creation, by balancing the major role of motor cars industry which is the main economic activity of the town, by creating opportunities for technologic transfer to small and medium enterprises, by stimulation of local services, by increasing of houses values and renting prices; stimulation of education and training and increasing the visibility and reputation of local community, were important elements concerning the social-economic impact at local level.

The debate offered some interesting conclusions as follows: a real interest of LDG members exists concerning the process, tools and methods to create the competences; the people were interested on the approach to stimulate the local recruitment for the new created jobs; planning of the phases of the implementation in connection with the problem of funding was an important issue raised and also the possible difficulties and blockage concern in relation with the development of the investment on Mioveni platform.

**ALFRED Local Dialog Group evolution towards a Local Committee**

ALFRED LDG is intending to play an interface function between the local community and the implementer. The main role is to inform and to discuss all issues in connection with ALFRED, starting with the strategic planning and continuing with the implementation aspects. Actually the LDG is mainly oriented to the information purpose, but in the future it is possible to enlarge the activity and to focus on the DMP, and consequently the transformation of LDG into a LC.

Construction of a LC supposes a set of elements concerning [6]: local context, mission and role, legitimacy (composition and representativeness), funding and resources, organization, products and outputs and external communications (relations with other parties).

*Local context*

The reference site for ALFRED construction is Mioveni nuclear platform, actually incorporating 4 organizations: Institute for Nuclear Research (RATEN ICN), Nuclear Fuel Factory (SNN-FCN), National Company for Nuclear Power Technologies (RATEN) and a branch of National Nuclear Agency and for Radioactive Waste (AN&DR). Mioveni town is located at around 2.5 Km away from the reference site. In this context Mioveni is the most affected community and the key stakeholder in the public participation.

*Mission and role of the LC*

The identity of the LC will be reflected in local context, mission and role. The mission of LC is necessary to correspond to community needs. Taking into account the local democratic culture and experience, population characteristics, social and economic context, the proposed mission of LC is to gather information from the implementer concerning ALFRED investment and project’s evolution, to disseminate the information to the population, to formulate recommendations for ALFRED implementer, and to participate in DMP.
The role we proposed for the future LC is to be an active voice of community in DMP concerning ALFRED implementation, to act as a negotiator in possible disputes, and to propose an integrated vision for the local development taking into consideration the research infrastructure.

Legitimacy
The legitimacy of a LC supposes elements regarding composition and representativeness. The LC should be composed by individuals who can represent their community and cooperate over many years. They could represent different point of view, can even provide their professional experience that could help the LC discussions. The representativeness means to what extent LC can speak for the local people. Persons elected in a democratic process by the community’s people (elected representatives), citizens from local population who work outside of nuclear platform, people with professional experience in nuclear field, members of different local organizations, people who can represent certain point of view, retirees, people from NGOs, etc., can be members of the future LC.

Funding and resources
Sufficient funds should be allocated to finance the involvement of local community. With limited resources, the community voice could be difficult to make heard. The LC has to decide if its members are going to be paid for their time spent. Some LCs consider that activities based on volunteer members could maintain the credibility of community. For our case, an amount of money dedicated to LC activities could come from Local Council budget, taking into account the applicable national law.

Organization
The organization of any LC is necessary to reflect its mission and role. A statute will ensure an increased credibility of the LC, a clearer expression of the public voice and a correct management of the group, including the budget. LC members have to be able to define the role and mission of LC, to elect a chairman or an executive committee, to create groups (e.g. environment aspects, community development, risks) depending on the ALFRED evolution, to define rules of functioning (the frequency for general meetings, for groups meetings, the links between groups, how the information will be spread for community, media, etc.). The members’ responsibility must be clearly defined.

Products and outputs
The LC activity has to produce visible and concrete work on community interests. A web page of the LC, newsletters, posters, brochures, television transmission, issues from the LC groups on the impact of ALFRED construction and operation to be addressed to the implementer/operator, etc., could be considered important products and outputs of the future LC activity. These elements can increase the LC visibility and the community confidence in LC.

Interaction with other stakeholders
A special and careful attention on the relations with other parties (media, stakeholders.) is recommended. The relation with media (local/national) could to be a sensitive and important one. It is good to ensure the fact the information gathered by media reflect the real aspects of the LC activity. An increased recognition of the future LC can be obtained by its links with other stakeholders involved or not in ALFRED implementation: local public, local or/and national NGOs, the mayor of Mioveni, the Local Council (which is a key decision maker by the issuing of the urbanism certificate and by approval of the strategy for local development), the County Council (which elaborates the strategy for the county development), the local politicians, the ALFRED implementer, the Environment Agency (which gives the environment accord), the Romanian Regulatory Body (who gives the licensee permit for siting and operation of ALFRED), RATEN ICN (which can inform the people on the evolution of ALFRED project), etc. The interaction needs to be continuous and from the positions of equal partners in order to accommodate the general interests and the existing specificities of the local context.
Conclusions

(C1) Mioveni town being the community which will host ALFRED, a process to involve the representatives of the community in the implementation decision was started.

(C2) After the first year of functioning, ALFRED LDG members understood the role to support an efficient communication between implementer and citizens, and the interface role in maintaining a permanent contact with the local community concerning ALFRED implementation, transmitting messages and capturing feedback from citizens. Important point of views concerning ALFRED implementation on Mioveni nuclear platform, are expected from the group. The activity will continue this year based on meetings and other activities.

(C3) Actually, the LDG is mainly oriented to the information purpose. The possible evolution of actual LDG towards a LC will focus its activity on the DMP concerning ALFRED implementation.

(C4) The potential solutions regarding the main elements for ALFRED LC constructions are exemplificative, the final decision have to be taken following the discussions with community and other interested stakeholders.

References

CORONA PROJECT – CONTRIBUTION TO VVER NUCLEAR EDUCATION AND TRAINING

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ABSTRACT

CORONA Project is established to stimulate the transnational mobility and lifelong learning amongst VVER end users. The project aims to provide a special purpose structure for training of specialists and to maintain the nuclear expertise by gathering the existing and generating new knowledge in the VVER area.


The selected form of the CORONA Academy, together with the online availability of the training opportunities will allow trainees from different locations to access the needed knowledge on demand. The project will target also new-comers in VVER community like Vietnam, Turkey, Belarus, etc.

Key words: Nuclear Education and Training, VVER Technology, ECVET, lifelong learning, transnational mobility

Introduction

Within European Union (EU) there is a strong need for maintaining and preserving knowledge and nuclear competence including VVER competence. Russian technology is very popular amongst the European countries but except in Russia, it is operated mainly in small countries, which have no enough resources to maintain individually the whole necessary knowledge. In addition, there are approximately 30 VVER units under construction in 10 countries in Europe and Asia. Some of these countries will operate nuclear power plants for the first time and will spend significant amount of resources in education and training in the near future.

With regard to Education and Training (E&T) the key challenge is still to raise the attractiveness for qualified young people of studies and professions related to nuclear technologies. Systematic approaches are under preparation to develop solutions tailored to meet the challenges that nuclear E&T is facing in the near future [1].

The investment of European Commission (EC) in VVER technology competence preservation is beneficial for the entire European community. The integration of VVER knowledge is a must. For this
reason the establishment of VVER Training Center (also called CORONA Academy) is the solution that comes in the exact moment of time.

**Objectives**

CORONA I project showed the sustainability of the idea of establishment and maintenance of a Regional Centre of competence in VVER technology (RCC). It revealed some important advantages of such centre by:

- Contributing to the enhancement of safety and performance of nuclear installations with VVER technology through specialized initial and continuous training of personnel involved;
- Keeping the adequate level of safety culture;
- Contributing to the development of Knowledge Management System for VVER technology;
- Preserving and further developing nuclear competencies, skills and knowledge related to VVER technology, as a technology used in the EU.

After successful completion of the CORONA project most of the Consortium partners expressed their will to continue cooperation with the purpose to make the project idea viable and applied for additional funding from the EC, under the EURATOM program 2014-2015 [2].

The CORONA II project specific objectives are:

- To elaborate a harmonized approach to education in the nuclear science and nuclear engineering in VVER countries to support improving the safety of nuclear installations;
- To achieve co-operation and sharing of academic resources and capabilities at the national and international level;
- To accelerate and optimize the development of competences in the nuclear area to ensure the high quality of nuclear education and training in VVER area;
- To further develop the VVER training infrastructure;
- To promote the implementation of modern training methodologies and technologies, dissemination of experience and best practices in Europe in the field of training;
- To promote the establishment and development of national training systems for the nuclear power sector in the new-coming countries;
- To establish a framework for mutual recognition: Implementation of European Credit System for Vocational Education and Training (ECVET), which is one of the mail goals of EC in education and training area, will be supported through testing of its elements and pilot implementations;
- To integrate VVER education and training with the European education and training in nuclear safety and radiation protection;
- To foster and strengthen the relationship with technology platforms, networks and other organisations in the nuclear education and training sector;
- To enhance knowledge sharing, dissemination and online collaboration through an advanced knowledge management portal.

**Implementation**

The proposed CORONA Academy will maintain the nuclear expertise by gathering the existing and generating new knowledge in the VVER area. It will bring together the most experienced trainers in the different aspects of the area within EU and abroad, thus overcoming the mobility challenge that stands ahead the nuclear education and training community. The selected form of the CORONA Academy, together with the online availability of the training opportunities will allow trainees from different locations to access the needed knowledge on demand. The selected courses will cover the whole range of training of VVER specialists from the university until reaching high professional skills and competences in the area. The CORONA Academy will meet the social goals of EURATOM program by providing
training and source of knowledge also for non-nuclear specialists including school teachers, journalists and government officials.

Networking has been widely recognized as a key strategy for capacity building and better use of available educational resources. Via networking the available expertise, resources, information and facilities can be easily exchanged. In practice, its benefits have been acknowledged, and networks are being established at all levels i.e. national, regional and global levels. Networking might even become more important in the future, both in terms of the extent and depth of co-operation.

Networking establishes and promotes national and international collaboration in educational and/or training programmes, bringing a key benefit for many private and public organizations. In addition, national as well as international institutions are aware of the possible lack of nuclear experts and plant operational staff (at both higher education and technical levels) in the coming years. There is a general concern about the maintenance of qualified staff in the areas of reactor systems, spent fuel and radioactive waste management as well as radiation protection. As a consequence, education and training is becoming a key issue for many private and public organizations.

In the frame of CORONA project the training needs of different types of personnel operating and maintaining nuclear installations with VVER technology were analysed in order to develop training schemes and programs, training materials and training tools required to meet the identified needs, to assess the results and recommend additional training tools and equipment for RCC sustainable development.

The following training needs (and target groups accordingly) were identified:

Group A: Specialized training on specific VVER technology aspects for nuclear professionals and researchers
Group B: Basic training on VVER technology specifics for non-nuclear professionals and subcontractors
Group C: Specialized technical training on VVER technology for students studying nuclear disciplines.
Group D: Safety culture and Soft skills training for nuclear professionals and personnel of nuclear facilities suppliers and contractors.

After the identification of the training schemes the objective was to develop training programs and training materials for the target groups as well as to conduct a pilot training. For each target group the following was done:

- Development of training programs and training materials;
- Deliver a pilot training;
- Validate the training program.

Part of the work was the development of the concept for the Knowledge Management (KM) portal, which collects information related to the operational experience of VVER reactors, outcome of the scientific research and its applications in the nuclear industry, technologies and safety requirements and rules, which will contribute to its wide dissemination and application in various countries, operating that type of reactors. The portal is intended to provide information relevant to the key stakeholders of the nuclear power industry such as scientific organizations, academia, industry and government.

Based on the results of CORONA I project it was concluded that the idea for VVER Education and Training Center (now called CORONA Academy) has a great potential for development and has to be explored further. After a numerous discussions held during the meetings of CORONA between partners the idea was transformed and enhanced.
The CORONA II project has started in September 2015. Nine organizations from seven European countries are involved in the project, which is scheduled for a total duration of 3 years. It consists of the following activities:

1. Evaluation of the training schemes and training programs for possible improvement, review and enhancement of the training materials developed during the previous project, evaluation of the courses and providers, and revision, exclusion, inclusion of new ones. Implementation of corrective measures identified during the previous project.
2. Extension of KM Portal.
3. Selection of appropriate qualification and development and application of the set of activities towards pilot implementation of ECVET.
5. Establishment of CORONA Academy.
6. Increased outreach, inclusion of partner training institutions and establishment of regional training hubs
7. Development and pilot implementation of public outreach programmes to increase awareness of VVER technology

It commenced with analysis of the pilot trainings’ evaluation and proposed corrective measures developed as part of CORONA I project. Based on this analysis the partners are in the process of elaborating a list of training schemes, programs and courses which should be improved or newly developed in order to make an explicit and comprehensive set of training programs, which cover all areas of training courses necessary for training of the target groups.

A pilot implementation of ECVET system [3, 4, 5] is planned as part of the work on the project. ECVET is a system of accumulation and transfer of credits designed for vocational education and training in Europe. It enables recognizing and recording of the learning achievement/ outcomes of the individual engaged in a learning pathway leading to a qualification, a vocational diploma or certificate. ECVET is based on the description of qualifications in terms of knowledge, skills and wider competences, organised into units (that can be transferred and accumulated), and the allocation of credit points to qualifications and units depending on their relative weight. The tasks in question are the design of qualifications, the design of formal or non-formal programmes, the design of assessment/validation processes and procedures, delivery of education and training programmes, assessment and recognition of learning outcomes [4].

ECVET is applicable for all learning outcomes which should in principle be achievable through a variety of education and learning paths at all levels of the European Qualifications Framework (EQF) for lifelong learning. EC recommends that member states create the necessary conditions and adopt measures, in accordance with the national legislation and practice and on the basis of trials and testing for ECVET to be gradually applied to VET qualifications at all level of the EQF and used for the purpose of transfer, recognition and accumulation of individuals’ learning outcomes, achieved in formal and where appropriate non-formal and informal contexts [3].

Leadership has been demonstrated to be one of the most important aspects in the enhancement of safety and culture in nuclear organizations. The way management behaves and transmits their beliefs has a huge impact on others’ behaviours. This is a key factor in the effort of improving safety. The goal is to create an academy for managers, from field supervisors to Chief Executive Officers (CEO), in which they can develop and grow their capabilities as safety leaders. In order to achieve this goal it will be necessary to identify the existing development programs on leadership for safety. Then, taking into account the best practices and theories on leadership for safety, specific leadership development schemes will be elaborated. The development process will be designed to include the particular necessities for each of the leadership levels. Special attention will be paid on training capacities to become the next level leader and
how the transition will be successfully achieved. This activity will result in a leadership pipeline for nuclear organizations.

Based on the evaluation of the results of CORONA I project it was concluded that is necessary to strengthen focus on practical training. Human factors simulator (HFS) becomes a powerful tool for training of plant personnel to communicate expectations and practical expertise. After achievement of the optimal knowledge and skills during the theoretical training, the second step is to start with on the job training independently of the training level (basic, intermediate or advanced).

One of the objectives that the consortium wants to achieve with the HFS is to have a specific and complementary resource oriented to practice human performance tools and human behaviours modelling tools. Strengthening and improving attitudes, practices and behaviours of staff to fulfil the expectations and standards established in the plant have to be a result achieved by the practical training conducted at the HFS.

The main objective of using the Training Station is to know and train the different expectations of the plant in a dynamic way, also the objectives and procedures of different areas and organizational units and also make an assessment to see if the nuclear professionals really understand the objectives of each task of the plant. Several indicators that will be defined during the project will help to measure the results after training using the HFS.

The Consortium is focusing its efforts on using the most advanced ways of providing training to the trainees saving cost and time – distance learning and e-learning approaches will be tested in CORONA II Project.

The knowledge management portal, which development started within CORONA I project will be enhanced to include new features and various information for user’s benefit. It will integrate the information on VVER web into a single communication system and develop and implement a semantic web structure to achieve mutual recognition of authentication information with other databases. The structure of the knowledge portal will ensure the possibility of creation, filling-up and further development of the database on available training programmes, training materials, training and methodological documents to be shared by all the countries using VVER technology in the process of nuclear facilities personnel training and advanced training. That will enable the partners to share the materials available in each specific training centre.

The key components of the knowledge management portal are

<table>
<thead>
<tr>
<th>Information</th>
<th>Education</th>
</tr>
</thead>
<tbody>
<tr>
<td>News</td>
<td>i. Training resources from Project CORONA</td>
</tr>
<tr>
<td>VVER Reactor Information</td>
<td>ii. Links to external training and education providers</td>
</tr>
<tr>
<td></td>
<td>iii. Knowledge Resources</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Collaboration</th>
<th>About Project CORONA</th>
</tr>
</thead>
<tbody>
<tr>
<td>Discussion Forums</td>
<td>i. Overview</td>
</tr>
<tr>
<td>Blogs</td>
<td>ii. Work Packages</td>
</tr>
</tbody>
</table>
Social networks

iii. Project Participants

iv. Achieved Results

v. Reference Documents

vi. Contact Us

The involvement of lecturers and trainers representing different institutions and countries will take into account the different approaches applied in education and training in different countries thus providing the trainees with broader competences and enhanced flexibility.

The project will apply the most advanced approaches to coordinate the project specific tasks and actions with the similar European actions taken in the field (by other European projects).

In order to meet the end users requirements and needs and the proposed solutions for the E&T framework in VVER area the following actions are intended to be done in coordination by the working groups:

- assess the training needs of the VVER field;
- consolidate E&T programmes necessary to meet stakeholder requirements and addresses the human resources required to deliver E&T and KM;
- evaluate the need for facilities to support Education & Training;
- perform a gap analysis between demand and offer;
- perform pilot implementation of ECVET;
- outline the basic requirements to improve the current situation;
- ensure co-ordination within EURATOM and HORIZON 2020 E&T activities.

The preservation of the greater part of CORONA Consortium partners will ensure succession of the project activities and initiatives and will serve as a base for the members to continue and strengthen the networking and coordination of their training programmes and policies. On the other hand the newly involved partners will contribute to the implementation of the envisaged activities with their solid knowledge and experience in the area.

The proposed structure of CORONA Academy is shown in Fig. 1.
Dissemination activities

A large amount of the dissemination activities is necessary to be done in order to promote and to present the CORONA II project objectives and results, as well as the Community’s contribution, and to raise awareness of the project activities in order to make it successful and sustainable.

For the purposes of CORONA II project the following main target groups were identified:

- CORONA II beneficiaries;
- European Commission;
- Basic stakeholders: IAEA, WANO, regulatory bodies, vendors, utilities, technical support organizations, universities involved in the VVER operation;
- Non-governmental organizations;
- Academic circles;
- General public - a heterogeneous group consisting of the countries’ population;
- Media representatives.

This separation is based on the specific role of each group in relation to CORONA II project, as well as on the difference in the level of information, the level of interest, and the preferred communication channels.

The main objective of the exploitation of CORONA II project results is to guarantee the transfer of project outcomes beyond its lifetime, and among and beyond the consortium members. The developed and tested training schemes in the frames of the project will take into account the different approaches applied in VVER education and training thus achieving synchronization between the training policies of the partners. The difference in the education and training approaches and the cultural attitudes across the European Union and Russia, being main reactor designer and vendor, will be considered during the development of the programmes. The training schemes will create an environment where the partners can discuss and evaluate how to integrate the different perspectives as an integral part in the curriculum.

As a result the project will unify the existing VVER related training schemes according to the IAEA standards and commonly accepted criteria recognized in the EU. Furthermore this will aid to the cross-
border mobility within the EU contributing to the flagship initiatives within the Europe 2020 Strategy for smart, sustainable and inclusive growth [6].

The pilot implementation of ECVET within CORONA II project will enable recognition of learners’ achievements during periods of mobility. In the context of international mobility but also mobility within countries, ECVET implementation is aimed to support recognition of learning outcomes without extending learners’ education and training pathways.

The knowledge management portal will integrate the information on VVER web into a single communication system providing the prerequisites to capture the existing VVER related knowledge and to store and retain the verified one, allowing timely and easy computer based access.

References

[1] Strategic Research and Innovation Agenda, February 2013
[4] Necessary conditions for ECVET implementation, CEDEFOP
ASSESSMENT OF THE IMPLEMENTATION RISKS FOR ALFRED DEMONSTRATOR

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ABSTRACT

The risks related to the implementation of ALFRED demonstrator in Romania are discussed. The assessment is based on the risk matrix approach. Two groups of experts were used in the investigation: participants in FP7 ARCADIA projects and members of the FALCON consortium. The results consist of the hierarchy of the risks obtained based on the appreciations of the two groups. They are comparatively presented and discussed in terms of the identified critical risks and possible measures for prevention and mitigation. Additionally, some elements derived from the experience of similar project such as SUSEN and ELI-NP are discussed.

Key words: ALFRED, risk assessment, prevention, mitigation strategy

Introduction

According the consortium agreement of FALCON [1] Romania is considered as the reference site for the LFR demonstrator, ALFRED. Due to a set of advantages such as the existing research infrastructure, more than 40 years experience in the nuclear field, the proximity of specialists, the good connection to networks, and the acceptance of the public the nuclear platform Mioveni, Arges County, is considered as the reference site for current stage of the Viability Phase of ALFRED [2].

The phases of ALFRED project as defined by [2] are: Viability Phase, Preparation Phase, Construction Phase, and Operation Phase. The paper is devoted to the assessment of the risks for the Viability, Preparation, and Construction Phase. Therefore, the risks resulted from the operational phase are not included in the present analysis.

Generally, the risks of nuclear investment are discussed from different perspective (return of investment, strategic national interests, sustainable development, etc.) In [3] the discussion of the risks is devoted to nuclear power plants concluding “the investors in new NPP are exposed to financial risks from inaccurate estimates and from changes in markets in the future” and it is based on some facts such as: the cost of the produced electricity is dominated by the cost of the capital for the NPP; duration of the implementation phase is greater than 7 years; payback times can be 30 years or more.

In [4] it is appreciated that “long-term electricity price risk creates bias against high capital cost technologies such as nuclear”. The case of demonstrators is quite more difficult since the nature of such projects (novelty of the technology, still open issues, reliability of innovative components and equipment, licensing duration susceptible to delays, etc.). Moreover, the demonstrators introduce important difficulties in financing process since the nature of the investment.

In [5] the main barriers for raising finance for First-Of-A-Kind demonstration projects were identified as:
(1) high risk nature of the First-Of-A-Kind projects;
(2) lack of supporting policy and regulatory frameworks that would allow to build a commercially viable business case;
(3) lack of coordination and complementarities between financing instruments from EU, Member States, and technology promoters;
(4) lack of financial and technical advice to technology developers and investors, respectively

Even in such favorable conditions nuclear demonstrators will not be built unless there is investment. According with [6] “governments support new build, but do not create the criteria for success”. Comparing the investment in nuclear, coal and gas plant [6] concludes: “nuclear projects are capital intensive and have long project schedules, gas plants are fuel intensive, and coal plant are balanced”.

The methodology

The list of risks was built based on brainstorming approach in the frame of ARCADIA project [7]. The risks were codified and grouped into 7 categories as follows:

- Technical risks (1.1 to 1.23),
- Financial risks (2.1 to 2.15),
- Political risks (3.1 to 3.10),
- Market risks (4.1 to 4. 4),
- Management risks (5.1 to 5.26),
- Relationship risks (6.1 to 6.12),
- Governance risks (7.1 to 7.9).

All the 99 risks were considered in the investigation. The definition of them is detailed presented in [8].

The assessment approach is based on matrix of the risks [8]. It consists of a matrix with 5x5 elements corresponding of the level 1 to 5 for the likelihood (perceived frequency of the risk) and the impact (perceived importance of the effect of the risk). Each respondent attributed a mark (1 to 5) for each risk according with own perception based on the experience and expertise in the field:

- likelihood (1 to 5 from, from Non-credible risk to Highly likely),
- impact (1 to 5, from Negligible to Very high impact).

The risks were evaluated by 48 experts [8]. The data were collected by the representative of each organization in ARCADIA-WP3 by using face to face method. In few cases the questionnaire was answered by each respondent based on the supplied questionnaire and associated Excel sheet; in this case the instructions were discussed before, in details, with each respondent in order to reduce any misunderstandings. Data were collected between April and July 2015. For each set of data, a separate Excel sheet was produced. For each risk the two scores - Lik(i,j) and Imp(I,j) -were transformed into a mark based on the relation: $S(i,j)=2^{(Lik(i,j)-1)}5^{(Imp(i,j)-1)}$

Finally, the answers were converted in a total score by summing all the appreciations for each analyzed risk, and into an average score. Since for some risks, some respondents didn’t offer any answer in the averaging operation these answers were eliminated. Therefore, the number of total respondents were replaced with the number of respondents offering non-zero answers.

Results and discussions

The results were grouped into categories: (C1) ARCADIA group, including all 48 respondents; (C2) FALCON group, including only the members of FALCON participating in the investigation. The analysis of the results has produced:
- the hierarchy of risks,
- the analysis of the explanations for high marks
- the identification of some prevention and mitigation measures.

For C1 the detailed results are presented in [8]. In Table 1 the more important risks according with the results of the investigation are presented (the critical risk group). A complete hierarchy for C1 is presented in [8]

**Table 1** The critical risks for ALFRED implementation in Romania – Comparative results: (C1) ARCADIA and (C2) FALCON

<table>
<thead>
<tr>
<th></th>
<th>(C1) ARCADIA</th>
<th>Risk (averaged)</th>
<th>Corresp.</th>
<th>(C2) FALCON</th>
<th>Risk (averaged)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Difficulties to ensure the pre-financing planned amounts</td>
<td>51.13</td>
<td>1-1</td>
<td>Difficulties to ensure the pre-financing planned amounts</td>
<td>68.00</td>
</tr>
<tr>
<td>2</td>
<td>Delays and costs overruns</td>
<td>46.04</td>
<td>2-9</td>
<td>Unavailability (in time) of the equipment, components, materials</td>
<td>56.33</td>
</tr>
<tr>
<td>3</td>
<td>Maturity of the technology is not reached</td>
<td>43.78</td>
<td>3-20</td>
<td>Unavailable infrastructure for pre-licensing and licensing</td>
<td>56.00</td>
</tr>
<tr>
<td>4</td>
<td>Underestimation of expenses</td>
<td>41.34</td>
<td>4-5</td>
<td>Lack of some funding sources</td>
<td>53.67</td>
</tr>
<tr>
<td>5</td>
<td>International crisis introducing disturbances in supporting of projects</td>
<td>39.91</td>
<td>5-16</td>
<td>Underestimation of expenses</td>
<td>51.00</td>
</tr>
<tr>
<td>6</td>
<td>Uncertainties concerning the spent fuel and RWM decisions</td>
<td>37.53</td>
<td>6-14</td>
<td>Vulnerability to political action</td>
<td>44.33</td>
</tr>
<tr>
<td>7</td>
<td>Lack of some funding sources</td>
<td>36.62</td>
<td>7-4</td>
<td>Excessive bureaucracy in reimbursement of the eligible expenses for each step</td>
<td>40.33</td>
</tr>
<tr>
<td>8</td>
<td>Licensing approval issues - GIV insufficient expertise at regulatory body</td>
<td>35.97</td>
<td>8-15</td>
<td>Uncertainty in the costs for the equipment, components and materials</td>
<td>38.67</td>
</tr>
<tr>
<td>9</td>
<td>Vulnerability to political action</td>
<td>35.29</td>
<td>9-6</td>
<td>Delays and costs overruns</td>
<td>38.00</td>
</tr>
<tr>
<td>10</td>
<td>Loss of experienced managers and experts due to limited bonus</td>
<td>35.21</td>
<td>3.1</td>
<td>Decreasing of the political support due to inadequate</td>
<td>37.33</td>
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<tr>
<td>11</td>
<td>5.25</td>
<td>Excessive bureaucracy in reimbursement of the eligible expenses for each step</td>
<td>34.72</td>
<td>11-7</td>
<td>3.4</td>
</tr>
<tr>
<td>12</td>
<td>5.27</td>
<td>Inadequate “Human resources” management</td>
<td>34.59</td>
<td></td>
<td>3.3</td>
</tr>
<tr>
<td>13</td>
<td>2.3</td>
<td>Uncertainty in the costs for the equipment, components and materials</td>
<td>34.04</td>
<td>13-8</td>
<td>2.2</td>
</tr>
<tr>
<td>14</td>
<td>5.23</td>
<td>The complexity of the project</td>
<td>33.28</td>
<td>14-19</td>
<td>5.26</td>
</tr>
<tr>
<td>15</td>
<td>3.1</td>
<td>Decreasing of the political support due to inadequate evolution of the project</td>
<td>33.00</td>
<td>15-10</td>
<td>1.2</td>
</tr>
<tr>
<td>16</td>
<td>5.21</td>
<td>Unavailability of personnel</td>
<td>32.90</td>
<td></td>
<td>2.13</td>
</tr>
<tr>
<td>17</td>
<td>5.20</td>
<td>Unavailability (in time) of the equipment, components, materials</td>
<td>32.85</td>
<td>17-2</td>
<td>2.10</td>
</tr>
<tr>
<td>18</td>
<td>3.6</td>
<td>Considering ALFRED as a NPP</td>
<td>32.49</td>
<td></td>
<td>2.14</td>
</tr>
<tr>
<td>19</td>
<td></td>
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<td>23</td>
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</tbody>
</table>

The middle column (between ARCADIA and FALCON results) show the correspondence between the rankings resulted from the averaged perceptions of the two groups. For example, 2-9 means the second ranked risk for ARCADIA is find on the 9th place in the FALCON perception. The cell of the risks that are present in both lists are dashed.

From Table 1 it can be seen that the most critical risk is the same for the both groups: “difficulties to ensure the pre-financing planned amounts”. At the same time, it can be observed that in the ARCADIA list there are 18 critical risks compared with 23 in FALCON list. Due to the point of view of the intersection of the two lists there are 4 risks in the first list that don’t appear in the second one. Three of them are directly related with the human resources: “Loss of experienced managers and experts due to
limited bonus system”, “Inadequate human resources management”, “Unavailability of personnel”. ARCADIA group seems to be more concerned on the competence issues. The last is “Considering ALFRED as a NPP” that is considered by FALCON group as a not critical issue.

Conversely there are 9 risks in the second list that are not present in the first one. The most important difference consists of the risks “Unavailable infrastructure for pre-licensing and licensing” ranked by FALCON group on the 3rd position and not seen as critical by the ARCADIA group. Also it can be seen that the second group is more concerned on the possible loss of the support from different policy-makers (European, national) and changing in the acceptance landscape such as “Escalation of green visions” and financial aspects such as “Overestimation of funding sources” and “Underestimation of operation and maintenance cost”.

In Figure 1 a graphical representation of the ranking produced by ARCADIA group is presented. Similar in Figure 2 the results for the critical risk category, for FALCON group, are presented. As a general trend the appreciations of FALCON respondents are more pessimistic than those of ARCADIA group at least for the first 10 critical risks.

![Fig. 1 ARCADIA group – Relative ranking for the category of critical risks](image-url)
The previous investigations revealing the concerns of the 48 involved experts were completed by desk research and discussion aimed to elicit the difficulties appeared in the implementation of similar projects, mainly in SUSEN [9] and ELI-NP [10]. Both are devoted to research infrastructures and are in relation with the nuclear field. Moreover, ELI project introduces some peculiarities of the Romanian context for projects supported by structural funds taking into consideration the most probable financing scheme for ALFRED demonstrator. The feedback is described in [8].

Important difficulties are mentioned for the procurement process (due to the fact that many devices of the proposed research infrastructure are unique) and delays in the implementation (generated by negotiation process, procurement difficulties, and by an optimistic planning of the steps and activities).

At the same time some good practices were captured such as: the appropriate phasing negotiated from the beginning including the approval of the requested funds, international hiring for scientists as a good solution for human resources needs.

Some elements form the practice of other projects implemented in Romania is presented below:

- generally, in Romania, the projects funded by European funds suffers of delays produced by excessive bureaucracy - many approvals and time consuming from different local and national authorities, postponed decisions and frequent changes of the rules,

- in Romanian context the implementers of projects based on public funds faces difficulties of bids (such as the criterion of the lowest price as the dominant element of the decision; it influences the quality of works, products, services; the frequent litigations delay the implementation).
insufficient personnel for the implementation causing delays or decline of the quality of the results,
underestimation of the planned costs determining the increase of the non-eligible costs and consequently the efforts of the implementers,
sometimes the sub-contractors produce delays and also low quality activities, including the supply of equipment,
- frequent changes in the legislation affecting the project, including the financial planning,

Based on the previous elements (risk identification and ranking, good practices and lesson learnt) the list of critical risks and threats was developed. At the same time mitigation and prevention strategies were developed for ALFRED demonstrator implementation in Romania.

The main risks and threats resulted from the questionnaire based investigation and from the analysis of other project are connected with:

(R1) difficulties with funding, especially the pre-financing resources,
(R2) overruns of the planned costs with significant impact on the local efforts to cover the investment costs in order to achieve the final objectives,
(R3) delays of the project, of some steps or delays in availability of the equipment, components, materials,
(R4) difficulties in the procurement process created by the uniqueness of some equipment (lead pumps, steam generators) and national legal context, stimulated by uncertainties in the costs of the equipment, components and materials,
(R5) maturity of the technology is not reached (present technological limits such as coating of materials for large surfaces, pump fabrication, qualification of the equipment),
(R6) large investment needs for ALFRED and vulnerability to political action,
(R7) decreasing of national and international commitment in case of postponing of tasks and objectives,
(R8) international crisis introducing disturbances in support of the project,
(R9) siting and licensing issues (novelty, request for experimental installations and tests, insufficient expertise at regulatory body, etc.)
(R10) uncertainties concerning the spent fuel and RWM decisions
(R11) insufficient coordinated planning of national and local resources,
(R12) excessive bureaucracy for the projects (approval, monitoring, reimbursement),
(R13) inadequate “human resources” management (unavailability of personnel, quality of the hiring process, loss of experienced personnel),
(R14) considering ALFRED as a NPP

The prevention strategy is presented in Table 2 and the mitigation strategy in Table 3.
Table 2 The prevention strategy

<table>
<thead>
<tr>
<th>Measure</th>
<th>Risks</th>
<th>Phase</th>
</tr>
</thead>
<tbody>
<tr>
<td>1  Transfer of good practices and also analysis of the difficulties in the procurement process (SUSEN, ELI-NP, and from other available international experience)</td>
<td>R2, R3, R4, R11, R12, R13</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>2  Reducing as much as possible the misinformation about costs at the level of the planning and act for a realistic approach in all estimations</td>
<td>R2,</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>3  Phasing for ALFRED and also a stepwise approach for the other supporting facilities, in direct relation with the specific conditions of the national calls for structural funds</td>
<td>R1, R4, R6, R9, R11,</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>4  Organizing real debates involving all stakeholders (including general public and local communities) to allow the civil society to voice the criticism and to support of forecasts; knowledge generated in this way will be integrated in planning and decision making</td>
<td>R14, R6, R7</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>5  Keep a strong connection between implementer, local and national decision-makers in order to produce a better coordination</td>
<td>R14, R6, R7</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>6  Produce realistic planning of the tasks and deadlines in accordance with the predicted resources and a strictly respect of the proposed timeline</td>
<td>R11, R3, R4, R9, R13</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>7  Intensify the RD efforts to clarify the existing open issues and to identify appropriate solutions; keep the interest of international community (including Horizon 2020 framework) on the progress on the field</td>
<td>R5, R9</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>8  Early identification of the peculiarities of the national regulations for licensing and siting, systematic communication with the regulators in the process, initiation of the pre-licensing process, early involvement of the public in the debate and decision making process</td>
<td>R9, R3</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>9  Consistent approach to performance and progress monitoring based on key milestones</td>
<td>R3, R11, R13</td>
<td>Viability Preparatory</td>
</tr>
<tr>
<td>10 Competence building, including the cooperation with universities, started as early as possible; adaptation on national Curricula for nuclear engineering</td>
<td>R13</td>
<td>Viability</td>
</tr>
<tr>
<td>11 Develop a consistent communication programme and promote ALFRED as a demonstrator and research infrastructure</td>
<td>R14</td>
<td>Viability</td>
</tr>
</tbody>
</table>
12. Plan sufficient time resources to analyze the risks and opportunities
   - Risks: All
   - Phase: Viability
   - Preparatory

13. Backup solution for the solution financing based mainly on structural funds
   - Risks: R1, R6
   - Phase: Viability

14. Prepare draft of the main documents before the official start of a call
   - Risks: R12
   - Phase: Preparatory

Table 3 The mitigation strategy

<table>
<thead>
<tr>
<th>Measure</th>
<th>Risks</th>
<th>Phase</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Nominate the risk manager (responsible for risk mitigation plan)</td>
<td>All</td>
<td>Preparatory</td>
</tr>
<tr>
<td>2 Identify and plan the resources to implement the plan for the management of the risks</td>
<td>All</td>
<td>Preparatory</td>
</tr>
<tr>
<td>3 Management team will immediately analyze any deviation from the planning and estimated costs and will identify the appropriate corrective measures</td>
<td>R3, R4, R9, R11, R12, R13</td>
<td>Preparatory, Construction</td>
</tr>
<tr>
<td>4 If necessary, a re-planning of the tasks will be rapidly performed and implementation measures will be ensured</td>
<td>R3, R4, R9, R11, R12, R13</td>
<td>Preparatory, Construction</td>
</tr>
<tr>
<td>5 Creating an effective and flexible team to monitor the evolution of the impacts of the risks</td>
<td>All</td>
<td>Preparatory, Construction</td>
</tr>
<tr>
<td>6 Appointment of a cost control consultant if a significant impact of the escalation of the costs will occur. Performing a continuously updating of the market study. Identification of the solutions that allow reducing the cost while keeping the project within the intended level of quality.</td>
<td>R2</td>
<td>Preparatory, Construction</td>
</tr>
<tr>
<td>7 Solving disputes taking into account good practices in similar projects and situations</td>
<td>All</td>
<td>Preparatory, Construction</td>
</tr>
<tr>
<td>8 Prepare training programme for the new personnel</td>
<td>R13</td>
<td>Preparatory, Construction</td>
</tr>
</tbody>
</table>

Conclusions

(C1) The most important risks are grouped into the category of “high risks”. For ARCADIA respondents the first five positions are associated with: “Difficulties to ensure the pre-financing planned amounts”, “Delays and costs overruns”, “Maturity of the technology is not reached”, “Underestimation of expenses”, and “International crisis introducing disturbances in supporting of projects”. It can be seen a strong connection with financial aspects. For FALCON group the same positions are of the following risks: “Difficulties to ensure the pre-financing planned amounts”, “Unavailability (in time) of the equipment,
components, materials”, “Unavailable infrastructure for pre-licensing and licensing”, “Lack of some funding sources”, and “Underestimation of expenses”. Beside the similar financing concerns the risks associated with delays in preparatory phase appeared.

(C2) Good practices and difficulties resulted from the implementation of some similar projects was identified by desk research and by direct discussions with the management team of SUSEN and ELI-NP projects. These elements were integrated with the list of most important and represent the basis to identify appropriate preventive and mitigation measures.

(C6) Based on the investigations the prevention strategy and mitigation strategy were developed. They are in direct relation with the peculiarities of the implementation in Romanian national context and taking into consideration the reference site.

References


